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Question #26 Supporting References

- C51-1070-E-001, Rev. 27, Elem Diag – Startup Range Neutron Mon Sys Sht. 1 (1 page)
- LGSOPS0074, Rev. 3, Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems (Student Handout) (35 pages)
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- BWR Components Ch. 7, Sensors and Detectors (Pages 122 and 123) (2 pages)
- LGSOPS0074, Rev. 3, Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems (Lesson Plan) (100 pages)
- LGS Daily Chemistry Data (1 page)
- SRM Voltage Regulator Print (1 page)
- SRM Vendor Manual (4 pages)
- LRS-55, Regulated Power Supplies (20 pages)

Question #28 Supporting References

- LGS UFSAR Section 7.5, Information Systems Important to Safety (68 pages)

Question #81 Supporting References

- ST-6-043-320-1, Daily Jet Pump Operability Verification for Two Recirculation Loop Operation, Rev. 45 (14 pages)
- ON-100, Failure of a Jet Pump, Rev. 8 (3 pages)
- LGS Lesson Plan, LLOT-0041A, Reactor Vessel Internals, Rev. 07-05-11 (44 pages)

Question #93 Supporting References

- T-102 Primary Containment Control Bases, Rev. 24 (130 pages)
- LGSOPS0077, Drywell Ventilation Lesson Plan (18 pages)
- P&ID M-77, Sheet 1, Drywell HVAC, Rev. 12 (1 page)

2015 Limerick ILT NRC Exam Post-Exam Challenges

Question #19

Unit 1 plant conditions:

- RPV water level is -150", steady
- Core Spray is maintaining RPV water level
- LPCI 'A' Injection Valve is overridden CLOSED (P-T-L)
- RHR 'B' is operating in Drywell Spray
- Suppression Pool level is 30 feet

PRO is directed to lower Suppression Pool level using T-233 (RHR to Radwaste).

WHICH ONE of the following describes an action that the PRO performs in the MCR to complete this task?

A. OPEN the RHR 'A' HX Inlet (Outlet) Valves, F047A (F003A)

B. START 'A' or 'C' RHRSW Pump

C. Re-align Drywell Sprays to RHR 'A'

D. START 'B' or 'D' RHRSW Pump

Answer: B

Immediately following this Discussion is the Answer Explanation for Question #19 as it appeared on the RO portion of the Approved Exam Key.

Discussion

Of the twelve (12) Applicants, seven (7) chose 'B', the correct answer, while four (4) chose distractor 'A' and one (1) chose distractor 'C'. Distractor 'A' is written as follows:

"OPEN the RHR 'A' HX Inlet (Outlet) Valves, F047A (F003A)"

The procedure which the question is based upon, T-233, "Dumping Suppression Pool Inventory to Radwaste By Way of RHR Loop A", provides direction on lowering Suppression Pool level by aligning 'A' RHR Pump discharge to the Equipment Drain Tank. The first step in the procedure, 4.1, is provided below:

"OPEN the following valves to ensure flow path to Radioactive Waste Tap-Off Point at 10C601 (Main Control Room):

- HV-51-1F047A, "1A RHR Heat Exchanger Inlet" (INLET)
- HV-51-1F003A, "1A RHR Heat Exchanger Outlet" (OUTLET)

The explanation for why distractor 'A' is incorrect is centered around both of these valves being normally open and no information provided in the stem would indicate that the valves had been closed. However, procedure T-233 requires the operator to perform the action "OPEN" in order to correctly complete the procedure step and align RHR to the Equipment Drain Collection Tank. In order to properly implement the procedure use and adherence standards for this step, an operator performing T-233, is required to ensure that the valves have been opened and if not in the desired position then take the appropriate actions to place the valves in the desired position.

Since the stem of the question indicates that the PRO was directed to lower Suppression Pool level using T-233, then the PRO is required to utilize procedure use and adherence standards in accordance with HU-AA-104-101, Procedure Use and Adherence. Other allowances to operate the plant from memory in order to stabilize a plant transient and then follow up with a procedure would NOT be applicable as described HU-AA-104-101 Section 4.8, "Transient Conditions" and OP-LG-103-102-1002, "Strategies for Successful Transient Mitigation" Step 2.8.3.

The standard from HU-AA-104-101, Procedure Use and Adherence, step 4.4.4 covers the execution of procedure steps where the condition of the plant is already in the desired configuration when the procedure step is executed. Step 4.4.4 is written as follows:

> "IF an action or condition is called for by a step is found to already exist, THEN perform the following:
>
> 1. Evaluate any unexpected actions or conditions.
> 2. IF the condition is clearly understood and does not invalidate the intent or scope of the procedure, THEN sign off the step as being completed (or mark N/A if directed by the procedure being performed).

So, an operator working from T-233 would be expected to sign off step 4.1 as *having been performed* after evaluating the conditions and determining that the valves are open. Therefore to the operator executing this procedure there is no fundamental difference in the execution of the step whether a physical manipulation of a component occurs or not. In both cases the "action" in the procedure is the process of verifying the valves open and opening the valves if not already open.

Answer "B" to start the "A" or "C" RHRSW Pump per T-233 step 4.4 is equally correct as "an action that the PRO performs in the MCR to complete this task". The execution of Step 4.4 also requires conformance to HU-AA-104-101 performance requirements and is an executable step of the procedure as noted in the Answer Explanation attached.

Distractors C and D remain plausible but incorrect because both Answers imply the need to utilize "B" RHR to execute T-233. This is not a true condition so both answers remain plausible and incorrect.

Applicant Comments:

Discussions with the applicants revealed that their answers to the question were based on (1) their knowledge that the "A" RHR System was the only loop that could be used to let down the Suppression Pool to Radwaste and (2) the understanding of procedure use requirements made both answer A and answer B correct. Since answer A occurred earlier in T-233 than answer B the four applicants picked answer A.

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Facility Recommendation

Based on (1) identifying that the governing Procedure Use and Adherence standards made the execution of Step 4.1 (Answer A) and Step 4.4 (Answer B) equally correct answers and (2) the question stem did not request the Applicants to select the first action to execute in the procedure (which would make Answer A the only correct Answer) or to identify the first step where a component needed to be repositioned (which would make Answer B the only correct answer), the facility agrees with the applicants and recommends that both Answers A and B be taken as correct answers to this question.

References:

1. T-233, "Dumping Suppression Pool inventory to Radwaste by way of RHR Loop A"
2. HU-AA-104-101, "Procedure Use and Adherence"
3. OP-LG-103-102-1002, "Strategies for Successful Transient Mitigation"

19 **ID: 1098470** **Points: 1.00**

Answer Explanation

Refer to T-233 (its use being directed from T-102 (Primary Containment Control, Step SP/L-11)...Only RHR 'A' is capable of being aligned to Radwaste in order to reduce Suppression Pool water level. Reviewing the actions of Section 4.0, we discover the following (with respect only to the answer choices for this question):

Step 4.1 is a MCR action but is not required because the RHR 'A' HX Inlet (Outlet) Valves are already open (normally-open valves), and nothing in the stem conditions indicates any reason why operators would have been directed to close them before the CRS directed this evolution.

Step 4.4 is a MCR action and will have to be performed. The RHRSW Pumps have no auto-start features; therefore, they did not auto-start in response to the -129" LOCA signal. MCR operators will have to place either the 'A' or 'C' RHRSW Pump in service.

Step 4.5 is a MCR action but is not required because the 'A' RHR Pump is already running...it auto-started on the -129" LOCA signal. The fact that operators have overridden CLOSED (Pull-To-Lock) its associated LPCI Injection Valve (F017A) indicates that the RHR 'A' Pump is still running (on min flow).

'B' is correct: START 'A' or 'C' RHRSW Pump. Correct for the reasons described above.

'A' is wrong: OPEN the RHR 'A' HX Inlet (Outlet) Valves, F047A (F003A). Plausible to the examine who forgets that these are normally-open valves or who believes that one or both have auto-closed in response to the LOCA signal.

'C' is wrong: Re-align Drywell Sprays to RHR 'A'. Plausible to the examinee who cannot recall whether T-233 uses RHR 'A' exclusively (it does) or rather RHR 'B' exclusively (it doesn't). That examinee believes it to be RHR 'B'. Recognizing that RHR 'B' is currently spraying the drywell, the operator would have to swap drywell sprays over to RHR 'A'.

'D' is wrong: START 'B' or 'D' RHRSW Pump. Plausible for reasons similar to those of choice 'C'. In this case, however, the examinee also forgets that one of the 'B' Loop RHRSW pumps was already started when RHR 'B' was placed in drywell sprays.

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Question 19 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	1098470
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LLOT0051.IL10
Topic:	Determine actions to lower Supp Pool level with LOCA signal present using T-233
Num Field 1:	2.9
Num Field 2:	3.0
Text Field:	295029 EA1.03
Comments:	Level: RO Tier: 1 Group: 2 KA # and Rating: 295029 EA1.03 (2.9/3.0) KA Statement: 295029 High Suppression Pool Water Level Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: EA1.03 RHR/LPCI References: T-102, Rev.24 T-225 U/1, Rev.22 T-233 U/1, Rev.14 Examinee References: None Learning Objective: LLOT0051.IL10 Question source: New Question history: None Cognitive level: Higher 10 CFR 55: 41.7, 41.10 Comments: None

Question #19

The licensee recommended two correct answers for Q#19; Choice B – the key answer and Choice A – a 2nd correct answer. The NRC agrees that Choices A and B are both correct. **Choice A and Choice B will each be accepted as a correct answer for Q#19.**

The question provides plant conditions, states that the balance of plant operator (the PRO) is directed to lower suppression pool level using procedure T-233, and then asks the applicant to select the choice which describes an action that the operator *performs* in the control room to complete the task. It is apparent the question author expected applicants to determine that the action described in Choice A did not need to be "performed" since the component would already be in the stated alignment for the given plant conditions. The word, "performed," was intended to mean a required physical manipulation. However, the station administrative guidance on procedure implementation explicitly states that an operator is to sign off a step ***as being completed*** if an action or condition called for by a procedure step is found to already exist. Since the question stem stated the task was being performed using the procedure, it is appropriate for applicants to consider the requirements of the procedure use and adherence standard.

Step 4.1 of T-233, "Dumping Suppression Pool Inventory to Radwaste by way of RHR Loop A" directs the operator to open the RHR HX inlet and outlet valves. These valves should already be open for the conditions given in the stem. It is correct for applicants to consider the action in Choice A (to open the RHR HX valves) as "an action the [operator] ***performs*** ... to complete this task" in the sense described, where the action should be considered performed and complete if the valve is open when implementing the T-233 procedure step. Therefore Choice A is a correct answer to the question.

Step 4.4 of T-233, "Dumping Suppression Pool Inventory to Radwaste by way of RHR Loop A" directs the operator to place 'A' or 'C' RHRSW pump in service. This action has not yet been performed and is required. Therefore key answer Choice B (start A or C RHRSW pump) remains a correct answer to the question.

Choices C and D remain wrong answers.

2015 Limerick ILT NRC Exam Post-Exam Challenges

Question #26

Unit 2 reactor startup is in progress, with the following:

- All IRMs are on Range 4
- All SRMs are reading 600 CPS
- Rod withdrawals are in progress

An RPV instrument dry tube cracks and fills with water; as a result:

- Interior of SRM 'C' detector fills with water

WHICH ONE of the following identifies the SRM 'C' recorder and RMCS Rod Block response?

	SRM 'C' recorder	RMCS Rod Block
A.	Fails UPSCALE	Yes
B.	Fails UPSCALE	No
C.	Fails DOWNSCALE	Yes
D.	Fails DOWNSCALE	No

Answer: A

Immediately following the Answer Explanation is for Question #26 as it appeared on the RO portion of the Approved Exam Key.

Discussion

Of the twelve (12) Applicants, two (2) chose 'A', the correct answer, while nine (9) chose distractor 'D' and one (1) chose 'B'. The question asks the impact of water leakage into the dry tube and subsequently into the fission chamber of an SRM detector on (1) SRM recorder indication and (2) the associated RMCS Rod Block response. The correct answer identified from the answer key is that the SRM recorder will fail upscale and a rod block will occur. This is based upon the BWR Generic Fundamentals (GFE) lesson plan for "Sensors/Detectors" wherein the lesson describes the "wetting" of a fission chamber causing a detector to fail high. With the assumed failure upscale, the SRM recorder would indicate UPSCALE and initiate a rod block due to the all of the IRM's operating below range 8 as given in the question stem.

The information provided in the 'failures of fission chambers' section of the GFE lesson is as follows:

"Wetting – Short circuits due to leaking of water into detector causes detector to fail high"

However, within the same section of the lesson can be found the following:

"Loss of voltage – A loss of high voltage DC to the detector causes the detector to fail to low value"

AND

"Loss of detector gas pressure – The indication is dependent on the number of primary and secondary ionizations that occur from the ionizing event. If the gas pressure decreases, there will be less gas atoms to undergo ionization, therefore less electrons collected. This means a smaller electrical current will exist and the indication will fail downscale."

Based on the information provided in the question stem, the instrument dry tube "cracks and fills with water". The dry tube and the area surrounding the SRM detector are open to the drywell and maintained at drywell pressure which is normally just above atmospheric pressure. Similarly, the SRM detector sits within the SRM shuttle tube which is also open to drywell pressure. In order for the SRM detector to fill with water the SRM detector would have to develop a crack or flaw which allows water to enter the SRM chamber as described in the question stem. This crack or flaw would then allow all of the Argon Gas which is pressurized within the detector to leak out. As noted in both BWR Generic Fundamentals (GFE) lesson plan for "Sensors & Detectors" and LGSOPS0074, FLGS Operations Initial Lesson Plan Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems, the loss of the Argon gas in the detector would cause a lack of ionization in the fission detector and would result in a downscale condition from the SRM detector prior to and concurrent with water entering the fission chamber. (Refer to LGSOPS0074 Page 9 for a description of the fission chamber operation and the function of the Argon gas.)

As water enters and eventually fills the fission chamber the water would cause a short between the high voltage center electrode and the grounded outer case of the SRM detector. Although the GFE materials indicate that wetting of the inside of fission chamber would cause the detector to fail high it is a simplistic approach to this particular failure mode. The GFE material does not take into account the electrical protection for the high voltage power supply for the SRMs. The GFE material is written assuming a constant power supply regardless of the condition of the detector. (Refer to LGSOPS0074 page 23 and 24 for a description of the voltage regulator and high voltage power supply and Reference 2 for a drawing of the fuse protection for each SRM power supply from 1AY-160).

The shorting of the center electrode to the case would induce a current which would pass through the Pulse Height Discriminator and Pre-Amplifier assembly and directly impact the High Voltage Power Supply causing it to fail. With the High Voltage Power Supply impacted by the water intrusion, the SRM would continue to produce a downscale condition. This failure mode was reviewed with the LGS subject matter expert as well as the Exelon Corporate SME for nuclear instrumentation who agreed that the determination that a downscale condition would be produced under the conditions presented in the stem of the question.

With the SRM in a downscale condition, the SRM would <u>not</u> produce a control rod block because as noted in the following locations, Reference 1 on Page 13, Reference 1 Attachment 2 (Page 89) and Reference 1 Attachment 6 (Page 93), the Control Rod Block due to SRM downscale <u>is</u> bypassed when the IRMs are on Range 4 as noted in the question stem.

Therefore the correct response of the SRM to the question conditions is the SRM recorder will display downscale condition and a Control Rod Block will <u>NOT</u> be generated.

Answer "A", the original correct answer, is now incorrect but plausible due to the original reasoning that supported this as the correct answer.

Distractor 'B' is plausible and incorrect since the upscale failure of the SRM detector is a plausible failure mode given the information in the GFE materials. The plausibility of the Applicant not recalling that an

Upscale SRM condition is bypassed with the IRMs on Range 4 remains intact from the original question explanation.

Distractor 'C' is plausible and incorrect since it contains the correct failure mode of the SRM (downscale) and contains a plausible distractor that a Rod Block would occur due to the downscale condition as noted previously.

Answer 'D' is the new correct answer since it has the correct failure mode for the SRM detector and recorder (downscale) and with IRM's on Range 4, they are above the range (Range 2) at which the SRM downscale rod block will be bypassed so a Control Rod block would NOT occur with the downscale condition.

Applicant Response:

The nine applicants who got the question wrong believed that the SRM power supply would trip if the detector was full of water as described in the stem of the question. They all recognized that without the voltage difference in the fission chamber the SRM would fail downscale. Each of these applicants also recognized that with the IRMs on Range 4 an SRM downscale condition would not result in control rod block condition.

Facility Recommendation:

The facility agrees with the Applicants and recommends that the correct answer be changed to Answer 'D' from the original Answer 'A'.

References:

1) LGSOPS0074, Rev. 3,"Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems"
2) C51-1070-E-001, Rev. 27, "Elem Diag – Startup Range Neutron Mon Sys Sht. 1"

26 **ID: 1102330** **Points: 1.00**

Answer Explanation

Per BWR Generic Fundamentals (GFE) lesson plan for "Sensors & Detectors", learning objective #24, the "wetting" of a fission chamber detector causes the detector to fail HIGH (recorder fails UPSCALE). The SRM Upscale (>1 x 10^5 CPS) Rod Block is enabled so long as its associated IRMs are below Range 8. block).

'A' is correct: Fails UPSCALE; Yes. Correct for the reasons described above. All IRMs are on Range 4, so the Upscale rod block function is enabled.

'B' is wrong: Fails UPSCALE; No. Plausible to the examinee who fails to recall the SRM Upscale rod block or who mistakenly believes the Upscale rod block is bypassed with IRMs on Range 4.

'C' is wrong: Fails DOWNSCALE; Yes. Plausible to the examinee who doesn't recall the GFE knowledge regarding the failure modes for fission chamber detectors. That examinee believes the detector (and therefore the recorder) fails downscale, and as such results in an SRM DOWNSCALE (<3 CPS) Rod Block.

'D' is wrong: Fails DOWNSCALE; No. First part is plausible for the same reason as for choice 'C'. The second part is plausible to the examinee who either confuses the SRM DOWNSCALE rod block function (which is NOT dependent on the IRM Range) with the SRM NOT FULL INSERTED AND <100 CPS rod block function (which IS bypassed when the IRMs are on Range 3 or higher), or who simply believes that the SRM downscale rod block function is bypassed with the IRMs on Range 4.

2015 Limerick ILT NRC Exam Post-Exam Challenges

Question 26 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	1102330
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LGSOPS0074.IL6
Topic:	Predict SRM and RMCS response to SRM detector wetting
Num Field 1:	2.9
Num Field 2:	2.9
Text Field:	215004 K6.04
Comments:	Level: RO Tier: 2 Group: 1 KA # and Rating: 215004 K6.04 (2.9/2.9) KA Statement: 215004 SRMs Knowledge of the effect that a loss or malfunction of the following will have on the SOURCE RANGE MONITOR (SRM) SYSTEM: K6.04 Detectors References: ARC-MCR-107, G4, Rev.0; ARC-MCR-107, H4, Rev.1; BWR Fundamentals Chapter 7 (Components), Rev.4 Examinee References: None Learning Objective: LGSOPS0074.IL6 Question source: New Question history: None Cognitive level: Higher* 10 CFR 55: 41.7 Comments: *Justification for HIGHER COGNITIVE categorization: First part requires comprehension of GFE "Components" theory regarding fission chamber "wetting".

Additional Correspondence LGS Operations Training Manager to the Chief Examiner, Related to Post-Exam Comments on Question #26

02/23/2015:

Peter

Attached is the clarification of the following paragraph:

"The shorting of the center electrode to the case would induce a current which would pass through the Pulse Height Discriminator and Pre-Amplifier assembly and directly impact the High Voltage Power Supply causing it to fail. With the High Voltage Power Supply impacted by the water intrusion, the SRM would continue to produce a downscale condition. This failure mode was reviewed with the LGS subject matter expert as well as the Exelon Corporate SME for nuclear instrumentation who agreed that the determination that a downscale condition would be produced under the conditions presented in the stem of the question."

Downstream of the fuses in the SRM Chassis which are seen on Reference 2 of the original submittal there is a voltage regulator which supplies the 15 volt DC to the High Voltage (HV) power supply.

The HV power supply feeds voltage to the Pre-amp.

The pre-amp applies the High Voltage to the SRM detector.

Attached are 4 documents which demonstrate the power supply to the SRM detector. The original GE power supply which feeds the entire SRM drawer was replaced with a Lambda 25705. The Lambda 25705 consists of a Lambda LRS-55 power supply with mounting modifications. The vendor manual for the LRS-55 is attached. Lambda does not supply detailed internal electrical schematics with the vendor manual. The manual does provide a description of the protection features associated with the SRM drawer power supply.

The drawings show the connection of the voltage regulator, the high voltage power supply and SRM pre-amp wiring.

The SRM Pre-amp drawing shows the two 24K resistors (Labelled R3 and R4 at C-2 on the attached drawing). These resistors are in series on the HV input to limit current draw on the HV power supply.

With a normal set-up on the HV Power Supply of **approximately** 480vAC and the detector was shorted to ground, the current would be limited to approximately 10 milliamps (480V /48000 ohm). Operating voltage is routinely kept at approximately 400V.

So with a short circuit on the detector, the HV power supply input would produce a continuous maximum current output. The 480 volt output of HV power supply would be dropped across the 48K resistance resulting in the lowest possible voltage (approximately zero volts DC) supplied to the detector.

With a low voltage supplied to the detector, the detector cannot produce any pulses (counts) because the constant discharge and resultant low voltage does not produce an **additional** voltage change inside the detector due to ionization in the fission chamber. Without a means to create a pulse on the high voltage pin in the detector the SRM would read downscale.

Additionally, the power-supply does have a short-circuit protection feature (page 5 of the LR-55 Manual attachment under "Current Limit and Frequency Shift") which limits the overall current into the SRM chassis should a short circuit exist.

You also asked a question as to how this was different from an APRM where more fissions or more current would result in a larger and potentially upscale condition. For an APRM (which is the section of the GF Materials where the original discussion of a wetted fission chamber was located) because of the power level and number of fissions in the chamber, the total current from the APRMS is measured since individual pulses can no longer be discriminated. So whereas an APRM with a constant discharge would result in a measure of power being high (upscale), for an SRM the result is the exact opposite with a constant current flow. This difference in the operating characteristics between and SRM and an APRM is covered on Page 119 Item 4 for SRMs and Page 120 Item 7 for IRMs and APRMs of the Generic Fundamentals Materials Chapter 7 which were provided previously.

These conclusions were verified by the Limerick Station, Exelon Corporate and General Electric (GEH) SMEs and they all agreed that all 3 factors (p/s protection, pre-amp circuitry & dc signal with no pulses) would occur if water were to enter a SRM detector resulting in a downscale condition.

If you need additional information please let me know and we will get it for you.

Dave Molteni

03/09/2015:

Peter

Here are the responses to your questions from Friday.

1. [Chief Examiner's Question] Your previous comments do not appear to take coolant conductivity into consideration. Please explain the effect of filling an SRM with reactor coolant, with respect to its normally low conductivity. Would the detector voltage drop go to zero as previously described due to short circuit current or would the low conductivity limit the current flow? If current flow were limited by low conductivity, how would the detector behave when filled with water?

   [Response] Unit 1 and Unit 2 have a conductivity goal of <.15 μS/cm and an administrative limit of .30 μS/cm. I have attached our latest Plan of the Day Meeting Info to show you where both units are currently trending.

   The Tech Spec Limit per TRM Section 4.4.4 is 1.0 μmhos/cm. However an immediate action to commence a required shutdown action is not required until conductivity reaches 10 µmhos/cm. The two units are different however conductivity converts at 1 µS/cm to 1 µmhos/cm so the orders of magnitude are the same.

I'm assuming you are asking this in order to confirm / refute the shorting of the SRM detector using reactor water. The item that needs to be considered is if you put a hole in the dry tube (reactor) and allow it to flow and flash to atmosphere any crud or particulate coated in the area of the break would be entrained in the flow through the break. Obviously the quantity of material has too many variables (size of the break, crud levels around the area of the break, etc..) to specifically quantify that value but it would certainly contribute some amount of contaminant into the fluid around the SRM detector in question.

More importantly however is the fact that the assumption / understanding that the water that enters the fission detector is sufficiently conductive is an essential of the original Generic Fundamentals write-up which provided the original basis for the response of the fission detector. Without that condition a high current flow could not be established in the fission chamber.

Therefore establishing that the water which enters the detector is sufficiently conductive to allow current to rise and the voltage to drop across the 48K resistance prior to the high voltage probe in the detector itself is reasonable based on the conditions through which the fluid must pass prior to filling the detector and is consistent with the conditions assumed in the Generic Fundamentals materials which describes the response of the fission chamber when it is filled with water.

2. [Chief Examiner's Question] The discussion in your correspondence dated 2/23/15 stated "the 480 volt output of HV power supply would be dropped across the 48K resistance resulting in the lowest possible voltage (approximately zero volts DC) supplied to the detector." Please explain whether or not an INOP TRIP would actuate RCMS (as shown in LGSOPS0074, pages 88 and 89) on the SRM loss of detector high voltage less than 90% and why or why not.

   [Response] The "Loss of High Voltage" INOP Trip is generated within the SRM draw. A voltage divider within the HV power supply is used to develop a small voltage which is compared to a reference voltage within the INOP trip unit (see the attached block diagram and vendor manual description, the location of the HV power supply is L-4 and the INOP trip unit is F-9). The 48K resistor is located down at the pre-amp. Shorting the detector doesn't cause an INOP trip since the voltage on the HV power supply remains and the entire voltage is dropped across the 48K resistor (current limiter) located at the Pre-amp.

   Without a Loss of High Voltage determination there would NOT be an INOP trip and no RMCS Rod Block would occur.

   If the INOP trip were to occur for another reason, the RMCS Rod Block would occur since IRMs need to be above range 7 to block the INOP trip.

Additional Correspondence LGS Operations Training Manager to the Chief Examiner, Related to Post-Exam Comments on Question #26

> As we discussed, this failure mode of the power supply (supplying current and voltage at the output of the power supply but with voltage dropped across the inline resistors) is the designed failure mode of the power supply with a shorted condition inside the fission detector. This was considered to be one of the primary means that the power supply / SRM would fail as initially noted in our first write-up and this analysis was verified both internally and externally through GE. There are other less likely ways for the power supply to fail but all would require some additional component weakness or failure to become true which we did not assume not was that condition built into the question stem.

I apologize for not being able to turn these around sooner but please feel free to call me if you need anything else to help close the review.

Dave Molteni

NRC Post Exam Comment Resolution
2015 Limerick Initial License Exam

Question #26

The licensee recommended changing the key answer for Q#26 from Choice A to Choice D. The NRC does not agree. The correct answer cannot be determined from the information provided. **Q#26 will be deleted from the exam.**

Q#26 was developed from BWR Generic Fundamentals Sensors and Detectors training material which generally stated that failures in fission chambers includes wetting, where "short circuits due to leaking of water into detector causes detector to fail high." This material was not specifically tied to local power range monitor (LPRM) fission chambers, but appears to apply to LPRMs in that they operate in a current mode where increased current flow across the detector chamber results in higher indicated power. In contrast, source range monitor (SRM) fission chambers, the type referenced in the question stem, operate in a pulse counting mode where the fission fragments produced in the detector ionize a relatively high pressure argon fill gas to produce a large pulse that can be counted by the detector circuitry. A continuous high current in a SRM should not by itself result in the detector failing high. The lesson material does not address differences in response of a SRM as compared to a LPRM to increased current flow.

The licensee argues that wetting of the SRM as described in the question stem will result in the loss of the high voltage drop across the detector. They explain the coolant that fills the detector will cause a short circuit between the center electrode and the outer can, such that all voltage will be dropped across two 24 kohm resistors in the circuitry powering the detector. With all the voltage dropped across these two resistors, they explain that there will be no potential to develop any pulses in the detector. This argument is based on an assumption that the reactor coolant filling the detector is a high conductivity substance able to short out the detector. However, Limerick Technical Requirements Manual Specification 3.4.4 requires coolant conductivity to be less than 2 μmhos per cm at 25°Cin the Operational Conditions given in the question. Typically, the reactor coolant conductivity is at least a factor of ten below this limit.

Conductivity is equal to 1 divided by resistance. Therefore typical coolant resistance is approximately 5 Megohms per cm, resulting in a very high value of resistance even when considering the small, 1/4 inch diameter of the typical SRM detector.

Conductivity is sensitive to temperature variations, with conductivity of ultrapure water increasing approximately 5.5% for every 1°C increase in temperature. The question stem does not provide coolant temperature other than to state a startup is in progress with SRMs reading 600 cps and IRMs on range 4. Boiling water reactors are generally heated up on reactor heat after attaining criticality so it is reasonable to assume in this case that the reactor coolant is at a relatively low temperature. Conservatively assuming a coolant temperature of 60°C (140°F) would yield a maximum increase in conductivity of 200%. Under these assumptions, the resistance could be reduced by as much as 50%, to an approximate value of 2.5 Megohms per cm, still a substantial resistance to current flow. The resistance provided by the coolant would therefore appear to make it unlikely at best that the high voltage drop across the chamber would be lost.

The licensee argues the coolant would become contaminated as it entered the detector, resulting in the detector filling with a highly conductive fluid. This is purely supposition and would depend on a unique set of conditions not specified in the stem. No data is provided to support this hypothesis.

If the detector high voltage remains, then one must question how the detector will behave in the presence of a neutron flux with the fill gas either replaced by coolant or absorbed by coolant. Existing lesson source material cannot be relied upon because, as previously described, it assumes the wetting is the result of a highly conductive fluid.

It seems certain the detector would not fail upscale. It is not at all certain the detector would fail downscale. The ionization behavior of the water will not be the same as that of the argon gas. However, ionization will still occur.

The premise of the question appears flawed as stated by the licensee in their post exam comment. SRMs and Intermediate Range Monitors (IRMs) travel in dry guide tubes, which are vented to the primary containment building. The detectors are housed in shuttle tubes which allow the detectors to be inserted and retracted from the core during operation. The SRMs in particular have a higher pressure fill gas, pressurized to about 220 psi as opposed to IRMs, which are only pressurized to about 18 psi. Given the location of the detectors, in a dry vented tube, and the higher pressure fill gas, it is highly unlikely that the detector would ever become wetted internally with reactor coolant. A coolant leak into the guide tube would leak to the containment atmosphere and the tube would likely remain at a lower pressure than the gas space of the detector such that coolant would/could not leak into the detector.

This question has little operational validity as the cause of detector failure cannot be diagnosed from its response and the cause would have no immediate bearing on operational decisions.

The NRC thinks this question is deeply flawed for the reasons stated above, that 1) the reference source material was not intended to apply to SRMs, 2) low conductivity coolant would not cause a short circuit effect in a detector, 3) source materials do not accurately describe how the detector would behave if filled with low conductivity coolant, 4) detector behavior in the presence of this different fill material cannot be readily predicted, 5) physical construction and placement of SRMs make the described failure mode improbable at best, and 6) the question has low operational validity. Accordingly, the question will be removed from the exam.

<u>References:</u>

1) Limerick Lesson Plan LGSOPS0074, Rev. 3,"Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems"

2) BWR Fundamentals Components Chapter 7 – Sensors and Detectors, Rev 4

2015 Limerick ILT NRC Exam Post-Exam Challenges

Question #28

Unit 1 LOCA is in progress, with the following:

- RPV level is -135", down slow
- RPV pressure is 490 psig, down slow
- 1B and 1D RHR Pumps are running

WHICH ONE of the following will provide **DEFINITIVE** evidence that some amount of RHR Loop 'B' water is actually injecting into the RPV?

A. FI51-1R603B (LOOP B) reads 4,000 gpm

B. HV51-1F041D (INBOARD CHECK) "Disc Pos" indicates OPEN, "Stem Pos" indicates CLOSED

C. DIV 2 LPCI INJECTION VALVE DP PERMISSIVE alarm

D. 1B RHR PUMP DISCH HI/LO PRESS alarm

Answer: B

Immediately following this Discussion is the Answer Explanation for Question #28 as it appeared on the RO portion of the Approved Exam Key.

Discussion

Of the twelve (12) Applicants, nine (9) chose 'B', the correct answer, while three (3) chose distractor 'A'. The question provides a set of plant conditions following a LOCA signal with Reactor Pressure dropping and asks the candidate to evaluate conditions that would provide *definitive evidence* that 'B' RHR is injecting into the vessel. The correct answer identified on the answer key is the INBOARD CHECK valve disc position indicates open, which would identify flow through the piping. Since the question stem provides no indication of RHR operation in any other mode, it is logical to believe that the check valve would actuate to indicate injection as Reactor Pressure lowers below RHR Pump shutoff head with the LPCI valve open. The LPCI valve will open when differential pressure between the reactor and RHR Pump discharge is less than 74 psid.

Distractor 'A' provides a flowrate of 4,000 gpm, indicated on FI51-1R603B (LOOP B), as evidence that 'B' RHR Loop is injecting. The answer key states that the position of the flow indicator is upstream of the injection valve and is not definitive evidence of injection into the vessel. This is due to the fact that there are multiple flowpaths in which flow would be indicated on FI51-1R603B, including Suppression Pool Cooling, Suppression Pool Spray and Drywell Spray. However, the stem provides no indication that 'B' RHR has been manually aligned into any of those alternate alignments (i.e. Suppression Pool cooling, Suppression Pool Spray or Drywell spray). Additionally, no additional failure modes of the RHR system were provided or should have been assumed which would create a diversion of flow from the RHR system which would cause flow to be indicated on the referenced flow indicator (i.e. pipe leak / break, relief valve opening).

With RPV level at -135", a LOCA signal would have been initiated which would start the 'B' and 'D' RHR Pumps and close the suppression pool cooling and spray valves. The other flowpaths which could result in flow through the RHR System all require manual alignment (Drywell Spray, Suppression Pool Cooling, Suppression Pool Spray) and as noted previously the question does not identify operator actions having been taken to place any of those alternate alignments in service.

Without operator action, the pump flow path would be through the Min Flow Valve, HV-51-1F007B, until pump discharge flow exceeded 1,300 gpm as determined by the same flow element that provides for the flow indication on FI51-1R603B. With flow greater than 1,300 gpm, the Min Flow Valve will automatically close and provide water to the reactor vessel when reactor pressure has dropped low enough to allow the LPCI Injection valve, HV-51-1F017B, to automatically open and allow injection into the reactor vessel.

Additionally, the procedurally established flow rates for the aforementioned RHR modes (with identified procedures) are as follows:

- Suppression Pool Cooling – 8,000 to 8,500 gpm (S51.8.A)
- Suppression Pool Spray - 8,000 to 8,500 gpm prior to opening the spray valve (with spray valve fully open flow will indicate approx. 9,000 gpm) (T-225)
- Drywell Spray – 9,250 to 10,500 gpm (T-225)

Since 4,000 gpm is well below any of the established flowrates for these modes, it reinforced to the Applicants that RHR is not operating in any of these modes and can only be injecting to the vessel through the LPCI Injection valve.

Operationally, verification of flow to the vessel with low pressure injection ECCS is performed by licensed operators and the Applicants using both the discharge check valve indication (as a gross indication of flow to the vessel) and the RHR loop flow indicator (as the available indicator to provide numerical values of flow to the reactor vessel). Use of the flow indicator provides a redundant indication of flow to the vessel under the conditions provided in the stem of the question and would be the instrument used to monitor actual flow values for vessel injection. The use of the flow indicator in this way is supported by UFSAR Tables 7.5-1 and 7.5-3 which list flow indicator FI51-1R603B (LOOP B) as a credited instrument for LPCI Flow verification during accident mitigation.

Distractors C and D still remain plausible but incorrect for the reasons listed in the original Answer Explanation. The basis for their plausibility or why they are still incorrect is not impacted by the determination that the system aligns to the vessel to inject, injects automatically and the flow is displayed on the loop flow indications as reactor pressure drops.

Applicant Response

The applicants provided input that the RHR Flow indication was an equivalent indication of flow to the vessel as the position indication on the inboard check valve and was the instrument used to monitor system flow once system flow has been established. Additionally, the applicants indicated that since no other flowpaths appeared to be possible due to the question not indicating any manual alignment of the "B" RHR Loop AND flow was higher than the value that the min flow valve would close then the only place for the flow of water to go was into the vessel.

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Facility Recommendation:

Based on the question construction which has the RHR Loop responding automatically to the LOCA signal and lowering reactor pressure there was only one flowpath aligned for the RHR loop to establish system flow and either (1) the position indication on the discharge check valve or (2) the indications of 4000 gpm system flow provide equivalent evidence that the system is discharging into the reactor vessel. Therefore the facility agrees with the Applicants and recommends that both answers A and B should be accepted for Question #28.

References:

1) M-0051, Sheet 3, Rev. 68, "P&ID Residual Heat Removal"
2) S51.8.A, Rev. 45, "Suppression Pool Cooling Operation (Startup and Shutdown) and Level Control
3) T-225, Rev. 22, "Start and Shutdown of Suppression Pool and Drywell Spray Operation
4) LGS UFSAR Section 7.5, "Information Systems Important to Safety"

28 **ID: 1102409** **Points: 1.00**

Answer Explanation

Refer to RHR P&ID M-0051, Sheet 3, Coordinates F/G-7, which shows the LPCI 'D' Inboard [Testable] Check Valve HV-1F041D. On MCR panel 10C601, this valve has a remote-position indicator lamp with two halves. The top half of the lamp is labeled "Disc Pos", the bottom half is labeled "Stem Pos". Only by actual flow pushing against the check valve's disc will the "Disc Pos" half of the lamp indicate OPEN (i.e., Green light OFF, Red light ON). The "Stem Pos" half of the lamp will indicate OPEN only when operators apply a TEST signal (TEST pushbutton) to surveill the check valve's operability; in that case, the "Stem Pos" is indicative of the position of the pneumatic operator used to open the valve. The LPCI 'D' injection line flow indicator (FI-1R603D) is shown at P&ID Coordinate G-3. This instrument senses flow upstream of the line's injection valve (1F017D); therefore, this indication is NOT definitive evidence of injection into the RPV.

'B' is correct: HV51-1F041D (INBOARD CHECK) "Disc Pos" indicates OPEN, "Stem Pos" indicates CLOSED. Correct for the reasons described above.

'A' is wrong: FI51-1R603B (LOOP B) reads 4,000 gpm. Plausible to the examinee for three primary reasons: 1) at LGS, use of the INBOARD CHECK valve indication is given very little emphasis, if any, during the course of the Simulator Phase; 2) students are accustomed to using the injection line flow indicators to determine when RHR is injecting; and 3) many examinees have never given a thought to exactly where in the system the flow element for FI51-1R603B is located.

'C' is wrong: DIV 2 LPCI INJECTION VALVE DP PERMISSIVE alarm. Refer to alarm resposne card ARC-MCR-115, F4 (DIV 2 LPCI INJECTION VALVE DP PERMISSIVE). This alarm simply alerts operators to the fact that all conditions are satisfied for the automatic opening of the LPCI 'B' injection valve (F017B). Under expected LOCA conditions, this alarm is expected at an RPV pressure well above the RHR pump shutoff head pressure (~270 psig). Plausible to the examinee who confuses the opening of the injection valve with conditions that would allow actual injection flow into the RPV.

'D' is wrong: 1B RHR PUMP DISCH HI/LO PRESS alarm. Refer to alarm response card ARC-MCR-115, F3. This is NOT an expected alarm so long as the RHR pump has an unobstructed flowpath at its discharge. Plausible to the weaker examinee who believes the alarm to be indicative of sufficient pump discharge pressure so as to cause sufficient injection flow into the RPV.

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Question 28 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	1102409
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	LGSOPS0051
Topic:	Recognize definitive MCR indication of LPCI injection flow into the RPV
Num Field 1:	2.7
Num Field 2:	2.9
Text Field:	203000 K5.01

NRC Post Exam Comment Resolution
2015 Limerick Initial License Exam

Question #28

The licensee recommended two correct answers for Q#28; Choice B – the key answer and Choice A – a 2nd correct answer. The NRC agrees that Choices A and B are both correct. **Choice A and Choice B will each be accepted as a correct answer for Q#28.**

The question stem describes a LOCA in progress and asks the applicant to select the choice which provides definitive evidence that some RHR Loop "B" water is actually injecting into the RPV.

Key answer Choice B, open indication on the LPCI injection line inboard check valve, is a correct answer. This check valve will indicate open only if there is forward flow in the line, from RHR Loop "B" into the RPV.

Choice A, 4000 gpm flow indicated on the RHR Loop "B" flow instrument, was originally proposed by the authors as a distractor under the assumption that this indication was not definitive indication of injection flow. This was because the location of the flow sensing element in the system is such that this device will also indicate flow associated with suppression pool cooling, suppression pool spray, and/or drywell spray alignments. But the question stem does not state that the system is lined up for any of these other purposes. Applicants are briefed in advance of taking the NRC initial license written exam using NUREG 1021, Examiner's Standard, Appendix E, "Policies and Guidelines for Taking NRC Examinations. The Appendix states:

> "When answering a question, ***do not make assumptions regarding conditions that are not specified in the question*** unless they occur as a consequence of other conditions that are stated in the question. For example, you should not assume that any alarm has activated unless the question so states or the alarm is expected to activate as a result of the conditions that are stated in the question. ***Similarly, you should assume that no operator actions have been taken, unless the stem of the question or the answer choices specifically state otherwise.***"

If the described event actually occurred, operator actions would likely be directed in accordance with emergency operating procedures to mitigate effects of the LOCA on the primary containment. However, the stem does not provide any specifics on containment conditions and does not indicate that any operator actions have been performed. With the lack of information provided and the instructions in the Examiner's Standard regarding assumptions, it is reasonable for the applicant to assume that emergency core cooling systems were in a normal standby alignment prior to the event and that systems are functioning as designed. As the event progresses, without operator action, low pressure injection systems will inject as RPV pressure falls below shutoff head conditions for the injection pumps. Any flow indicated on F151-1R603B, RHR Loop B Flow Instrument, will be the result of injection flow into the RPV because the system will not be aligned for any of the other potential flowpaths.

Therefore, Choice A is a 2nd correct answer to the question. Choices C and D remain wrong answers for the reasons given in the choice justifications as originally submitted.

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Question #81

SRO

Unit 2 is operating at 100% power.

The CRS reviews the latest "Daily Jet Pump Operability" surveillance (ST-6-043-320-2) which reveals the following Conditions:

1. Indicated recirc loop flow differs by 11% from the established pump speed-loop flow characteristics
2. Indicated total core flow differs by 6% from the established total core flow derived from recirc loop flow measurements
3. Indicated diffuser-to-lower plenum d/p of a single jet pump (#3) differs from the established patterns by 14%
4. Indicated diffuser-to-lower plenum d/p for all other jet pumps differ from the established patterns by 8%

WHICH ONE of the following identifies (from the above list) the test data that requires the plant must be in HOT SHUTDOWN within 12 hours, per Tech Spec 3.4.1.2 (Jet Pumps)?

A. Condition 3, alone

B. Conditions 3 AND 4, combined

C. Conditions 1 AND 3, combined

D. Conditions 1, 2, AND 3, combined

Answer: C

Immediately following this discussion is the Answer Explanation to from Question #81 as it appeared on the SRO portion of the Approved exam.

Discussion

Of the five (5) Applicants, three (3) chose 'C', the correct answer, while two (2) chose distractor 'A'.

Question 81 is centered around the determination of LCO impact based on surveillance requirements not being inside their allowable band. This question is an application of Tech Specs requiring the evaluation of multiple surveillance requirements and making a determination which of the surveillance requirements were out of specification and the correct combination which results in a Tech Spec LCO entry.

To answer the question as written, the Applicant needs to have memorized either the surveillance requirements of Tech Spec 3.4.1.2 or the acceptance criteria of surveillance test ST-6-043-320-2, "Daily Jet Pump Operability Verification for Two Recirculation Loop Operation", from memory in order to answer the question. As noted in Reference 1 the goal of the examination process is not to memorize Tech Specs but the operators are required to recognize Tech Spec entry conditions, immediate actions and bases. Additionally Reference 2 identifies that Tech Specs with an action of one hour or less also need to be known from memory.

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Since neither the surveillance requirements of Tech Spec 3.4.1.2 nor the acceptance criteria of the aforementioned surveillance test do not fit in the generally accepted items that must be known from memory, the question content must be evaluated to determine if a reference is required in to order to answer the question. Reference 1 provides guidance that if a question involves application of Tech Specs and requires a reference then a reference should be provided if it does not compromise the integrity of the question.

The facility acknowledges that Question 81 was <u>not</u> presented as involving the application of Tech Spec which may have impacted the decision to determine it was appropriate to administer without a reference. Based on the fact that the question requires memorization of information which is beyond the accepted norm and beyond the guidance provided for the portions of Tech Specs which must be memorized by the applicants, this question presents an unreasonable set of conditions to answer the question correctly.

<u>Applicant Comments:</u>

Discussions with the applicants revealed that their answers to the question were based on their memory of the Jet Pump Surveillance test noted previously. Of the three applicants who chose the correct answer all three indicated that they could not recall the exact values limitations for each of the conditions on the stem and used deductive reasoning to come to the correct answer.

<u>Facility Recommendation</u>

Based on developing a question which involves the memorization of acceptance criteria in a surveillance test or the surveillance requirements of a Tech Spec action and applying those items to determine a required Tech Spec action, this question the facility believes that a reasonable opportunity was not provided to the Applicants to answer the question correctly. Therefore the facility recommends that question 81 be removed from the examination.

<u>References:</u>

1) Operator Licensing Program Feedback Summary Item 401.11
2) Operator Licensing Program Feedback Summary Item 401.52

81 **ID: 1104650** **Points: 1.00**

Answer Explanation

Refer to Tech Spec 3.4.1.2, SR 4.4.1.2.a, which shows only three distinct "conditions" (as they are called in the SR language) that are used to determine the OPERABILITY of the jet pumps (i.e., to determine the existence of any failed jet pump). Those "conditions" are the first three conditions given in the question stem. The 4th "condition" in the stem doesn't exist in the SR as a stand-alone item. Per the SR, if any two (never just one) of the three conditions exist, together, the presence of a failed jet pump is presumed and the ACTION of Tech Spec 3.4.1.2 (12-hour HOT SHUTDOWN) must be taken.

'C' is correct: Conditions 1 AND 3, combined. Both Condition1 and Condition 3 exceed the 10% limit of the SR.

'A' is wrong: Condition 3, alone. Plausible to the examinee who recognizes that Condition 1 exceeds the associated 10% limit, but who forgets that it takes at least two of the three Conditions to be UNSAT before having to take the plant shutdown ACTION.

'B' is wrong: Conditions 3 AND 4, combined. Plausible to the examinee who "seems to recall" something about "at least two" Conditions needing to be UNSAT in order to determine a failed jet pump, but who doesn't recall exactly which two Conditions. That examinee is distracted by the fact that a single jet pump (#3) is behaving significantly different from the other 19 jet pumps, in that its d/p differs by more than 5%. To the uncertain examinee, the suggestion of 5% is plausible considering the 5% recirc loop flow mismatch limit (of Tech Spec 3.4.1.3) that exists right now, with the plant operating at 100% power (i.e., well above 70% of rated Core Flow).

'D' is wrong: Conditions 1, 2, AND 3, combined. Plausible to the examinee who forgets that it takes only two UNSAT Conditions in order to have to take the plant shutdown ACTION. That examinee is distracted by the fact that Condition 2 is greater than 5%, confusing that with the 5% recirc loop flow mismatch limit (of Tech Spec 3.4.1.3) that exists right now, with the plant operating at 100% power (i.e., well above 70% of rated Core Flow).

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Question 81 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	1104650
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	NONE
Topic:	(SRO) Recall Tech Spec 3.4.1.2 SR requirements for indications of a failed jet pump
Num Field 1:	3.7
Num Field 2:	4.1
Text Field:	295001 2.2.12
Comments:	Level: SRO Tier: 1 Group: 1 KA # and Rating: 295001 2.2.12 (3.7/4.1) KA Statement: 295001 Partial or Complete Loss of Forced Core Flow Circulation 2.2.12 Knowledge of surveillance procedures. References: U/2 Tech Spec 3.4.1.2 (latest) U/2 Tech Spec 3.4.1.3 (latest) Examinee References: None Learning Objective: No specified objective Question source: New Question history: None Cognitive level: Lower 10 CFR 55: 43.2 Comments: None

NRC Post Exam Comment Resolution
2015 Limerick Initial License Exam

Question #81

The licensee recommended removing Q#81 from the exam. The NRC does not agree. Q#81 will remain on the exam. Choice D is a 2nd correct answer. **Choices C and D will each be accepted as a correct answer for Q#81.**

The licensee argues that the question requires memorization of information beyond the accepted norm. The NRC does not agree. TS 3.4.1.2 states that all jet pumps shall be OPERABLE in Operational Conditions (OPCONs) 1 and 2. The LCO directs that the unit be placed in at least HOT SHUTDOWN within 12 hours if any jet pump is determined to be INOPERABLE. The surveillance requirement for demonstrating jet pump OPERABILITY is listed immediately below the LCO, in the same format that the related information was provided in the question stem. Per the licensee's Surveillance Frequency Control Program, this surveillance is performed daily in the applicable at-power Operational Conditions.

The surveillance requires that no two of the three listed conditions occur. Each of the three criterions in the TS surveillance requirement has a stated acceptance threshold value of 10%.

This question listed values for the specific conditions stated in the jet pump surveillance in the same format and language that is used in TS Surveillance Requirement 4.4.1.2. The applicants were required to evaluate the values and apply understanding of the 10% threshold and the "no two of" requirements to determine that Choice C met the conditions for declaring a jet pump INOPERABLE. As structured, this question tests the ability of the applicant to apply technical specifications and does not require a reference. It is reasonable to expect a SRO license applicant to be able to answer this question without references, given that the LCO requires component operability, the LCO has a relatively short action time (12 hours) and requires unit shutdown, the jet pump operability is determined by operators daily in the applicable at-power OPCONs, and the surveillance criterion were listed in the question.

While reviewing this licensee comment, the NRC noted that Choice D, a choice not selected by any applicant, is a 2nd correct answer to the question. Choice D lists two conditions that exceed the surveillance criterion threshold along with a third condition that does not exceed the threshold. This arrangement of conditions would also require a plant shutdown per the tech spec and is therefore a 2nd correct answer. Accepting Choice D as a second correct answer will not change the scores for any applicant because none of the applicants selected Choice D.

Choices A and B remain wrong answers for the reasons given in the choice justifications as originally submitted.

2015 Limerick ILT NRC Exam Post-Exam Challenges

Question #93
SRO

Unit 1 is operating at 100% power, with the following:

- Drywell cooling is maximized
- MCC D114-R-G is tagged out for repairs
- Drywell air temperature is 125°F

A loss of MCC D134-R-E occurs; as a result, drywell air temperature begins to rise.

Assume drywell air temperature rises at a constant rate of 5°F/hr for each drywell fan that is no longer operating.

WHICH ONE of the following identifies:

(1) the approximate time it will take for drywell air temperature to exceed its Tech Spec limit, and
(2) the Tech Spec time allowed for restoring drywell air temperature to within its Tech Spec limit?

A. (1) 1 hour
(2) 8 hours

B. (1) 2 hours
(2) 8 hours

C. (1) 1 hour
(2) 24 hours

D. (1) 2 hours
(2) 24 hours

Answer: B

Immediately following this Discussion is the Answer Explanation for Question #93 as it appeared on the SRO portion of the Approved Exam Key.

Discussion
All five (5) SRO Applicants chose distracter 'A' as the correct answer, based on a common interpretation of the following stem condition statement:

> "**Assume** drywell air temperature rises at a constant rate of 5°F/hr for each drywell fan that is no longer operating."

Each Applicant correctly understood the following aspects of the question:

1) Drywell unit cooler fans 1A1V212 and 1E1V212 were not running due to the D114-R-G MCC outage. They also recognized that additional fans were not in operation but that aspect did not impact their answer selection.
2) The Applicants also recognized that based on the information in the stem, the alternate fans in each unit drywell unit cooler where one fan had tripped was operating. In this case the 1A2V212 and 1E2V212 fans were the ones which would be used in their answer selection.

3) Each Applicant additionally recognized that the loss of MCC D134-R-E resulted in the trip of two fans (specifically, 1A2V212 and 1E2V212) which resulted in the loss of both fans in two drywell unit coolers.
4) Each Applicant understood that the four fans previously identified to be out of service rendered two drywell unit coolers out of service.
5) Each Applicant recognized that the intent of question was to calculate a heat up rate only for those unit coolers where both fans were out of service. This was not written in the stem but was obvious to the students since the other drywell unit coolers where one fan was in operation would still be providing cooling and would not be contributing to heat up of the drywell.

The question stem, however, did not use standardized terms in order to clearly identify what measure the students were to use as the basis for their calculation of drywell air temperature heat up rate. The question asks the student to calculate the heat up rate based on the number of fans out of service. As noted above, the students recognized that there were a total of four fans out of service which rendered the 1AV212 and 1EV212 drywell unit coolers non-operational.

The correct term to have used in this question stem, based on the original intent of the question, would have been "drywell unit cooler" versus "drywell fan". The term "drywell unit cooler" is the Tech Spec description used in LGS Tech Spec 3.6.6.2. The Tech Spec also clearly establishes that each "drywell unit cooler ... shall be OPERABLE with each subsystem consisting of one unit cooler fan". The same differentiation between a unit cooler and a fan is provided on P&ID M-0077 Sheet 1, "Drywell HVAC" which has a tabulation of drywell unit coolers and the two associated fans per unit cooler. Therefore the use of the term "fan" in the question stem led each of the applicants to determine the number of individual fans out of service not the number of unit coolers out of service as was originally intended by the question.

Since the terms "unit cooler(s)" or alternately "pair of fan(s)" was not used in the stem, the Applicants performed their heat up calculations based four fans no longer operating in the 1AV212 and 1EV212 drywell unit coolers. Using four fans as the basis for the heat up rate resulted in a 20°F/hr heat-up rate as a result of four (4) non-operating *fans* versus a 10°F/hr heat-up rate as a result of two (2) non-operating *unit coolers* as originally intended by the question.

With a 20°F/hr drywell heat up rate each of the applicants recognized that it would take 1 hour to reach the Tech Spec limit of 145°F in the drywell.

Additionally all of the applicants recognized that per the TS 3.6.1.7 ACTION, operators have 8 hours to restore the temperature to within the limit.

It should be noted that none of the SRO Applicants had any question, or needed any clarification, regarding Question #93 during the administration of this exam. It was quite clear in the mind of each Applicant that the question calculation was to be based on individual fan losses and not a loss of unit coolers or pair of fans.

Based on drywell temperature reaching the limit of 145°F in one hour and the LCO ACTION allowing 8 hours to restore drywell temperature all of the applicants selected Answer 'A' as the correct answer.

Answer 'B' remains a plausible distractor but incorrect since the Tech Spec action time is correct for exceeding drywell temperature under Tech Spec 3.6.1.7 but the time to reach the temperature limit is incorrect.

Answers C and D are still plausible and incorrect. Both Answers have the incorrect Tech Spec action time as noted in the original Answer Explanation when applying Tech Spec 3.6.1.7. Answer 'C' has the correct heat-up rate per the new understanding of the term "fan" versus unit cooler and Answer 'D' has the new incorrect time based on the discussion above.

Applicant Response

The applicants indicated that they believed they understood the failure mechanism of the unit coolers and that the intent of the question was asking to calculate a heat up rate based for those unit coolers where both fans were no longer in operation. They assumed that since the stem of the question stated to calculate the heat-up rate based on the number of fans that were no longer operating, the total number of non-operating fans was four and the heat up rate was 20F/hr.

Some of the candidates recognized that there were additional fans out of service as a result of the loss of D114-R-G MCC. However they did not use those additional individual fan loses in their heat-up calculations since (as noted previously) they correctly recognized that heat up would only occur in unit coolers where both fans were out of service.

All of the applicants knew that the time to correct the high temperature condition was 8 hours once the tech spec action limit was reached.

Therefore all five Applicants chose Answer "A".

Facility Recommendation

Based on the term "fan" used in the question stem, the question stem directed the students to calculate the heat-up rate of the drywell using a different methodology than the question originally intended. The term "fan" in the question led all the students to determine a drywell heat-up rate that did not match the original question design. Therefore the facility agrees with the Applicants that the question provided guidance to calculate the heat-up rate based on the total number of out of service fans NOT the total number of out of service drywell unit coolers which was the original intent of the question. Therefore the new correct response is Answer 'A' which replaces Answer 'B' in the original key.

References:

1) LGS Tech Spec 3.6.6.2, "Drywell Hydrogen Mixing System"
2) P&ID M-0077 Sheet 1, "Drywell HVAC"

93	ID: 1104389	Points: 1.00

Answer Explanation

All operators and ILT Candidates understand that the phrase "drywell cooling is maximized" means only the following: All 8 Unit Coolers are in service, with 1 fan operating in each of the Unit Coolers, with 1 Drywell Chilled Water (DWCW) Chiller operating and 2 DWCW Circ Pumps operating.

The following identifies the power supplies for the drywell unit cooler fans:

MCC D114-R-G feeds 1A1V212, 1C1V212, 1E1V212, 1G1V212 (a total of 4 fans)
MCC D124-R-G feeds 1B1V212, 1D1V212, 1F1V212, 1H1V212 (a total of 4 fans)
MCC D134-R-E feeds 1A2V212, 1E2V212 (a total of 2 fans)
MCC D134-R-H feeds 1C2V212, 1G2V212 (a total of 2 fans)
MCC D144-R-H feeds 1D2V212, 1F2V212 (a total of 2 fans)
MCC D144-R-E feeds 1B2V212, 1H2V212 (a total of 2 fans)

NOTE - all of these power supplies can be validated by reviewing 1S77.1.A (COL), pages 1 - 6.

Stem conditions indicate that MCC D114-R-G is out of service, yet "drywell cooling is maximized." this necessarily means that the fans fed from MCC's D134-R-E (2 fans) and D134-R-H (2 fans) are operating in place of the 4 fans that are not operating because of the MCC outage.

When MCC D134-R-E is lost, two fans trip (they are 1A2V212 and 1E2V212). The stated "assumption" that drywell air temperature rises at a constant rate of 5°F/hr for each fan that has been lost (2 fans) means that the air temperature is rising at a rate of 10°F/hr.

The Tech Spec 3.6.1.7 Drywell Average Air Temperature limit is 145°F. Therefore, starting at a temperature of 125°F, it will take 2 hours to reach the Tech Spec limit.

Per the TS 3.6.1.7 ACTION, operators have 8 hours to restore the temperature to within the limit.

'B' is correct: (1) 2 hours; (2) 8 hours. Correct for the reasons described above.

'A' is wrong: (1) 1 hour; (2) 8 hours. Part (1) assumes a total of 4 fans are lost when MCC D134-R-E is lost. This is plausible to the examinee who fails to recall that the Division 3 and Division 4 associated MCCs (i.e., D134 and D144, respectively) each power only two fans, as opposed to the Div 1 and Div 2 MCCs (i.e., D114 and D124, respectively) that each power 4 fans.

'C' is wrong: (1) 1 hour; (2) 24 hours. Part (1) is plausible for the same reasons as that for choice 'A'. Part (2) suggests the restoration time allowed by Tech Spec 3.6.2.1, ACTION 'b' when the 95°F suppression pool temperature limit is exceeded; plausible for that reason.

'D' is wrong: (1) 2 hours; (2) 24 hours. Part (2) is plausible for the same reason as that for choice 'C'.

2015 Limerick ILT NRC Exam Post-Exam Challenges

Question 93 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	3.00
System ID:	1104389
User-Defined ID:	REV 00, 11/17/14
Cross Reference Number:	NONE
Topic:	(SRO) Predict DW air temperature rise time and recall Tech Spec AOT to restore to within limits
Num Field 1:	3.6
Num Field 2:	3.8
Text Field:	223001 A2.10
Comments:	Level: SRO Tier: 2 Group: 2 KA # and Rating: 223001 A2.10 (3.6/3.8) KA Statement: 223001 Primary CTMT and Auxiliaries Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.10 High drywell temperature References: 1S77.1.A (COL), Rev.3; U/1 Tech Spec 3.6.1.7 (latest); U/1 Tech Spec 3.6.2.1 (latest) Examinee References: None Learning Objective: No specified objective Question source: New Question history: None Cognitive level: Higher 10 CFR 55: 41.5, 43.2 Comments: None

NRC Post Exam Comment Resolution
2015 Limerick Initial License Exam

Question #93

The licensee recommended changing the key answer for Q#93 from Choice B to Choice A. The NRC does not agree. Multiple conflicting answers can be correct based on different interpretations of information in the question step. Conflicting correct answers are not acceptable, per guidance in the Examiner's Standard (NUREG-1021, ES-303, Section D.1.c). **Q#93 will be deleted from the exam.**

Limerick Unit 1 has 8 Drywell (DW) coolers. Each cooler has 2 fans. The stem has 1 fan in each of 4 coolers tagged for maintenance. Initial conditions indicate (with the phrase "DW cooling maximized") that 1 fan is running on each of the 8 coolers. One fan is not running on each of the 8 coolers, for a total of 8 fans initially not running. Then, an event occurs, the loss of a MCC, which powers 1 of the previously running fans on each of 2 coolers. The plant is left with 1 fan running on each of 6 coolers. No fans are running on 2 of the coolers. After the event, a total of 4 fans are not running on the 2 coolers that are no longer functioning and a total of 10 of the 16 DW cooler fans are not running.

The licensee comment relates to how the applicants should interpret the question stem statement that "air temperature rises at a constant rate of 5°F/hr for each drywell fan that is no longer operating." They contend the applicants appropriately discounted the possibility of any heatup effect from the 6 coolers that still had 1 fan each in service and correctly attributed the heatup to the loss of cooling from the 2 coolers that had no fans operating. The licensee contends it is therefore reasonable for the applicants to count a total of 4 fans no longer operating (2 fans on each of the 2 non-functioning coolers) to determine a heatup of 20°F/hr such that the key answer should be changed to Choice A.

The confusion surrounds the meaning of the statement that "each drywell fan that is no longer operating." There are at least the 3 following possible interpretations of the statement:

1. 8 fans were initially operating. 6 fans are now operating. **Therefore one can conclude that 2 fans are no longer operating.** Heatup rate is 10°F/hr. It will take 2 hours to hours to reach the DW temp TS limit of 145°F, making Answer Choice B, the key answer, correct.
2. 8 coolers were initially operating, now 6 coolers are operating. The two non-operating coolers have a total of 4 fans. **Therefore one can conclude that 4 fans are no longer operating.** Heatup rate is 20°F/hr. DW TS limit will be reached in 1 hour, making Choice A the correct answer.
3. 8 coolers have 16 fans. Only 6 fans are now running. **Therefore one can conclude that 10 fans are no longer operating.** Heatup rate is 50°F/hr. DW TS limit will be reached in 24 minutes. Under this interpretation, none of the choices is correct.

The stem wording is not clear. The question authors intended the statement to apply to each fan that stops as a result of the MCC failure. However the stem does not provide context or a reference point for the words "no longer operating." The question will be removed from the exam since Choices A and B can both be correct and since Choices A and B are in conflict with each other.

TITLE ELEM DIAG
START-UP RANGE NEUTRON MON SYS
FIRST MADE FOR LIMERICK 1 256X550ADG1

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING— APPLIED PRACTICES / SURFACES ✓

	(A)	(B)	(C)	(D)
(1)	UPSC TRIP	UPSC TRIP	UPSC TRIP	UPSC TRIP
	UPSC AL OR INOP	UPSC AL OR INOP	UPSC AL OR INOP	UPSC AL OR INOP
(2)	PERIOD	PERIOD	PERIOD	PERIOD
(3)	DNSC	DNSC	DNSC	DNSC
	BYPASS	BYPASS	BYPASS	BYPASS
	SRM A	SRM B	SRM C	SRM D

Z1

	(A)	(B)
(1)	UPSC TR OR INOP	UPSC TR OR INOP
	UPSC ALARM	UPSC ALARM
(2)	DNSC	DNSC
	BYPASS	BYPASS
	(IRM)	(APRM)

C51B-Z3A-D

	(A)	(B)
	UPSC TR OR INOP	UPSC
	UPSC ALARM	INOP
	DNSC	DNSC
	BYPASS	BYPASS
	(IRM)	(RBM)

C51B-Z4A,B

(1)	UPSC TR OR INOP	
	UPSC ALARM	
(2)	DPSC	
	BYPASS	
	(IRM)	(APRM)

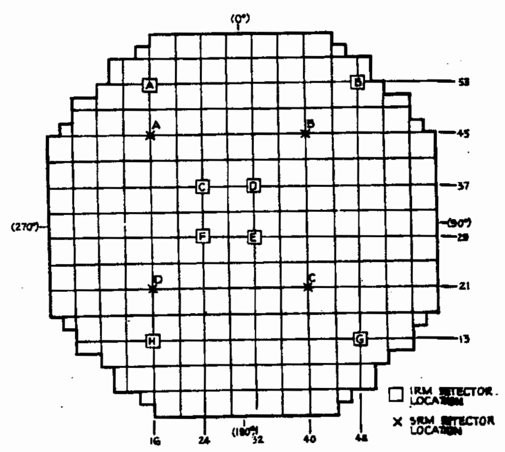
C51B-Z3E,F

FIGURE 1
REMOTE INDICATORS
MASTER SPECIALTIES SERIES 800

FIELD CABLING CONNECTORS

PENETRATION	SRM DETECTOR	PULSE PREAMP	IRM DETECTOR	VOLTAGE PREAMP	SRM	IRM	RANGE SWITCH	PNL P603	PNL P606	PNL P633	CATALOG NO.	VENDOR
JX	JX	J1	JX	J1							UG-1213/U	—
		J2		J2	J1	J1					UG-1213/U PLUS 11485421P	—
		J3		J3	J7	J7					UG-932/U	—
		J4			J6						UG-260/U	—
				J4							MS3106A-18-8S	—
						J42					MS3106A-18-8P	—
							J1				MS3106A-24-12S	—
									J43	J43	MS3106A-28-12P	—
								J27			69-6R20-29S(137)	AMPHENOL
								J31-J34			69-6R22-44S(137)	AMPHENOL

(Header group: MATING CONNECTORS TO PARTS DESIGNATED — first eleven columns; CONNECTOR — CATALOG NO., VENDOR)



TOP VIEW OF REACTOR CORE PLAN

0° CH A; 270° CH D + CH C 90° (NEUT); CH B 180°
SQUARE D PUSHBUTTON JOY STICK
S5 SRM BYPASS SELECTOR SWITCH

HANDLE POSITION	CONTACTS 1-2	3-4	5-6	7-8
NEUT				
0°		X		
90°				X
180°	X			
270°			X	

SWITCH DESIG	CONT	SH NO
S5	1 - 2	3
	3 - 4	3
	5 - 6	3
	7 - 8	3

0° CH A; 270° CH G + CH E 90° (NEUT); CH C 180°
S6 IRM BYPASS SELECTOR SWITCH

0° CH B; 270° CH H + CH F 90° (NEUT); CH D 180°
S7 IRM BYPASS SELECTOR SWITCH

SQUARE D PUSH BUTTON JOY STICK

HANDLE POSITION	CONTACTS 1-2	3-4	5-6	7-8	9-10	11-12	13-14	15-16
NEUT								
0°		X				X		
90°				X				X
180°	X				X			
270°			X				X	

SWITCH DESIG	CONT	SH NO
S6	1 - 2	3
	3 - 4	6
	5 - 6	3
	7 - 8	6
	9 - 10	6
	11-12	3
	13-14	6
	15-16	3

SWITCH DESIG	CONT	SH NO
S7	1 - 2	3
	3 - 4	7
	5 - 6	3
	7 - 8	7
	9 - 10	7
	11-12	3
	13-14	7
	15-16	3

DESIGNATES DOUBLE OR TRIPLE SHIELDED COAXIAL CABLE WITH ALL SHIELDS CONNECTED TOGETHER AT EACH END

DESIGNATES CONNECTION TO PANEL GROUND BUS

DESIGNATES COMPUTER INPUT

MATCH NO.
ZONE
SHEET NO.
MATCH CIRCLE

FCF: UNIT1, 258X121ADG1 (C51-1070)

TABLE 1

DESIGNATOR CHART

REFERENCE DESIGNATION	MPL NO	SUFFIX
IRM DETECTOR	C51 N002	A,B,C,D,E,F,G,H
PULSE PREAMPLIFIER	C51 K001	A,B,C,D
VOLTAGE PREAMPLIFIER	C51 K002	A,B,C,D,E,F,G,H
SOURCE RANGE MON	C51 K603	A,B,C,D
INTERMEDIATE RANGE MON	C51 K601	A,B,C,D,E,F,G,H
SRM DETECTOR	C51N001	A,B,C,D
PERIOD METER	C51 R601	A,B,C,D
SRM RECORDER	C51 R602	A,B
IRM/APRM RECORDER	C51 R603	A, B
IRM/APRM/RBM RECORDER	C51 R603	C,D

INDEX

SH NO.	CONTENT
1	POWER DISTRIBUTION
2	SRM CHANNEL A,B,C,D
3	SRM AUX RELAYS, REMOTE INDICATORS
4	IRM CHANNEL A,C,E,G
5	IRM CHANNEL B,D,F,H
6	IRM AUX RELAYS CH. A,C,E,G AND REMOTE INDICATORS CH. A,C,E,G
7	IRM AUX RELAYS CH. B,D,F,H AND REMOTE INDICATORS CH. B,D,F,H
8	ANNUNCIATOR OUTPUTS
9	COMPUTER OUTPUTS
10	ROD WITHDRAWAL BLOCK OUTPUTS (TO RMCS) TRIP OUTPUT (TO RPS)
11	CABLING DIAGRAM

NOTES:

1. UNLESS OTHERWISE INDICATED, THE FOLLOWING REFERENCE DESIGNATIONS SHOWN ON THIS DIAGRAM ARE PREFIXED WITH C51A:

REFERENCE DESIGNATION	NAME
ZXX	SUBASSEMBLY
FXX	FUSE
SXX	SWITCH
JXX	JACK
PXX	PLUG
XXX	LINE CODE SEE LEGEND

2. SEE FIG. NO. 1 FOR INDICATOR MARKING
3. FOR REACTOR PROTECTION SYSTEM ROUTING SEE REF. DWG. 13. FOR SPECIAL CABLE & CONDUIT REQUIREMENTS SEE REF DWG. 14.
4. COMPONENTS NOT IDENTIFIED WITH A REF. DESIGNATION ARE IDENTIFIED BY MPL NO. IN TABLE 1.
5. REMOTE RANGE SWITCH IS A 12 POSITION (15° INDEXING) 4 DECK (A,B,C,D) SWITCH. MECHANICAL STOP PREVENTS OPERATION OF SWITCH IN POSITIONS 11 & 12
6. PREFIXES FOR PANEL DESIGNATIONS ARE AS FOLLOWS: H12-P603, P606, P633, P789, P790 H22-P008, P030 P051 P032 P033
7. REMOVED
8. REMOVED
9. SHIELDED TWISTED PAIR CABLING IS SUPPLIED AS PART OF PGCC WIRING IN ANNUNCIATOR CIRCUITS.
10. THESE NUMBERS ARE FOR REFERENCE ONLY, NOT FOR PGCC CABLING.

REFERENCE DRAWINGS:

1. 846D581 PULSE PREAMPLIFIER ELEM DIAG
2. 729E388 SOURCE RANGE MONITOR ELEM DIAG
3. 127D1861 TRIP AUX UNIT ELEM DIAG
4. 846D871 VOLTAGE PREAMPLIFIER ELEM DIAG
5. 729E384 INTERMEDIATE RANGE MONITOR ELEM DIAG
6. REMOVED
7. 136B1755 REMOTE RANGE SWITCH ELEM DIAG
8. C51-1080 POWER RANGE NEUTRON MON SYS ELEM DIAG
9. C71-1020 REACTOR PROTECTION SYS ELEM DIAG
10. C11-1080 REACTOR MANUAL CONTROL SYS ELEM DIAG
11. C51-1050 START UP RANGE DET DRIVE CONTROL SYS ELEM DIAG
12. C91-4030 INPUT/OUTPUT LISTING (PROCESS COMPUTER)
13. C71-1040 RPS INTERCONNECTION SCHEME
14. A61-4010 SPECIAL WIRE & CABLE SPEC
15. H12-1060 POWER & UTILITY DISTR INTCON ELEM DIAG
16. D12-1020 PROCESS RAD MON SYSTEM ELEM DIAG

LEGEND
XXX — INDICATES PGCC LINE CODES
LINE CODE — LOCATION CODE
XXX (AR) — SITS CABLE IN AUX ROOM
XXX (CR) — SITS CABLE IN CONTROL ROOM
XXX (PXXX) — FITS CABLE IN AUX ROOM TO TERMINATION CABINET INDICATED
XXX (CR/PXXX) — SITS CABLE FROM CONTROL ROOM TO AUX ROOM, ROUTED THROUGH BUT NOT TERMINATED INTERMINATION CABINET INDICATED

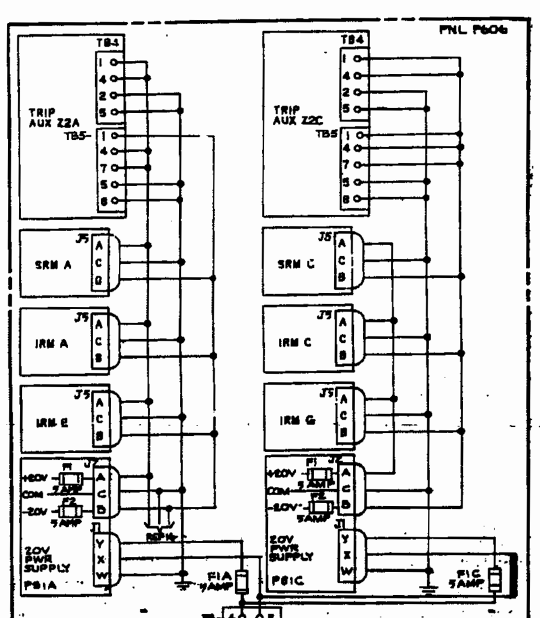
DESIG	RECOMMENDED WIRE AND CABLE TYPE
1	(COAXIAL-TRIPLE SHIELDED)(SRM).
2	(COAXIAL-TRIPLE SHIELDED)(IRM).
3	(COAXIAL-DOUBLE SHIELDED).
4	(COAXIAL-SINGLE SHIELDED).
5	7/C #20 AWG SHIELDED WIRE.
6	#20 AWG TWISTED SHIELDED PAIR.
7	15/C #20 AWG SHIELDED WIRE.
8	#14 OR #16 AWG SINGLE CONDUCTOR SHIELDED WIRE.

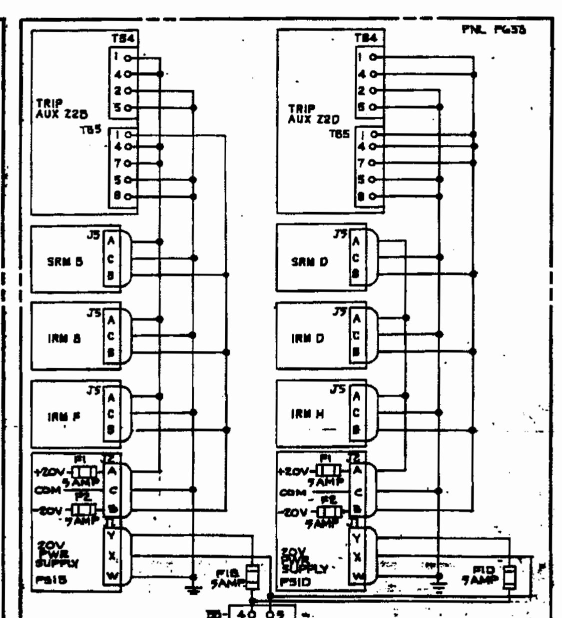
EMERGENCY RESPONSE INFORMATION SYS (ERIS)
FIRST LINE: BECHTEL SCHEME NO.
SECOND LINE: INPUT NO.

LINE CODES:
LAST USED 143
THIS SHEET NONE

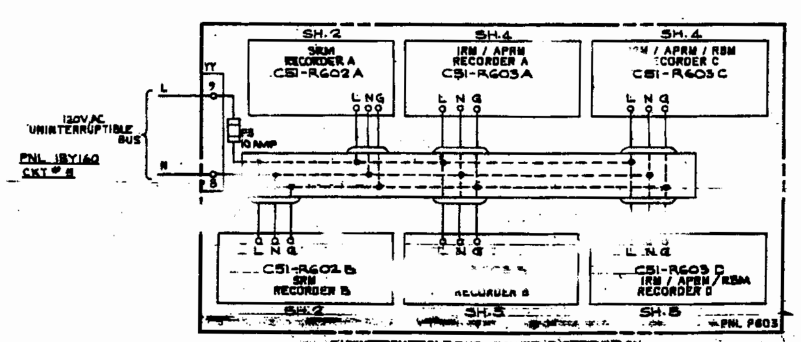
TRIP AUXILIARY UNIT RELAYS COIL LOCATIONS

TRIP AUXILIARY Z2A OR Z2C RELAY	SHEET	TRIP AUXILIARY Z2B OR Z2D RELAY	SHEET
K1 THRU K6	2	K1 THRU K6	2
K7 THRU K14	3	K7 THRU K14	3
K16	6	K16	6
K17 & K18	3	K17 & K18	3
K30 THRU K33	4	K30 THRU K33	5
K34 THRU K44	6	K34 THRU K44	7
K50 THRU K53	4	K50 THRU K53	5
K54 THRU K61	6	K54 THRU K61	7
K90	8	K90	8
K15	10	K15	10
K63	6	K63	7





120 VAC RPS BUS A (REF 15)
PNL 1AY160 CKT #2
120 VAC RPS POWER DISTRIBUTION & 20VDC POWER DISTRIBUTION
120 VAC RPS BUS B (REF 15)
PNL 1BY160 CKT #2



UNINTERRUPTIBLE BUS POWER DISTRIBUTION

8031-M-1-C51-1070-E-001, 27

LGSOPS0074

Source Range Monitoring (SRM) And Intermediate Range Monitoring (IRM) Systems

1

Please turn off all cell phones, pagers, and other communications devices

Thank You

2

Lesson Objectives

EO1. Determine the need for overlap between ranges of Nuclear Instrumentation, and how overlap is accomplished.

EO2. Identify statements which describe the operation of a fission chamber.

EO3. Determine the function/purpose of any given Source Range Monitoring (SRM) or Intermediate Range Monitoring (IRM) System component.

3

Lesson Objectives

EO4. Identify the inputs from the Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems to:

a. Reactor Protection System (RPS)

b. Reactor Manual Control System (control rod withdrawal blocks)

c. Auxiliary Equipment Room Alarm indicators

4

Lesson Objectives

EO5.	***Identify the control functions and protective actions associated with the Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems.***
EO6.	***** Given applicable references and a set of plant conditions, identify the appropriate EO actions (For Continuing Training Only).***
EO7.	***** Given applicable references and a set of plant conditions, identify the system response (For Continuing Training Only).***

5

Lesson Objectives

IL1.	***Identify the purpose of the Source Range Monitoring (SRM) System.***
IL2.	***Select the range of the Source Range Monitoring (SRM) System in terms of % power.***
IL3.	***Identify the physical arrangement of the Source Range Monitoring (SRM) System detectors within the Reactor Pressure Vessel (RPV).***

6

Lesson Objectives

IL4. ***Identify the function of the following Source Range Monitoring (SRM) System components:***

a. ***Pulse Preamplifier***

b. ***Pulse Height Discriminator***

c. ***Log Integrator***

d. ***DC Amplifier***

e. ***Period Circuit***

7

Lesson Objectives

IL5. ***Determine the power supply to the Source Range Monitoring (SRM) System channels.***

IL6. ***Identify the conditions, including setpoints, in the Source Range Monitoring (SRM) System that will generate control rod withdrawal blocks and reactor scrams, and identify the bypasses for each.***

IL7. ***Given a drawing of controls, indications, and/or alarms located in the Main Control Room:***

a. ***Determine the status of the Source Range Monitoring (SRM) System***

b. ***Predict the effect of manipulation of system controls***

8

Lesson Objectives

IL8. Predict the effect that a loss or malfunction of the Source Range Monitoring (SRM) System will have on:

a. Reactor Protection System (RPS)

b. Reactor Manual Control System (RMCS)

c. Reactor Power Indication

IL9. Predict the effect that a loss or malfunction of the Reactor Protection System (RPS) will have on the Source Range Monitoring (SRM) System.

9

Lesson Objectives

IL10. Given a copy of Technical Specifications and various plant conditions:

a. Identify any condition that does not comply with applicable Limiting Conditions for Operation (LCOs)

b. Select the bases for applicable Limiting Conditions for Operation (LCOs) which pertain to the Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems

c. (SRO ONLY) Determine the actions required for each condition of non-compliance

10

Lesson Objectives

IL11. ***(SRO Only) Given a copy of Emergency Action Levels (EALs), recognize any condition which meets the criteria to declare an event.***

IL12. ***Identify the purpose of the Intermediate Range Monitoring (IRM) System.***

IL13. ***Select the range of Intermediate Range Monitoring (IRM) sensitivity in terms of % power.***

IL14. ***Identify the mechanism for gamma compensation in the Intermediate Range Monitoring (IRM) System.***

IL15. ***Identify the physical arrangement of the Intermediate Range Monitoring (IRM) detectors within the Reactor Pressure Vessel (RPV).***

11

Lesson Objectives

IL16. ***Determine which Intermediate Range Monitoring (IRM) channels provide inputs to Reactor Protection Trip Systems "A" and "B".***

IL17. ***Identify the function of the following Intermediate Range Monitoring (IRM) System components:***

a. ***Voltage Preamplifier***

b. ***Amplifier And Attenuator Unit***

c. ***Inverter***

d. ***Mean Square Analog (MSA) Unit***

e. ***Output (operational) Amplifier***

12

Lessuon Objectives

IL18. *Identify the power supplies to each Intermediate Range Monitoring (IRM) channel.*

IL19. *Identify the conditions, including setpoints, in the Intermediate Range Monitoring (IRM) System that will generate control rod withdrawal blocks and reactor scrams, and identify the bypasses for each.*

IL20. *Given a drawing of controls, indications, and/or alarms located in the Main Control Room:*

a. Determine the status of the Intermediate Range Monitoring (IRM) System

b. Predict the effect due to the manipulation of system controls

13

Lesson Objectives

IL21. *Predict the effect that a loss or malfunction of Intermediate Range Monitoring (IRM) System will have on:*

a. Reactor Protection System (RPS)

b. Reactor Manual Control System (RMCS)

c. Reactor Power Indication

IL22. *Predict the effect that a loss or malfunction of the following would have on the Intermediate Range Monitoring (IRM) System:*

a. Reactor Protection System (RPS) power supply

b. Detectors

c. Trip Units

14

Lesson Objectives

IL23. ***Identify the means of changing Intermediate Range Monitoring (IRM) detector position and the reason for doing so.***

IL24. ***Given a copy of Technical Specifications and various plant conditions:***

a. ***Identify any condition that does not comply with the applicable Limiting Conditions for Operation (LCOs)***

b. ***Summarize the bases for those Limiting Conditions for Operation (LCOs) which pertain to the Intermediate Range Monitoring (IRM) System***

c. ***(SRO ONLY) Determine the actions required for any condition of non-compliance***

15

Lesson Objectives

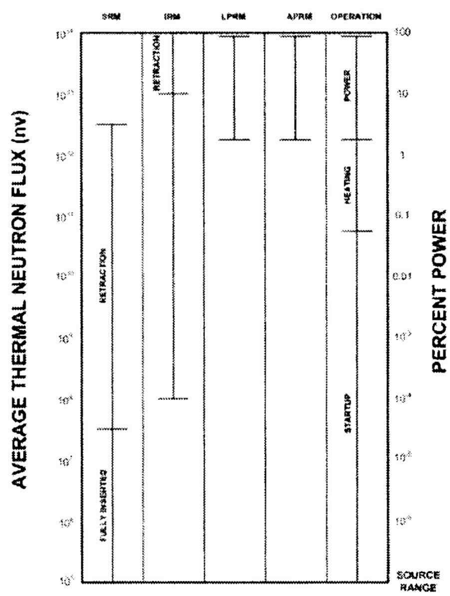
IL25. ***(SRO Only) Given a copy of Emergency Action Levels (EALs), recognize any condition which meets the criteria to declare an event.***

16

Nuclear Instrumentation Review And Overview

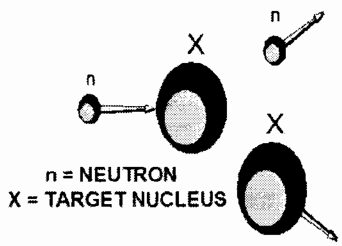
17

RANGES OF NEUTRON MONITORING SYSTEM

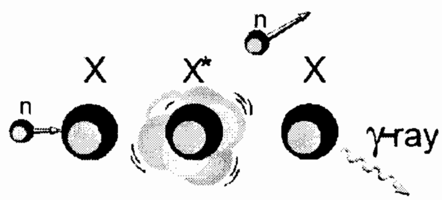


18

ELASTIC SCATTERING



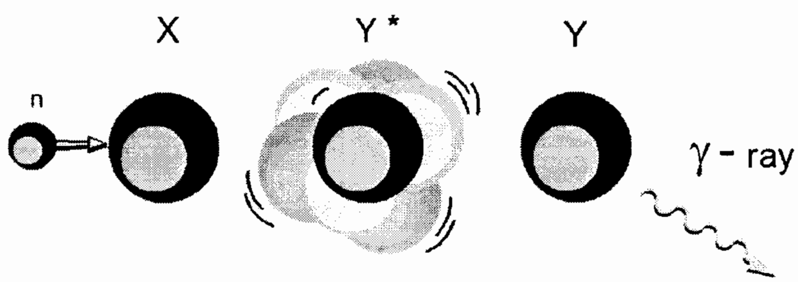
INELASTIC SCATTERING



X* = EXCITED TARGET NUCLEUS

19

RADIATIVE CAPTURE

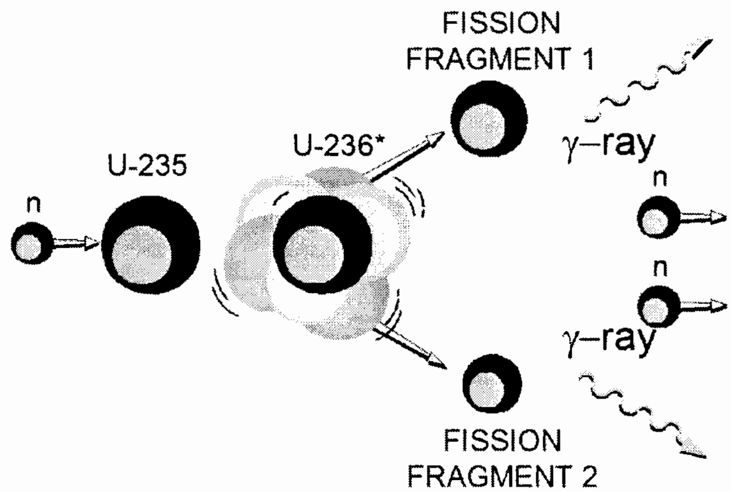


Y = ISOTOPE OF TARGET NUCLEUS

Y* = EXCITED ISOTOPE

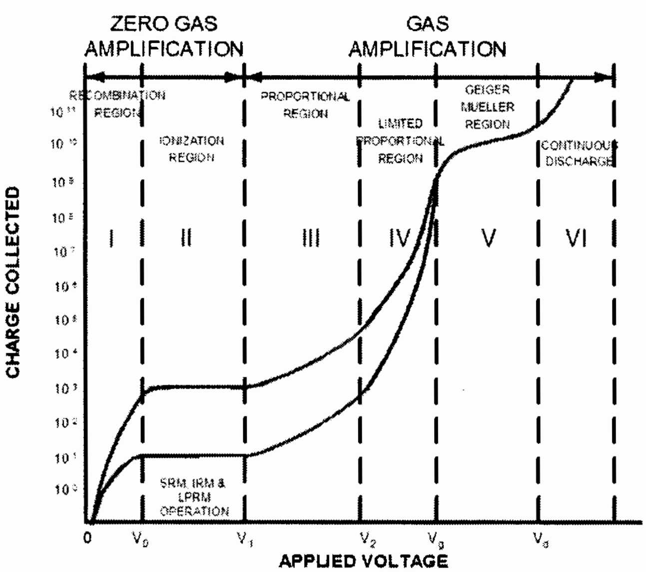
20

FISSION OF U-235 ATOM



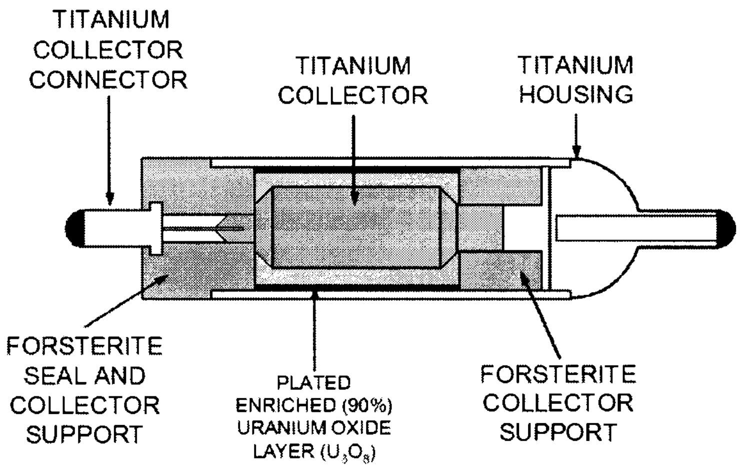
21

GAS AMPLIFICATION CURVE



22

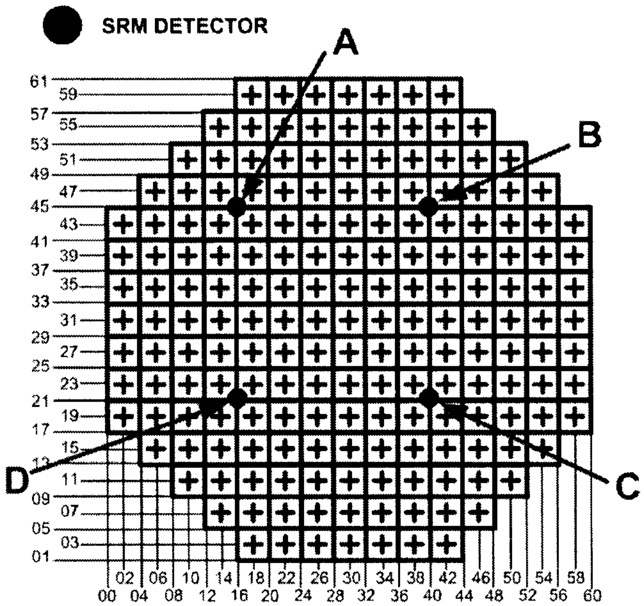
FISSION CHAMBER DETECTOR



23

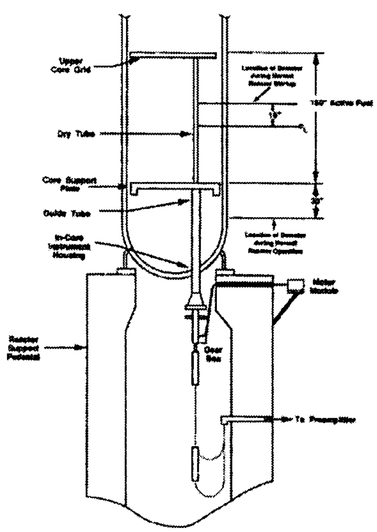
SRM System

24



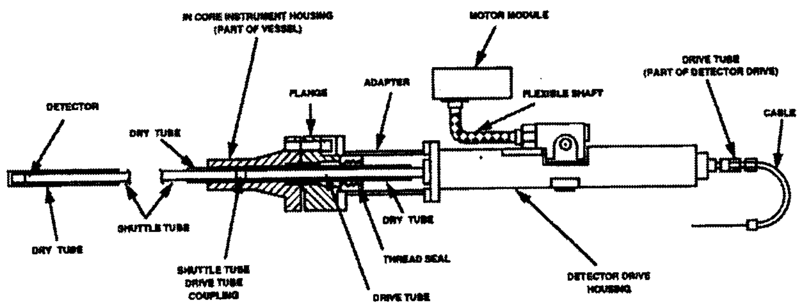
25

DETECTOR IN-CORE DETAIL



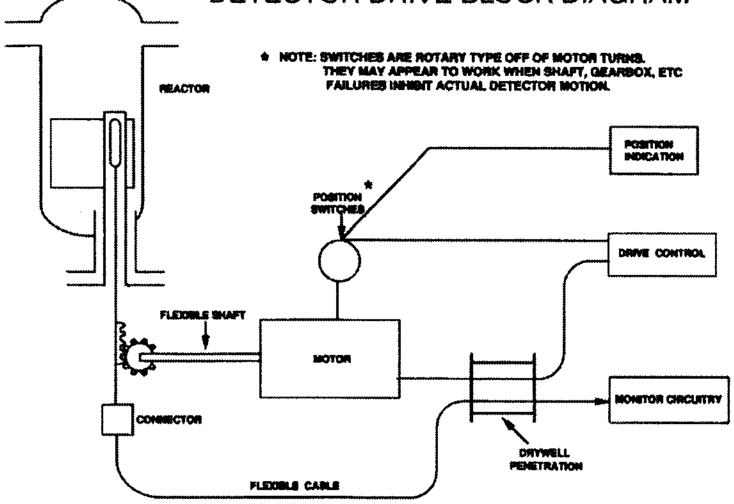
26

DETECTOR DRIVE ASSEMBLY

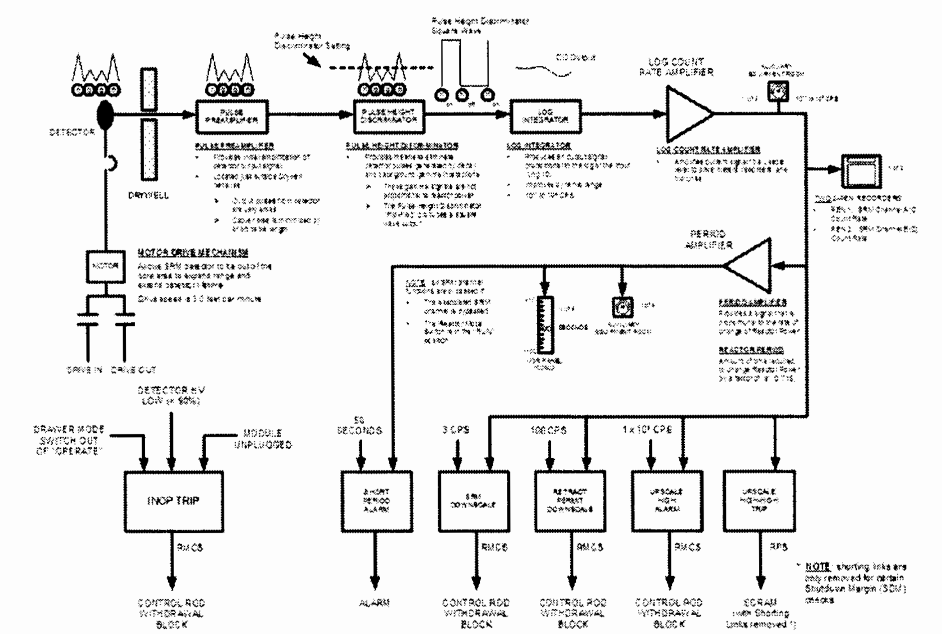


27

DETECTOR DRIVE BLOCK DIAGRAM

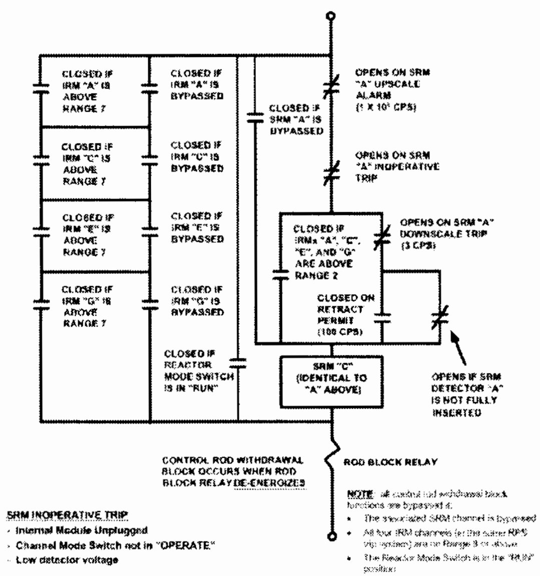


28



29

SRM CHANNEL "A"
CONTROL ROD BLOCK SIGNALS AND BYPASSES



SRM INOPERATIVE TRIP
- Internal Module Unplugged
- Channel Mode Switch not in "OPERATE"
- Low detector voltage

NOTE: all control rod withdrawal block functions are bypassed if:
- The associated SRM channel is bypassed
- All four IRM channels (in the same RPS trip system) are on Range 8 or above
- The Reactor Mode Switch is in the "RUN" position

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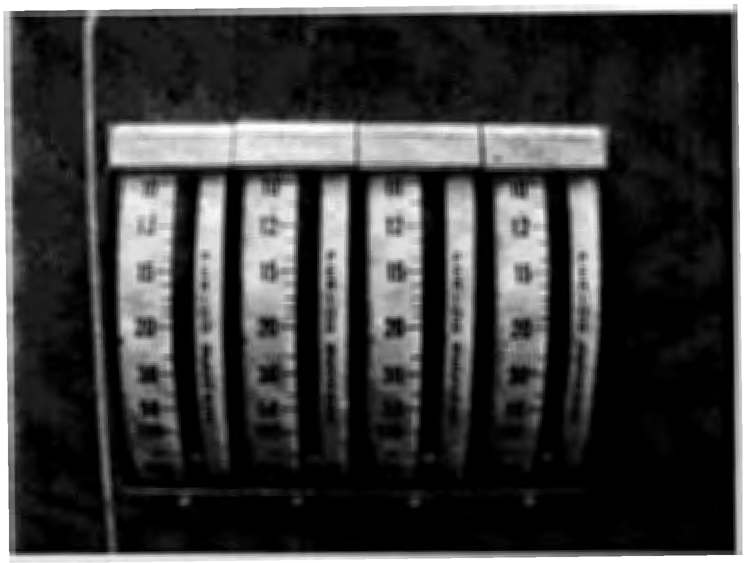
MAIN CONTROL ROOM PANEL 10C603

31

SRM RECORDERS (MCR PANEL 10C603)

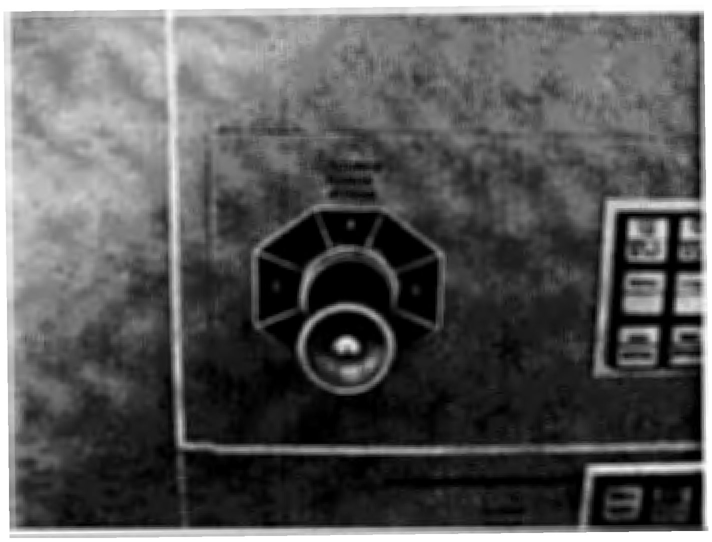
32

SRM REACTOR PERIOD METERS (MCR PANEL 10C603)



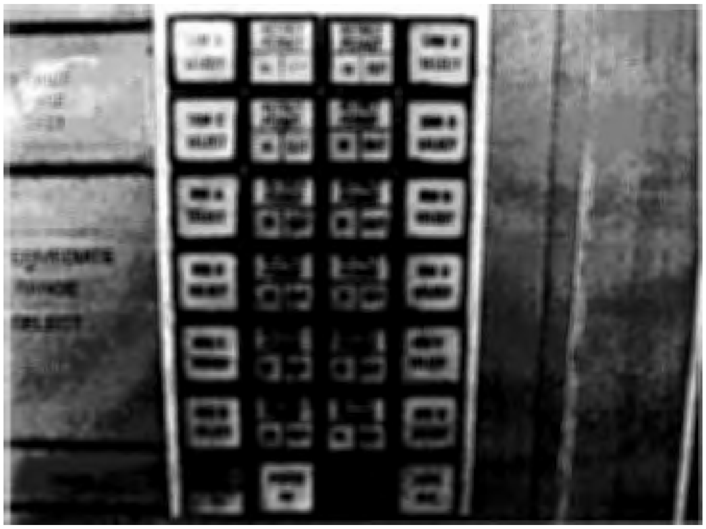
33

SRM BYPASS SWITCH (MCR PANEL 10C603)



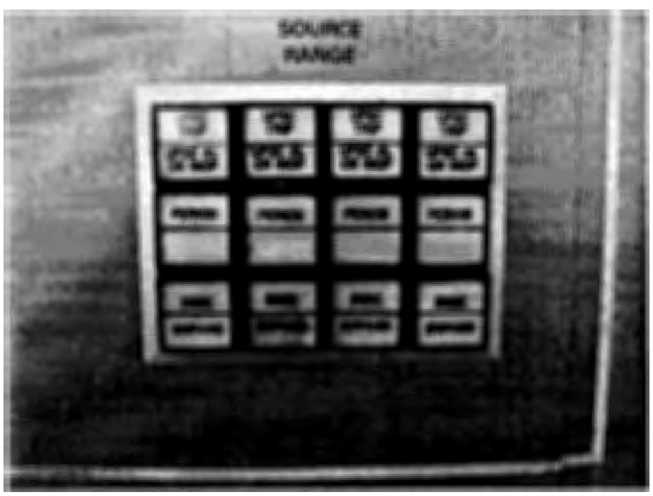
34

SRM DETECTOR DRIVE CONTROLS AND INDICATING LAMPS (MCR PANEL 10C603)



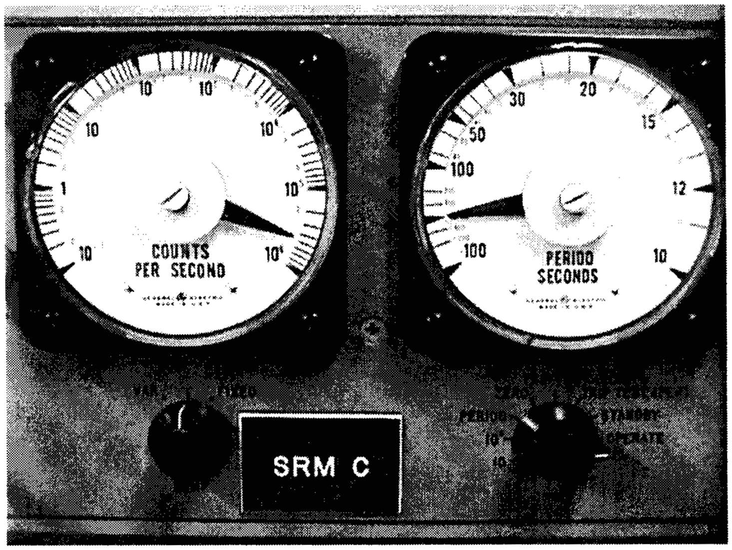
35

SRM ALARM LAMPS (MCR PANEL 10C603)



36

SRM METERS (AER)



37

SRM LIGHTS (AER)

38

IRM System

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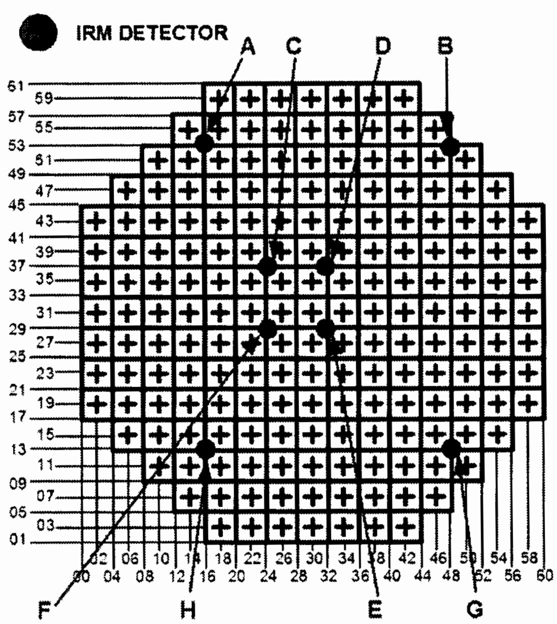
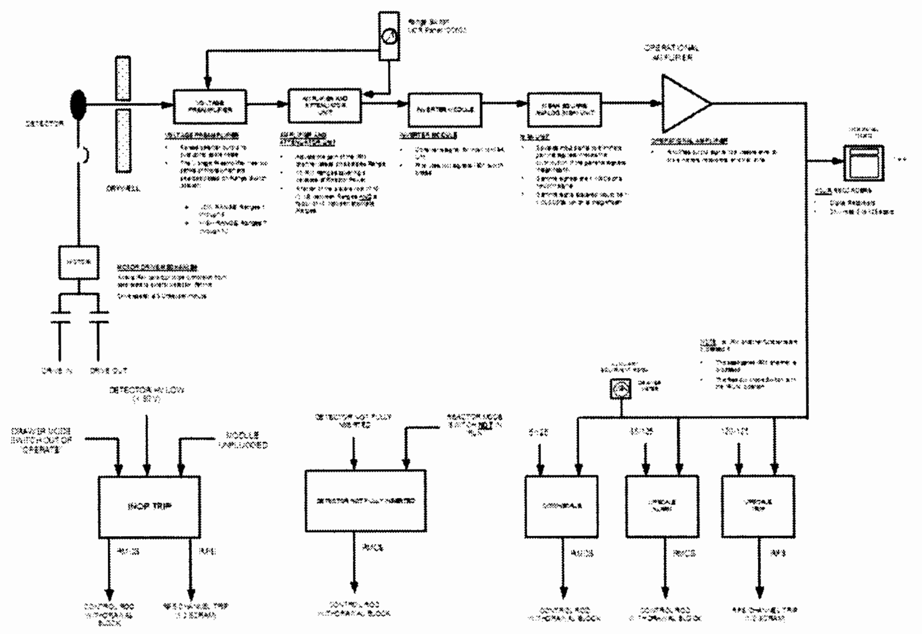
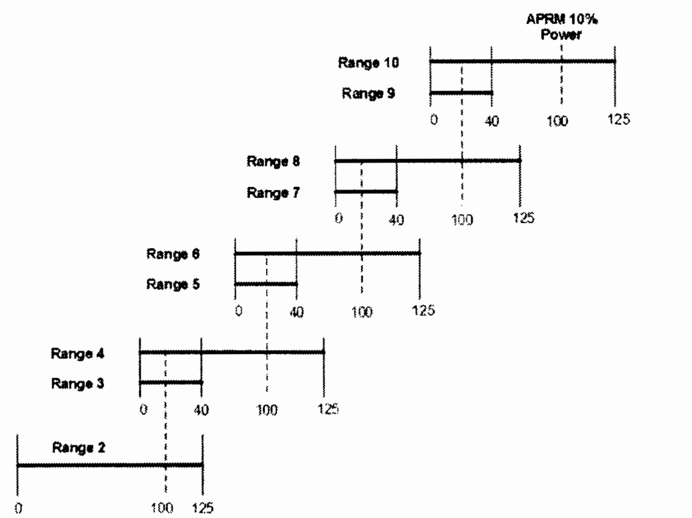


40



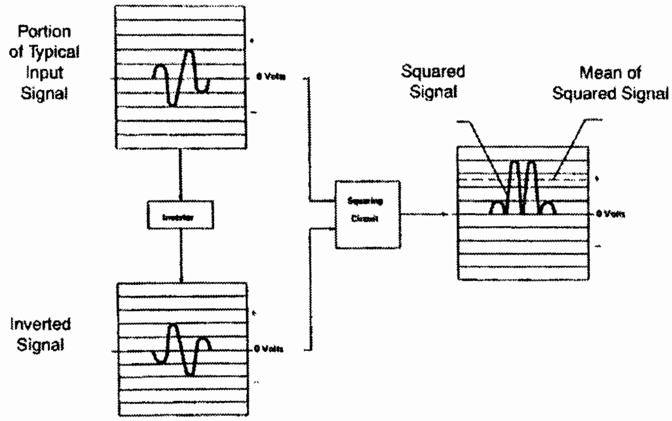
41

IRM RANGE AND REACTOR POWER RELATIONSHIP



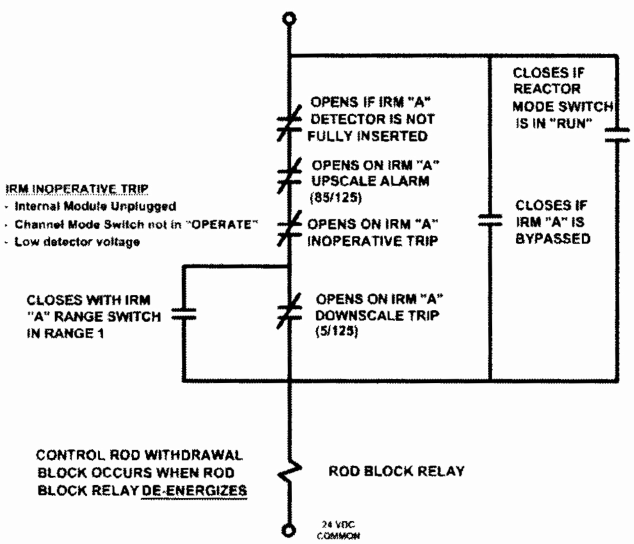
42

MEAN SQUARE ANALOG OPERATION



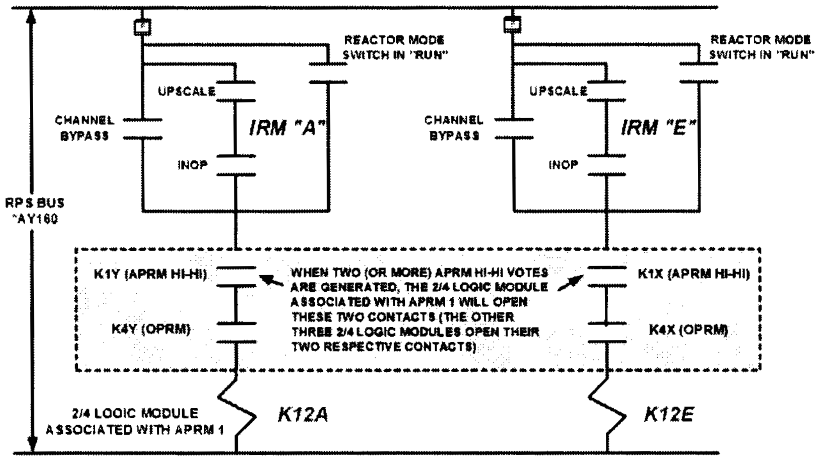
43

IRM CHANNEL "A"
CONTROL ROD BLOCK SIGNALS AND BYPASSES



44

APRM 1 2/4 LOGIC MODULE INTERFACE WITH SCRAM CONTACTS IN RPS TRIP CHANNEL "A1"



45

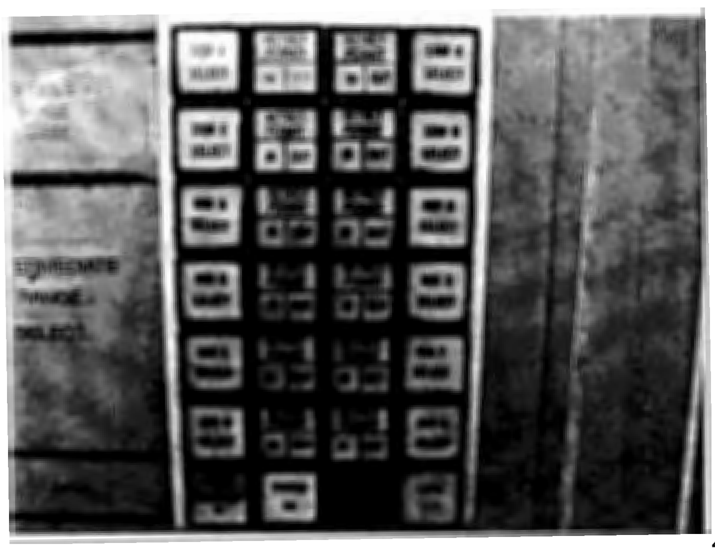
IRM RECORDERS (MCR PANEL 10C603)

46

IRM CHANNEL BYPASS SWITCH (MCR PANEL 10C603)

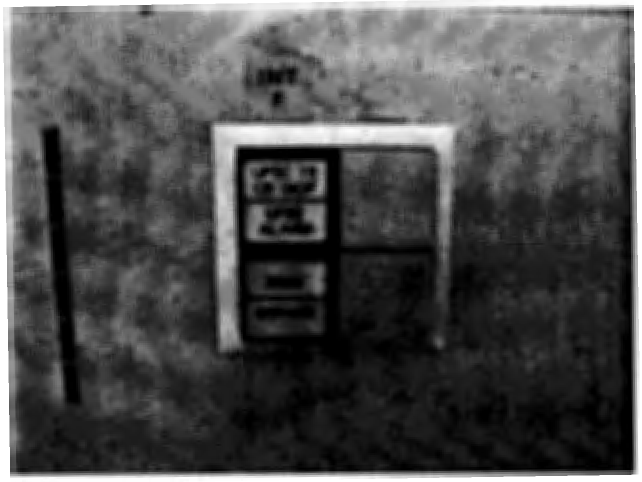
47

IRM DETECTOR DRIVE CONTROLS AND INDICATING LAMPS (MCR PANEL 10C603)



48

IRM ALARM LAMPS (MCR PANEL *0C603)



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IRM METERS (AER)

50

IRM LIGHTS (AER)

51

SRM System Technical Specifications

- **3/4.3.6, Control Rod Block Instrumentation**
- **3/4.3.7, Monitoring Instrumentation - Source Range Monitors**
- **3/4.9.2, Refueling Operations - Instrumentation**
- **3/4.9.10, Refueling Operations - Control Rod Removal**
- **Table 4.3.1.1-1, Reactor Protection System Instrumentation Surveillance Requirements**
- **3/4.3.7.5, Accident Monitoring Instrumentation *(reference LER 2009-001-00)***

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IRM System Technical Specifications

2.2, Limiting Safety System Settings, Reactor Protection System Instrumentation Setpoints

3/4.3.1, Reactor Protection System Instrumentation - Intermediate Range Monitors

3/4.3.6, Control Rod Block Instrumentation - Intermediate Range Monitors

3/4.3.7.5, Accident Monitoring Instrumentation

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SRM System EALs

- MU2/CU2
- CG6
- CS6

54

IRM System EALs

- None

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Lesson Objectives Review

EO1. *Determine the need for overlap between ranges of Nuclear Instrumentation, and how overlap is accomplished.*

EO2. *Identify statements which describe the operation of a fission chamber.*

EO3. *Determine the function/purpose of any given Source Range Monitoring (SRM) or Intermediate Range Monitoring (IRM) System component.*

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Lesson Objectives Review

EO4. Identify the inputs from the Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems to:

a. Reactor Protection System (RPS)

b. Reactor Manual Control System (control rod withdrawal blocks)

c. Auxiliary Equipment Room Alarm indicators

57

Lesson Objectives Review

EO5. Identify the control functions and protective actions associated with the Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems.

EO6. ** Given applicable references and a set of plant conditions, identify the appropriate EO actions (For Continuing Training Only).

EO7. ** Given applicable references and a set of plant conditions, identify the system response (For Continuing Training Only).

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Lesson Objectives Review

IL1. *Identify the purpose of the Source Range Monitoring (SRM) System.*

IL2. *Select the range of the Source Range Monitoring (SRM) System in terms of % power.*

IL3. *Identify the physical arrangement of the Source Range Monitoring (SRM) System detectors within the Reactor Pressure Vessel (RPV).*

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Lesson Objectives Review

IL4. *Identify the function of the following Source Range Monitoring (SRM) System components:*

a. *Pulse Preamplifier*

b. *Pulse Height Discriminator*

c. *Log Integrator*

d. *DC Amplifier*

e. *Period Circuit*

60

Lesson Objectives Review

IL5. ***Determine the power supply to the Source Range Monitoring (SRM) System channels.***

IL6. ***Identify the conditions, including setpoints, in the Source Range Monitoring (SRM) System that will generate control rod withdrawal blocks and reactor scrams, and identify the bypasses for each.***

IL7. ***Given a drawing of controls, indications, and/or alarms located in the Main Control Room:***

- ***a. Determine the status of the Source Range Monitoring (SRM) System***
- ***b. Predict the effect of manipulation of system controls***

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Lesson Objectives Review

IL8. ***Predict the effect that a loss or malfunction of the Source Range Monitoring (SRM) System will have on:***

- ***a. Reactor Protection System (RPS)***
- ***b. Reactor Manual Control System (RMCS)***
- ***c. Reactor Power Indication***

IL9. ***Predict the effect that a loss or malfunction of the Reactor Protection System (RPS) will have on the Source Range Monitoring (SRM) System.***

62

Lesson Objectives Review

IL10. ***Given a copy of Technical Specifications and various plant conditions:***

a. ***Identify any condition that does not comply with applicable Limiting Conditions for Operation (LCOs)***

b. ***Select the bases for applicable Limiting Conditions for Operation (LCOs) which pertain to the Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems***

c. ***(SRO ONLY) Determine the actions required for each condition of non-compliance***

63

Lesson Objectives Review

IL11. ***(SRO Only) Given a copy of Emergency Action Levels (EALs), recognize any condition which meets the criteria to declare an event.***

IL12. ***Identify the purpose of the Intermediate Range Monitoring (IRM) System.***

IL13. ***Select the range of Intermediate Range Monitoring (IRM) sensitivity in terms of % power.***

IL14. ***Identify the mechanism for gamma compensation in the Intermediate Range Monitoring (IRM) System.***

IL15. ***Identify the physical arrangement of the Intermediate Range Monitoring (IRM) detectors within the Reactor Pressure Vessel (RPV).***

64

Lesson Objectives Review

IL16. Determine which Intermediate Range Monitoring (IRM) channels provide inputs to Reactor Protection Trip Systems "A" and "B".

IL17. Identify the function of the following Intermediate Range Monitoring (IRM) System components:

a. Voltage Preamplifier

b. Amplifier And Attenuator Unit

c. Inverter

d. Mean Square Analog (MSA) Unit

e. Output (operational) Amplifier

65

Lesson Objectives Review

IL18. Identify the power supplies to each Intermediate Range Monitoring (IRM) channel.

IL19. Identify the conditions, including setpoints, in the Intermediate Range Monitoring (IRM) System that will generate control rod withdrawal blocks and reactor scrams, and identify the bypasses for each.

IL20. Given a drawing of controls, indications, and/or alarms located in the Main Control Room:

a. Determine the status of the Intermediate Range Monitoring (IRM) System

b. Predict the effect due to the manipulation of system controls

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Lesson Objectives Review

IL21. ***Predict the effect that a loss or malfunction of Intermediate Range Monitoring (IRM) System will have on:***

a. ***Reactor Protection System (RPS)***

b. ***Reactor Manual Control System (RMCS)***

c. ***Reactor Power Indication***

IL22. ***Predict the effect that a loss or malfunction of the following would have on the Intermediate Range Monitoring (IRM) System:***

a. ***Reactor Protection System (RPS) power supply***

b. ***Detectors***

c. ***Trip Units***

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Lesson Objectives Review

IL23. ***Identify the means of changing Intermediate Range Monitoring (IRM) detector position and the reason for doing so.***

IL24. ***Given a copy of Technical Specifications and various plant conditions:***

a. ***Identify any condition that does not comply with the applicable Limiting Conditions for Operation (LCOs)***

b. ***Summarize the bases for those Limiting Conditions for Operation (LCOs) which pertain to the Intermediate Range Monitoring (IRM) System***

c. ***(SRO ONLY) Determine the actions required for any condition of non-compliance***

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Lesson Objectives Review

IL25. ***(SRO Only) Given a copy of Emergency Action Levels (EALs), recognize any condition which meets the criteria to declare an event.***

69

What Questions Do You Have?

70

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
	f. By selecting proper ratios of Uranium-235 to Uranium-234, life of detector is extended g. The uranium coating makeup of typical power range detector is 18% Uranium-235, 78% Uranium-234, and 4% Uranium-238
	C. Failures in Fission Chambers 1. Fission chambers are periodically calibrated in core, with amplifier gain adjustments made to offset decreased detector output over its life, due to uranium depletion
Objective 24	2. Failures of fission chambers can include following: a. Spiking - Very short, high signal pulses that occur infrequently, due to bridging of electrode gap by hair like fibers of casing lining 1) Newer Local Power Range Monitors (LPRMs) have more stable linings, and problem is not very common 2) LPRM spikes can cause trips of neutron monitoring channels b. Wetting - Short circuits due to leaking of water into detector causes detector to fail high c. Loss of voltage - A loss of high voltage DC to detector causes detector to fail to low value d. Loss of detector gas pressure - The indication is dependent on the number of primary and secondary ionizations that occur from the ionizing event. 1) If the gas pressure decreases, there will be less gas atoms to undergo ionization, therefore less electrons collected.

KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
	2) This means a smaller electrical current will exist and the indication will fail downscale.
	D. BWR Nuclear Instrumentation
	1. Boiling Water Reactors designed by General Electric use incore fission chambers for Neutron Monitoring System
	a. This system consists of following five major portions:
	1) Source Range Monitor (SRM) System
	2) Intermediate Range Monitor (IRM) System
	3) Local Power Range Monitor (LPRM) System
	4) Average Power Range Monitor (APRM) System
	5) Traversing Incore Probe (TIP) System
Objective 23	b. The Neutron Monitoring System employs following ranges and probe system to monitor reactor power level:
	1) The SRM System, which provides power indication while shutdown; during startup and shutdown; and refueling
	2) The IRM System, which provides power indication during startup and shutdown
	3) The LPRM System, which provides power indication during power operation and at specific locations in core
	4) The APRM System, which uses outputs from LPRM detectors to compute average core power

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Course/Program:	LGS Operations Initial Training	Module/LP ID:	**LGSOPS0074** LEOT0074 LLOT0240 LLOT0250
Title:	Source Range Monitoring (SRM) And Intermediate Range Monitoring (IRM) Systems ©	Course Code:	Per LMS Coding
Author:	Samuel M. Cohen	Revision/Date:	003/04-25-12
Prerequisites:	NONE	Revision By:	smc
OPEX Included:	Internal / External / Both / None (circle one)	Est. Duration:	2/50 Minute Periods

TERMINAL COGNITIVE OBJECTIVES (LICENSED OPERATORS)

(671067/TCO-2000090401)	(ON-109) Loss Of All Neutron Monitoring Instrumentation [B-15 A nuclear instrumentation failure]
(671459/TCO-2151050401)	Bypass An IRM Channel
(671449/TCO-2150020101)	Perform startup checks and Monitor SRM Indication
(671453/TCO-2150040401)	Bypass SRM Channel
(672267/TCO-2950030101)	Withdraw SRM/IRM Detectors
(672271/TCO-2952020401)	Insert IRM SRM Detectors

TERMINAL COGNITIVE OBJECTIVES (EQUIPMENT OPERATORS)

(671452/TCO-2150040104)	Verify APRM Operability
(671451/TCO-2150030104)	Bypassing an LPRM
(671940/TCO-2670120104)	Perform APRM Gain Adjustment
(671454/TCO-2150050104)	Reset an APRM to Single Loop or Dual Loop Operation

ENABLING OBJECTIVES

Upon successful completion of this lesson, the trainee shall perform the following using references (as appropriate), and from memory in accordance with the lesson materials.

The trainee shall:

Objective #	Objective Description	Pg. #
EO1.	Determine the need for overlap between ranges of Nuclear Instrumentation, and how overlap is accomplished.	3
EO2.	Identify statements which describe the operation of a fission chamber.	5
EO3.	Determine the function/purpose of any given Source Range Monitoring (SRM) or Intermediate Range Monitoring (IRM) System component.	11, 37
EO4.	Identify the inputs from the Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems to:	
	a. Reactor Protection System (RPS)	15, 44
	b. Reactor Manual Control System (control rod withdrawal blocks)	13, 46
	c. Auxiliary Equipment Room Alarm indicators	21, 52
EO5.	Identify the control functions and protective actions associated with the Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) Systems.	13, 44
EO6.	** Given applicable references and a set of plant conditions, identify the appropriate EO actions (For Continuing Training Only).	
EO7.	** Given applicable references and a set of plant conditions, identify the system response (For Continuing Training Only).	

Objective #	Objective Description	Pg. #
IL1.	Identify the purpose of the Source Range Monitoring (SRM) System.	6
IL2.	Select the range of the Source Range Monitoring (SRM) System in terms of % power.	6
IL3.	Identify the physical arrangement of the Source Range Monitoring (SRM) System detectors within the Reactor Pressure Vessel (RPV).	6
IL4.	Identify the function of the following Source Range Monitoring (SRM) System components:	
	a. Pulse Preamplifier	11
	b. Pulse Height Discriminator	11
	c. Log Integrator	12
	d. DC Amplifier	12
	e. Period Circuit	13
IL5.	Determine the power supply to the Source Range Monitoring (SRM) System channels.	23
IL6.	Identify the conditions, including setpoints, in the Source Range Monitoring (SRM) System that will generate control rod withdrawal blocks and reactor scrams, and identify the bypasses for each.	13, 15
IL7.	Given a drawing of controls, indications, and/or alarms located in the Main Control Room:	
	a. Determine the status of the SRM System	15
	b. Predict the effect of manipulation of system controls	15
IL8.	Predict the effect that a loss or malfunction of the Source Range Monitoring (SRM) System will have on:	
	a. Reactor Protection System (RPS)	31
	b. Reactor Manual Control System (RMCS)	31
	c. Reactor Power Indication	31
IL9.	Predict the effect that a loss or malfunction of the Reactor Protection System (RPS) will have on the Source Range Monitoring (SRM) System.	30

Objective #	Objective Description	Pg. #
IL10.	Given a copy of Technical Specifications and various plant conditions: a. Identify any condition that does not comply with applicable Limiting Conditions for Operation (LCOs). b. Select the bases for applicable Limiting Conditions for Operation (LCOs) which pertain to the Source Range Monitoring (SRM) System. c. (SRO ONLY) Determine the actions required for each condition of non-compliance.	72
IL11.	(SRO Only) Given a copy of Emergency Action Levels (EALs), recognize any condition which meets the criteria to declare an event.	77
IL12.	Identify the purpose of the Intermediate Range Monitoring (IRM) System.	35
IL13.	Select the range of Intermediate Range Monitoring (IRM) sensitivity in terms of % power.	35
IL14.	Identify the mechanism for gamma compensation in the Intermediate Range Monitoring (IRM) System.	42
IL15.	Identify the physical arrangement of the Intermediate Range Monitoring (IRM) detectors within the Reactor Pressure Vessel (RPV).	36
IL16.	Determine which Intermediate Range Monitoring (IRM) channels provide inputs to Reactor Protection Trip Systems "A" and "B".	37, 44
IL17.	Identify the function of the following Intermediate Range Monitoring (IRM) System components:	
	a. Voltage Preamplifier	40
	b. Amplifier And Attenuator Unit	41
	c. Inverter	42
	d. Mean Square Analog (MSA) Unit	42
	e. Output (operational) Amplifier	44
IL18.	Identify the power supplies to each Intermediate Range Monitoring (IRM) channel.	54
IL19.	Identify the conditions, including setpoints, in the Intermediate Range Monitoring (IRM) System that will generate control rod withdrawal blocks and reactor scrams, and identify the bypasses for each.	44, 46, 47

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Objective #	Objective Description	Pg. #
IL20.	Given a drawing of controls, indications, and/or alarms located in the Main Control Room:	47
	a. Determine the status of the Intermediate Range Monitoring (IRM) System	
	b. Predict the effect due to the manipulation of system controls	
IL21.	Predict the effect that a loss or malfunction of Intermediate Range Monitoring (IRM) System will have on:	
	a. Reactor Protection System (RPS)	67
	b. Reactor Manual Control System (RMCS)	67
	c. Reactor Power Indication	67
IL22.	Predict the effect that a loss or malfunction of the following would have on the Intermediate Range Monitoring (IRM) System:	
	a. Reactor Protection System (RPS) power supply	66
	b. Detectors	66
	c. Trip Units	66
IL23.	Identify the means of changing Intermediate Range Monitoring (IRM) detector position and the reason for doing so.	38
IL24.	Given a copy of Technical Specifications and various plant conditions:	74
	a. Recognize any condition that does not comply with the applicable Limiting Conditions for Operation (LCOs).	
	b. Summarize the bases for those Limiting Conditions for Operation (LCOs) which pertain to the Intermediate Range Monitoring (IRM) System.	
	c. (SRO ONLY) Determine the actions required for any condition of non-compliance.	
IL25.	(SRO Only) Given a copy of Emergency Action Levels (EALs), recognize any condition which meets the criteria to declare an event.	77

References:

1. LGS Updated Final Safety Analysis Report (UFSAR), Volume 9, Sections 7.1, 7.6, 7.7
2. LGS SRM Operation And Maintenance Instructions (GEK-7326A)
3. LGS Operation And Maintenance Instructions (GEK-75680, GEK-75760)
4. Elementary Diagrams, 8031-M-1-C51-1010, I.E.D. Neutron Monitoring System
5. Elementary Diagrams, 8031-M-1-C51-1050-E, Start-Up Range Detector Drive Control System
6. Elementary Diagrams, 8031-M-1-C51-1070-E, Start-Up Range Neutron Monitoring System
7. LGS Schematic Diagrams E-0032, E-0108, E-0120, E-0126, E-0620
8. LGS Technical Specifications
9. E-*AY160, Loss Of *A RPS UPS Power
10. E-*BY160, Loss Of *B RPS UPS Power
11. GP-2, Normal Reactor Startup
12. GP-2 Appendix 1, Reactor Startup And Heatup
13. IC-11-00443, Operational Adjustment Of Source Range Monitors
14. ON-109, Total Loss Of The SRM, IRM Or APRM Systems
15. S74.0.G, Abnormal SRM/IRM Indications
16. ST-2-074-642-*, Source Range And Intermediate Range Neutron Monitor Pre-Shutdown Functional Test
17. ST-6-107-882-*, IRM Operability Verification
18. ST-6-107-883-*, SRM Operability Verification
19. ST-6-107-884-*, Neutron Monitoring System Overlap Verification On Startup
20. ST-6-107-886-*, Neutron Monitoring System Overlap Verification On Shutdown
21. T-100, Scram/Scram Recovery
22. T-101, Reactor Pressure Vessel (RPV) Control
23. LGS PEP I0006614, Loss Of 2BY160
24. LGS PEP I0010989, SRM Channel "B: Fuses Incorrectly Installed On The "B" Channel IRM, Due To Incorrect Clearance
25. LER 86-013-00, Clinton Power Station, Operator Error Resulting In RPS Actuation
26. LGS CR 150243, Event Investigation, Inability To Immediately Demonstrate Neutron Monitoring System Overlap During Limerick-2 BOC-8 Startup
27. IR 00165221, Assignment 94
28. IR 00493128, Assignment 42
29. IR 00718791, TS Bases For 3.3.7.5 Requires Enhancement

30. SEN-185, Significant Event Notification: Recurring Event, Inappropriate Continuous Control Rod Withdrawal From Subcritical Conditions
31. SEN-228, Recurring Event, Reactor Overpower Condition - Improper Adjustment Of Excore Power Range Channels Due To A Personnel Error
32. OE 9121, Reactor Scram During Startup Due To IRM Upscale
33. LER 88-012-01, Reactor Protection System Actuation On Intermediate Range Monitor Upscale Due To A Personal Error
34. IR 00493132, Assignment 70
35. LER 2009-001-00, Neutron Flux Accident Monitoring Instrumentation Inoperable

Materials:

A. Instructor:

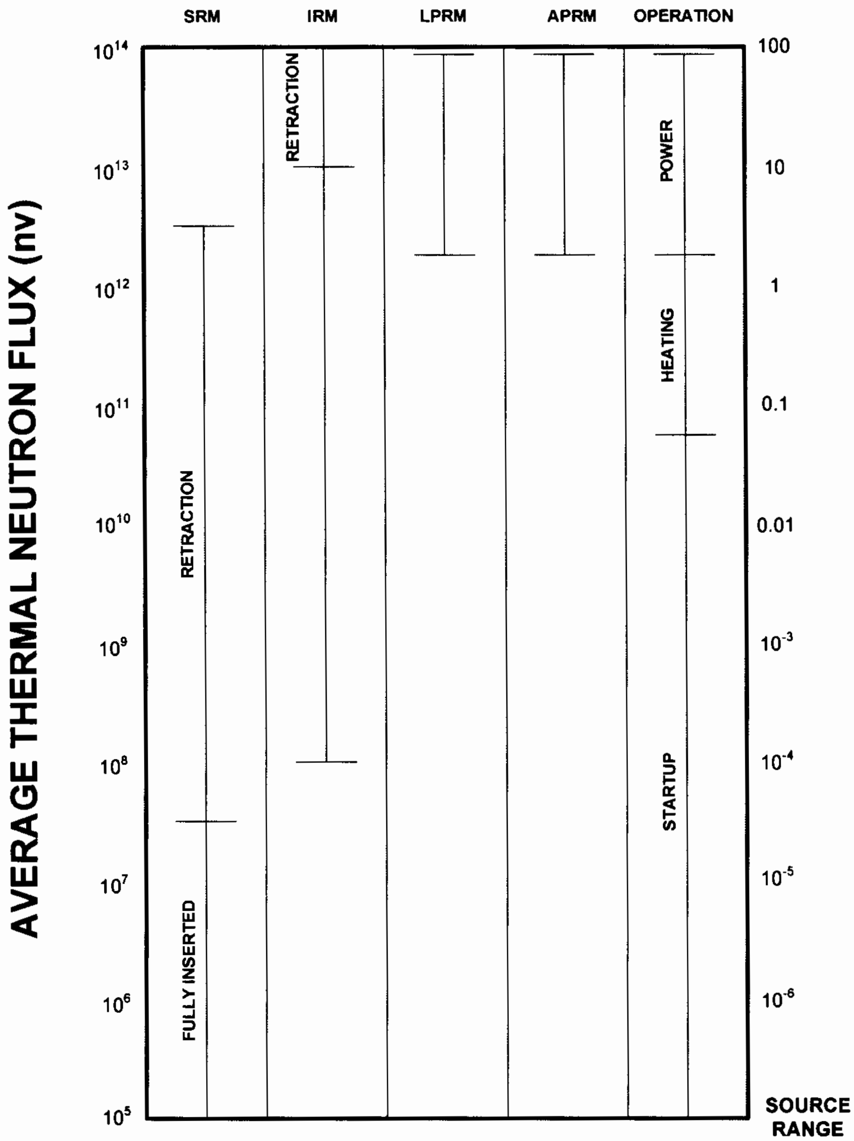
1. Whiteboard, Markers, and Erasers
2. PowerPoint Presentation LGSOPS0074 and one gun projector
3. Student Handout(s)

B. Student:

1. Pen/Pencil
2. Notepad
3. Student Handout(s)

Content/Skills	Activities/Notes
I. INTRODUCTION	***Suggested Instructional Methods And Media***
A. Instructor Introduction	*Classroom lecture with facilitated discussions, using appropriate questioning techniques. Suggested media includes the associated PowerPoint presentation and visual aids attached to this lesson plan*
B. Instructional Method Lecture utilizing questioning, discussion, and visual aids	
C. Read Objectives	
II. OVERVIEW	
A. Purpose Of Nuclear Instrumentation	
1. Provides continuous indication of: a. Reactor Power level b. Rate of change of Reactor Power	Instructor should emphasize that Nuclear Instrumentation is the only indication for the status of the controlled fission reaction occurring inside the reactor core
2. Provides indication from: a. COLD SHUTOWN conditions to b. Rated power	**INSTRUCTOR NOTE** LGSOPS0074 presentations for Licensed and Equipment Operator candidates should focus on areas of this Lesson Plan that are important to the respective job position and Objectives, and need not address all sections of this Lesson Plan
3. Provides inputs to: a. Reactor Protection System (RPS) b. Reactor Manual Control System (RMCS) c. Process Computer	
III. PRESENTATION	
A. System Overview	
1. Three ranges of Nuclear Instrumentation are required:	
a. Source Range	From ~ 10^{-9}% to 10^{-3}% power
b. Intermediate Range	From ~ 10^{-4}% to 40% power
c. Power Range	From ~ 0% to 125% power

RANGES OF NEUTRON MONITORING SYSTEM



2. Each system consists of:

   a. In-core detectors (fission chambers) which develop a signal indicative of neutron flux

   b. Signal conditioning circuitry which converts the detector output to a usable signal

c. Provides outputs for:

1) Continuous indication of Reactor Power level for Main Control Room (MCR) Operators

2) Trip circuits to provide alarms and protective functions

3) Process Computer for core performance evaluation

3. Continuous indication is necessary to ensure that the Operators are aware of Reactor Power level and that protective features are available to back up the Operator should an unsafe condition occur.

Objective EO1

a. Continuous indication is verified by performing overlap Surveillance Test (ST) Procedures.

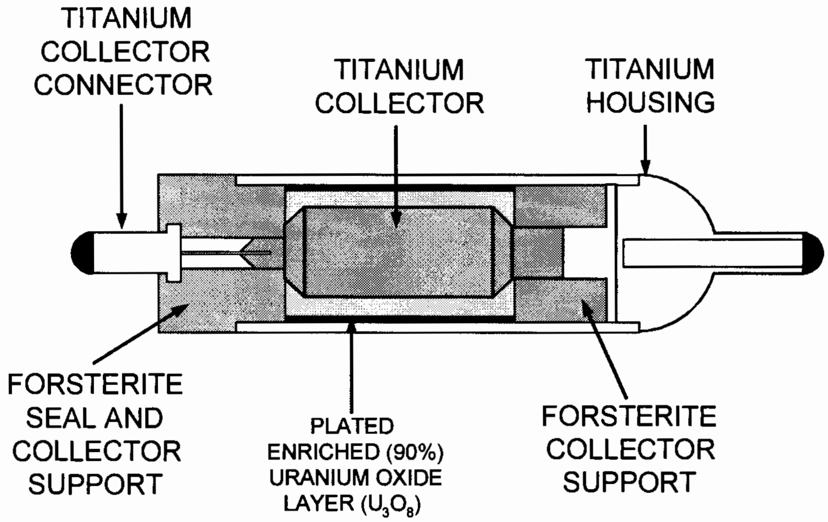
ST-6-107-884-*, Neutron Monitoring System Overlap Verification On Startup, verifies proper overlap between the SRM and IRM Systems

b. These ST procedures ensure the next range of neutron flux monitoring is responding before the currently used channels go either upscale (during startup) or downscale (during shutdown).

ST-6-107-886-*, Neutron Monitoring System Overlap Verification On Shutdown, verifies proper overlap between the IRM and APRM Systems

IV. FISSION CHAMBER DETECTORS

FISSION CHAMBER DETECTOR



1. Dimensions
   a. 1.0-inch long
   b. 0.25 inches in diameter
2. Outer Can
   a. Forms the outer boundary of the detector
   b. Constructed of Titanium
   c. Inner surface coated with Uranium Oxide
      1) Form of U_3O_8
      2) Enriched to > 90% U-235
3. Inner Electrode
   a. Constructed of Titanium
   b. Electrically insulated from Outer Can
4. Detector filled with Argon gas
5. Direct Current (DC) voltage applied to the detector
   a. Voltage varied according to application

B. Fission Chamber Operation

1. **Neutron Pulse**: generated by a thermal neutron causing a fission within the U-235 lining

   a. Resulting fission products have high mass, kinetic energy (20 to 22 MeV) and charge

   b. Cause ionization of the Argon gas

   c. Free electrons are accelerated towards positively charged Center Electrode, generating an output pulse

   d. Positive Argon ions accelerate towards Outer Can, generating an output pulse

2. **Gamma Pulse**: gamma rays can enter the chamber and directly ionize the Argon gas, thus creating an output pulse

3. Gamma pulses have a smaller pulse height than neutron pulses

C. Detector Utilization

1. Detector characteristics are varied according to application

   a. Weight and thickness of U_3O_8 layer

   b. Type and pressure of fill gas

   c. Applied voltage: as applied voltage is raised, the sensitivity of the detector goes up.

2. SRM Detectors

   a. Designed to operate in Proportional Region, but are currently being operated in the Ionization Region

   b. Gas amplification raises detector sensitivity

3. IRM And Local Power Range Monitor (LPRM) Detectors

   a. Designed to operate in Ionization Region

   b. Due to:

      1) Reduced sensitivity requirements

      2) More linear output characteristics

Objective EO2

NOTE: due to the high applied voltage differential, the electrons are accelerated towards the Center Electrode. Their high kinetic energy will cause secondary ionization. This is called "Gas Amplification" and raises SRM detector sensitivity.

Gamma Sources:

1. Fission related capture gammas
2. Decay and background gammas

Fission Fragment Energy:

1. 160 to 170 MeV pulses
2. Results in large pulse

Gamma Energy:

1. 7.0 to 8.0 MeV
2. Results in small pulse

V. SRM SYSTEM

A. Purpose

1. The SRM System provides neutron flux information during reactor startup, refueling, and low neutron flux level operations. The rate of change of neutron flux is provided as Reactor Period indication.

   Objective IL1

2. SRM detectors provide neutron flux/reactor power indication from COLD SHUTDOWN conditions to the Intermediate Range, over a range equivalent to 10^{-9} % power to 10^{-3} % power.

   Objective IL2

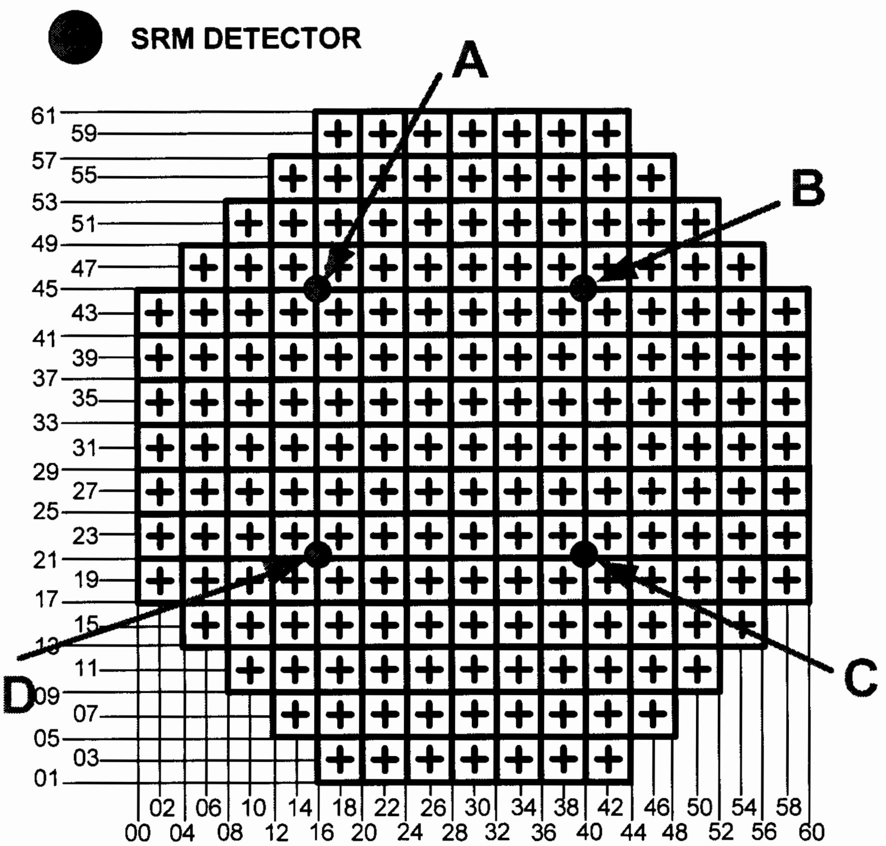
B. Detectors And Detector Drives

1. Location and Construction

   Objective IL3

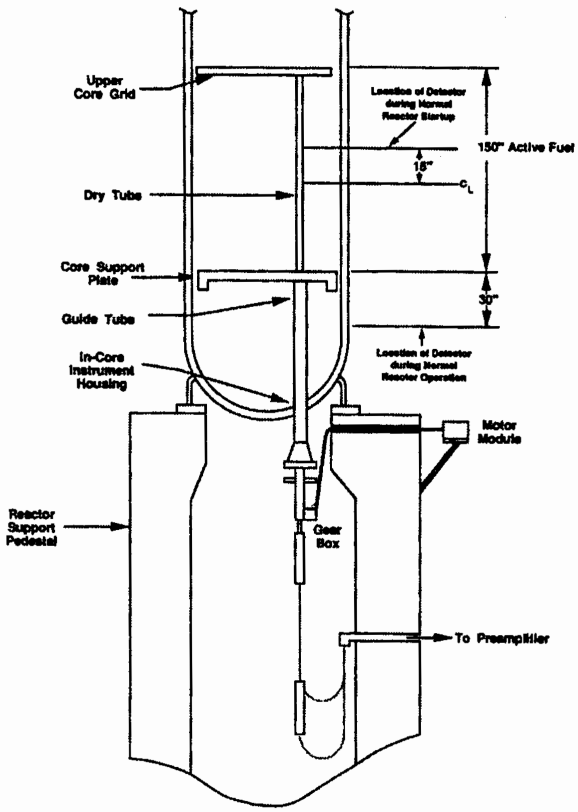
   a. Each of the four SRM detectors is located within a Dry Tube in the center of each core quadrant, with one detector located in each quadrant of the core.

   b. The Dry Tube serves as a pressure boundary.



Content/Skills	Activities/Notes
c. The axial position of each detector is variable between 30 inches below the bottom of the active core to 15 inches above core mid-plane (based on 150 inch fuel length).	These positions are determined by mechanical travel stops, electrical travel stop controls "full in" at 15 inches above mid-plane and "full out" at 30 inches below the core

DETECTOR IN-CORE DETAIL



d. The detector itself is located within a Shuttle Tube

1) The Shuttle Tube is enclosed by the Dry Tube

2) The Shuttle Tube allows axial positioning of the detector by means of the drive assembly

2. Drive Assembly

    Reference: C51-1050, Sheet 2

    a. Each SRM detector is driven by a separate motor

    Powered from Lighting Panel 1L34(2L107) (208 VAC) through Fuse Panel *0C008 located on the 253 foot elevation of the Reactor Enclosure

        1) Engages drilled detents in shuttle tube by a gear drive assembly

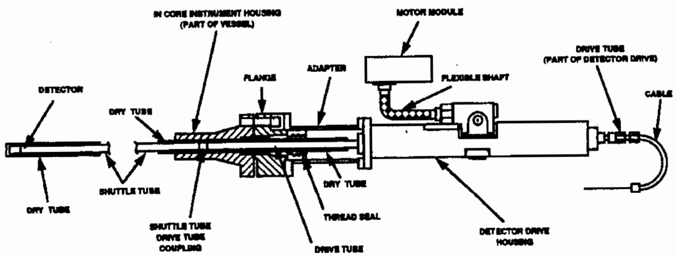
    Remote motor to:

    1) Reduce undercore crowding

    2) Reduce radiation aging of components

        2) Drive modules drive the SRM detectors into/out of the core at a rate of 3.0 feet per minute.

DETECTOR DRIVE ASSEMBLY



<u>Detector drive speed</u>: 3.0 feet/minute

It takes ~ 3 minutes, 20 seconds to drive a fully withdrawn SRM detector fully into the core.

C. SRM Detectors

1. <u>Purpose</u>: detect neutron flux in the reactor core

2. Each SRM detector is a fission chamber type detector operated in the Ionization Region of the Gas Amplification Curve.

    a. The SRM detectors are approximately 1.0-inch long and 0.25 inches in diameter.

    Gas space is ~ 0.020 inches

    b. Case and collector are fabricated from Titanium.

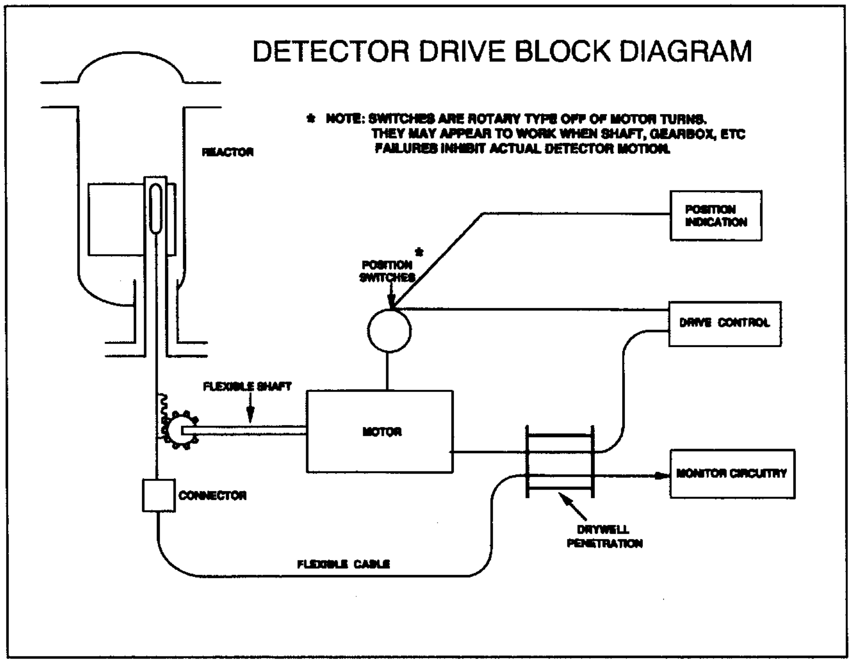
    Forsterite is the insulating material

    c. Inner surface of the case is coated with highly enriched Uranium Oxide to provide neutron detection.

    > 90% enriched with U-235

    d. Inner volume is pressurized with Argon gas, an inert noble gas.

    Pressurized to about 220 psia



3. A high voltage (potential) is applied across the center electrode (collector) and the case during normal operating conditions.

   a. When ionizing radiation enters the cylinder, a finite number of ion pairs are formed in the Argon gas as a result of high energy particles stripping away electrons from the Argon atoms.

   b. Under the influence of the applied high voltage, the positive ions will migrate toward the case and the negative ions (primarily electrons) will migrate towards the center electrode.

   c. The result is that the electrons and positive ions will be collected and be evidenced in the detector circuitry as a small pulse of electrical charge.

Voltage is operationally determined, but typically about 400 VDC (reference IC-11-00443, Operational Adjustment Of Source Range Monitors). This potential causes the center electrode to be positive with respect to the case.

4. **<u>Neutron Pulses</u>**: generated by thermal neutrons causing fissions within the U-235 lining

   a. Resulting fission products have high mass, kinetic energy (20 to 22 MeV), and charge

   b. Cause ionization of the Argon gas

c. Free electrons are accelerated towards positively charged center electrode, generating an output pulse

d. Positive Argon ions are accelerated towards the outer can, generating an output pulse

5. **Gamma Pulses**: gamma rays can enter the chamber and cause direct ionization of the Argon gas, thus creating output pulses

   a. Gamma pulses have a smaller pulse height than neutron pulses

   b. This characteristic allows for detection of neutron pulses alone.

Gamma Sources:

- Fission related capture gammas
- Decay and background gammas

Gamma Interactions With Matter:

- Photo-electric effect
- Compton Scattering
- Pair Production

D. Electronic components process and amplify detector developed signals

1. Consist of:

   a. Pulse Preamplifier

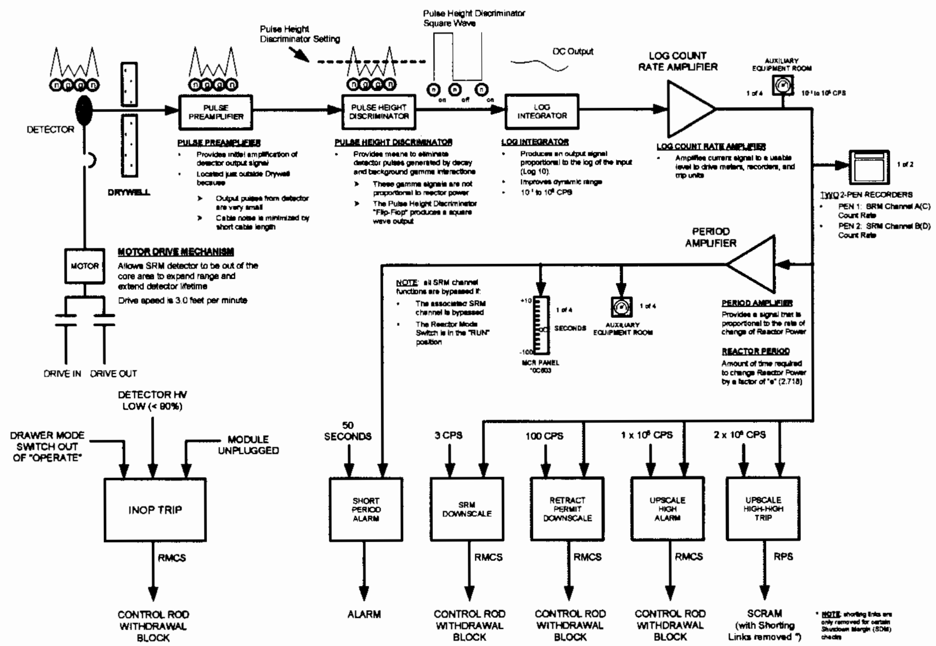
   b. Pulse Height Discriminator

   c. Log Integrator

   d. DC Amplifier (Log Count Rate Amplifier)

   e. Reactor Period Circuit

ATTACHMENT 1



E. Pulse Preamplifier

1. Provides the initial amplification of SRM detector output. Due to the small magnitude of the energy pulses produced by the SRM detectors, a significant stage of pulse amplification must be introduced in order for the pulses to be useable.

Objective EO3
Objective IL4.a

2. The Pulse Preamplifier converts the pulse energy to a current signal, and amplifies this signal.

3. The Pulse Preamplifier is located immediately outside the Drywell

   a. As close to the SRM detectors as possible, while still remaining accessible during operation

   b. Must be close to the SRM detectors since:

      1) Output pulses from the SRM detectors are very small (signal must be amplified prior to transmission to the SRM Equipment Drawer in the Auxiliary Equipment Room (AER).

      2) Cable noise level would be excessive if the pulses had to travel all the way to the AER before being amplified.

<u>**NOTE**</u>: even with the Pulse Preamplifier, arc welding in the Reactor Enclosure and/or lower part of the Drywell can cause erroneous signals

<u>**Operations Fundamentals**</u>

SOLID UNDERSTANDING OF PLANT DESIGN AND SYSTEM INTERACTIONS

- **Technical Human Performance** *(Questioning Attitude)*

4. The output of the Pulse Preamplifier is connected to the SRM Equipment Drawer in the AER through a shielded cable.

F. Pulse Height Discriminator

Objective IL4.b

1. At the low Reactor Power levels of operation found in the Source Range, it is necessary to discriminate against the pulses produced by gamma interactions, since at these levels of neutron flux, decay or background gamma signals can overshadow the signals produced by neutrons.

2. The Pulse Height Discriminator provides the means to eliminate detector pulses generated by gamma interactions.

3. Discrimination Section

   a. The Pulse Height Discriminator receives current pulses from the output of the Pulse Preamplifier and performs the discrimination function based upon the amplitude (height) of the current pulses.

   b. Pulses generated by neutron events are much higher in amplitude than those generated from gamma events.

   c. The Pulse Height Discriminator is set to pass only those pulses with a height greater than the discrimination level.

   d. Because the height of gamma generated pulses is less than the discrimination level, they are eliminated.

   e. Neutron pulses, being greater than the discrimination level, are sent forward.

4. The output of the Pulse Height Discriminator is a voltage pulse signal proportional to the neutron count rate. It is sent to the Log Integrator for further processing.

NOTE: the Pulse Preamplifier raises the signals from both gamma and neutron pulses. This results in a larger difference between the two and aids the discrimination process.

G. Log Integrator

Objective IL4.c

1. The Log Integrator converts the linear input signal to a DC output signal, which is proportional to the common log (log 10) of the input signal.

2. Logarithmic scaling provides an indication with an equal amount of deflection for each of the six decades.

10^{-9}% to 10^{-3}% power

3. Displaying the required decades of the Source Range on a linear meter would be highly impractical.

4. The output of the Log Integrator is sent to a final stage of DC amplification.

H. DC Amplifier

Objective IL4.d

1. The DC Amplifier converts the DC current signal from the output of the Log Integrator to a value of 0 to 10 VDC, a value which can be used to drive meters, recorders, and trip relays

2. The output of the DC Amplifier drives the following:

   a. Local meters (AER)

   b. Remote count rate indicators (MCR)

   c. Trip units

   d. Reactor Period circuit

I. Reactor Period Circuit

1. Converts the log input signal to a rate signal

Objective IL4.e

2. Uses an Operational Amplifier connected as a Differentiator

3. The Reactor Period meters indicate the amount of time required to change Reactor Power by a factor of "e," where "e" is the base of the natural logarithm, and is equal to ~ 2.718.

This is the definition of Reactor Period

The factor for a one-half decade rise/drop in Reactor Power is 3.16

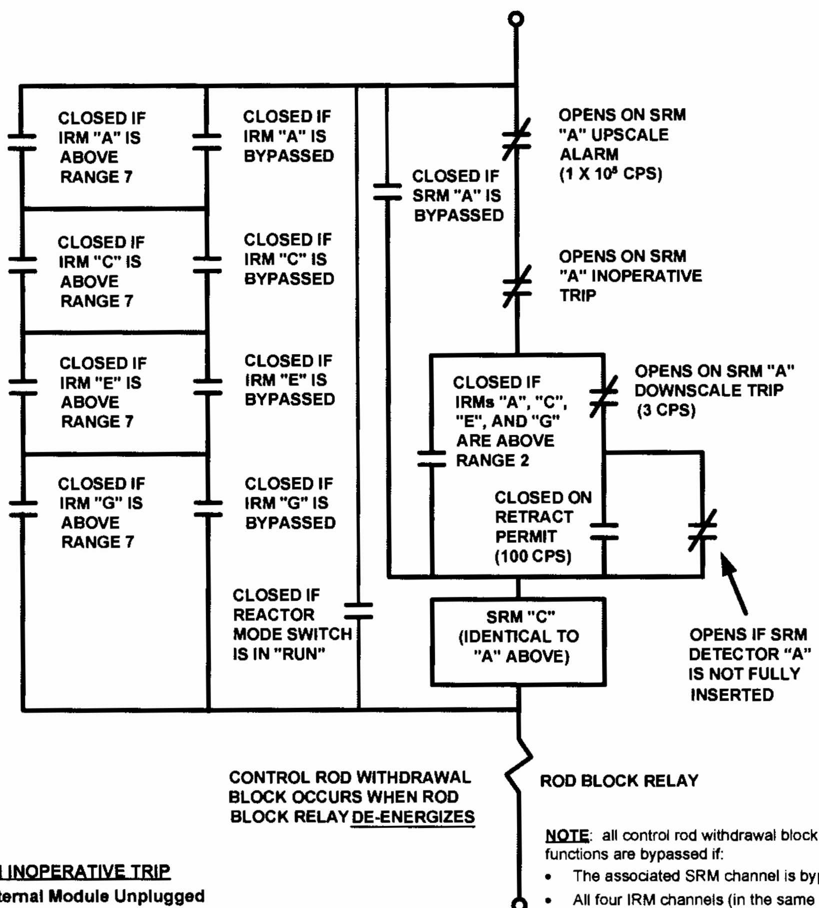
4. Reactor Period indication allows the Reactor Operator to monitor and control the rate of change of Reactor Power.

J. Control Rod Withdrawal Blocks

Objective EO4.b
Objective EO5
Objective IL6

ATTACHMENT 2
<u>AND</u> ATTACHMENT 6

SRM CHANNEL "A"
CONTROL ROD BLOCK SIGNALS AND BYPASSES



<u>NOTE</u>: all control rod withdrawal block functions are bypassed if:

- The associated SRM channel is bypassed
- All four IRM channels (in the same RPS trip system) are on Range 8 or above
- The Reactor Mode Switch is in the "RUN" position

<u>SRM INOPERATIVE TRIP</u>
- **Internal Module Unplugged**
- **Channel Mode Switch not in "OPERATE"**
- **Low detector voltage**

Content/Skills	Activities/Notes
1. The SRM channels will generate control rod withdrawal block signals under the following conditions:	
a. Upscale: channel output ≥ 1 x 10^5 CPS	
b. Inoperative (INOP): channel INOP caused by any one of the following:	
1) Low detector high voltage (< 90% of actual high voltage value)	
OR	
2) Internal module unplugged; other hardware failures	
OR	
3) Channel Mode Switch **not** in "OPERATE" (on Instrument Drawer in AER)	
c. Downscale: channel output ≤ 3 CPS	
d. SRM Detector Not Fully Inserted and ≤ 100 CPS: bypassed if all four associated IRM channels (in the same trip system) are on Range 3 or higher (or are bypassed)	This prevents premature withdrawal of the SRM detectors
2. All SRM channel control rod withdrawal block signals will be bypassed if:	One SRM channel at a time can be bypassed in the MCR
a. The SRM channel is bypassed	
b. All four associated IRM channels (in the same trip system) are on Range 8 or higher (or are bypassed)	
c. The Reactor Mode Switch is in the "RUN" position	

K. RPS Trip Signals (Scram Signals)

1. The SRM channels generate a scram signal and send it to the RPS on High-High count rate at ($\geq 2 \times 10^5$ CPS) or on an INOP condition (loss of power only)

2. This scram signal is only activated during initial fuel load and startup, and during the performance of Shutdown Margin (SDM) demonstration following refueling

    a. This trip is normally bypassed by installation of shorting links

    b. Shorting links are removed as specified above (during SDM demonstration)

        1) The shorting links are physically located in Panels *0C609 and *0C611 (AER)

        2) The SRM Upscale High-High Trip is bypassed when the RPS shorting links are installed

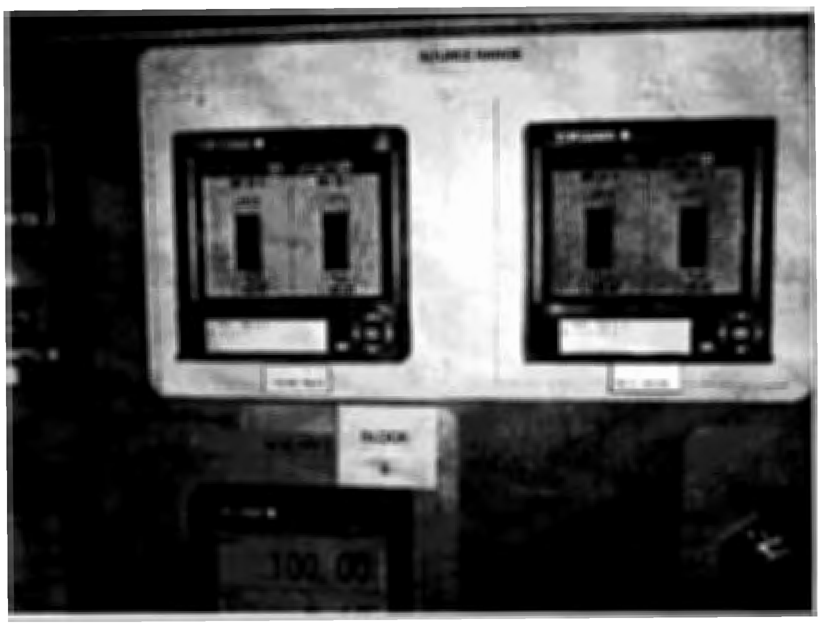
Reference: C71-1020-E-008, Sheet 1

Objective EO4.a
Objective IL6

SRM channel RPS trips are active when the shorting links are removed

Removal of shorting links also places all SRM and IRM channel trips in the non-coincident mode (i.e., any one SRM or IRM channel trip causes a full scram). Average Power Range Monitor (APRM) and Oscillation Power Range Monitor (OPRM) channel trips remain in an "any two (or more) out of four" logic configuration

L. SRM System MCR Instrumentation And Alarms



Objective IL7.a
Objective IL7.b

1. SRM System Controls, Instrumentation, and Alarms located on MCR Panel *0C603
   a. Count Rate Indication
      1) Two Yokogawa DX1000N model paperless recorders
         a) SRM Channels "A" and "C"
            - XRX-M1-*R602A
            - Red Display: SRM Channel "A"
            - Green Display: SRM Channel "C"
         b) SRM Channels "B" and "D"
            - XRX-M1-*R602B
            - Red Display: SRM Channel "B"
            - Green Display: SRM Channel "D"
         c) Each recorder has a range from 10^{-1} to 10^{6} CPS

      2) SRM count rate indications are also provided on PMS Format 135

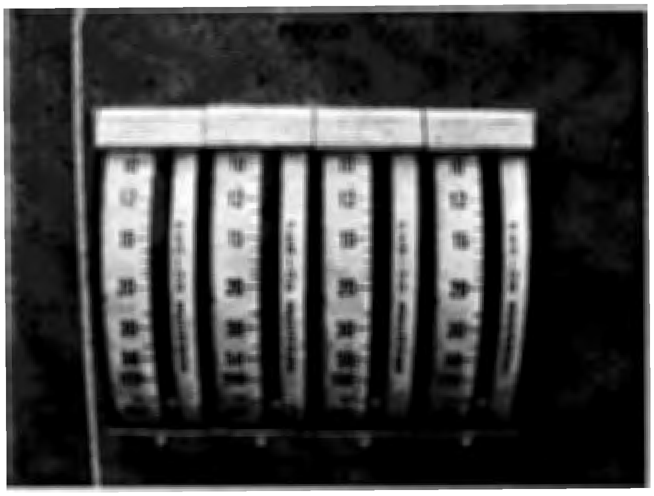
<u>SRM Channel "A"</u>: E*109
<u>SRM Channel "B"</u>: E*133
<u>SRM Channel "C"</u>: E*110
<u>SRM Channel "D"</u>: E*134

b. Reactor Period Indication

1) Four Reactor Period meters

2) Each has a range of +10 to -100 seconds, with an electrical zero at infinity

3) Each channel also has a Reactor Period meter on its associated instrument drawer in the AER

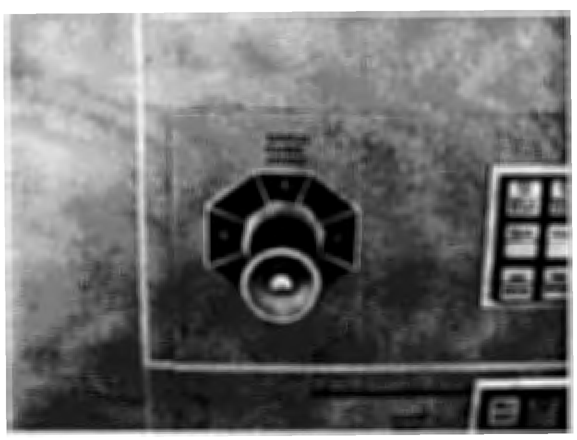


c. SRM Channel Bypass Joystick

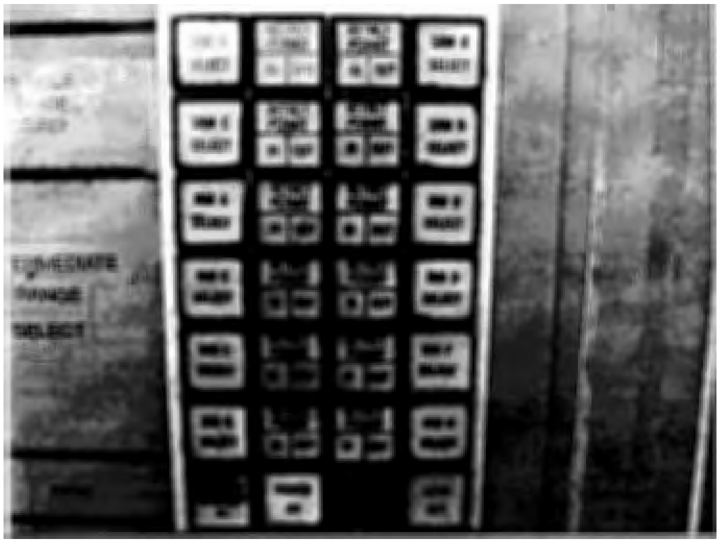
1) 5-position joystick: "SRM A," "SRM B," "SRM C," "SRM D," and neutral (no SRM channel bypassed)

2) Only one SRM channel may be bypassed at a time

3) Channel bypass defeats all control functions, but does not affect channel indications



d. Detector Drive Controls and Indicating Lamps



1) Four Detector Selector Switches, used for selecting an SRM detector to be moved
   a) "ON/OFF," maintained-contact pushbutton for each detector
   b) Backlight illuminates when associated detector is selected
2) One Drive-In Pushbutton, with two indicating lamps
   a) "ON/OFF," maintained-contact pushbutton
   b) If depressed again while the selected detectors are not yet fully inserted, detector motion will stop
   c) If depressed again, detector motion stops
   d) Also used for inserting IRM detectors, if selected
   e) One "DRIVE IN" indicating lamp on pushbutton, illuminated when selected detectors have been selected for movement into the core.
   f) One "DRIVING IN" indicating lamp, on pushbutton. When illuminated, indicates selected detectors are in the process of being driven into the core.

The "DRIVE IN" indicating lamp remains illuminated after the selected detectors have reached the fully inserted position

The "DRIVING IN" indicating lamp extinguishes when the selected detectors have reached the fully inserted position

3) One Drive Out Pushbutton and indicating lamp

   a) "ON/OFF," momentary-contact pushbutton

   b) Push and HOLD: selected detectors drive out of the core as long as the pushbutton is depressed

   c) Released: selected detectors stop moving

   d) Also used for withdrawing IRM detectors, if selected

   e) "DRIVE OUT" indicating lamp on pushbutton, illuminated while selected detectors are moving out of the core (pushbutton depressed)

4) SRM Position Indicating Lamps, one set of lamps for each SRM detector

   a) "IN": detector is full-in

   b) "OUT": detector is full-out

   c) "RETRCT PERMIT": illuminated when detector is not fully withdrawn **AND** any one of the following exists.

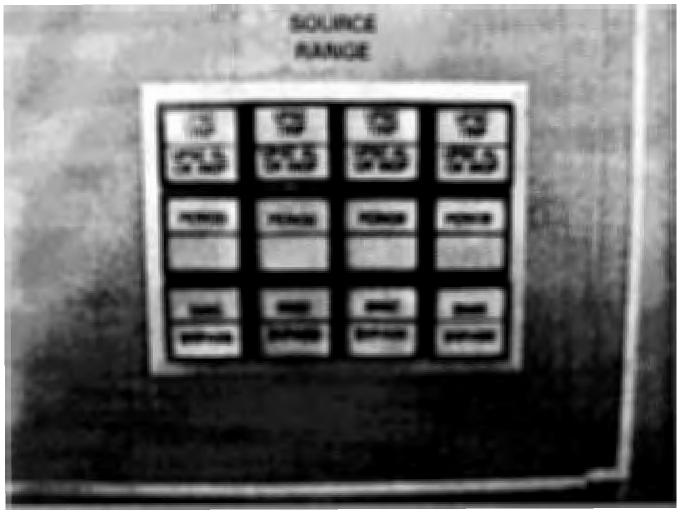
      (1) SRM channel is bypassed

      (2) SRM channel indicating $\geq$ 100 CPS

      (3) All four associated IRM channels (in same trip system) are on Range 3 or above (or bypassed).

e. SRM Channel Alarm Lamps

MCR Panel *0C603
Same setpoints as Alarms



1) Five indications for each detector

a) "UPSC TRIP" (red): setpoint $\geq 2 \times 10^5$ CPS

b) "UPSC AL OR INOP" (amber): setpoint $\geq 1 \times 10^5$ CPS or channel INOP

c) "PERIOD" (amber): setpoint ≤ 50 seconds

d) "DNSC" (white): setpoint ≤ 3 CPS

e) "BYPASS" (white): Manual Channel Bypass

2) Used in conjunction with annunciator to determine which SRM detector is causing the alarm

f. Annunciators on Panel *07 REACTOR

NOTE: These annunciators have the same setpoints and bypasses as discussed previously for the SRM Alarm Lamps

1) "SRM PERIOD" (≤ 50 seconds) — Window F-4

2) "SRM DOWNSCALE" (≤ 3 CPS) — Window G-4

3) "SRM UPSCALE/INOPERATIVE" ($\geq 1 \times 10^5$ CPS or channel INOP) — Window H-4

4) "SRM RETRACTED WHEN NOT PERMITTED"

Window I-4

a) SRM detector not fully inserted and associated channel indicating $\leq$ 100 CPS, except when bypassed as previously discussed

b) This alarm is associated with a control rod withdrawal block, it does <u>not</u> restrict detector motion

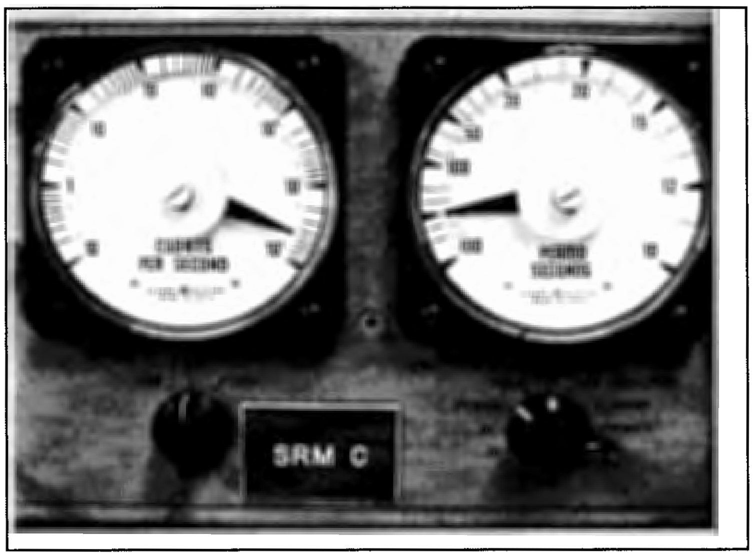
M. SRM Channel AER Controls And Indications

1. Meters

Objective EO4.c

a. Count Rate Meter, supplies indications of counts from 10^{-1} to 10^{6} CPS.

b. Reactor Period Meter, indicates a Reactor Period of -100 to +10 seconds, with an electrical zero at infinity



2. Indicating Lamps, all lamps on the panel seal in, and must be reset at the panel to clear them.

   a. "DOWNSCALE" lamp, illuminates when count rate drops to ≤ 3 CPS.

   b. "RETR PERM DWNSCL" lamp, illuminates when count rate is ≤ 100 CPS.

   c. "UPSCALE ALARM" lamp, illuminates when count rate is ≥ 1×10^5 CPS.

   d. "PERIOD" lamp, illuminates when Reactor Period is ≤ 50 seconds (getting shorter).

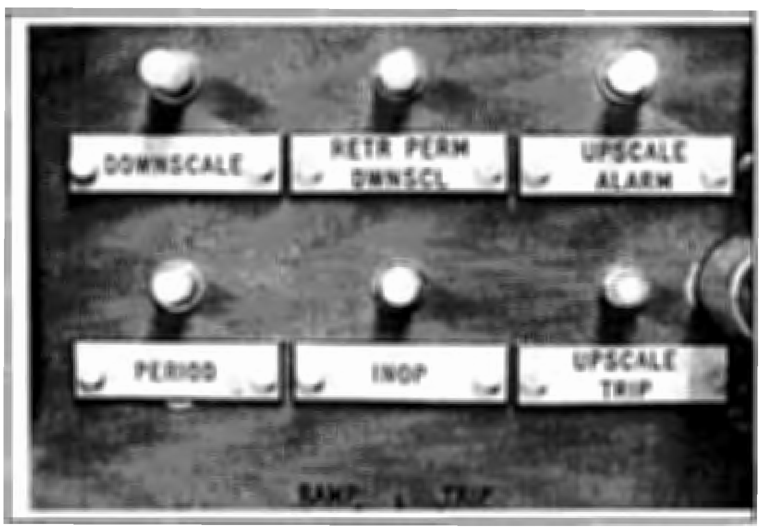
   e. "INOP" lamp, turns on under <u>any</u> of the following conditions:

      1) Low detector high voltage (< 90% of actual high voltage value)

      2) Internal module unplugged; other hardware failures

      3) Channel Mode Switch **<u>not</u>** in "OPERATE" position

   f. "UPSCALE TRIP" lamp, illuminates when count rate is ≥ 2×10^5 CPS.



   g. <u>"BYPASS" lamp (white)</u>: located above SRM channel drawer; illuminates if SRM Channel Bypass Joystick is positioned to bypass the associated SRM channel

3. Switches

   a. Channel Mode Switch, normally in "OPERATE" position; other positions used for various testing purposes.

   b. Channel Ramp/Trip/Reset Switch

      1) The reset function is used to reset the lights on the panel.

      2) The ramp function is used to test the meter functions of the system as well as the Reactor Period function.

N. Power Supplies

Objective IL5

1. Power to the + 20 VDC power supplies required by the SRM channels is supplied by 120 VAC RPS/UPS

2. RPS Bus *AY160 (Circuit #2) supplies power to:

   <u>NOTE</u>: also supplies power to Nuclear Steam Supply Shutoff System (NS^4)

   a. SRM Channel "A" and Channel "C" circuits and amplifiers

   b. SRM Channel "A" and Channel "C" Main Control Room (MCR) alarm lamps, including detector drive "RETRCT PERMIT"

3. RPS Bus *BY160 supplies power to:

   <u>NOTE</u>: also supplies power to NS^4

   <u>NOTE</u>: there are two separate SRM recorders

   a. SRM Channels "A" through "D" MCR recorders (Circuit #5)

   b. SRM Channel "B" and Channel "D" circuits and amplifiers (Circuit #2)

   c. SRM Channel "B" and Channel "D" MCR alarm lamps, including detector drive "RETRCT PERMIT" (Circuit #2)

4. Voltage Regulator

   a. Converts + 20 VDC input power to a well regulated + 15 VDC output

   b. Output is provided for use by:

      1) SRM channel circuitry (including Pulse Preamplifier)

      2) High Voltage Power Supply

5. High Voltage Power Supply
   a. Receives + 15 VDC from the Voltage Regulator
   b. Converts this to a high voltage output
   c. This high voltage output, normally 400 VDC, is supplied to the SRM detector, and provides the necessary detector polarizing potential.
   d. If the output of the High Voltage Power Supply lowers below a setpoint, an SRM channel INOP trip will occur.
6. Miscellaneous Power Supplies
   a. 120 VAC *0Y202 (Circuit #13) supplies power to:

      Reference: C51-1050, Sheet 2

      1) All SRM detector drive relays

         Loss of this bus prevents SRM detectors from being moved

      2) All SRM detector drive indicating lamps, except detector drive "RETRCT PERMIT" indicating lamp
   b. Non-Safeguard Load Center *24B (Lighting Panel 1L34, 2L107) powers SRM Channel "A" through "D" detector drive motors.

      Loss of this bus also prevents SRM detectors from being moved

O. System Operation

1. <u>Reactor Startup</u>

   <u>NOTE</u>: Reactor Startups are performed in accordance with GP-2, Normal Plant Startup

   a. Initial Conditions
      1) SRM detector drives are fully inserted and all operational channels are indicating > 3 CPS (**<u>NOTE</u>**: an indication of > 3 CPS per channel ensures all channels are functioning properly).
      2) The "> 3 CPS" value may be reduced, provided the associated SRM channel has an observed count rate and the signal to noise ratio is <u>on or above</u> Technical Specification Figure 3.3.6-1, SRM Count rate Versus Signal To Noise Ratio.

         Instructor should ensure that students can locate this curve in Technical Specifications

      3) SRM and IRM channel "INOP" lamps are extinguished ("OFF").
      4) IRM channel Range Switches are on Range 1.

5) APRM channel "DNSC" lamps are illuminated ("ON").

b. Record initial SRM channel count rates in GP-2 Appendix 1, Reactor Startup And Heatup, for determination of count rate after three doublings.

c. Withdraw control rods while monitoring count rate and Reactor Period.

NOTE: control rods are withdrawn in accordance with GP-2 Appendix 1, Reactor Startup And Heatup

1) When any SRM channel exceeds three doublings, then single notch control rod withdrawal between position 04 to 36

2) Reactor critical (positive Reactor Period **AND** sustained rise on SRM channel instrumentation with no control rod motion indicates the reactor is critical)

Safety Emphasis

Criticality should be expected at any time. Close attention must be paid to all SRM channel indications when pulling control rods.

3) Record Critical Count Rate Data

4) Do not permit Reactor Period to be < 50 seconds during control rod withdrawal, as indicated on SRM channel Reactor Period meters

Reference: GP-2 Appendix 1, Reactor Startup And Heatup

d. Withdraw additional control rods, as necessary, to obtain a stable positive Reactor Period (150 seconds to 50 seconds).

Safety Emphasis

Maintain plant parameters within operational limits

e. Continue monitoring SRM and IRM channel response. When IRM channel response is observed, record unbypassed SRM channel count rate as required per ST-6-107-884-*, Neutron Monitoring System Overlap Verification On Startup.

f. Verify proper overlap by completing ST-6-107-884-*, Neutron Monitoring System Overlap Verification On Startup

1) At least one-half decade during each startup after entering OPERATIONAL CONDITION (OPCON) 2

2) Technical Specification Table 4.3.1.1-1, Footnote (b)

g. When SRM/IRM System overlap has been verified, then withdraw SRM detectors to maintain count rate 100 to 100,000 CPS

   1) If SRM detectors are withdrawn too soon, a control rod withdrawal block may occur ($\leq$ 100 CPS **and** detector not fully inserted)

   2) If SRM detectors are withdrawn too late, a control rod withdrawal block may occur ($\geq 1 \times 10^5$ CPS)

h. Fully withdraw SRM detectors when the IRM channels are indicating on Range 3 or above. This protects the fission chamber from the high neutron flux occurring in the core during power operations, thus extending detector lifetime.

2. **Event Investigation**: Inability To Immediately Demonstrate Neutron Monitoring System Overlap during Limerick-2 BOC-8 Startup

   LGS CR 150243

   Instructor should emphasize that this event demonstrates correct decision making

   a. During the Limerick startup following 2R07, it was not immediately possible to demonstrate satisfactory overlap between SRM and IRM channels

      March 21, 2003; 1330 hours, Cycle 8 BOC Startup

      1) Required by Technical Specification Table 4.3.1.1-1, Footnote (b)

         At least one-half decade during startup following entry into OPCON 2

      2) Required by ST-6-107-884-2, Neutron Monitoring System Overlap Verification On Startup

   b. It became necessary to stop startup activities in order to investigate and troubleshoot the problem.

   c. The consequences of this event were:

      1) No nuclear safety concern, however, it clearly complicated reactor startup by causing unplanned delays

         a) A delay in power ascension of approximately 2 hours

         b) Corresponding loss of production

         c) Neutron flux asymmetries such as were observed during this event can give rise to high notch reactivity worths

d. Significance of this event is that it had a negative impact on efficient reactivity management

e. Detailed Event Description

1) During 2R07 reactor startup, reactor declared critical (control rod 42-47)

2) SRM Channels "A" and "B" responded normally, but SRM Channels "C" and "D" were indicating approximately a factor of 10 in magnitude below SRM Channels "A" and "B."

Instructor should refer to detector location map

a) The reactor went critical with control rod 42-47, which is immediately adjacent to SRM detector "B."

b) The prior withdrawn control rod was control rod 18-47, which is located immediately adjacent to SRM detector "A."

c) The first two control rods in Group 2 were bypassed in the RWM and left fully inserted due to a channel bowing issue.

d) Since Quadrants I and II had adequate reactivity insertions, a critical reaction was sustained.

e) Quadrants III and IV did not have sufficient reactivity inserted for subcritical multiplication to raise the neutron population to the same level as in Quadrants I and II.

f) A similar response was noted on the IRM channels in each respective core quadrant

g) It was not possible to demonstrate overlap for IRM Channel "G" and IRM Channel "H"

All eight IRM channels must respond on-scale to support performance of ST-6-107-884-2, Neutron Monitoring System Overlap Verification On Startup

h) It was <u>not</u> possible to withdraw control rods to raise those IRM channel readings because further control rod withdrawal would result in reaching the SRM Upscale setpoint (1 x 10^5 CPS) on SRM Channels "A" and "B," which were located in the high flux area of the core.

3) Control rod withdrawal was stopped.

4) The Operating Crew determined that the reactor was stable in the observed condition.

5) It was determined that all activities that could result in a change in heat load on the reactor would be suspended to avoid an unplanned cooldown with resultant reactivity change.

6) During a meeting convened in the OCC it was decided to allow reactor coolant temperature to go up enough to insert enough negative reactivity to allow pulling control rod 42-47 full out, and establish criticality by pulling control rod 42-15 in Quadrant III and control rod 18-15 in Quadrant IV.

a) This resulted in a more symmetric control rod pattern with a significantly flatter neutron flux.

b) Overlap was easily demonstrated.

f. <u>Causes Of This Event</u>

1) The LGS method of clockwise control rod withdrawal tended to allow these neutron flux tilts to be observed at LGS as well as other sites.

2) Newer, high energy cores tend to achieve criticality in RWM Group 2. These control rods are notched to the "FULL OUT" position before moving on to the next control rod.

<u>**Operations Fundamentals**</u>

PRECISELY CONTROLLING PLANT EVOLUTIONS

- **Human Error Prevention** *(Use OOPS: if Outside of Procedures, Parameters or Process, then STOP and contact your supervisor)*

CONSERVATIVE BIAS TO PLANT CONDITIONS

- **Reactivity Management** *(Respond conservatively in the face of prolonged uncertainty or unusual reactor behavior to include rod insertion or a reactor scram)*

Some sites, such as PBAPS, use a "zigzag" method of control rod withdrawal

a) During the following control rod groups (Group 3 and later), control rods are banked out with core symmetry established at intermediate positions (04, 08, 12)

b) Banking mitigates the potential for large cold neutron flux tilts

g. Corrective Actions

1) Best practices for control rod withdrawal on approach to criticality will be evaluated to mitigate flux tilt.

2) Evaluated employment of “zigzag” control rod withdrawal patterns

3) Evaluated banking of RWM Group 2 control rods instead of full withdrawal to position 48.

h. Significance To The Reactor Operator

1) Be aware that these types of situations can occur

2) Remember that large, cold neutron flux tilts in the Source Range can result in high control rod notch worths, which can result in excessively short Reactor Period values

i. **Human Performance Successes** during this event

1) Situation was evaluated and a good decision implemented. Possible Error Precursors that were evidently overcome:

a) Schedule pressure

b) Can-do attitude

c) Changes/Departure from routine

d) Unexpected equipment conditions

3. Power Operation: SRM detectors are fully withdrawn from the core.

4. Reactor Shutdown (Scram)

   a. Per T-100, Scram/Scram Recovery, fully insert all SRM and IRM detectors

   b. Per T-101, RPV Control, fully insert all SRM and IRM detectors

   c. Ensures decaying neutron flux is tracked on instrumentation

5. Reactor Shutdown (Soft Shutdown)

   Reference: GP-3, Normal Plant Shutdown

   a. ST-2-074-642-*, Source Range And Intermediate Range Neutron Monitor Pre-Shutdown Functional Test, is performed

   This Surveillance Test (ST) procedure verifies the operability of the SRM and IRM channels, with the exception of required CHANNEL CHECKS and the functionality of detector "IN" and "OUT" lamps

   b. When the first IRM channel is on Range 8, ST-6-107-883-*, SRM Operability, is performed to verify functionality of the SRM detector "IN" and "OUT" lamps, and to verify channel operability by the performance of a CHANNEL CHECK

   When ST-6-107-883-*, SRM Operability, has been completed satisfactorily, the SRM channels are verified OPERABLE

P. System Malfunctions/Losses

1. Loss of RPS/UPS buses (*AY160/*BY160)

   Objective IL9

   a. RPS/UPS buses provide power to the SRM System

   b. SRM channels are fail-safe, so that a loss of power will cause a trip of the SRM channel losing power.

      1) If power is lost to any SRM channel, a control rod withdrawal block will be generated depending upon the operating condition.

      2) If the shorting links are removed, loss of power to any SRM channel will generate a full scram.

c. Loss of one of these buses also results in other events:

1) Drywell Chill Water (DWCW) System isolation

2) Reactor Enclosure Cooling Water (RECW) System isolation

3) Reactor Water Cleanup (RWCU) System isolation

4) Reactor Enclosure Heating, Ventilation And Air Conditioning (RE HVAC) System isolation

5) Refuel Floor (RF) HVAC System isolation

d. A loss of the SRM System is of relatively low concern during this scenario.

Reference:

- E-*AY160, Loss of *A RPS UPS Power
- E-*BY160, Loss of *B RPS UPS Power

2. Loss of SRM System

a. Loss of the SRM System will prevent monitoring Reactor Power when in the Source Range. Actions should be taken in accordance with ON-109, Total Loss Of The SRM, IRM Or APRM Systems.

Objective IL8a
Objective IL8b
Objective IL8c

b. Failure of the SRM System to function properly may also prevent/generate control rod withdrawal blocks/scrams depending upon plant operating conditions and system alignment.

c. During Refueling Operations, in accordance with Technical Specifications:

1) Immediately suspend all operations involving CORE ALTERATIONS

Technical Specification 3.9.2

2) Insert all insertable control rods

3) Suspend removal of individual control rods and associated mechanisms from the core and/or RPV.

Technical Specification 3.9.10 requires the SRM System to be OPERABLE per Technical Specification 3.9.2 in order to remove a control rod from the core

3. **Operating Experience**: Loss Of 2BY160

a. On February 7, 1997, Unit 2 control rod blade (CRB) and drive (CRD) maintenance was in progress.

Ref: LGS PEP I0006614

b. At 0910, 2BY160 was de-energized for the performance of ST-2-036-420-2 and ST-2-036-422-2 (RPS - Electrical Power Monitoring Channel Calibration/Functional Tests). This caused the following:

1) SRM Channels "B" and "D" were rendered inoperable due to loss of power

2) SRM recorders on MCR Panel 20C603 were de-energized

3) All SRM channel annunciators on Panel 207 REACTOR were in alarm

Panel 207 REACTOR Annunciators

- Window F4, "SRM PERIOD"
- Window G4, "SRM DOWNSCALE"
- Window H4, "SRM UPSCALE/INOPERTIVE"
- Window I4, "SRM RETRACTED WHEN NOT PERMITTED"

c. It was not recognized that SRM Channels "A" and "C" were also inoperable, due to the loss of power to the recorders on MCR Panel 20C603.

d. No CORE ALTERATIONS were ongoing and all CRDs to be worked had been previously withdrawn and uncoupled. Four of these CRDs had been retorqued.

e. At approximately 1050, the Reactor Operator brought to the attention of Shift Supervision (SSV) that the SRM recorders were no longer energized.

Operations Fundamentals

CLOSELY MONITORING PLANT CONDITIONS

- **Control Board Awareness** *(Maintain continuous awareness of major plant parameters)*
- **Operator Rounds** *(Report abnormal conditions to supervision in a timely manner)*

f. Technical Specification 3.9.2 was reviewed and it was determined that the action for Technical Specification 3.9.2, which requires two OPERABLE SRM detectors, was complied with, because no CORE ALTERATIONS were in progress and all insertable control rods were inserted.

g. At approximately 1140, the Unit 2 SSV questioned compliance with Technical Specification 3.9.10.2, Item "b," which requires "The SRM System to be OPERABLE per Technical Specification 3.9.2."

h. At 1145, the SSV contacted NMD to suspend removal of CRBs/CRDs from the core and/or RPV.

i. NMD was given direction to finish CRD retorques (one drive); CRB 46-47 was allowed to be installed as work on cell restoration was in progress.

j. During the period that the SRM System was inoperable:

1) No CORE ALTERATIONS occurred

2) No CRDs were retorqued

3) Four CRDs, previously detorqued, were replaced

4) Two CRBs associated with offloaded cells were replaced

k. Upon recognition of a potential noncompliance with Technical Specification 3.9.10.2, all CRB/CRD work was suspended. Reactor Engineers, NMD, the SOD, and the 2BY160 Coordinator were contacted to investigate the issue and take action to return the SRM recorders to service.

l. Significance Of The Event

1) CRBs were removed from the RPV and CRDs were removed from the RPV while the SRM System was not capable of indicating in the MCR **and** while not capable of alarming in the MCR.

2) This is contrary to the requirements of Technical Specification 3.9.10.2.b.

3) Operation prohibited by Technical Specifications occurred.

The only indications available in the MCR were SRM Channel "A" and SRM Channel "C" count rate on PMS Format 135

All SRM channel annunciators were already in alarm; any subsequent alarm condition would have been masked

m. Corrective Actions

1) Revised E-*BY160 to note that electrical power will be lost to SRM recorders.

2) Surveillance Test procedures were revised to allow performance without de-energizing the RPS buses.

n. An evaluation was made of the possibility for making a Technical Specification change to eliminate the need for SRM System operability when removing multiple control rods in defueled cells. However, this Technical Specification change was not recommended.

1) Not desirable from a reactivity management perspective. The SRM System is the only mechanism available to monitor neutron flux in a subcritical BWR core.

2) Having the SRM System unavailable during CRB maintenance would remove a significant barrier to a reactivity event (e.g., accidentally withdrawing a CRB in a fueled cell.)

3) General Electric (GE) guidelines for reactivity control during refueling specifically recommend maintaining SRM System operability during fuel shuffles and CRB maintenance campaigns.

VI. IRM SYSTEM

Objective IL12

A. Purpose

1. The IRM System detects and indicates neutron flux level in the range between the capabilities of the:

   a. Source Range

   b. Power Range

IRM Sensitivity:

- 1×10^8 n/cm^2/sec to 1.5×10^3 n/cm^2/sec

Equivalent To:

- 10^{-4} to 40% power

Objective IL13

2. The IRM System provides the Operator with neutron flux level information from criticality, through the point of adding heat (POAH), and into the Power Range (from ~10^{-4} to 40% Reactor Power).

3. The IRM System generates control rod withdrawal blocks via the Reactor Manual Control System (RMCS) if a prescribed neutron flux level is exceeded or a system malfunction occurs.

4. The IRM System is also capable of producing a reactor scram via the Reactor Protection System (RPS).

B. The IRM System generates trip signals to prevent fuel damage (while operating in the Intermediate Range) from:

1. Single Operator error

2. Single equipment malfunction

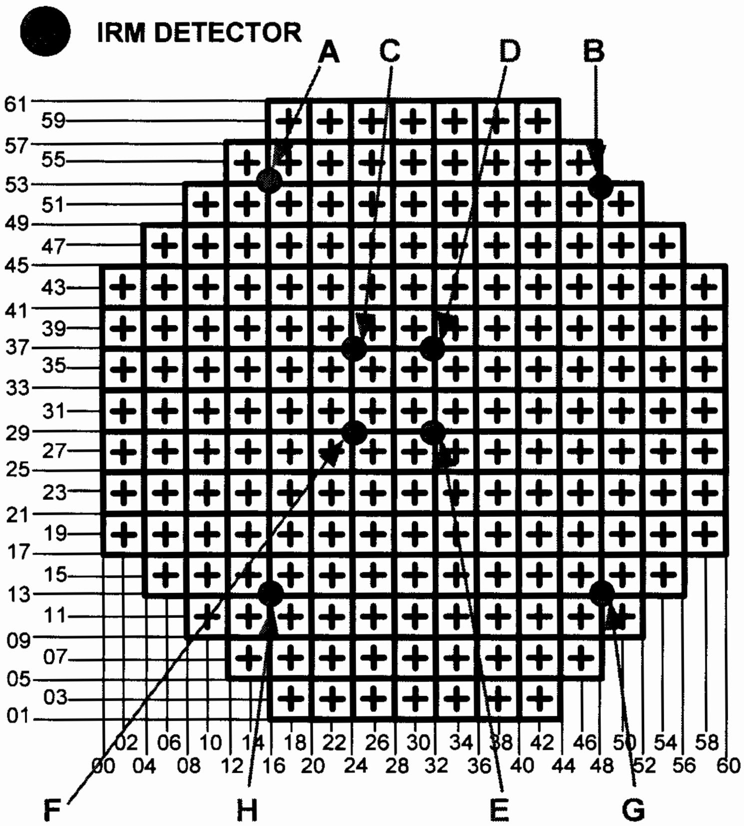
C. Eight Identical Neutron Detectors

1. Two detectors in each core quadrant

   Objective IL15

   a. One towards the center

   b. One towards the outer edge



2. Axial position of each detector is variable:

   a. From 15 inches above core mid-plane, to

   b. 30 inches below the bottom of the active core (based on 150 inch fuel length)

   Positions enforced by mechanical and electrical travel stops

   Same as SRM detectors

D. A drive assembly moves each detector into and out of the core

1. Driven by a rack and pinion arrangement

2. Connection made via a flexible drive cable

Content/Skills	Activities/Notes

E. Electronic components process and amplify detector developed signals

1. Consist of: — **Objective EO3**

   a. Voltage Preamplifier

   b. Amplifier and Attenuator

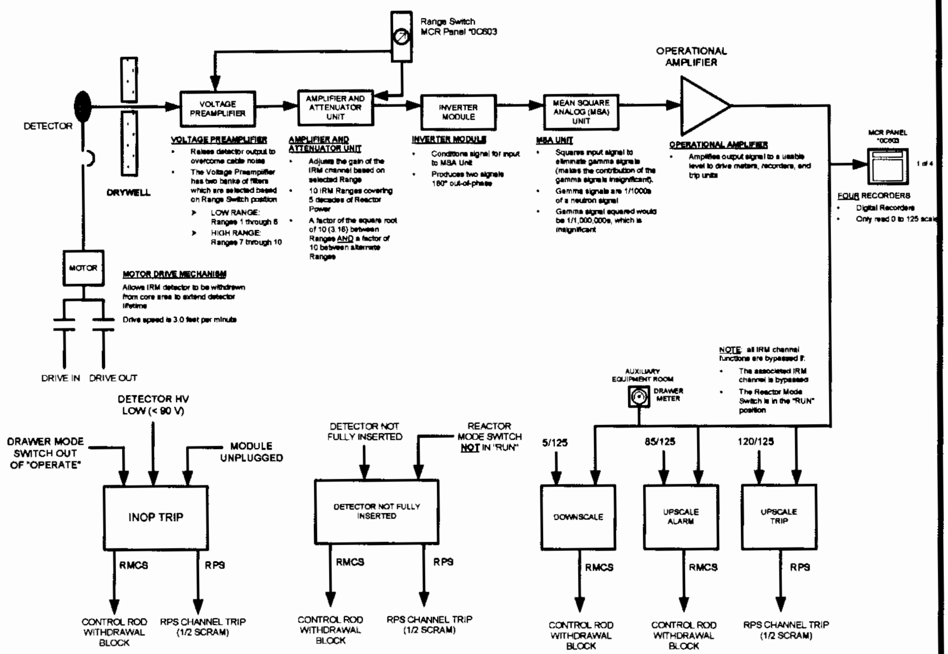
   c. Inverter

   d. Mean Square Analog (MSA) Unit

   e. Output Amplifier

   f. Power Supplies

ATTACHMENT 3



F. IRM Channels

1. Each detector and associated circuitry comprise one channel, designated as Channels "A" through "H"

2. IRM Channels "A", "C", "E", and "G" provide inputs to RPS Trip System "A" — **Objective IL16**

3. IRM Channels "B", "D", "F", and "H" provide inputs to RPS Trip System "B"

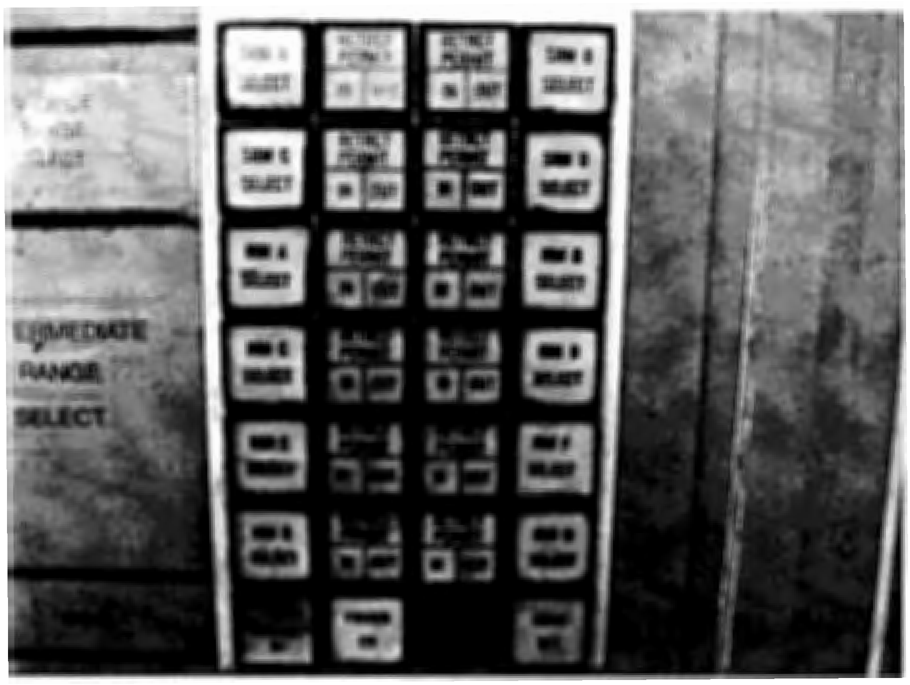
Content/Skills	Activities/Notes
G. Detector Assembly	
1. Consist of:	**Objective IL23**
a. Detector	
b. Shuttle Tube	
c. Dry Tube	
d. Detector drive	
2. Detector	
a. Eight fission chamber detectors	<u>Characteristics Differ From SRM Detectors</u>:
1) Operated in Ionization Region of the Gas Amplification Curve	1. Lower U_3O_8 loading 2. Lower Argon fill pressure 3. Lower operational voltage
2) Outer chamber lined with:	<u>Combination Of Above Results In</u>:
a) 2.7 mg U-235	1. Detector being less sensitive
b) Enriched to > 90% U-235	2. More suitable for high neutron flux levels
3) Pressurized with 18 psi Argon gas	
4) Operated with a 100 VDC potential	
b. Neutron pulse	
1) Thermal neutron causes fission within U-235 lining	
2) Results in fission fragments of:	
a) High kinetic energy	20 to 22 MeV
b) Large charge	
3) Produces large degree of ionization	
c. Gamma pulse	
1) Gammas also cause ionization	
2) Pulses created by non-fission gammas are removed as a side effect of the "Campbelling" process	
3. Shuttle Tube	
a. Movable portion of the detector assembly	
b. Detector fixed within the Shuttle Tube	

Content/Skills	Activities/Notes

4. Dry Tube
   a. Serves as pressure boundary of assembly
   b. Connected and sealed to detector drive
   c. Shuttle Tube moves inside Dry Tube
5. Detector Drive
   a. Each detector is positioned by individual drive units

      Powered from 208 VAC Lighting Panel 1L34 (2L107)
   b. Moves detector through rack and pinion arrangement
   c. Motor moves detectors in and out at a rate of 3.0 feet per minute

      <u>Remote motor reduces</u>:
      1. Undercore crowding
      2. Radiation aging of components
   d. Motors are located in the Drywell
   e. Connection between motor and drive assembly via flexible cable
   f. MCR Panel *0C603 Detector Drive Controls and Indicating Lamps



H. Voltage Preamplifier

1. The method of observing individual neutron pulses, used by the SRM System, becomes impractical in the high neutron flux regions where the IRM detectors operate.

Objective IL17.a

   a. Pulses are still produced at the detector output, but are not observable as discrete events.

   b. High neutron flux causes a "pile-up" of pulses at the detector output (i.e., "later" ionizing events occur within the detector before previous pulses have decayed).

   c. The rapid succession of pulses results in a changing AC signal superimposed on the DC voltage level at the detector output. This changing voltage is a result of the effect of both gammas and neutrons.

2. The Voltage Preamplifier amplifiers the output of the IRM detector

   a. This overcomes cable noise, **<u>AND</u>**

Located just outside Drywell

   b. Blocks the DC component of the detector signal from the IRM Instrument Drawer by converting the superimposed current pulses from the detector into voltage pulses of varying frequency

   c. The Voltage Preamplifier also passes one of two frequencies depending upon the position of the IRM Range Switch

      1) Lower Band (8 to 16 KHz)

Range 1 through 6:	8 to 16 KHz
Range 7 through 10:	300 to 600 KHz

         a) IRM Ranges 1 through 6

         b) Provides higher sensitivity

         c) Used for the lower ranges

      2) Higher Band (300 to 600 KHz)

         a) IRM Ranges 7 through 10

         b) Provides greater immunity to cable noise

         c) Used for the higher ranges where sensitivity is less important

I. Amplifier and Attenuator Unit

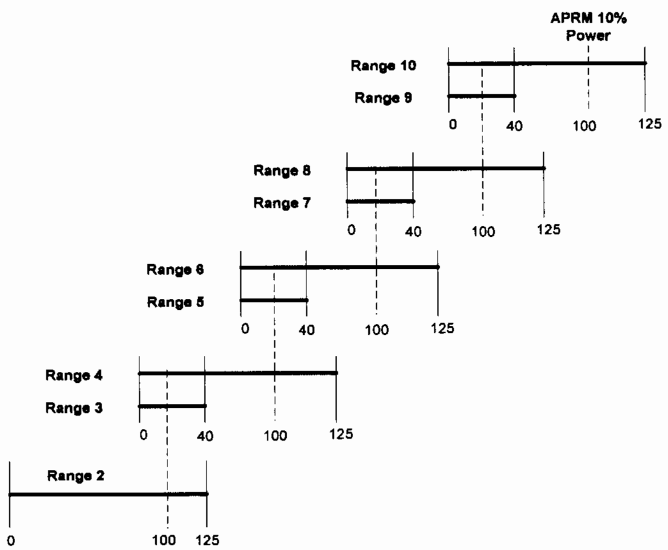
1. The Amplifier and Attenuator Unit receives the AC signal from the Voltage Preamplifier and, along with the IRM Range Switch, establishes the range of indication for the IRM channel

   Objective IL17.b

   a. 12-position Range Switch

   b. Although each Range Switch is provided with positions for IRM Ranges "11" and "12", these two positions can not be physically selected

IRM RANGE AND REACTOR POWER RELATIONSHIP



2. The Amplifier and Attenuator Unit provides six discrete gain ranges by switching different amounts of attenuation into the signal path

   Webster: **Attenuate** = to lessen the amount, magnitude or value of something

   a. Range Switch positions "1" through "6" insert a low frequency amplifier stage in service

   b. Range Switch positions "7" through "10" route the signal through a high frequency amplifier

   Combination of six attenuation ratios and two preamplification stages allows for 12 ranges of signal measurement

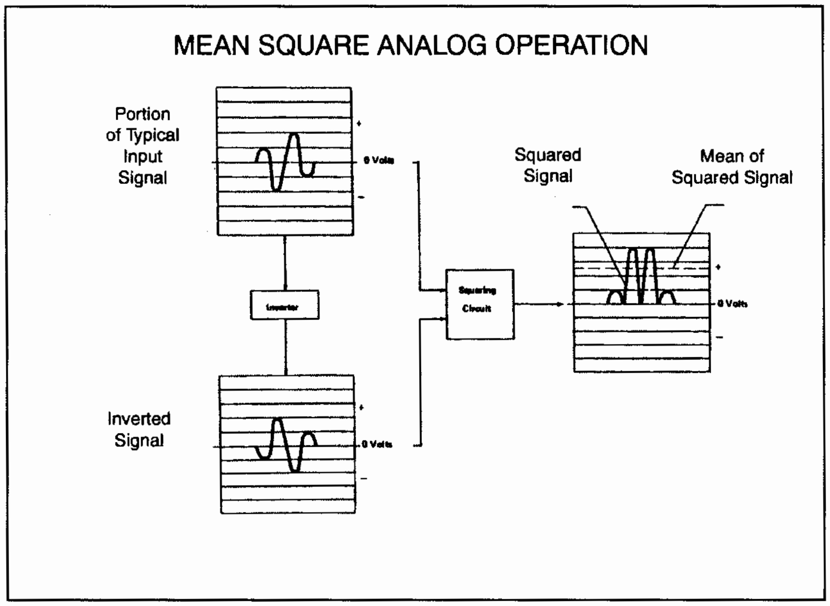
3. Every other (alternate) position on the Range Switch is a factor of one decade (x 10) of power different. As a result, the IRM System provides five decades of Reactor Power coverage.

   IRM Range Switch positions "11" and "12" can **not** be physically selected

4. Gain of 3.16 (the square root of 10) between each range

5. Ratio of 10:1 between alternate ranges

Content/Skills	Activities/Notes
J. Inverter Module	
1. The Inverter Module acts as a signal conditioning unit to prepare the IRM detector signal for input into the Mean Square Analog (MSA) Unit	**Objective IL17.c**
2. The Inverter Module receives a single input from the Amplifier and Attenuator Unit	
3. Produces two outputs, each being:	This is necessary for the MSA unit
a. Equal in magnitude	
b. 180° out-of-phase	
4. This allows the MSA Unit to respond to AC variations above and below the DC average	
K. Mean Square Analog (MSA) Unit	
1. The MSA Unit can be considered the "heart" of the IRM System.	**Objective IL14** **Objective IL17.d**
2. Here the composite detector signal is processed to remove the gamma signal	
a. Utilizes a discrimination method known as "Campbelling".	FYI: for Norman Campbell who, in 1910, proposed this statistical theory
b. Campbell's Theorem	
1) The average value of a statistical event is proportional to the size of the individual event	
2) The average value of an event is proportional to the average frequency of the event	
3) The variation from the average is proportional to the square root of the average frequency	
4) Therefore, if the size of the deviation from the average is squared, the result is fairly representative of the level of the events	



c. The MSA Unit receives the two outputs from the Inverter Module.

d. The MSA Unit then squares the amplitude of each output.

By squaring the difference in currents, the value of the gamma current is made insignificant. In effect, gamma discrimination has been performed

e. An Integrator circuit within the MSA Unit then sums both signals.

f. The output of the MSA Unit is a DC voltage proportional to the average (mean of the square) of the inputs.

g. The ratio that existed between neutron and gamma currents is also squared

   1) Gamma levels in the Intermediate Range are about 1/1000 that of neutrons

h. This process is called "Campbelling".

i. The objective of a discrimination process is to either remove the gamma contribution or significantly reduce its effect.

L. Output (Operational) Amplifier

1. The Output (Operational) Amplifier receives the DC output signal from the MSA Unit.

Objective IL17.e

2. The Output (Operational) Amplifier then amplifies the output to drive:

   a. IRM Instrument front panel meters in the AER

   b. MCR recorders

   c. Trip units

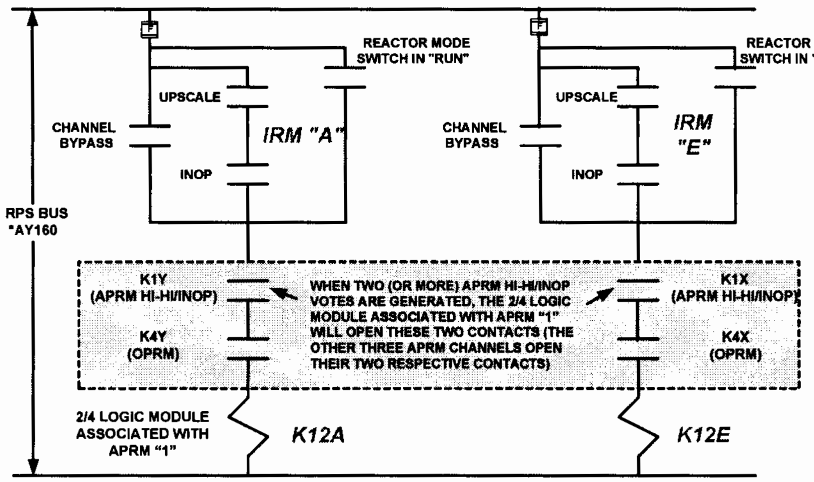
M. Protective Functions

Objective EO4.a
Objective EO5
Objective IL16
Objective IL19

ATTACHMENT 4
<u>AND</u> ATTACHMENT 6

1. <u>RPS Trips (scram signals)</u>

APRM "1" 2/4 LOGIC MODULE INTERFACE WITH SCRAM CONTACTS IN RPS TRIP CHANNEL "A1"



   a. IRM Channels "A", "C", "E", and "G" provide inputs to RPS Trip System "A"

   b. IRM Channels "B", "D", "F", and "H" provide inputs to RPS Trip System "B"

   c. A RPS trip (scram) signal is generated upon:

      1) IRM Upscale (≥ 120/125 of scale)

      2) IRM INOP

<u>IRM Channels "A" and "E":</u>
RPS Trip Channel "A1"

<u>IRM Channels "C" and "G":</u>
RPS Trip Channel "A2"

<u>IRM Channels "B" and "F":</u>
RPS Trip Channel "B1"

<u>IRM Channels "D" and "H":</u>
RPS Trip Channel "B2"

d. In order for a full RPS trip to occur (reactor scram):

1) One IRM channel in RPS Trip System "A" must be tripped ("A", "C", "E", or "G"), **AND**

2) One IRM channel in RPS Trip System "B" must be tripped ("B", "D", "F", or "H")

e. IRM System Trip Bypass Conditions

1) Two Manual Joysticks located on MCR Panel *0C603 allow manually bypassing of one IRM channel per RPS Trip System at a time.

2) All IRM channel Control Rod Withdrawal Block and Scram signals are bypassed when the Reactor Mode Switch is in "RUN"

3) IRM channel Downscale Control Rod Withdrawal Block signals are bypassed when the associated IRM channel Range Switch is on Range 1. This allows initial control rod withdrawals when in the Source Range.

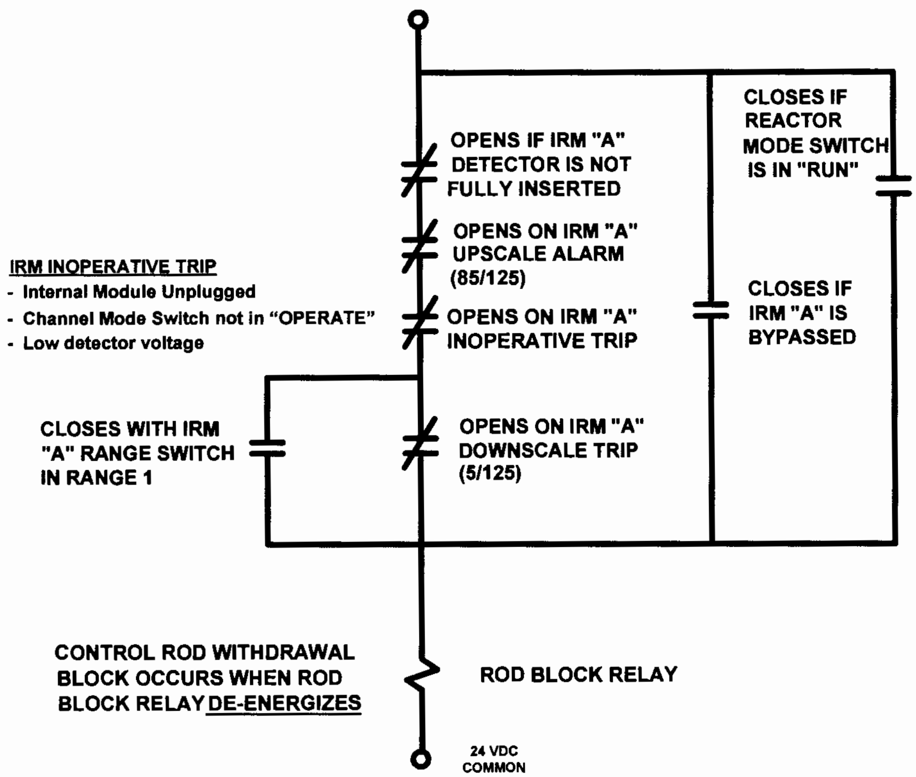
4) IRM channel alarm lamps and annunciators are also bypassed when the associated IRM channel is bypassed.

2. Control Rod Withdrawal Blocks

Objective EO4.b
Objective IL19

ATTACHMENT 5
AND ATTACHMENT 6

IRM CHANNEL "A"
CONTROL ROD BLOCK SIGNALS AND BYPASSES



3. Control Rod Withdrawal Blocks, through interface to RMCS. The IRM channels produce a control rod withdrawal block under the following conditions:

A control rod withdrawal block will occur if any of the indicated conditions is met for one (or more) IRM channel(s)

   a. IRM Upscale: channel output ≥ 85/125 of scale
   b. INOP
      1) Low detector voltage (< 90% of rated)

      < 90 V

      2) Internal module unplugged; other hardware failures
      3) Channel Mode Switch **not** in "OPERATE" (on Instrument Drawer in AER)
   c. IRM Downscale: channel output ≤ 5/125 of scale
   d. IRM Detector Not Fully Inserted

N. IRM System MCR Instrumentation And Alarms

1. MCR Panel *0C603

Objective IL19
Objective IL20

a. Percent Indication

1) Four Yokogawa DX1000N model paperless recorders

a) XRX-M1-*R603A

- Red Display: IRM Channel "A" or APRM Channel "1"
- Green Display: IRM Channel "C"

b) XRX-M1-*R603B

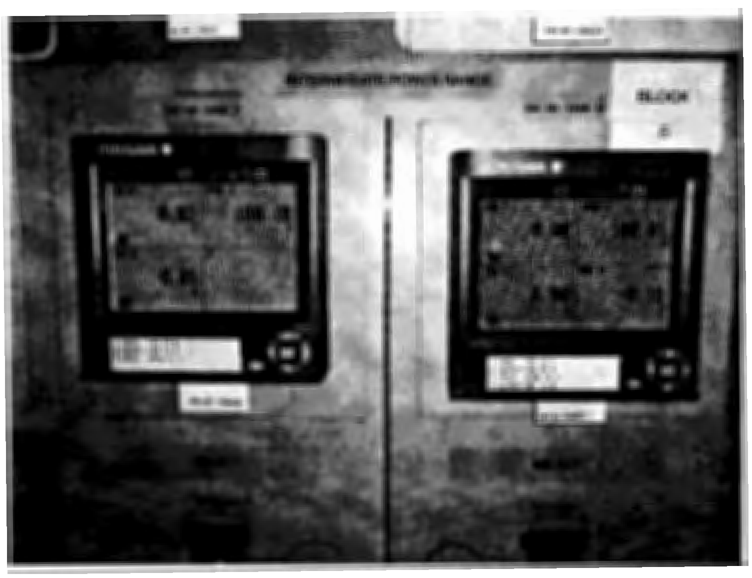
- Red Display: IRM Channel "B" or APRM Channel "2"
- Green Display: IRM Channel "D"

c) XRX-M1-*R603C

- Red Display: IRM Channel "E" or APRM Channel "3"
- Green Display: IRM Channel "G" or RBM Channel "A"

d) XRX-M1-*R603D

- Red Display: IRM Channel "F" or APRM Channel "4"
- Green Display: IRM Channel "H" or RBM Channel "B"



2) Each IRM/APRM recorder has two scales:

NOTE: the digital readout for each recorder selected to indicate an IRM channel reading provides indication of 0 to 125 (percent of scale)

a) Red bargraph has a scale of 0 to 125 (percent of scale)

b) Green bargraph has a scale of 0 to 40 (percent of scale); however, use the 0 to 125 (percent of scale) scale when reading IRM channels in the MCR

c) When selected to "APRM", the recorders read out in 0 to 125% of Reactor Power

d) When selected to "IRM", the recorders read out in 0 to 125 (percent of scale) units. These units have no absolute meaning. An IRM channel reading is simply referred to as "percent of scale" (e.g., a reading of 54/125 means an indication of 54 units on a scale of 0 to 125)

b. IRM Channel Bypass Joysticks (MCR Panel *0C603)

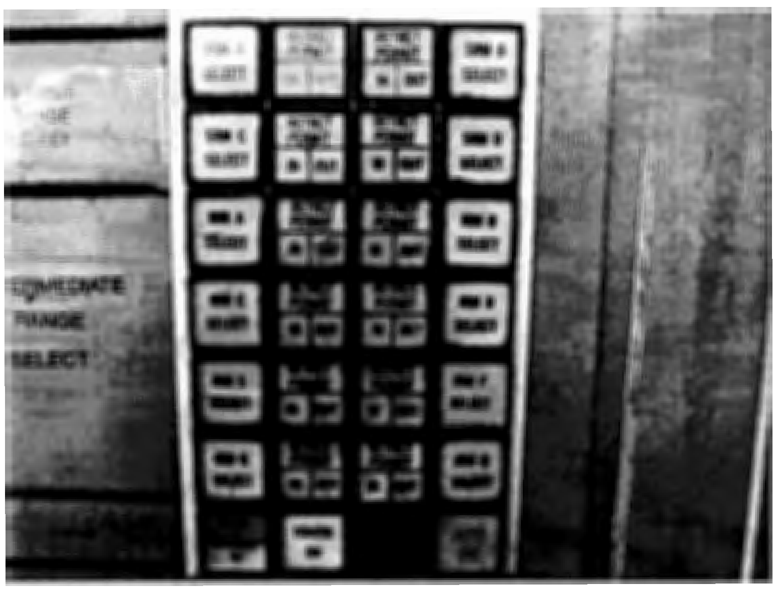
Operations Fundamentals

PRECISELY CONTROLLING PLANT EVOLUTIONS

- **Human Error Prevention** *(Use self check for component identification and equipment manipulation. Self-Check Technique is STAR: Stop, Think, Act, Review))*

1) Two separate joysticks with 5-positions
    a) Center "OFF", no channel bypass
    b) Left joystick, select one channel for bypass from IRM Channels "A", "C", "E", or "G" (associated with RPS Trip System "A")
    c) Right joystick, select one channel for bypass from IRM Channels "B", "D", "F", or "H", (associated with RPS Trip System "B")
2) Only one IRM channel per RPS Trip System may be bypassed at a time
3) Channel bypass condition defeats all control functions (including alarm lamps and annunciation), but does **not** affect channel power indication on meters and recorders.

c. Detector Drive Controls and Indicating Lamps



1) Eight detector Selector Switches, used for selecting the IRM detector to be moved
    a) "ON/OFF," maintained-contact pushbutton for each detector
    b) Backlight illuminates when associated detector is selected

2) One Drive-In Pushbutton, with two indicating lamps

   a) "ON/OFF," maintained-contact pushbutton

   b) Once depressed, selected detectors drive in to the core until fully inserted

   c) If depressed again while the selected detectors are not yet fully inserted, detector motion will stop

   d) Also used for inserting SRM detectors, if selected

   e) One "DRIVE IN" indicating lamp on pushbutton, illuminated when selected detectors have been selected for movement into the core.

   The "DRIVE IN" indicating lamp remains illuminated after the selected detectors have reached the fully inserted position

   f) One "DRIVING IN" indicating lamp, on pushbutton. When illuminated, indicates selected detectors are in the process of being driven into the core.

   The "DRIVING IN" indicating lamp extinguishes when the selected detectors have reached the fully inserted position

3) One Drive Out Pushbutton and indicating lamp

   a) "ON/OFF," momentary-contact pushbutton

   b) <u>Push and HOLD</u>: selected detectors drive out of the core as long as the pushbutton is depressed

   c) <u>Released</u>: selected detectors stop moving

   d) Also used for withdrawing SRM detectors, if selected

   e) "DRIVE OUT" indicating lamp on pushbutton, illuminated while selected detectors are moving out of the core (pushbutton depressed)

4) IRM Position Indicating Lamps, one set of lamps for each IRM detector

   a) <u>"IN"</u>: detector is full-in

   b) <u>"OUT"</u>: detector is full-out

c) "RETRCT PERMIT": illuminated when detector is not fully withdrawn **AND** any one of the following exists.

(1) IRM channel is bypassed

(2) Reactor Mode Switch in "RUN"

d. IRM Channel Alarm Lamps

1) Four indications for each channel

Panel *0C603

Same setpoints as Alarms

a) "UPSC TR OR INOP" (red): setpoint ≥ 120/125 or channel INOP

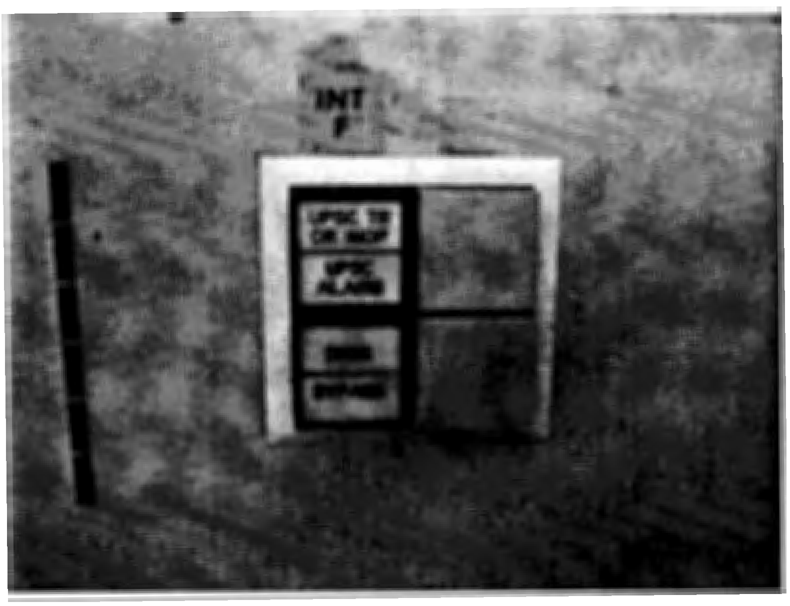
b) "UPSC ALARM" (amber): setpoint ≥ 85/125

c) "DNSC" (white): setpoint ≤ 5/125

d) "BYPASS" (white): Manual Channel Bypass

2) Used in conjunction with annunciator to determine which detector is causing the alarm

3) These alarm lamps are disabled during channel bypass conditions.



e. Annunciators on Panel *07 REACTOR

NOTE: these annunciators have the same setpoints and bypasses as discussed previously for the IRM channel alarm lamps:

1) "IRM DOWNSCALE" (≤ 5/125)

Window G-3

2) "IRM UPSCALE" (≥ 85/125)

Window F-3

3) "IRM UPSCALE/INOPERATIVE" (≥ 120/125 or channel INOP)

Window H-3

f. No annunciation on IRM Detector Retracted When Not Permitted (as exists in the case of the SRM detectors)

g. IRM channel annunciators are disabled during channel bypass conditions.

O. IRM Channel AER Controls and Indications

1. Meters: IRM channel indications will be essentially the same as in the MCR, and will change as the MCR IRM Range Switches are operated.

Objective EO4.c

2. Indicating Lamps: all lights on the panel seal-in, and must be reset at the panel to clear them.

a. "DOWNSCALE" lamp, illuminates at ≤ 5/125 of scale

b. "UPSCALE 1" lamp, illuminates at ≥ 85/125 of scale

c. "UPSCALE 2" lamp, illuminates at ≥ 120/125 of scale

d. "INOP" lamp, illuminates under any of the following conditions:

1) Low detector voltage (< 90% of rated)

OR

2) Internal module unplugged; other hardware failures

OR

3) Channel Mode Switch **not** in "OPERATE" (on Instrument Drawer in AER)

4) "BYPASS" lamp (white): located above IRM channel drawer; illuminates if IRM Channel Bypass Joystick is positioned to bypass the associated IRM channel

3. Switches

a. Channel Mode Switch, normally in "OPERATE" position; other positions used for various testing purposes.

b. Reset Switch

1) The reset switch is used to reset the lights on the panel.

P. Power Supplies

1. Input power to + 20 VDC power supplies and instrument recorders supplied by 120 VAC RPS/UPS

   Objective IL18

2. RPS Bus *AY160 (Circuit #2) supplies power to:

   <u>NOTE</u>: also supplies power to NS[4]

   a. IRM Channel "A", "C", "E", and "G" circuits and amplifiers

   b. IRM Channel "A", "C", "E", and "G" MCR alarm lamps, including detector drive “RETRCT PERMIT”

   c. Loss of Bus *AY160 results in an RPS Trip System "A" half-scram

3. RPS Bus *BY160 (Circuit #2) supplies power to:

   a. IRM Channel "B", "D", "F", and "H" circuits and amplifiers

   <u>NOTE</u>: also supplies power to NS[4]

   b. IRM Channel "B", "D", "F", and "H" MCR alarm lamps, including detector drive “RETRCT PERMIT”

   c. IRM Channel "A" through "H" MCR recorders (Circuit #5)

   <u>NOTE</u>: four separate recorders, shared with APRM and RBM channels

   d. Loss of Bus *BY160 results in an RPS Trip System "B" half-scram

4. Voltage Regulator

   a. Converts + 20 VDC input power to:

      1) Well regulated, + 15 VDC output

      2) Each instrument has its own Voltage Regulator

   b. This output is used by:

      1) IRM channel circuitry (including Voltage Preamplifier)

      2) High Voltage Power Supply

5. High Voltage Power Supply

   a. Receives + 15 VDC from Voltage Regulator

   b. Converts this to a high voltage output (normally 100 VDC)

   c. Output is supplied to the IRM detector and provides the necessary detector polarizing potential

d. If output of power supply drops below a setpoint, an IRM channel INOP Trip will occur	This results in a control rod withdrawal block and a half-scram signal on the affected RPS trip channel
6. Miscellaneous Power Supplies	
a. 120 VAC *0Y202 (Circuit #13) supplies power to:	
1) All IRM detector drive relays	Loss of this bus prevents detectors from being able to be moved
2) All IRM detector drive indicating lamps, <u>except</u> "RETRCT PERMIT"	
b. Non-safeguard Load Center *24B (Lighting Panel 1L34, 2L107) powers IRM Channel "A" through "H" detector drive motors.	Loss of this bus also prevents detectors from being able to be moved
Q. IRM System Operation	
1. <u>Reactor Startup</u>	
<u>NOTE</u>: Reactor Startups are performed in accordance with GP-2, Normal Plant Startup	
a. Initial Conditions	
1) SRM detector drives are fully inserted and all operational channels are indicating > 3 CPS (**<u>NOTE</u>**: an indication of > 3 CPS per channel ensures all channels are functioning properly).	
2) The"> 3 CPS" value may be reduced, provided the associated SRM channel has an observed count rate and the signal to noise ratio is <u>on or above</u> Technical Specification Figure 3.3.6-1, SRM Count rate Versus Signal To Noise Ratio.	Instructor should ensure that students can locate this curve in Technical Specifications
3) SRM and IRM channel "INOP" lamps are extinguished ("OFF").	
4) IRM channel Range Switches are on Range 1.	
5) APRM channel "DNSC" lamps are illuminated ("ON").	
b. Record initial SRM channel count rates in GP-2 Appendix 1, Reactor Startup And Heatup, for determination of count rate after three doublings.	

c. Withdraw control rods while monitoring count rate and Reactor Period.

NOTE: control rods are withdrawn in accordance with GP-2 Appendix 1, Reactor Startup And Heatup

1) When any SRM channel exceeds three doublings, then single notch control rod withdrawal between position 04 to 36

2) Reactor critical (positive Reactor Period **AND** sustained rise on SRM channel instrumentation with no control rod motion indicates the reactor is critical)

Safety Emphasis

Criticality should be expected at any time. Close attention must be paid to all SRM channel indications when pulling control rods.

3) Record Critical Count Rate Data

4) Do not permit Reactor Period to be < 50 seconds during control rod withdrawal, as indicated on SRM channel Reactor Period meters

Reference: GP-2 Appendix 1, Reactor Startup And Heatup

d. Withdraw additional control rods, as necessary, to obtain a stable positive Reactor Period (150 seconds to 50 seconds).

Safety Emphasis

Maintain plant parameters within operational limits

e. Continue monitoring SRM and IRM channel response. When IRM channel response is observed, record unbypassed SRM channel count rate as required per ST-6-107-884-*, Neutron Monitoring System Overlap Verification On Startup.

f. Verify proper overlap by completing ST-6-107-884-*, Neutron Monitoring System Overlap Verification On Startup

1) At least one-half decade during each startup after entering OPERATIONAL CONDITION (OPCON) 2

2) Technical Specification Table 4.3.1.1-1, Footnote (b)

g. When SRM/IRM System overlap has been verified, then withdraw SRM detectors to maintain count rate 100 to 10,000 CPS

   1) If SRM detectors are withdrawn too soon, a control rod withdrawal block may occur ($\leq$ 100 CPS **and** detector not fully inserted)

   2) If SRM detectors are withdrawn too late, a control rod withdrawal block may occur ($\geq 1 \times 10^5$ CPS)

h. Fully withdraw SRM detectors when the IRM channels are indicating on Range 3 or above. This protects the fission chamber from the high neutron flux occurring in the core during power operations, thus extending detector lifetime.

i. Continue control rod withdrawal to achieve and maintain target heatup rate (POAH)

   1) Human Performance Fundamentals: Control rod withdrawals at or near criticality are EXCELLENT opportunities to utilize Human Performance techniques.

   2) Many BWRs have experienced short Reactor Periods during reactor startups:

      a) Short Reactor Periods require fast reactions to operate IRM Range Switches

      b) These quick actions, made without utilizing STAR, have frequently led to reactor scrams

   3) Consider the effects of control rod manipulations, and continuously monitor Reactor Period for any radical changes in the rate of change of Reactor Power.

   4) Appraise plant conditions and consider their effects on reactivity.

   5) If in doubt about any indication or response, COMMUNICATE to Supervision.

j. Monitor neutron flux on SRM and IRM channel indications as Reactor Power rises

Safety Emphasis

Changing plant conditions can affect reactor coolant heatup rate and thereby affect reactivity

- Decay Heat load
- Steam users

k. Operate IRM Range Switches to higher ranges to maintain indication greater than the "IRM DOWNSCALE" alarm but < 75/125 of full scale (Review Note: when all IRM channels are above Range 2, the control rod withdrawal block associated with the SRM detector not being fully inserted with count rate ≤ 100 CPS is bypassed)

1) Human Performance Fundamentals: Operating the IRM Range Switches is an EXCELLENT opportunity to practice Self-Check (STAR)

a) **STOP**: pause and allow enough time to think about what you are going to do

b) **THINK**: in which direction do I want to operate the Range Switch? What will be the results?

c) **ACT**: operate the IRM Range Switch

d) **REVIEW**: did the actual results match the expected results?

2) Operating the Range Switch in an incorrect manner can result in a control rod withdrawal block or an RPS trip

l. Fully withdraw all SRM detectors when all IRM channels are on Range 3 or above.

m. Notch out control rods to maintain an approximate 90 °F per hour heatup rate

1) Do not exceed 100 °F per hour heatup

2) Record temperatures every 15 minutes per ST-6-107-640-*, Rx Vessel Temperature And Pressure Monitoring

n. Continue reactor heatup to rated temperature and pressure per GP-2, Normal Plant Startup

1) When all APRM channel "DOWNSCALE" alarms have cleared **AND** RPV pressure is > 800 psig, then place Reactor Mode Switch in "RUN."

Review Question

If an IRM Range Switch is accidentally ranged DOWN instead of UP, what is the minimum channel reading that will result in an RPS trip if the Range Switch is ranged DOWN one range:

IRM Scram Setpoint: ≥ 120/125

120 = "X" times 3.16

"X" = 120/3.16 = 38

2) When RPV pressure is 960 psig with two Main Turbine Bypass Valves open, then:

   a) Fully withdraw all IRM detectors

   b) Secure SRM/IRM detector drive power

2. **OPERATING EXPERIENCE**: Recurring Event, Inappropriate Continuous Control Rod Withdrawal From Subcritical Conditions

   Reference: SEN 185; OE9121

   a. Description Of The Event

      1) Early in the morning on July 2, 1998, at Susquehanna Unit 2, Operators inappropriately performed a continuous withdrawal of a control rod while conducting a reactor startup.

      2) The Operators had previously taken the reactor critical using the single notch method of control rod withdrawal, but did not recognize that the reactor had become subcritical.

      3) The reactor had become subcritical as a result of reactor coolant heatup from decay heat. This occurred while Main Steam System lineups were being conducted and while criticality data was being recorded.

         Error Precursor: tunnel vision

      4) Because there were no written requirements for using single notch withdrawal, and because they did not recognize that the reactor had gone subcritical, the Operators decided to continuously withdraw control rods until a 100 second Reactor Period was achieved.

      5) The control rod that had been used to achieve criticality was continuously withdrawn from notch 22 to notch 48, with small gains observed on SRM and IRM instrumentation.

      6) A second control rod was then withdrawn from the "FULL IN" position. The Operator stopped withdrawing the control rod at position 24, after observing a response on the SRM channels. (**NOTE**: SRM detectors had previously been withdrawn.)

7) This control rod had approximately five times the reactivity worth of the previous control rod, and a review of the data immediately following the event, showed that the control rod withdrawal caused an average Reactor Period of about 30 seconds.

8) Other Operators monitoring core conditions were observing a computer screen which had a several second delay in displaying core parameters.

9) Reactor Power went up quickly, requiring upranging of the IRM channels. While upranging, two Reactor Operators each operated one IRM Range Switch in separate divisions in the wrong direction, resulting in a reactor scram.

10) The SRO supervising the startup directed one operator to up-range IRM Channel "E." However, the operator mistakenly changed the IRM Range Switch from Range 5 to Range 7. He then responded to the error by downranging the channel to Range 6. The same Operator then intended to uprange IRM Channel "C," but instead, ranged it down. This resulted in a half-scram.

<u>HU Review</u>

Q. Why wasn't there enough time to practice STAR?

A. A 30-second Reactor Period doesn't allow that

11) Three seconds later another Operator made a similar error and downranged IRM Channel "D." This resulted in another half-scram on RPS Trip System "B," and a full scram resulted. These manipulations occurred within 15 seconds of each other.

b. Causes Of The Event

1) Station personnel did not exercise the same careful approach to reactivity control during low power operation as they did during approach to criticality.

2) Operators did not anticipate the effects of small temperature changes on reactivity that occurred during the period of secondary system alignments.

3) Operators did not recognize that the reactor had become subcritical.

Lack of questioning attitude

4) Operators did not realize the rapid power rise that would occur from continuously withdrawing a high worth control rod from a low Reactor Power level.

5) According to the Operators, continuous control rod withdrawals at that point were not unusual, and they wanted to avoid prolonging the startup by using single notch control rod withdrawal.

Error Precursor: can-do attitude

6) The personnel assigned to the start-up did not function effectively as a team.

7) Station personnel did not fully benefit from industry Operating Experience describing difficulties encountered during low Reactor Power level operations.

c. Corrective Actions

1) Control rod pull sheets were changed to impose notch withdrawal for all control rod withdrawals through Group 2.

2) Reinforcement of expectations on reactivity manipulations.

3) At Susquehanna, a single individual will now perform all manipulations. This includes any switch manipulations on the control rod control panel.

d. At Exelon/Limerick

1) Control Rod Notching at Limerick

a) RWM requirements (single notch control rod withdrawal movements are not normally required once a black and white control rod pattern has been established.)

b) Reactor approach to criticality after three doublings on SRM channel indications. Perform control rod withdrawal via single notch motion between notches 04 to 36, in addition to RWM requirements.

2) Reactivity Management at Limerick/Exelon

a) OP-AA-300, Reactivity Management

(1) Describes and defines operational reactivity management program within Exelon Nuclear

(2) Defines responsibilities of station personnel

(3) Acknowledge Operations owns reactivity management

(4) Respond conservatively in the face of prolonged uncertainty or unusual behavior to include control rod insertion or reactor scram

b) OP-AB-300-1001, BWR Control Rod Movement Requirements

(1) Defines responsibilities and provides guidance for BWR control rod maneuvers during non-transient conditions

(2) "Control rod movements shall be performed in a deliberate, carefully controlled manner while constantly monitoring nuclear instrumentation and redundant indications of reactor power level and neutron flux."

(3) "All control rod movements shall be performed in accordance with an approved procedure."

3. <u>Power Operation</u>: IRM detectors are fully withdrawn from core

4. <u>Reactor Shutdown (Scram)</u>

a. Per T-100, Scram/Scram Recovery, fully insert all SRM and IRM detectors

b. Per T-101, RPV Control, fully insert all SRM and IRM detectors

c. Ensures decaying neutron flux is tracked on instrumentation

Content/Skills	Activities/Notes
5. Reactor Shutdown (Soft Shutdown)	Reference: GP-3, Normal Plant Shutdown
a. ST-2-074-642-1, Source Range And Intermediate Range Neutron Monitor Pre-Shutdown Functional Test, is performed	This Surveillance Test (ST) procedure verifies the operability of the SRM and IRM channels, with the exception of required CHANNEL CHECKS and the functionality of detector "IN" and "OUT" lamps
b. When Reactor Power is ~ 21%, ST-6-107-882-*, IRM Operability, is performed to verify the functionality of the IRM detector "IN" and "OUT" lamps.	When both ST-6-107-882-*, IRM Operability, and ST-6-107-886-*, Neutron Monitoring System Overlap Verification On Shutdown, have been completed satisfactorily, the IRM channels are verified OPERABLE
c. Prior to Reactor Power dropping below 15%, ST-6-107-886-*, Neutron Monitoring System Overlap Verification On Shutdown, is performed to verify one-half decade overlap between APRM and IRM channel indications, and to perform the required IRM Channel CHANNEL CHECKS.	Reference Technical Specification Table 4.3.1.1-1, footnote (b)
6. **OPERATING EXPERIENCE**: Reactor Scram On IRM Upscale Due To Personnel Error	Ensure students have a copy of LGS LER 88-012-01
a. Conditions Prior To Event	
1) LGS Unit 1 was being shut down to facilitate work for a minor testing and maintenance outage.	
2) The shutdown was being accomplished by individual control rod insertions to facilitate a gradual depressurization to reduce offgas releases. This was a precaution being taken due to minor fuel cladding defects.	
3) A manually initiated half-scram was in place due to the IRM channels being out of surveillance.	
4) The plant was in OPCON 2, with Reactor Power < 1%.	

b. Description Of Event

1) On April 9, 1988 at 0415, a reactor scram occurred.

2) At 0400, the Reactor Operator had stopped inserting control rods in an effort to control the cooldown rate. At this time, Reactor Power was dropping steadily.

3) Approximately six minutes prior to the scram, the Reactor Operator had downranged the IRM channels from Range 3 to Range 2.

4) Approximately three minutes prior to the scram, Reactor Power began rising due to the positive reactivity effect of lowering moderator temperature.

Error Precursor: hidden system response

5) Reactor Power, initially indicated as 27/125 on Range 2, rose to 108/125 on Range 2, where an Upscale Alarm occurred.

NOTE: old Upscale alarm setpoint was 108/125, now set at 85/125.

6) The Reactor Operator, reacting to the alarm, upranged IRM Channel "A," but did not uprange IRM Channel "C" in time to prevent the scram.

7) The Reactor Operator was unable to uprange all of the IRM channels prior to Reactor Power rising to a value > 120/125. An RPS trip resulted six seconds after the Upscale Alarm.

c. Consequences Of Event

1) All systems actuated as designed.

2) Scram was reset at 0423.

3) There was no release of radioactive material to the environment as a result of this event.

d. Cause Of The Event

1) From the LER: "The cause of the event was a cognitive personnel error when the utility employed licensed reactor operator did not adequately anticipate and observe the effects of decreasing moderator temperature on core reactivity."

2) The Reactor Operator did not properly monitor IRM channel indications after downranging, several minutes prior to the scram.

3) Control rod insertion had progressed past the POAH and moderator temperature was dropping due to CRD cooling flow into the RPV, ambient RPV heat losses, and depressurization due to Main Steam loads.

4) When the positive reactivity, due to the falling moderator temperature, exceeded the negative reactivity from the control rod configuration, neutron flux began to rise, until terminated by the scram.

5) At the time of the event, the Reactor Operator was making necessary entries on the Reactor Operator Log Book, and did not properly anticipate the magnitude of the reduction in moderator temperature under the existing conditions.

e. Actions Taken To Prevent Recurrence

1) The Reactor Operator involved was cautioned to be constantly aware of conditions that affect core reactivity and act accordingly.

2) GP-3, Normal Plant Shutdown, was revised to caution Operators about the effects of lowering moderator temperature on core reactivity.

3) IRM Upscale Alarm setpoint was lowered.

Operations Fundamentals

PRECISELY CONTROLLING PLANT EVOLUTIONS

- **Briefs** (*Prepare and review material prior to conducting briefs*)

CONSERVATIVE BIAS TO PLANT CONDITIONS

- **Reactor Safety** (*Understand plant consequences of every action taken*)
- **Reactivity Management** (*Respond conservatively in the face of prolonged uncertainty or unusual reactor behavior to include rod insertion or a reactor scram*)

Safety Emphasis

Changing plant conditions can affect reactor coolant heatup rate and thereby affect reactivity

- Decay Heat load
- Steam users

Error Precursor: work environment - distractions/interruptions

f. <u>Other Factors In Events Like This</u>

1) With raised plant reliability and reduced scram frequencies, plant startups and shutdowns are very infrequent events.

2) Many Reactor Operators will only experience startups and shutdowns in the Simulator.

3) As a result, there are several Error Precursors around individual capabilities:

a) Unfamiliarity with task/first time

b) Lack of proficiency/inexperience

<u>Operations Fundamentals</u>

PRECISELY CONTROLLING PLANT EVOLUTIONS

- **Human Error Prevention** *(Use OOPS: if outside Procedures, Parameters, or Process, then STOP and contact your supervisor)*

7. Refueling Operations

a. Per Technical Specification Table 3.3.1-1, IRM Channels are required during OPCON 5

b. Footnote (i): with any control rod withdrawn

Review Technical Specification differences between SRM and IRM channels for Refueling operations

R. System Malfunctions/Losses

1. RPS/UPS Buses

a. Provides power to IRM channels

b. IRM channels are fail-safe so that a loss of electrical power will cause an RPS trip and control rod withdrawal block from the associated IRM channel

1) If electrical power to IRM channels located in both divisions is lost at the same time, a reactor scram will be generated depending upon the operating condition.

2) A loss of power to any single IRM channel will generate a control rod withdrawal block provided the Reactor Mode Switch is <u>not</u> in the "RUN" position.

Objective IL22.a
Objective IL22.b
Objective IL22.c

c. Loss of one of RPS buses also results in other events:

1) DWCW System isolation

2) RECW System isolation

3) RWCU System isolation

4) RE HVAC System isolation

5) RF HVAC System isolation

Reference:

- E-*AY160, Loss of *A RPS UPS Power
- E-*BY160, Loss of *B RPS UPS Power

d. A loss of the IRM System is of relatively low concern during this scenario

2. Loss of IRM System

Objective IL21.a
Objective IL21.b
Objective IL21.c

a. Loss of the IRM System will prevent monitoring Reactor Power when in the Intermediate Range. Actions should be taken in accordance with ON-109, Total Loss Of The SRM, IRM or APRM Systems.

b. Failure of the IRM System to function properly may also prevent/generate control rod withdrawal blocks/reactor scrams depending upon plant operating conditions.

VII. ADDITIONAL OPERATING EXPERIENCE

A. SRM Channel “B” Fuses Incorrectly Installed On IRM Channel “B,” Due To Use Of Incorrect Clearance

Reference: LGS PEP I0010989

1. Description Of The Event

While performing preventative maintenance on SRM Channel “B” using procedure M-074-012, SRM/IRM Drive Unit Maintenance (which requires installing detector drive motor module fuses), the fuses were incorrectly installed on IRM Channel “B” instead of SRM Channel “B”.

Instructor should emphasize, that, while the event occurred to Maintenance Technicians, the Error Precursors and Human Performance aspects apply to Operators as well

a. A Pre-Job Brief was held at the start of the shift in accordance with the Maintenance Pre-Job Brief Checklist.

b. When the Pre-Job Brief was completed, the team proceeded to Reactor Enclosure Elevation 253 feet. Some team members proceeded undervessel to perform work on SRM Channel “B.” Other team members remained outside the drywell on Elevation 253 feet.

c. As part of vibration testing on SRM Channel "B," the undervessel crew requested the Lead Maintenance Tech (LMT) to have the fuses for SRM Channel "B" installed under Clearance 00000025. The LMT contacted the MCR, requested and received permission to install the fuses for SRM Channel "B."

d. The LMT directed two technicians to install the fuses in accordance with M-074-012 and handed them a copy of a clearance. The technicians proceeded to SRM/IRM Fuse Panel 10C008. The technicians proceeded directly to the clearance steps that identified the fuses to be installed. The location and fuse number were double verified using the clearance in hand and the fuses were installed.

e. The technicians informed the LMT that the fuses were installed. The LMT contacted the MCR and requested that SRM detector "B" be withdrawn a short distance. The MCR performed this action, but the undervessel crew observed no movement of the drive tube.

f. The LMT informed the MCR that there had been no movement; the MCR requested a check of the fuses. The LMT informed the technicians to remove the fuses and check them for continuity. The fuses were found to be good.

g. At this time, the LMT and technicians re-grouped and reviewed the procedure and the clearances in hand and discovered that the technicians were on clearance 00000167, for IRM Channel "B." All work was halted and MCR and NMD management were notified.

<u>**Operations Fundamentals**</u>

PRECISELY CONTROLLING PLANT EVOLUTIONS

- **Human Error Prevention** *(Use OOPS: if Outside of Procedures, Parameters, or Process, then STOP and contact your supervisor)*

2. Importance Of The Event

a. When the rules for clearance and tagging are not followed, serious consequences can result.

Instructor should use questions to review safety consequences of <u>**not**</u> following clearance and tagging procedures

b. Although there are electronic and mechanical interlocks built into the clearance and tagging system, the fact remains that the system is based on trust in the fact that everyone is following the rules.

3. Causes of the Event and Corrective Actions

   a. When the LMT directed the two NMD technicians to install the fuses for SRM Channel "B," he handed them a clearance for IRM Channel "B" instead. The apparent cause is less than adequate Self Check on the part of the LMT prior to handing the clearance to the technicians.

   Instructor should emphasize: Self-Check, STAR

   Necessity for Adequate Pre-Job Brief, covering possible Error Precursors

   Corrective Action: The clearance holder checklist will be reviewed as part of the Pre-Job Brief.

   b. When the two NMD technicians went to install the fuses for SRM Channel "B," they did not adequately review the clearance nor did they review the clearance holder checklist for this Special Condition Tag (SCT) manipulation. The technicians opened the clearance to the points they were required to manipulate and proceeded from there without first reviewing the header, comments, special instructions, or the rest of the clearance points.

   Corrective Action:

   1) NMD Management has made it an expectation for technician's assigned SCT/Clearance Holder responsibilities to independently perform the Clearance Holder Checklist.

   2) Additionally, communications will be required between the control point and persons performing the manipulations using three-part communications.

   3) During a stand-down following this event, the importance of an adequate clearance review prior to performing actions as a clearance holder was stressed. If the technicians had reviewed the clearance thoroughly they would have identified that the clearance was for an IRM channel and not an SRM channel.

4. Possible Error Precursors

   a. Imprecise communication techniques

   Instructor should discuss how various Error Precursors led to this event

   b. Can-do attitude

   c. Assumptions

   d. Mental short-cuts

   e. Inaccurate risk perception

5. Other Corrective Actions

   a. Following the event a stand-down meeting was held:

      1) Compliance with all clearance and tagging rules was emphasized

      2) Review of responsibilities of clearance and tagging holders

6. Prevention

   a. Verification techniques, covered by HU-AA-101, Human Performance Tools And Verification Practices

   b. Clearance and Tagging, covered by OP-MA-109-101, Clearance And Tagging

   c. Briefs, covered by HU-AA-1211, Briefings - Pre-Job, Heightened Level Of Awareness, Infrequent Plant Activity And Post-Job Briefings

B. Operator Error Resulting In RPS Actuation

Clinton Power Station
LER: 86-013-00

1. Description Of The Event

   a. On October 19, 1986, at Clinton Power Station (CPS), during performance of the IRM Channel Functional Surveillance Test, the MCR Operator inadvertently placed the SRM Channel "C" Mode Switch out of the "OPERATE" position, instead of placing the IRM Channel Mode Switch in the "TEST" position. This resulted in a RPS trip (scram).

   At Limerick, SRM Channel Mode Switches are in the AER

   b. The plant was in Mode 5, performing CORE ALTERATIONS during initial fuel load. The RPS shorting links were removed. All control rods were inserted prior to the trip.

   Review: with shorting links removed, one SRM or IRM channel trip will result in a full scram

c. The Surveillance Test was stopped and the Reactor Mode Switch was taken to the "SHUTDOWN" position. The SRM Channel Mode Switch was placed back in the "OPERATE" position and the RPS trip was reset after 20 minutes.

d. The SRM Channel Mode Switch was out of the "OPERATE" position only momentarily.

2. Importance Of The Event

   a. Unplanned automatic RPS Initiation requires Nuclear Regulatory Commission (NRC) notification in accordance with Federal Regulations.

   b. Event was reportable as an Licensee Event Report (LER) in accordance with Federal Regulations.

   c. According to the LER, the event had no safety significance, since a scram places the plant in a safe condition, and the unit was already shutdown.

   d. However, this event most likely had effects on the outage:

      1) Schedule impact

      2) Possibility of water clarity problems due to high CRDH System flow rate

3. Causes Of The Event And Corrective Actions

   a. The cause of the event was the failure of a Licensed Operator to follow an approved procedure.

      Corrective Action:

      The Operator was counseled on the need to be more alert and careful of his control manipulations. If the Operator had followed the procedure, and had used self-check verification techniques (STAR), this event would not have happened.

Operations Fundamentals

PRECISELY CONTROLING PLANT EVOLUTIONS

- **Procedural Adherence** *(Adhere to procedures as written)*

4. Possible Error Precursors
   a. Distractions/Interruptions
   b. Changes/Departure from routine
   c. Time pressure
   d. Repetitive actions
   e. Stress

Instructor should discuss how various Error Precursors led or could have led to this event

5. Prevention
   a. Verification techniques, covered by HU-AA-101, Human Performance Tools And Verification Practices
   b. Procedural adherence, covered by HU-AA-104-101, Procedure Use And Adherence

VIII. TECHNICAL SPECIFICATIONS

A. SRM System Technical Specifications

Objective IL10.a
Objective IL10.b
Objective IL10.c

1. 3/4.3.6, Control Rod Block Instrumentation - Source Range Monitors (Technical Specification 3.3.6)
   a. Specifies required Control Rod Block Instrumentation channels (Table 3.3.6-1)
   b. Specifies Control Rod Block Instrumentation Setpoints (Table 3.3.6-2)
   c. Specifies allowable control rod block bypass conditions.
   d. Contains Figure 3.3.6-1, SRM Count Rate Versus Signal To Noise Ratio
   e. Specifies SRM Channel Surveillance Requirements (Table 4.3.6-1)

2. 3/4.3.7, Monitoring Instrumentation - Source Range Monitors (Technical Specification 3.3.7.6)
   a. Specifies required number of OPERABLE SRM channels for various OPCONs.
   b. **IMPORTANT NOTE**: prior to control rod withdrawal, SRM channel count rates must be verified to be > 3.0 CPS. This required count rate value can be reduced, provided the SRM detector has an observed count rate and a signal to noise ratio on or above Technical Specification Figure 3.3.6-1.
   c. Bases (3/4.3.7.6). The SRM System provides Operators with status of neutron flux at very low Reactor Power levels. Reactivity additions shall not be made without this information. SRM detectors can be retracted when the IRM System is providing needed information.
3. 3/4.9.2, Refueling Operations - Instrumentation (Technical Specification 3.9.2)
   a. Specifies required SRM channel operability during refuel operations.
   b. Unless adequate SDM has been demonstrated, shorting links must be removed for control rod withdrawal (not required for control rods removed per Technical Specification 3.9.10.1 or 3.9.10.2)
   c. Bases (3/4.9.2). Ensures redundant monitoring capability to detect core reactivity changes. During a core offload, the last fuel to be removed is that fuel adjacent to the SRM detectors.
4. 3/4.9.10, Refueling Operations - Control Rod Removal (Technical Specifications 3.9.10.1 and 3.9.10.2)
   a. Provides requirements for removal of a single or multiple control rods and/or associated mechanisms
   b. The SRM System must be OPERABLE per Technical Specification 3.9.2
   c. Bases (3/4.9.10). Ensures maintenance or repair of control rods or drives will be performed under conditions that limit the probability of inadvertent criticality.

5. Table 4.3.1.1-1, Reactor Protection System Instrumentation Surveillance Requirements
   a. Footnote (b)
   b. Requires verification of IRM/SRM System overlap by at least one-half decade during each startup after entering OPCON 2

ST-6-107-884-*, Neutron Monitoring System Overlap Verification On Startup

6. 3/4.3.7.5, Accident Monitoring Instrumentation (Technical Specification 3.3.7.5)
   a. Table 3.3.7.5-1, Accident Monitoring Instrumentation, Item 13, specifies requirements for "Neutron Flux" instrumentation channels in OPCONs 1 and 2
   b. The requirements specified for "Neutron Flux" are as follows:
      1) Required Number Of Channels:
         - Two SRM channels, and
         - Two IRM channels, and
         - Two APRM channels
      2) Minimum Channels Operable:
         - One SRM channel, and
         - One IRM channel, and
         - One APRM channel

Required Number of Channels: 2

Minimum Channels Operable: 1

Reference: IR 00718791 **AND** LER 2009-001-00, Neutron Flux Accident Monitoring Instrumentation Inoperable

B. IRM System Technical Specifications

1. 2.2, Limiting Safety System Settings, Reactor Protection System Instrumentation Setpoints (Technical Specification 2.2.1)
   a. Specifies RPS instrumentation trip setpoints (Technical Specification Table 2.2.1-1)
   b. Bases (2.2.1.1). The IRM System provides protection against local control rod errors and continuous withdrawal of control rods in sequence
      1) UFSAR analyzes control rod withdrawal accidents. Most severe cases involve a Thermal Power of approximately 1%.
      2) Peak power is limited to 21% of Rated Thermal Power with peak fuel enthalpy well below fuel failure threshold of 170 calories/gram.
      3) The IRM System provides backup protection for the APRM System.

Objective IL24.a
Objective IL24.b
Objective IL24.c

NOTE: the RWM System protects against out-of-sequence control rod movement

2. 3/4.3.1, Reactor Protection System Instrumentation - Intermediate Range Monitors (Technical Specification 3.3.1)
   a. Specifies required Reactor Protection Trip System Instrumentation Channels (Technical Specification Table 3.3.1-1)
   b. Specifies required Reactor Protection Trip System Response Times (Technical Specification Table 3.3.1-2)
   c. Specifies IRM channel Reactor Protection Instrumentation Surveillance Requirements (Technical Specification Table 4.3.1.1-1)
3. Table 4.3.1.1-1, Reactor Protection System Instrumentation Surveillance Requirements
   a. Footnote (b)
   b. Requires verification of IRM/SRM System overlap of at least one-half decade during each startup after entering OPCON 2, and overlap between IRM/APRM Systems during each controlled shutdown if not performed within the previous 7 days.

ST-6-107-886-*, Neutron Monitoring System Overlap Verification On Shutdown

4. 3/4.3.6, Control Rod Block Instrumentation - Intermediate Range Monitors (Technical Specification 3.3.6)
   a. Specifies required Control Rod Block Instrumentation Channels (Technical Specification Table 3.3.6-1)
   b. Specifies required Control Rod Block Instrumentation Setpoints (Technical Specification Table 3.3.6-2)
   c. Specifies allowable Control Rod Block Bypass Conditions.
   d. Specifies required IRM Channel Control Rod Withdrawal Block instrumentation Surveillance Requirements (Technical Specification Table 4.3.6-1)

Content/Skills	Activities/Notes
5. 3/4.3.7.5, Accident Monitoring Instrumentation (Technical Specification 3.3.7.5)	
a. Table 3.3.7.5-1, Accident Monitoring Instrumentation, Item 13, specifies requirements for "Neutron Flux" instrumentation channels in OPCONs 1 and 2	Required Number of Channels: 2 Minimum Channels Operable: 1
b. The requirements specified for "Neutron Flux" are as follows: 1) Required Number Of Channels: • Two SRM channels, and • Two IRM channels, and • Two APRM channels 2) Minimum Channels Operable: • One SRM channel, and • One IRM channel, and • One APRM channel	Reference: IR 00718791 **AND** LER 2009-001-00, Neutron Flux Accident Monitoring Instrumentation Inoperable

IX. EMERGENCY ACTION LEVELS (EALs)

Objective IL11

A. SRM System EALs

1. **MU2 (Hot Matrix)/CU2 (Cold Matrix), Inadvertent Criticality**

   Hot and Cold Matrices

   Using current revision of EAL Matrices, Instructor should review each of the specified EALs

   a. Applicability: MODE 3 ONLY

   b. EAL Threshold Values: UNPLANNED sustained positive period observed on nuclear instrumentation

2. **CG6, Loss Of RPV Inventory Affecting Fuel Clad Integrity With Containment Challenged**

   Cold Matrix

   a. Applicability: MODES 4 and 5 ONLY

   b. EAL Threshold Values:

      1. a. RPV level < **-161 inches** for ≥ **30 minutes**

         AND

         b. Any Containment Challenge Indication (Table C5)

      OR

      2. a. RPV level unknown for ≥ **30 minutes**

         AND

         b. Loss of RPV inventory as indicated by any of the following:

         - Table C3 indication

           OR

         - Erratic Source Range Monitor indication

           OR

         - Any Table C4 Refuel Floor Area Radiation Monitor **> 3 R/hr**

         AND

         c. Any Containment Challenge Indication (Table C5)

3. **CS6, Loss Of RPV Inventory Affecting Core Decay Heat Removal Capability**

   Cold Matrix

   a. Applicability: MODES 4 and 5 ONLY

   b. EAL Threshold Values:

      1. With CONTAINMENT CLOSURE **not** established, RPV level **< -44 inches**

      OR

      2. With CONTAINMENT CLOSURE established, RPV level < **-161 inches**

      OR

      3. a. RPV level unknown for **≥ 30 minutes**

         AND

         b. Loss of RPV inventory as indicated by any of the following:

            - Table C3 indication

              OR

            - Erratic Source Range Monitor indication

              OR

            - Any Table C4 Refuel Floor Area Radiation Monitor **> 3 R/hr**

B. IRM System EALs

None

X. SUMMARY

A. Purpose Of Nuclear Instrumentation

B. SRM System

1. Detector
2. Detector Drive
3. Channel Description
4. Control Functions
   a. Control Rod Withdrawal Block Signals
   b. Scram Signals
5. MCR Indications and Alarms
6. AER Indications and Controls
7. Power Supplies
8. System Operation
9. System Malfunctions/Losses

C. IRM System

1. Detector
2. Detector Drive
3. Channel Description
4. Control Functions
   a. Control Rod Withdrawal Block Signals
   b. Scram Signals
5. MCR Indications and Alarms
6. AER Indications and Controls
7. Power Supplies
8. System Operation
9. System Malfunctions/Losses

D. Additional Operating Experience

E. Technical Specifications

F. Emergency Action Levels (EALs)

XI. EVALUATION

A. In accordance with Course Plan

XII. ATTACHMENTS

Sample Questions

1.

Plant conditions are as follows:

- Shorting links are installed
- SRM Channel "A" indicating 2.1×10^5 CPS
- IRM Channel "B" indicating 4/125

WHICH ONE of the following describes the condition of RPS/RMCS?

a. RMCS control rod block only

b. Half-scram only on RPS Trip System "A"

c. Half-scram on RPS Trip System "A" and a RMCS control rod block

d. Half-scram on RPS Trip System "B" and a RMCS control rod block

2.

Given the following conditions:

- LGS Unit 1 is in OPCON 3
- RPV Pressure is 850 psig
- Cooldown in progress

During the reactor shutdown, the Reactor Operator notified SSV that SRM Channels "A" and "C" were not responding correctly. I&C estimates that it will take 2.5 hours to return one of the channels to service.

WHICH ONE of the following ACTION statements is required by Technical Specifications?

a. Remove shorting links from RPS circuitry.

b. Ensure all four SRM detectors are fully inserted into the core.

c. None. Only two SRM channels are required to be OPERABLE in OPCON 3.

d. Lock the Reactor Mode Switch in the "SHUTDOWN" position within one hour.

3.

A reactor startup is in progress with the following plant conditions:

• Reactor Water Temperature:	172 °F
• Reactor Mode Switch Position:	START & HOT STBY
• RPV Pressure:	0 psig
• SRM Channels:	All indicating between 1 x10^4 and 1 x10^5 CPS
• IRM Channels:	All channels indicating on-scale on Range 3
• Control Rod Blocks:	None in effect

WHICH ONE of the following describes the effect of withdrawing all four SRM detectors at this time?

a. The SRM detectors will retract fully with no alarms occurring, regardless of whether the SRM channels are indicating downscale

b. The SRM detectors will retract until any <u>one</u> SRM channel indication drops below 100 CPS, at which time the "RETRACTED WHEN NOT PERMITTED" annunciator will alarm along with application of a control rod withdrawal block

c. The SRM detectors will retract until any <u>one</u> SRM channel indication drops below 100 CPS, at which time the "RETRACTED WHEN NOT PERMITTED" annunciator will alarm and SRM detector withdrawal will be blocked

d. The SRM detectors will retract until all four SRM channel indications drop below 100 CPS, at which time the "RETRACTED WHEN NOT PERMITTED" annunciator will alarm along with application of a control rod withdrawal block

4.

A reactor startup on Unit 2 is in progress. The Neutron Monitoring System Overlap Surveillance Test is complete and SRM detectors are being retracted with the following conditions:

IRM Channel	Range Switch Position
A	3
B	3
C	2
D	BYPASSED
E	3
F	4
G	3
H	5

WHICH ONE of the following describes the expected response if, when being retracted, SRM Channel "A" indication drops to 80 CPS?

a. No alarms and no control rod withdrawal blocks

b. "SRM DOWNSCALE" alarm and no control rod withdrawal blocks

c. "SRM RETRACTED WHEN NOT PERMITTED" alarm and a control rod withdrawal block

d. "SRM RETRACTED WHEN NOT PERMITTED" alarm and no control rod withdrawal block

5.

Given the following conditions:

- Reactor Power is 7%
- Reactor Mode Switch in "RUN"
- IRM channels are indicating 25/125 on Range 10

A welder, on the 283 foot elevation of the Reactor Enclosure draws an arc, which causes IRM Channels "A" and "E" to spike full upscale.

WHICH ONE of the statements describes the status of RPS/RMCS?

a. No RPS scram and no RMCS rod block

b. Half-scram on RPS "A" only

c. Full scram and RMCS rod block

d. Half-scram on RPS "A" and RMCS rod block

6.

Given the following:

- Reactor startup is in progress
- Shorting links have been removed from RPS
- The Reactor has just been declared critical
- IRM Channel "F" High Voltage Power Supply fails low

WHICH ONE of the statements describes the status of RPS?

a. Half-scram on RPS "A" only

b. Half-scram on RPS "B" only

c. No RPS scram, no control rod motion

d. Full scram, all control rods filly inserted

7.

Unit 1 plant conditions are as follows:

- OPCON 2, with a reactor startup in progress
- All IRM channel Range Switches are on Range 7, and indicating as follows:

IRM Channel	Indication
A	32/125
B	36/125
C	17/125
D	19/125
E	24/125
F	29/125
G	30/125
H	26/125

WHICH ONE of the following describes the plant status if all IRM channel Range Switches are positioned to Range 6?

a. No control rod withdrawal block, no half-scram, and no full-scram

b. Control rod withdrawal block only

c. Control rod withdrawal block and RPS Trip System "B" half-scram

d. Control rod withdrawal block and full-scram

8.

Given the following plant conditions:

- Reactor Mode Switch is in "START & HOT STBY"
- All IRM channels are indicating mid-scale on Range 3

IRM Channel "B" fails downscale coincident with IRM Channel "C" failing upscale.

WHICH ONE of the following describes the expected status of RPS to the above conditions?

	RPS Trip System "A"	RPS Trip System "B"
a.	De-energized	De-energized
b.	Energized	De-energized
c.	De-energized	Energized
d.	Energized	Energized

Answers To Practice Questions

1. **A**

   Shorting links prevent SRM System scram signal. SRM System control rod withdrawal block signal initiation occurs at ≥ 1×10^5 CPS. IRM System control rod withdrawal block occurs when signal is < 5/125 and IRM Range Switch is <u>not</u> on Range 1.

2. **C**

   Reference Technical Specification 3.3.7.6

3. **A**

   All IRM Channel Range Switches are on Range 3

4. **C**

   IRM Channel "C" Range Switch is on Range 2, so SRM Channel "A" will still generate a downscale control rod withdrawal block

5. **A**

   IRM channel Upscale trip signals to RPS and RMCS are bypassed when the Reactor Mode Switch is in the "RUN" position

6. **D**

   Failure of the IRM Channel "F" High Voltage Power Supply causes an INOP trip signal to be generated to RPS Trip System "B." Removal of the shorting links enables RPS to complete a full scram on a single IRM channel trip condition

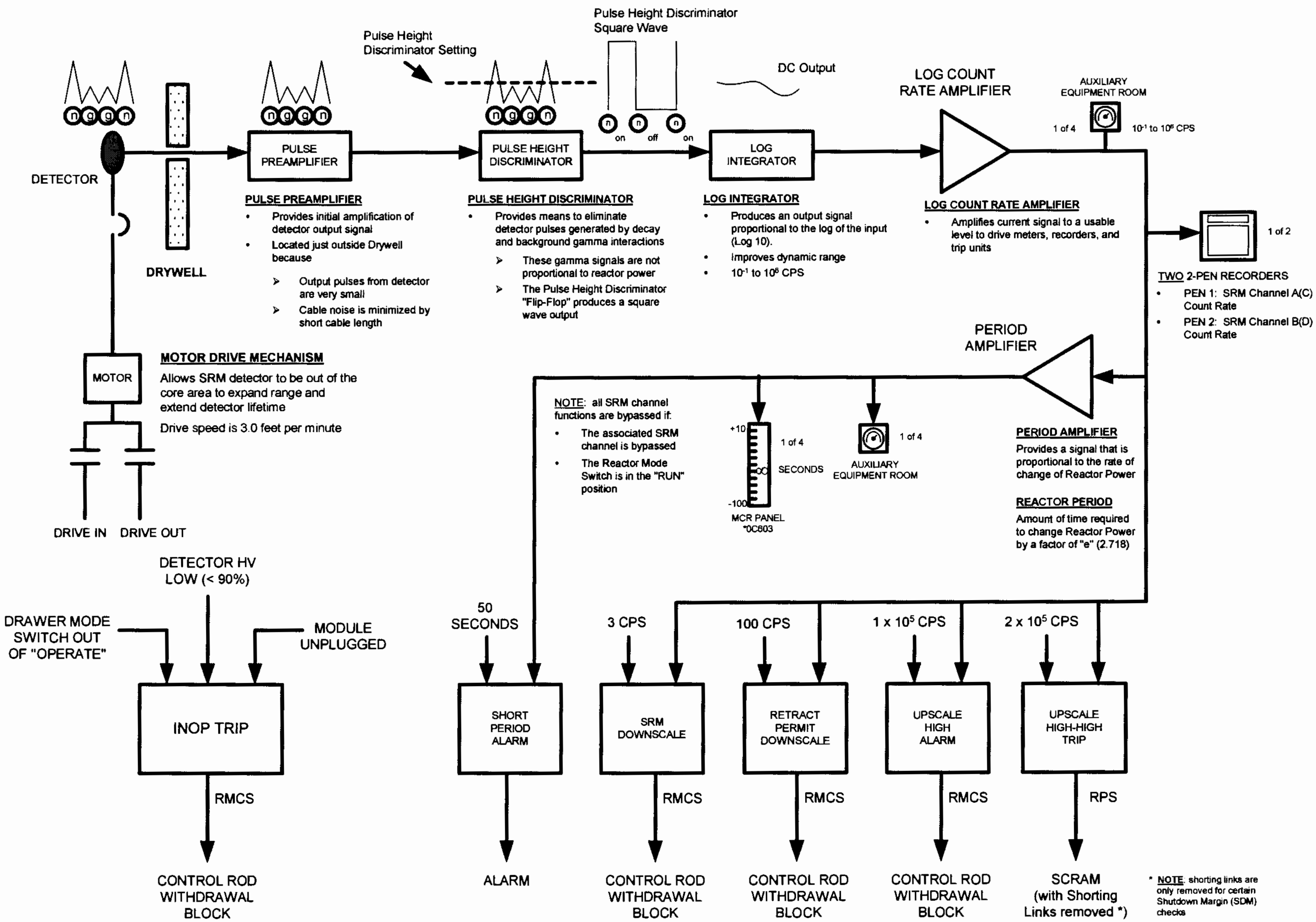
7. **B**

   Ranging down causes IRM channel indications to rise by a factor of 3.16. The highest reading IRM channel indication is 36/125. When the Range Switch is downranged, channel indication will change to (36 x 3.16 = 114), which is above the control rod withdrawal block setpoint (≥ 85/125), but below the RPS trip setpoint (≥ 120/125)

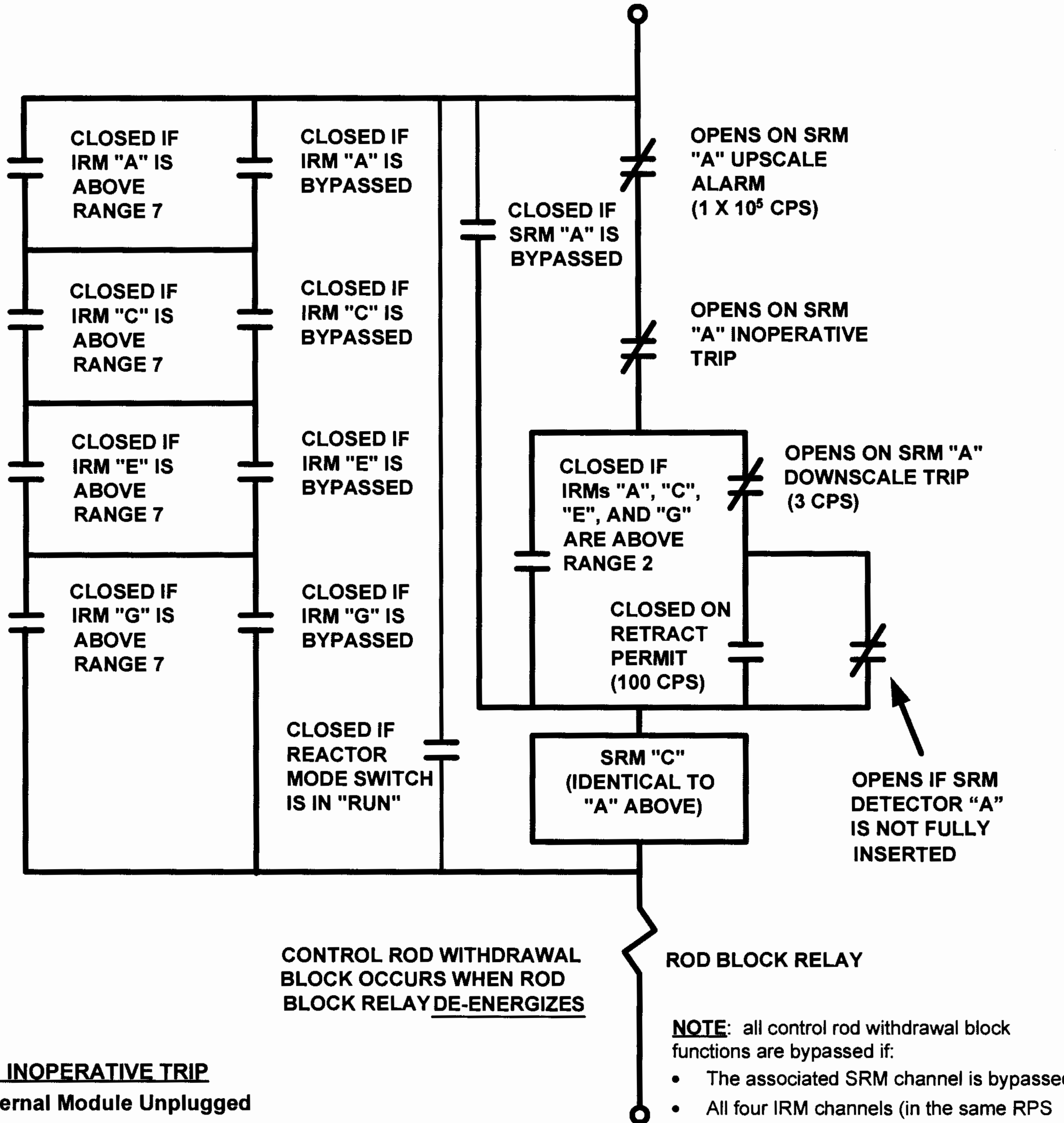
8. **C**

   The failure of IRM Channel "B" is downscale, which will have no affect on RPS Trip System "B" (i.e., it will remain energized). The failure of IRM Channel "A," however, is upscale, which will result in a half-scram signal being generated by RPS Trip System "A" (i.e., it will de-energize).

ATTACHMENT 1
SRM SYSTEM SIMPLIFIED BLOCK DIAGRAM



SRM CHANNEL "A"
CONTROL ROD BLOCK SIGNALS AND BYPASSES



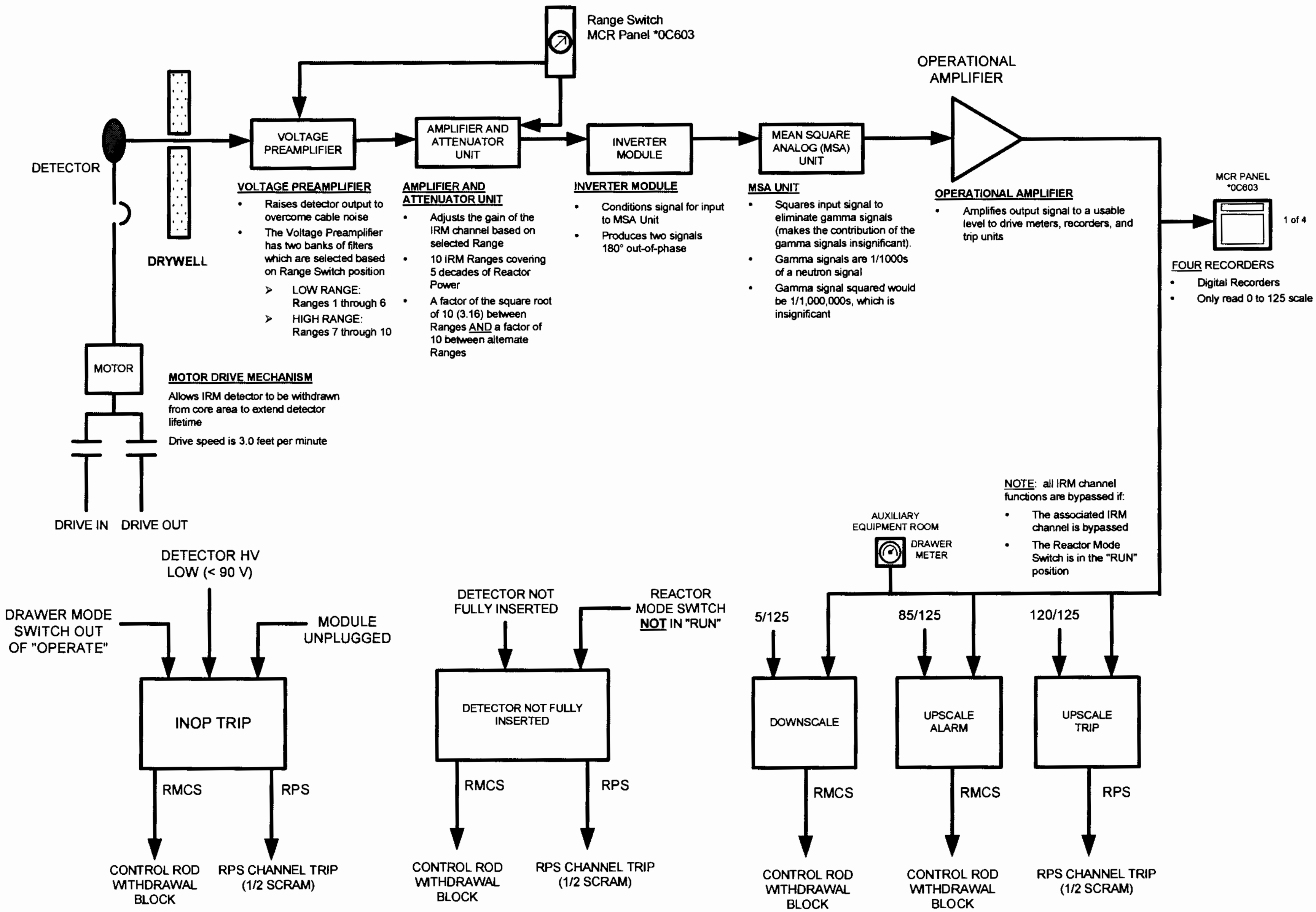
NOTE: all control rod withdrawal block functions are bypassed if:

- The associated SRM channel is bypassed
- All four IRM channels (in the same RPS trip system) are on Range 8 or above
- The Reactor Mode Switch is in the "RUN" position

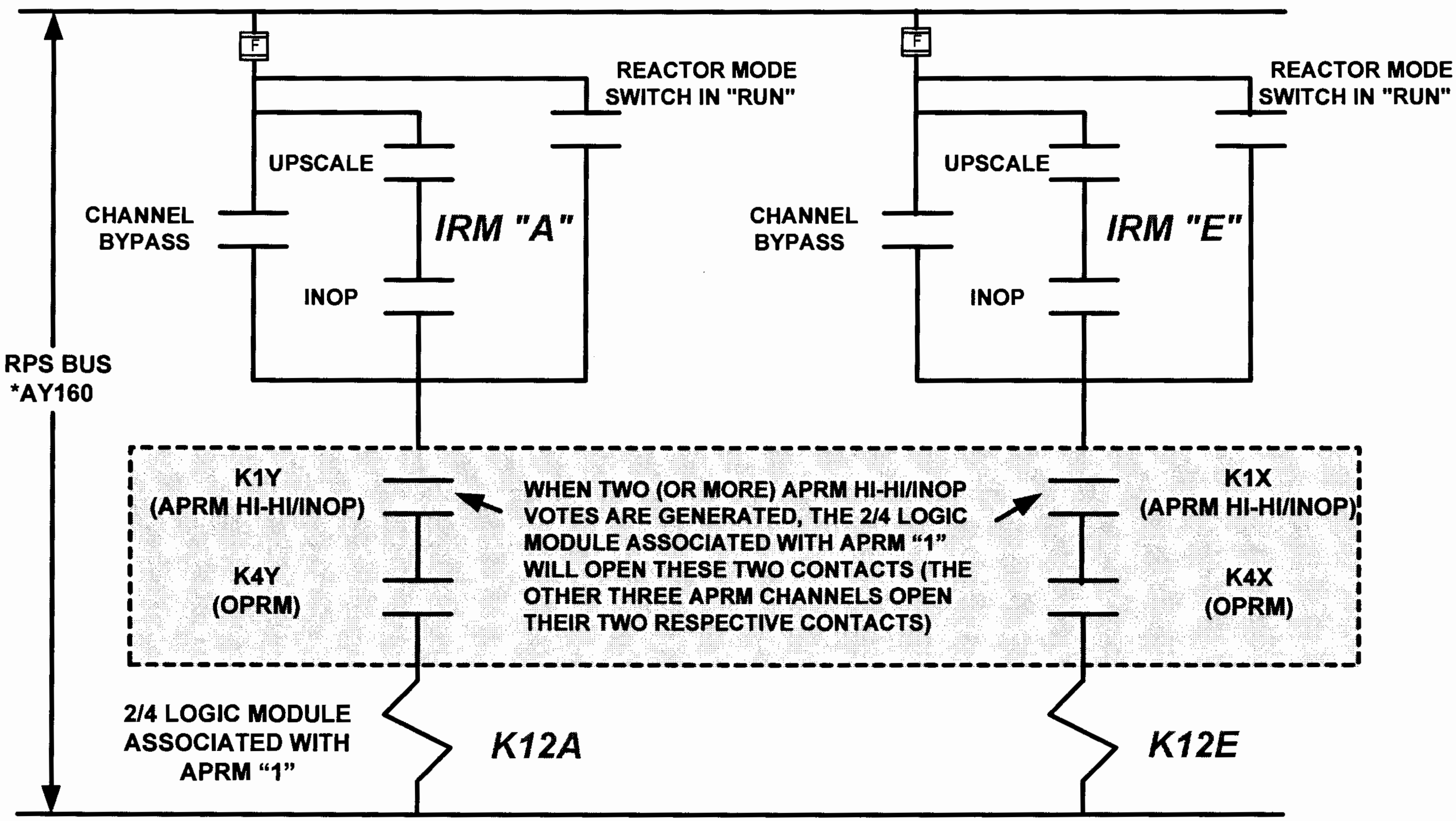
SRM INOPERATIVE TRIP

- **Internal Module Unplugged**
- **Channel Mode Switch not in "OPERATE"**
- **Low detector voltage**

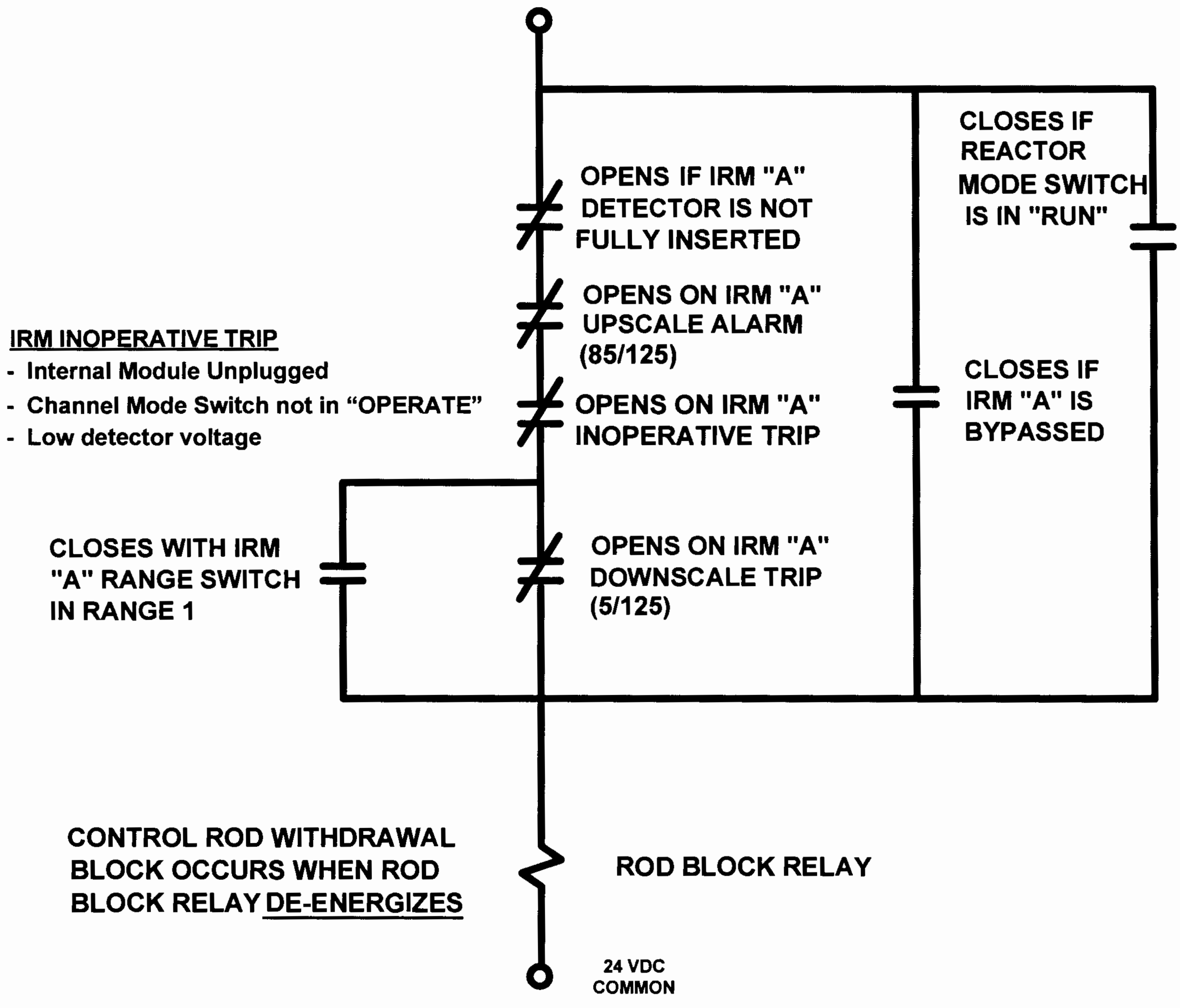
IRM SYSTEM SIMPLIFIED BLOCK DIAGRAM



APRM "1" 2/4 LOGIC MODULE INTERFACE WITH SCRAM CONTACTS IN RPS TRIP CHANNEL "A1"



IRM CHANNEL "A" CONTROL ROD BLOCK SIGNALS AND BYPASSES



SRM CONTROL ROD BLOCK AND SCRAM SIGNALS (INCLUDING BYPASSES)

ROD BLOCK	SETPOINT	WHEN BYPASSED
SRM Upscale	≥ $1x10^5$ CPS	• Any time associated SRM channel is bypassed • Any time all four associated IRM channels are on Range 8 <u>or</u> above • Reactor Mode Switch in "RUN"
SRM Inoperative (INOP)	• Channel module unplugged <u>or</u> other hardware failure • Channel Mode Switch (AER) <u>not</u> in "OPERATE" • Low detector high voltage (<90%)	• Any time associated SRM channel is bypassed • Any time all four associated IRM channels are on Range 8 <u>or</u> above • Reactor Mode Switch in "RUN"
SRM Downscale	≤ 3 CPS	• Any time associated SRM channel is bypassed • Any time all four associated IRM channels are on Range 3 <u>or</u> above • Reactor Mode Switch in "RUN"
SRM Not Fully Inserted And Indicating ≤ 100 CPS	Associated SRM channel count rate ≤ 100 CPS <u>AND</u> all four associated IRM channels on Range 2 or below	• Any time associated SRM channel is bypassed • Any time all four associated IRM channels are on Range 8 <u>or</u> above • Reactor Mode Switch in "RUN" • Any time associated SRM channel count rate >100 CPS • Any time all four associated IRM channels are on Range 3 <u>or</u> above
SCRAM	**SETPOINT**	**WHEN BYPASSED**
SRM Upscale (High-High)	≥ $2x10^5$ CPS	• When associated shorting links are installed • Any time associated SRM channel is bypassed
SRM Inoperative (INOP)	Loss of power to associated SRM channel <u>ONLY</u>	• When associated shorting links are installed

IRM CONTROL ROD BLOCK AND SCRAM SIGNALS (INCLUDING BYPASSES)

ROD BLOCK	SETPOINT	WHEN BYPASSED
IRM Upscale	≥ 85/125	• Any time associated IRM channel is bypassed • Reactor Mode Switch in "RUN"
IRM Inoperative (INOP)	• Channel module unplugged <u>or</u> other hardware failure • Channel Mode Switch (AER) <u>not</u> in "OPERATE" • Low detector high voltage (< 90V)	• Any time associated IRM channel is bypassed • Reactor Mode Switch in "RUN"
IRM Downscale	≤ 5/125	• Any time associated IRM channel is bypassed • Reactor Mode Switch in "RUN" • Any time associated IRM channel is on Range 1
IRM Not Fully Inserted	Associated IRM detector not fully inserted	• Any time associated IRM channel is bypassed • Reactor Mode Switch in "RUN"
SCRAM	**SETPOINT**	**WHEN BYPASSED**
IRM Upscale (High-High)	≥ 120/125	• Any time associated IRM channel is bypassed • Reactor Mode Switch in "RUN"
IRM Inoperative (INOP)	• Channel module unplugged <u>or</u> other hardware failure • Channel Mode Switch (AER) <u>not</u> in "OPERATE" • Low detector high voltage (<90V)	• Any time associated IRM channel is bypassed • Reactor Mode Switch in "RUN"

III. DAILY REPORT

3/6/2015	Goal	Unit 1 Value	Unit 2 Value
CEI (24 Mo. Rolling Avg)	**3.00**	**2.06**	**1.50**
CEI (EOY Projection)	**3.00**	**1.99**	**1.00**

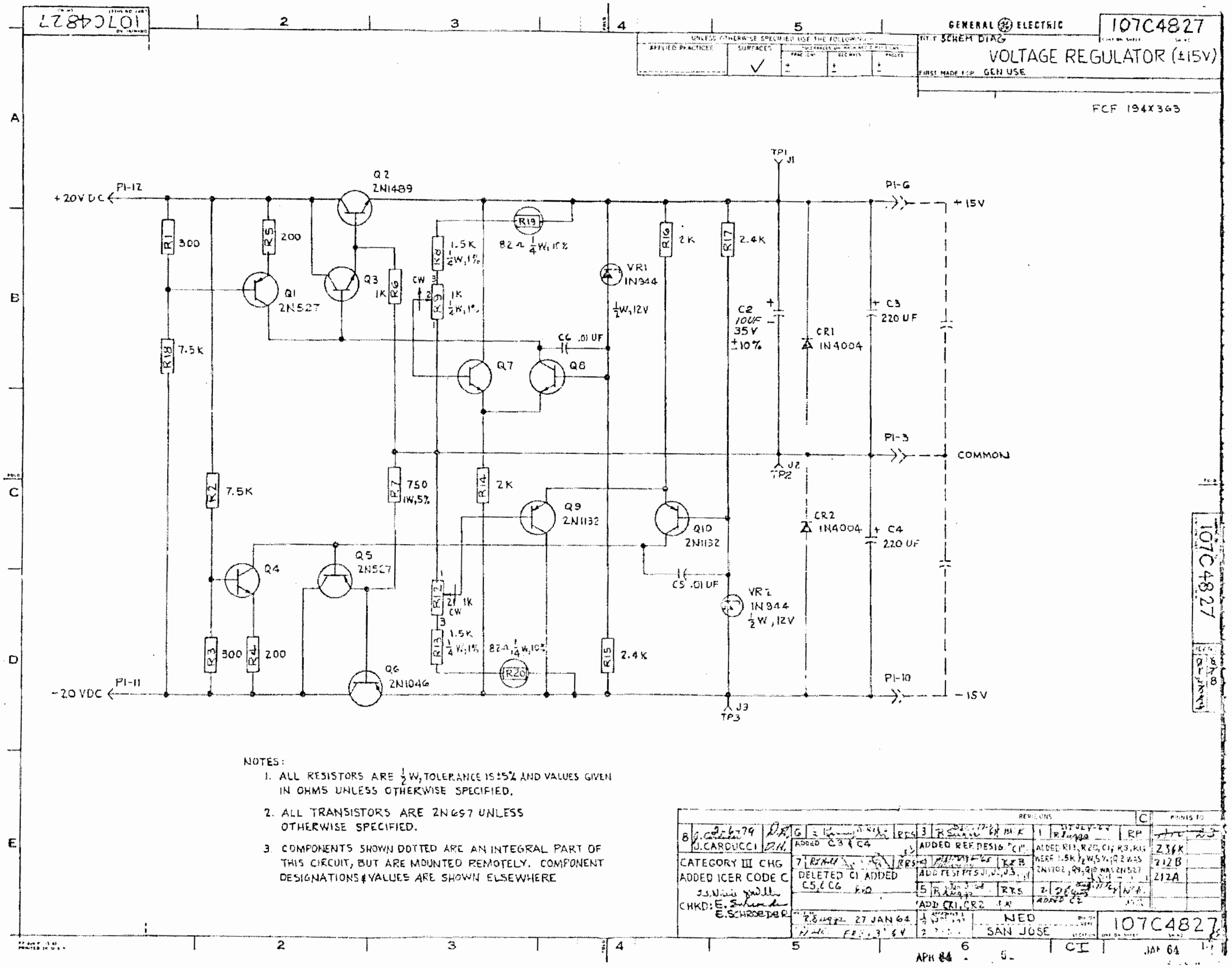
		Goal	Limit	Result Date	Unit 1 Values	Trend	Result Date	Unit 2 Values	Trend
	REACTOR WATER								
*1	Conductivity, µS/cm	**≤0.15**	**>0.30**	3/5/2015	0.13	Stdy	3/5/2015	**0.17**	Stdy
*1/4	Chloride, ppb	**≤1.0**	**>5**	3/5/2015	0.12	Stdy	3/5/2015	0.10	Stdy
*1	Sulfate, ppb	**≤2.0**	**>5**	3/5/2015	0.38	Stdy	3/5/2015	0.30	Stdy
	Zinc, Sol, ppb	**≥5 ppb**	N/A	2/19/2015	17.5	Stdy	2/19/2015	15.4	Stdy
*5	Co60, Sol, µCi/ml	**≤1.0E-4**	N/A	3/5/2015	8.23E-05	Stdy	3/5/2015	**1.86E-04**	Stdy
	Co60, InSol, µCi/ml	**≤1.0E-4**	N/A	3/5/2015	2.80E-05	Stdy	3/5/2015	2.32E-05	Stdy
*5	Co60/Zinc, Sol, Ratio	**≤1.0/2.0E-05**	N/A		4.72E-06	Stdy		1.21E-05	Stdy
	FEEDWATER								
	Conductivity, µS/cm	**≤0.060**	**>0.065**	3/5/2015	0.056	Stdy	3/5/2015	0.058	Stdy
*2	Cu, ppb	**≤0.05**	**≤0.10**	2/23/2015	0.003	Stdy	2/25/2015	0.004	Stdy
	DO_2, ppb	**≥30 - ≤80**	**<30 or >200**	3/5/2015	60.0	Stdy	3/5/2015	64.0	Stdy
*2	Iron, ppb (Cycle Avg)	**≥0.1 - ≤1.0**	**>5.0**	2/23/2015	0.206	Stdy	2/25/2015	0.264	Stdy
	Zinc, ppb (Cycle Avg)	**≤0.40 / ≤0.60**	**≤1.3**	2/23/2015	0.340	Stdy	2/25/2015	0.357	Stdy
	OFFGAS PRE-TREATMENT								
	Sum of 6, µCi/s calculated		**1000/1500**	3/5/2015	136	Stdy	3/5/2015	341	Stdy
	Sum of 6, µCi/s grab		**1000/1500**	3/4/2015	138	Stdy	3/4/2015	337	Stdy
	CONDENSATE								
	CDI Conductivity, µS/cm	**≤0.08**	**>0.10**	3/5/2015	0.067	Stdy	3/5/2015	0.062	Stdy
	Moisture Carryover (%)		**≤0.15**	3/2/2015	0.010	Stdy	3/2/2015	0.008	Stdy
	SPRAY POND								
	pH, su	**7.5-9.5**	**7.0-10.0**	3/5/2015	8.02	Stdy			
	Manganese, ppb	**<100**	**<200**	3/3/2015	74	Stdy			
	SYSTEM AVAILABILITY								
*3	HWC Avail. (12M Rolling Avg)	**≥99%**	N/A		98.6%	Stdy		99.5%	Stdy
	HWC Trips (Monthly/YTD)	**0/0**	N/A		0/2	Stdy		0/0	Stdy
*3	HWC (Monthly), hrs	**0 hrs**	**<7 hrs**	3/2/2015	0.00	Stdy	3/2/2015	0.00	Stdy
*3	Noble Metal (Monthly), hrs	**<36 hrs**	**<144 hrs**	3/2/2015	0.00	Stdy	3/2/2015	0.00	Stdy

U2 Rx Conductivity is above goal due to sodium leakage from the 5 oldest condensate deep beds. These beds were all in service during the U2 condenser leak. They are loaded with sodium, and equilibrium leakage will continue until the beds have been replaced.

U2 Soluble Co60 is increasing as the unit approaches end-of-cycle.

* **CEI Input**
1 CEI Condition 1 - > limit impacts CEI
2 CEI Condition 2 - > limit impacts CEI
3 CEI Condition 3 - HWC unavailability > limit impacts CEI
NMMS unavailability > limit impacts CEI
4 CEI Condition 4 - > goal impacts CEI
5 CEI Condition 5 - Combination of Sol Co-60 and ratio (Co-60 <5E-5 no impact on CEI)

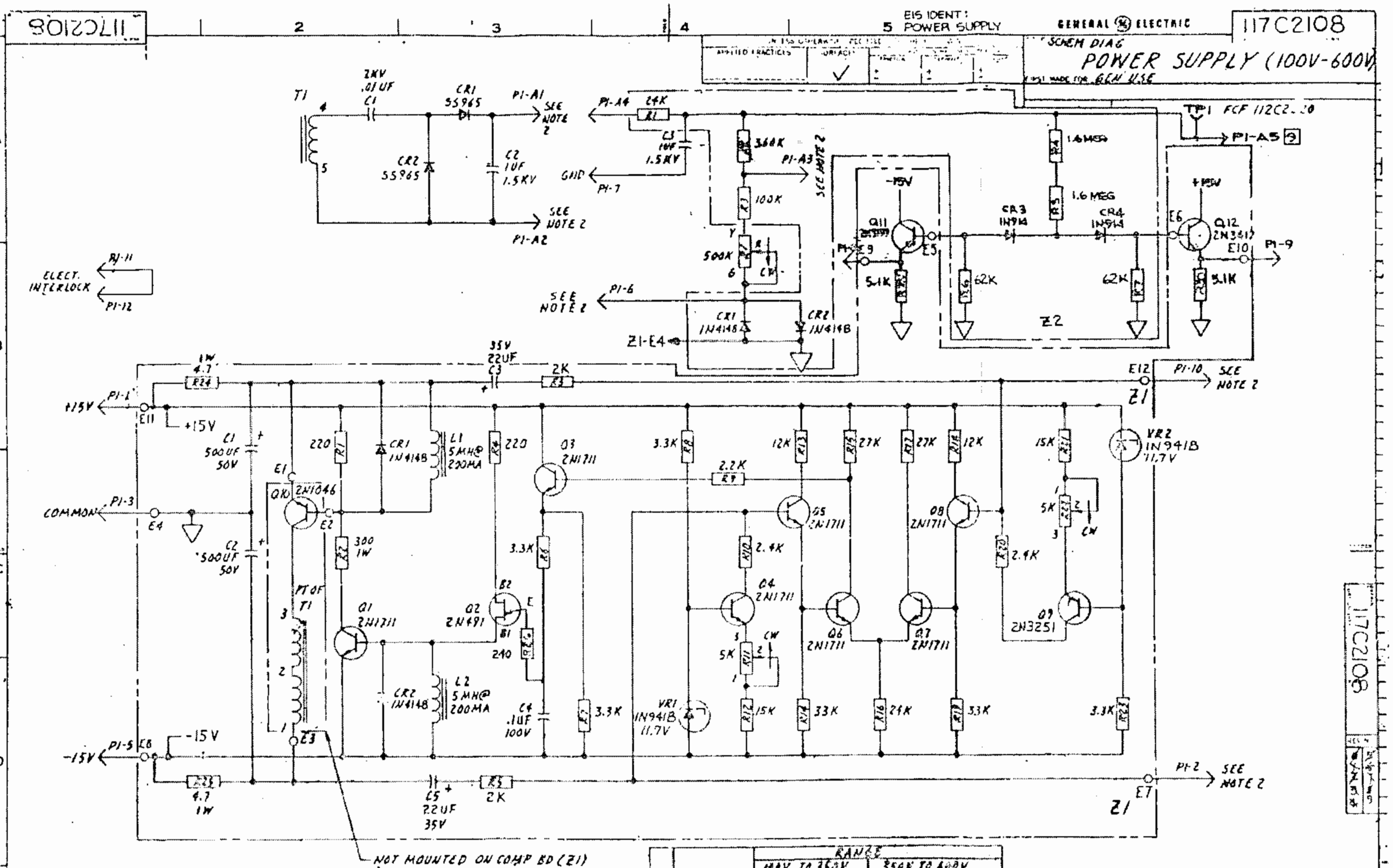
5RM Voltage Regulator



NOTES:
1. ALL RESISTORS ARE ½W, TOLERANCE IS ±5% AND VALUES GIVEN IN OHMS UNLESS OTHERWISE SPECIFIED.
2. ALL TRANSISTORS ARE 2N697 UNLESS OTHERWISE SPECIFIED.
3. COMPONENTS SHOWN DOTTED ARE AN INTEGRAL PART OF THIS CIRCUIT, BUT ARE MOUNTED REMOTELY. COMPONENT DESIGNATIONS & VALUES ARE SHOWN ELSEWHERE

J. CARDUCCI
CATEGORY III CHG
ADDED ICER CODE C
CHKD: E. SCHROEDER
ADDED C3 & C4
DELETED C1 ADDED C5 & C6
ADDED REF DESIG "C1"
ADD CR1, CR2
27 JAN 64
NED
SAN JOSE
107C4827
CI
JAN 64

JRM High Voltage Power Supply



NOTE

1. ALL RESISTORS ARE 1/2 W, ±5% TOLERANCE & VALUES GIVEN IN OHMS UNLESS OTHERWISE SPECIFIED.
2. THIS POWER SUPPLY IS CAPABLE OF SUPPLYING 100V TO 600V POSITIVE OR NEGATIVE. SEE TABLE 1 FOR CONNECTIONS TO BE MADE EXTERNALLY FOR DIFFERENT VOLTAGE RANGES & POLARITIES.

POLARITY	RANGE: 100V TO 350V		RANGE: 350V TO 600V	
POSITIVE	A1 TO	A4	A1 TO	A4
	A2	GND	A2	GND
	A3	A5	A3	NOT CONNECTED
	PIN 2	PIN 6	PIN 2	PIN 3
	PIN 10	GND	PIN 10	GND
NEGATIVE	A1	GND	A1	GND
	A2	A4	A2	A4
	A3	A5	A3	NOT CONNECTED
	PIN 2	GND	PIN 2	GND
	PIN 10 TO	PIN 6	PIN 10 TO	PIN 5

TABLE 1

9	8-27-80 T.A. ROGERS RSB
8	3-27-79 W.B. WALDROP RRS
	NJ18851 T/T
	NJ07183
CHK'D BY:	NORMAN NEUMAN
CHKD BY:	R.L. INMAN

NPD SAN JOSE

117C2108

MASTER 117C2223

GENERAL ELECTRIC

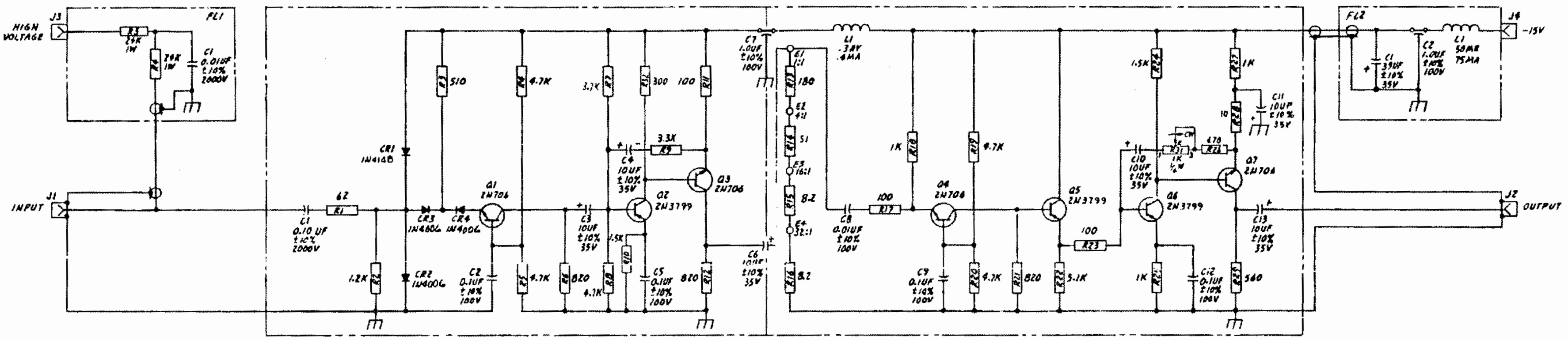
SCHEM DIAG

PULSE PREAMPLIFIER

FIRST MADE FOR LOGARITHMIC COUNT RATE MONITOR

846D581

5Rm Pre-Amp



NOTE

1. ALL RESISTORS ARE 1/2W, ±5% TOLERANCE & VALUES GIVEN IN OHMS, UNLESS OTHERWISE SPECIFIED.
2. POSITIVE LEAD OF CAPACITOR C6 IS USED AS GAIN ADJUSTMENT STRAP. UNIT IS FACTORY-SET FOR 1:1 ATTENUATION. OTHER ATTENUATION RATIOS ARE OBTAINED BY MOVING POSITIVE LEAD OF C6 TO POINT E2, E3 OR E4 AS REQUIRED.

RECEPTACLE	DESCRIPTION
J1	UG-496/U
J2	UG-496/U
J3	UG-931/U
J4	UG-447/U

NED SAN JOSE

846D581

SECTION II

THEORY OF OPERATION

2-1 GENERAL

2-2 The Source Range Monitor (SRM) may be separated into the following circuits: log count rate circuits, period circuits, integral test and calibration circuits, trip circuits, power circuits, and selector circuits. The following paragraphs describe each of these circuits in detail. Refer to drawing 807E578BB at the rear of this manual.

2-3 LOG COUNT RATE CIRCUIT

2-4 The SRM receives input current pulses from the remote pulse amplifier. (See Figure 2-1.) The input pulses are applied to the pulse height discriminator, which discriminates at a preset current level to eliminate the bulk of the low amplitude pulses caused by gamma radiation or noise. The discriminator action provides a current-to-voltage transformation. The pulse height discriminator provides two outputs; a rectangular voltage pulse, of constant amplitude with a repetition rate equal to the input pulse rate, is provided as an output to drive a scaler, while a rectangular voltage pulse, with a constant amplitude and a repetition rate of one-half the input pulse rate, is provided as an input to the logarithmic integrator. The frequency division of the output to the logarithmic integrator is obtained by using a bistable multivibrator, which changes state once for every two pulses, as an output stage. The logarithmic integrator produces a dc current output which is directly proportional to the common logarithm ($\log_{10}$) of the input pulse rate. The logarithmic integrator is followed by a linear operational amplifier stage, the log count rate amplifier, which amplifies the integrator output to a level (0 to +10 Vdc) sufficient to drive the front panel and remote log count rate meters. Since the voltage applied to the log count rate meters is proportional to the logarithm of the input count rate, the meter scales are graduated accordingly and are marked off in decades from 10^{-1} counts-per-second to 10^{6} counts-per-second. The log count rate recorder output is developed across a dropping resistor in series with the front panel meter.

2-5 PERIOD CIRCUIT

2-6 The output of the count rate circuit (0 to +10 Vdc) is applied to the period amplifier circuit. The period amplifier circuit consists of a voltage divider, a differentiating operational amplifier circuit, and meter networks.

Potentiometer R2 provides an adjustment to change the amplitude of the input signal, to compensate for small variations in the size of the differentiator components (R5 and C3). The input signal is applied to the differentiating operational amplifier circuit which is designed so that an input rate of change of 45 millivolts per second will produce a -10 Vdc output. The output from the period amplifier circuit is limited between +2 and -12 Vdc. The output from the period amplifier drives two parallel PERIOD meters, one of which is mounted on the front panel; the other is mounted remotely. These two meters are graduated in a hyperbolic manner, from -100 seconds through infinity, to +10 seconds. These meters indicate the rate of change of the input count rate (analogous to the rate of change of the neutron flux level). See Figure 2-1 for a block diagram of the period circuit.

2-7 INTEGRAL TEST AND CALIBRATION CIRCUITS

2-8 Signal Generator. The SRM contains its own internal signal generator which can be used for test and calibration purposes. The signal generator develops test pulses at frequencies of 10 or 10^5 pps. The test frequencies are applied as simulated inputs to the SRM when the front panel selector switch is placed in the 10 or 10^5 position.

2-9 Ramp Generators. The SRM also contains internal ramp generators to be used for test and calibration. The ramp generators produce voltages that increase at fixed or variable rates. A ramp voltage is applied to the period circuit when the front panel selector switch is set in the PER CAL position, and the RAMP switch is set in the FIXED or the VAR position. When the front panel selector switch is set in the PER TRIP TEST position, a ramp voltage is generated for use in testing the period trip circuit.

2-10 TRIP CIRCUITS

2-11 There are three dual trip units (Z16, Z17, and Z18) in the SRM. Each dual trip unit contains two separate independent trip circuits. One of the circuits trips when the drawer is inoperative, two are downscale log count rate level trips, two are upscale log count rate level trips, and the sixth trip circuit is a negative input upscale period trip circuit.

2-12 Drawer Inoperative Trip Circuit (Z16-Trip A). The drawer inoperative trip circuit is connected as a positive input, downscale trip circuit. The input applied at pin 26 is a positive voltage between 5 and 10 volts, derived in the high voltage power supply by dividing down the high voltage output. Thus this voltage can be expected to change as the high voltage output changes. Since the trip circuit is connected as a positive-input downscale trip, it will trip when the input voltage at pin 26 becomes less positive than the input at pin 25. Consequently, malfunctions in the high voltage power supply, which cause a decrease in the voltage applied at pin 26, result in a trip. The trip circuit also trips when a module is removed or the front panel switch is placed in a position other than OPERATE. Either of these actions opens the circuit to pin 25, causing 15 volts to be applied through R1 to pin 25. This voltage is much more positive than the input voltage from the high voltage power supply applied at pin 26, so the trip circuit trips. The trip that occurs as a result of placing selector switch S1 in a position other than OPERATE can be inhibited by depressing the INOP INHIBIT switch.

2-13 When trip circuit A trips, the output at pin 23 decreases from 12 volts to less than 1 volt, the differential output across pins 19 and 21 decreases from 20 volts to less than 3 volts. In addition to these outputs, the voltage at pin 1 decreases from approximately 20 volts to less than 2 volts, causing DS16A to light. The decrease of the 20 volt differential output across pins 19 and 21 can be utilized to actuate an alarm circuit.

2-14 Trip circuit Z16-Trip A is connected for automatic reset from a trip caused by opening the module interlock circuit. This is accomplished by connecting pin 3 directly to the +20 volt bus. The connection of pin 3 to the +20 volt bus means that the +20 volt reset voltage is applied continuously to the reset input on the bistable portion of the trip circuit. This will cause the bistable to reset as soon as the trip input circuit returns to the untripped condition.

2-15 <u>Downscale Level Trip Circuit (Z16-Trip B)</u>. Trip circuit Z16-Trip B is connected as a positive downscale trip circuit. The 0 to +10 Vdc output from the Log Count Rate (LCR) circuit is applied as an input at pin 31 and the reference voltage is connected as the other input at pin 32. When the voltage at pin 31 becomes less positive than the reference voltage, the trip circuit trips and the following outputs occur: the differential voltage between pins 19 and 36 decreases from 20 volts to less than 3 volts; the output voltage at pin 34 decreases from 12 volts to less than 1 volt; and the output at pin 37 decreases from 20 volts to less than 3 volts, causing DS16B to light.

2-16 Trip circuit Z16-Trip B is wired for automatic reset by connecting pin 17 directly to the 20 volt bus. Thus, the trip circuit will automatically return to the untripped configuration when the dc amplifier output becomes more positive than the reference voltage.

2-17 Downscale Level Trip Circuit (Z17-Trip C). Trip circuit Z17-Trip C is connected as a positive downscale trip circuit. Trip circuit Z17-Trip C will trip when the input voltage at pin 26 (the output of LCR circuit) becomes less positive than the reference voltage applied at pin 25. When the trip circuit trips, the output at pin 23 decreases from 12 volts to less than 1 volt, the differential output across pins 19 and 21 decreases from 20 volts to less than 3 volts, and the output at pin 20 increases from less than 3 volts to 20 volts. In addition to these outputs, the voltage at pin 1 decreases from approximately 20 volts to less than 2 volts, causing DS17A to light.

2-18 Trip circuit Z17-Trip C is connected for automatic reset by connecting pin 3 to the 20 Vac bus. After the trip circuit is tripped, it will reset all the outputs to the untripped configuration as soon as the input at pin 26 becomes more positive than the reference voltage applied at pin 25.

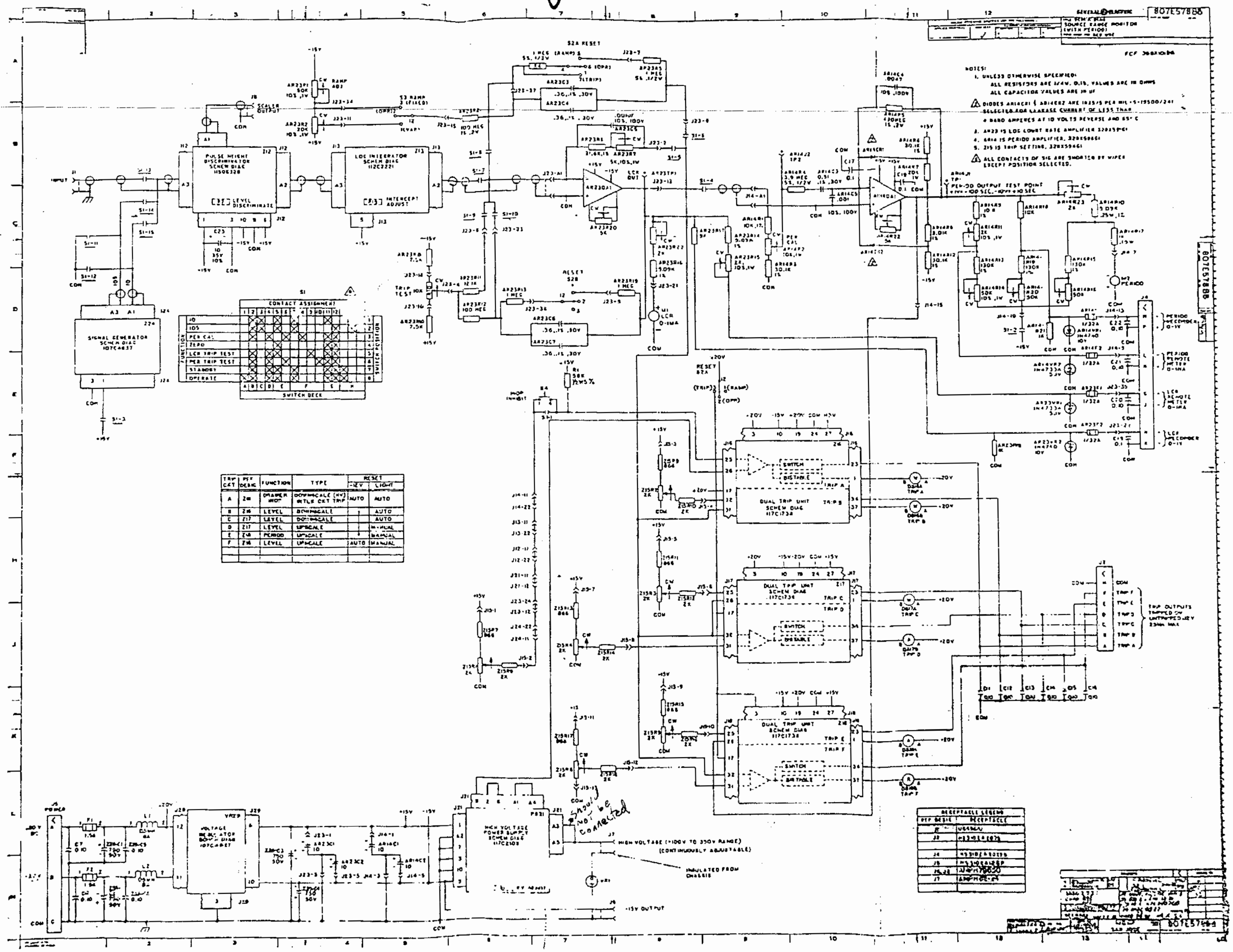
2-19 Upscale Level Trip (Z17-Trip D). Trip circuit Z17-Trip D is connected as a positive upscale trip. When the voltage applied at pin 32 (the output of the LCR circuit) becomes more positive than the reference voltage applied to pin 31, the trip circuit trips. When the trip circuit trips, the differential voltage between pins 19 and 36 decreases from 20 volts to less than 3 volts, the output voltage at pin 34 decreases from 12 volts to less than 1 volt, and the output voltage at pin 37 decreases from 20 volts to less than 3 volts, causing DS17B to light.

2-20 Trip circuit Z17-Trip D is connected for manual reset by connecting pin 17 through the reset switch to the 20 Vdc bus. Consequently, the outputs will return to the untripped configuration as soon as the voltage at pin 32 becomes less positive than the reference voltage applied at pin 31 and the reset switch is actuated.

2-21 Upscale Period Trip (Z18-TRIP E). Trip circuit Z18-Trip E is connected as a negative upscale period trip. It is connected for manual reset by connecting pin 3 through the reset switch to the 20 Vdc bus. Trip circuit Z18-Trip E will trip when the input voltage (output of the period circuit) at pin 26 becomes less positive than the reference voltage applied at pin 25. When trip circuit Z18-Trip E trips, its outputs will be the same as the tripped outputs for Z17-Trip C.

2-22 Upscale Level Trip (Z18-Trip F). Trip circuit Z18-Trip F is connected as a positive upscale trip. When the voltage at pin 32 (the output from AR23) becomes more positive than the reference voltage applied at pin 31, the trip circuit trips. Trip circuit Z18-Trip F is connected for manual reset by connecting pin 17 through the reset switch to the 20 Vdc bus. When trip circuit Z18-Trip F trips its outputs will be identical to the tripped outputs of Z17-Trip D.

SRm Block Diagram



INSTRUCTION MANUAL

FOR

REGULATED POWER SUPPLIES

LRS-55 SERIES

8/95

TABLE OF CONTENTS

SPECIFICATIONS AND FEATURES

DC OUTPUT — Voltage regulated for line and load. See table I for voltage and current ratings.

TABLE I
VOLTAGE AND CURRENT RANGES

MODELS	VOLTAGE RANGE	MAXIMIM CURRENT (AMPS) AT AMBIENT TEMPERATURE				INPUT POWER*
		40°C	50°C	60°C	71°C	
LRS-55-2	2 ± 5%	60A	51A	41A	30A	210W.
LRS-55-5	5 ± 5%	60A	51A	41A	30A	420W.
LRS-55-6	6 ± 5%	52	44	36	26	437W.
LRS-55-12	12 ± 5%	30	26	22	16	491W.
LRS-55-15	15 ± 5%	25	22	19	13	511W.
LRS-55-20	20 ± 5%	19	16.5	14	10	500W.
LRS-55-24	24 ± 5%	16	14	12	8	505W.
LRS-55-28	28 ± 5%	14	12	10	7	515W.
LRS-55-48	48 ± 5%	8.2	7.2	6.2	4.2	517W.

*With output loaded to full current rating and input voltage at 95 V AC.

Current range must be chosen to suit the appropriate maximum ambient temperature. Current ratings apply for entire voltage range.

REGULATED VOLTAGE OUTPUT

Regulation. . . . . .0.1% line or load with input variations from 95-132 or 132-95 volts AC and load variations from no load to full load.

Ripple and Noise. . . . . .10mV RMS, 35m V peak-to-peak for LRS-55-2, 5 and 6 volt models; 15mV RMS. 100m V peak-to-peak for LRS-55-12 thru LRS-55-28; 35m V RMS, 150m V p-p for LRS-55-48, with either positive or negative terminal grounded.

Temperature Coefficient . . . . .Change in output voltage 0.03%/°C max.

Remote Programming

External Resistor. . . . . .Nominal 1000 ohms/volt output.

Programming Voltage . . . . .One-to-one voltage change.

Remote Sensing. . . . . Provision is made for remote sensing to eliminate effect of power output lead resistance on DC regulation.

Remote On/Off Control . . . . The DC output may be turned on and off remotely by means of digital command. (See page 6).

HOLD-UP Time. . . . . 2V, 5V and 6V models will remain within regulation limits for at least 16.7 msec, after loss of AC power, when operating at full load, maximum output voltage and 105 VAC input at 60 Hz.

OVERSHOOT — No overshoot under conditions of power turn-on, turn-off, or power failure.

INPUT

AC Input. . . . . 95-132 VAC at 47-440 Hz. Input power: 517W max*. Input Current: 12.0A max*. For operation at frequencies above 440 Hz, consult factory. Where applicable, regulatory agency approval applies only for input frequencies in the range 47-63 Hz. Leakage current in the ground connection may exceed the limits allowed by these agencies at frequencies above this range.

*With output loaded to full current rating and input voltage 95 VAC, 60 Hz.

DC Input. . . . . 145 VDC ±10%. Input current: 3.8† Amperes.

†With output loaded to full current rating and input voltage 130 VDC.

SOFT START — The turn-on inrush current will not exceed 40 amps.

INPUT FUSE — Fuse F1 protects the input wiring to the power supply. Overload of power supply does not cause fuse failure.

OVERLOAD PROTECTION

Thermal. . . . . Thermostat resets automatically when over-temperature condition is eliminated.

Electrical. . . . . Automatic electronic current limiting circuit, limits output current to a preset safe value. Automatic current limiting protects the load and power supply when external overloads and direct shorts occur.

OVERVOLTAGE PROTECTION — All LRS-55 models include fixed non-crowbar built-in overvoltage protection circuits which prevent damage to the load caused by excessive power supply output voltage. See table on the following page for overvoltage protector firing ranges.

MODEL	OVERVOLTAGE PROTECTOR MAXIMUM TRIP POINT
LRS-55-2	4.2V
LRS-55-5	6.8V
LRS-55-6	7.7V
LRS-55-12	14.9V
LRS-55-15	19.0V
LRS-55-20	26.1V
LRS-55-24	31.2V
LRS-55-28	36.3V
LRS-55-48	59.8V

INPUT CONNECTIONS — Terminal block on front of chassis.

OUTPUT CONNECTIONS — 10/32 Threaded bus bars.

OPERATING AMBIENT TEMPERATURE RANGE AND DUTY CYCLE — Continuous duty from -10°C to 71°C ambient with corresponding load current ratings for all modes of operation.

STORAGE TEMPERATURE (non-operating) — -55°C to 85°C

FUNGUS — All LRS-55 power supplies are fungus inert.

VDC ADJ. Control — 10-Turn voltage adjust control permits adjustment of DC output over entire operating range.

PHYSICAL DATA

Size .... 3-3/4" x 4-7/8" x 10-1/2"
Weight .... 8 1/2 lbs. net; 10 lbs. shipping
Finish .... Grey, FED. STD. 595 No. 26081

MOUNTING — One mounting surface and one mounting position.

ACCESSORIES -

Rack adapters .... Rack adapters LRA-15, and LRA-17, used for ruggedized mounting with or without chassis slides, are available.

THEORY OF OPERATION

GENERAL

The LRS-55 Series consists of the following circuit elements:

Input Circuit

Pulse Width Modulated Inverter

Auxiliary Rectifier And Filter

Output Rectifier And Filter

Regulator Control Integrated Circuit; consisting of a timing oscillator, pulse width modulator, output drive, undervoltage detect and shutdown, current limit and frequency shift.

FUNCTIONAL DESCRIPTION

Input Circuit

Since phase AC input power is applied to diode bridge circuit CR201-CR204 through EMI suppression chokes L201 and L202. Fuse F201 protects the power supply against excessive currents due to internal failures.

The input bridge and filter capacitors C201, C202 provide power to the power MOSFET switch and provide power via R101, C101 and CR101 to the regulator control integrated circuit, IC101. At turn-on C201, C202 charges through inrush current limiting resistor R203. When the AC input reaches approximately 95 VAC, the U.V. detect circuit will enable control IC101 to supply pulse width modulated drive to the power MOSFET switch, Q101. As the power MOSFET switch begins to conduct, the voltage on gate capacitor C218 will rise until SCR201 conducts and shorts out R203 thus completing the inrush current limiting cycle.

Pulse Width Modulated Inverter

The inverter is a single-ended configuration using a power MOSFET switch, Q101. During the "on" time, a positive pulse is delivered to the gate of the power switch from integrated circuit controller IC101, pin 8, via C115. A positive pulse turns Q101 "on". During "off" time pin 8 shorts Q101 gate to C101(-) and turns Q101 off. Switching action of Q101 delivers positive and negative pulses to T1 primary.

Output Rectifier And Filter

The transistor "on" time primary pulse produces a positive secondary pulse which causes CR1A to conduct and deliver current to the load. Because output current is returned to T1 secondary through filter choke L1, energy is stored in L1 during Q101 "on" time. The Q101 "off" time primary pulse produces a negative secondary pulse which causes CR1A to turn off and CR1B to turn on. Energy stored in L1 is then delivered to the load through CR1B during Q101 "off" time. Output ripple attenuation is accomplished via the dual LC filter L1, L205, C208 and C209. Capacitors C206, C207, C203, C204, C205 and C220 furnish additional output noise suppression.

Regulator Control IC101

Timing oscillator section

IC101 pins 11, 12 and 13, R110 and R111 work in conjunction with C106 to set the main oscillator frequency, 50 khz, and the maximum duty ratio; $T_{on}/(T_{on} + T_{off})$. R110, C106 set maximum "on" time T_{on}; R111, C106 set maximum "off" time T_{off}.

Pulse width modulator section

Pulse width modulation, necessary to control the output voltage, is determined by the feed-forward comparator. Changes in the output are sensed and divided down by R128 and R129. This signal is fed into the reference of IC103. IC103 controls the current through optical coupler diode OC101, which is optically coupled to the base of OC101 internal transistor. Collector voltage of OC101 transistor is fed into IC101 pin 18. This voltage, when compared to a ramp signal developed by C105 by R109 and R135, sets the pulse width of the output drive through IC101 pin 8, as required to maintain the desired output voltage. The ramp on C105 is automatically reset each switching cycle by IC101. The slope of the ramp is proportional to the voltage across input capacitors C201 and C202.

Undervoltage detect; shutdown

Low input line detect comparator, IC101 pins 3 and 4, monitors changes in the input line. If the input line falls below approximately 95 VAC, the output drive will be inhibited.

At power up, C104, connected at IC101 pin 4, provides a delay before inverter operation to eliminate any secondary input inrush currents. At power down, C104 provides a delay at the undervoltage circuit to allow for hold-up time, where required.

Current limit and frequency shift

Current limit comparator IC101, pins 16 and 17, provide pulse termination current limit. When output current exceeds the preset safe value, the current-limit comparator disables the output drive, which remains disabled until the end of the switching cycle.

In order to maintain the foldback characteristic when both output voltage and current are decreasing, the comparator will parallel C107 with timer capacitor C106 increasing the discharge-time of C106. This increased discharge-time constant causes the operating frequency to decrease allowing a decreasing output current as the output voltage decreases. All LRS-55 units are designed with a current limit characteristic which presents constant current limiting from nominal output down to approximately 70% of Vo(nom). They then foldback to a current of approximately 70% full load at short circuit.

OPERATING INSTRUCTIONS

BASIC MODE OF OPERATION

This power supply operates as a constant voltage source provided the load current does not exceed the rated value at 40°C. For continuous operation, load current must not exceed the rating for each ambient temperature. When rated load current is exceeded, voltage decreases towards zero and the current at short circuit is held to a safe value.

CONNECTIONS FOR OPERATION

NOTE: Make all connections to the unit before applying input power.

Ground Connections. The supply can be operated either with negative or positive output terminal grounded. Both positive and negative ground connections are shown in the diagrams for all suggested output connections illustrated in this manual.

Connection Terminals. Make all connections to the supply at the terminal blocks and bus bars on the front of the supply. Apply input power to terminals 1 and 2; always connect the ungrounded (hot) AC or positive DC lead to terminal 1.

The supply positive terminal is brought out to terminal 6. The supply negative terminal is brought out to terminal 4. Recommended wiring of the power supply to the load and selection of wiring is shown in figures 1 through 10. Selection of proper wiring is made on the basis of load requirements. Make all performance checks and measurements of current or voltage at the front output terminals. Connect measuring devices directly to terminals or use the shortest leads possible.

Remote Turn-on/Turn-off. Make connections to Isolated pins 8 (+) and 9 (-), see figure 8.

Turn-off: Apply 2.8 → 5V (TTL "1") at terminals 8 and 9 observing polarity shown. There is an internal 1-KΩ limiting resistor in series with pin 8, causing approximately 2.8 → 5ma to be drawn from the TTL source.

Turn on: Apply 0 → 0.4V (TTL "0") or open circuit at terminals 8 and 9.

SUPPLY LOAD CONNECTIONS

Connections for Operation as a Constant Voltage Source

The output impedance and regulation of the power supply at the load may change when using the supply as a constant voltage source and connecting leads of practical length are used. To minimize the effect of the output leads on these characteristics, remote sensing is used. Recommended types of supply-load connections with local or remote sensing are described in the following paragraphs.

Refer to figure 1 to determine voltage drop for particular cable length, wire size and current conditions. Lead lengths must be measured from supply terminals to load terminals as shown in figure 2.

Local Sensing Connection, Figure 3. Local sensing is the connection suitable for applications with relatively constant load or for applications with short power output leads.

Remote Sensing Connection, Figure 4. Remote sensing provides complete compensation for the DC voltage drops in the connecting cables. Sensing leads should be a twisted pair to minimize AC pick-up. A 2.5 mf elect., capacitor may be required between output terminals and sense terminals to reduce noise pick-up.

Programmed Voltage Connections, Using External Resistor, Figure 5. Discrete voltage steps can be programmed with a resistance voltage divider valued at a nominal 1000 ohms/volt change and a shorting-type switch as shown in figure 5. When continuous voltage variations are required, use a variable resistor with the same 1000 ohms/volt ratio in place of the resistive voltage divider and shorting-type switch. Use a low temperature coefficient resistor to assure most stable operation.

Before programming, adjust programming resistor for zero resistance and set voltage adjust control to the minimum rated output voltage. Output voltage of programmed supply will nominally be minimum output voltage plus 1 volt per 1000 ohms.

As shown in figure 5, voltages can be programmed utilizing either local or remote sensing connections, as desired.

Programmed Voltage Connections Using Programming Voltage, Figure 6. The power supply voltage output can be programmed with an externally connected programming power supply. The output voltage change of the programmed supply will maintain a one-to-one ratio with the voltage of the programming supply. If the output voltage control of the programmed supply is set to minimum output voltage, output voltage of the programmed supply will be minimum output voltage plus voltage of programming supply.

The programming supply must have a reverse current capability of 1 ma. minimum.

Alternatively, when supplies with less than 1 ma. reverse current capability are used, a resistor capable of drawing 1 ma. at the minimum programming voltage must be connected across the output terminals of the programming supply. This programming supply must be rated to handle all excess resistor current at the maximum programming voltage.

Connections For Series Operation, Figure 7.

The voltage capability of LRS-55 power supplies can be extended by series operation. Figure 7 shows the connections for either local or remote sensing in a series connection where the voltage control of each unit functions independently to control the output.

A diode, having a current carrying capability equal to or greater than the maximum current rating of the supply, must be used and connected as shown in figure 7. The diode blocking voltage should be at least twice the maximum rated output voltage of the supply. See table I, of SPECIFICATIONS AND FEATURES, for power supply current and voltage ratings.

OPERATION AFTER PROTECTIVE DEVICE SHUTDOWN

Thermostat Shutdown. The thermostat shuts down inverter operation only when the temperature of the MOSFET switch heat radiator exceeds a maximum safety value. The thermostat will automatically reset when the temperature of the heat sink decreases to a safe operating value.

Fuse Shutdown. Fuse will blow when the maximum rated current value for the fuse is exceeded. Fatigue failure of fuses can occur when mechanical vibrations from the installation combine with thermally induced stresses to weaken the fuse metal. Many fuse failures are caused by a temporary condition and replacing the blown fuse will make the fuse protected circuit operative.

Overvoltage Shutdown. When the power supply output voltage increases above the overvoltage limit, the non-crowbar overvoltage circuit will shut down inverter operation. After eliminating the cause(s) for overvoltage, resume operation of the supply by interrupting the AC input circuit for a period of sixty seconds.

MAINTENANCE

GENERAL

This section describes calibration and test procedures that are useful for servicing the Lambda power supply. If the P.C. boards are removed from the chassis, the silicone rubber insulators must be replaced with new insulators: pt. nos. HSD-00-456, HSD-00-458 and HSD-00-464.

OPENING UNIT FOR TROUBLESHOOTING AND REPAIR

Whenever it is necessary to troubleshoot and repair the Lambda LRS-55 power supply, open the unit as follows:

1. Separate upper and lower enclosures by removing (5) 6-32 screws, see outline drawing, figure 10.

2. With unit resting on its 3.75" x 10.5" mounting surface, allow lower enclosure to drop down with PC board component side facing upward.

3. Power supply components are now accessible for troubleshooting and repair.

TROUBLE ANALYSIS

Whenever trouble occurs, systematically check all fuses, primary power lines, external circuit elements, and external wiring for malfunction before troubleshooting the equipment. Failures and malfunctions often can be traced to simple causes such as improper jumpers and supply-load connections or fuse failure due to metal fatigue.

Use the electrical schematic diagram and block diagram, figure 9, as an aid to locating trouble causes. The schematic diagram contains various circuit voltages that are averages for normal no load operation. Measure these voltages using the conditions for measurement specified on the schematic diagram. Use measuring probes carefully to avoid causing short circuits and damaging circuit components.

CHECKING CAPACITORS

The leakage resistance obtained from a simple resistance check of a capacitor is not always an indication of a faulty capacitor. In all cases the capacitors are shunted with resistances, some of which have low values. Only a dead short is a true indication of a shorted capacitor.

PRINTED CIRCUIT BOARD MAINTENANCE TECHNIQUES

1. If foil is intact, but not covered with solder, it is a good contact. Do not attempt to cover with solder.

2. Voltage measurements can be made from either side of the board. Use a needle-point probe to penetrate to the wiring whenever a protective coating is used on the wiring. A brass probe can be soldered to an alligator clip adapted to measuring instrument.

3. Always use a heat sink when soldering transistors; a transistor pad with mounting feet is an effective heat sink.

4. Broken or damaged printed wiring is usually the result of an imperfection, strain or careless soldering. To repair small breaks, tin a short piece of hook-up wire to bridge the break, and holding the wire in place, flow solder along the length of wire so that it becomes part of the circuitry.

5. When unsoldering components from the board never pry or force loose the part; unsolder the component by using the wicking process described below:

   a) Select a 3/16 inch tinned copper braid for use as a wick; if braid is not available, select AWG No. 14 or No. 16 stranded wire with 1/2 inch insulation removed.

   b) Dip the wick in liquid rosin flux.

   c) Place the wick onto the soldered connection and apply soldering iron onto the wick.

   d) When sufficient amount of solder flows onto the wick, freeing the component, simultaneously remove iron and wick.

PERFORMANCE CHECKS

Check the ripple and regulation of the power supply using the test connection diagram shown in figure 10. Use suggested test equipment or equivalent to obtain accurate results. Refer to SPECIFICATIONS AND FEATURES for minimum performance standards.

Set the differential meter, DC DVM (John Fluke Model 891A or equivalent) to the selected power supply operating voltage. Check the power supply load regulation accuracy while switching from the no load to load condition. Long load leads should be a twisted pair to minimize AC pick-up.

Use a Variac to vary the line voltage from 95-132 or 132-95 volts AC and check the power supply line regulation accuracy on the DVM differential meter.

Use a TVM, John Fluke Model 931B or equivalent, to measure rms ripple voltage of the power supply DC output. Use oscilloscope to measure peak-to-peak ripple voltage of the power supply DC output.

ADJUSTMENT OF CURRENT LIMIT CALIBRATION CONTROL R116

Whenever IC101, T101, CR104 or R117 are replaced, and voltage and current indications do not reflect maximum ratings, adjust R116 as follows. The adjustment procedure requires that the power supply is removed from associated equipment, is at an ambient temperature of 25-30°C, and is stabilized and not operating.

1. Remove AC power to the supply.

2. Break seal on wiper of R116 from resistor housing and turn to midrange position.

3. Operate power supply for constant voltage with local sensing connected as shown in figure 3, with no external load, and with AC input of 132 VAC, 60 Hz.

4. Turn voltage adjust control until maximum rated output voltage is obtained.

5. Apply load so that output current is 105% of 40°C rating for the unit. If output voltage drops, Adjust R116 to allow V_{out} max. at 105% of 40°C rating.

6. Using an oscilloscope, Tektronix 503 or equivalent, observe output voltage while adjusting R116 in CCW direction. Adjust R116 until output ripple increases sharply and output voltage begins to decrease.

7. After adjustment is completed, remove AC power input to the supply and use glyptol sealant to seal wiper of R116 to resistor housing.

8. After sealing, check setting and repeat adjustment procedure if required.

ADJUSTMENT OF FEED FORWARD CONTROL R135

Whenever IC101 is replaced it may be necessary to readjust R135 as follows. The adjustment procedure requires that the power supply is removed from associated equipment, is at an ambient temperature of 25-30°C, and is stabilized and not operating.

1. Remove AC input power to the supply.

2. Break seal on wiper of R135 from resistor housing and turn to midrange position.

3. Adjust R116 fully CW.

4. Operate power supply for constant voltage with local sense connected as shown in figure 3. Apply external resistive load equal to the 40°C current rating, with voltage input of 115 VAC at 60 Hz.

5. Remove AC input power to the supply and open the positive sense connection. While monitoring output voltage, reapply AC input power.

6. Adjust R135 until output voltage reaches 110% of the maximum output voltage rating. To increase output voltage turn R135 CW; to decrease output voltage turn R135 CCW.

7. After adjustment is completed, remove AC power input to the supply and use glyptol sealant to seal wiper of R135 to resistor housing. Reconnect positive sense connection.

8. Readjust R116; refer to ADJUSTMENT OF CALIBRATION CONTROL R116.

SERVICE

When additional instructions are required or repair service is desired, contact the nearest Lambda office where trained personnel and complete facilities are ready to assist you.

Please include the power supply model and serial number together with complete details of the problem. On receipt of this information, Lambda will supply service data or advise shipping for factory repair service.

All repairs not covered by the warranty will be billed at cost and an estimate forwarded for approval before work is started.

PARTS ORDERING

Standard components and special components used in the Lambda power supply can be obtained from the factory. In case of emergency, critical spare parts are available through any Lambda office.

The following information must be included when ordering parts:

1. Model number and serial number of power supply and purchase date.
2. Lambda part number.
3. Description of part together with circuit designation.
4. If part is not electronic part, or is not listed, provide a description, function, and location, of the part.

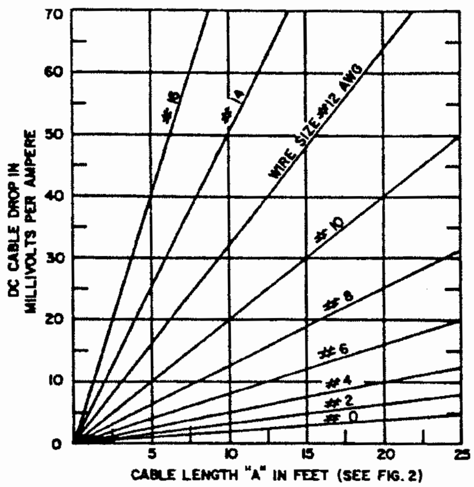


Figure 1. Cable Connection Chart.

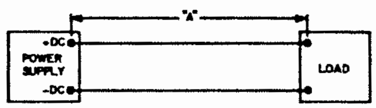
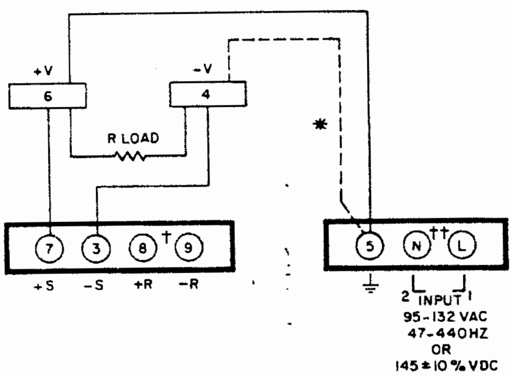


Figure 2. Cable Length "A" in Feet.

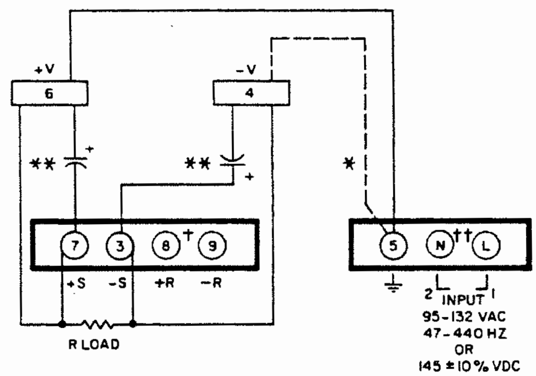


NOTE

* FOR NEGATIVE GROUND, DISCONNECT JUMPER FROM TERMINALS 5 AND 6 AND RECONNECT TO TERMINALS 5 AND 4.

† REMOTE ON/OFF TERMINALS; REFER TO FIGURE 8 FOR PROPER CONNECTION.

†† ALWAYS CONNECT UNGROUNDED (HOT) LEAD TO TERMINAL "L".



NOTE

* FOR NEGATIVE GROUND, DISCONNECT JUMPER FROM TERMINALS 5 AND 6 AND RECONNECT TO TERMINALS 5 AND 4.

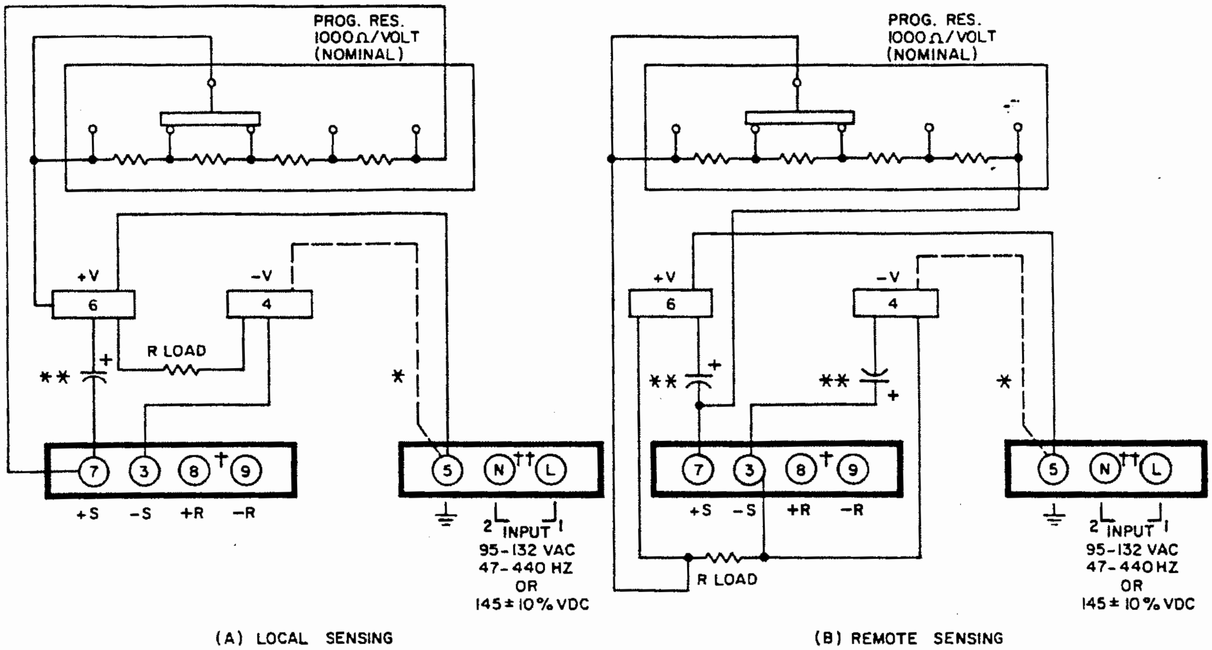
** A 2.5 MF, ELECT., CAP. MAY BE REQUIRED.

† REMOTE ON/OFF TERMINALS; REFER TO FIGURE 8 FOR PROPER CONNECTION.

†† ALWAYS CONNECT UNGROUNDED (HOT) LEAD TO TERMINAL "L".

Figure 3. Local Sensing Connections.

Figure 4. Remote Sensing Connections.



NOTES:

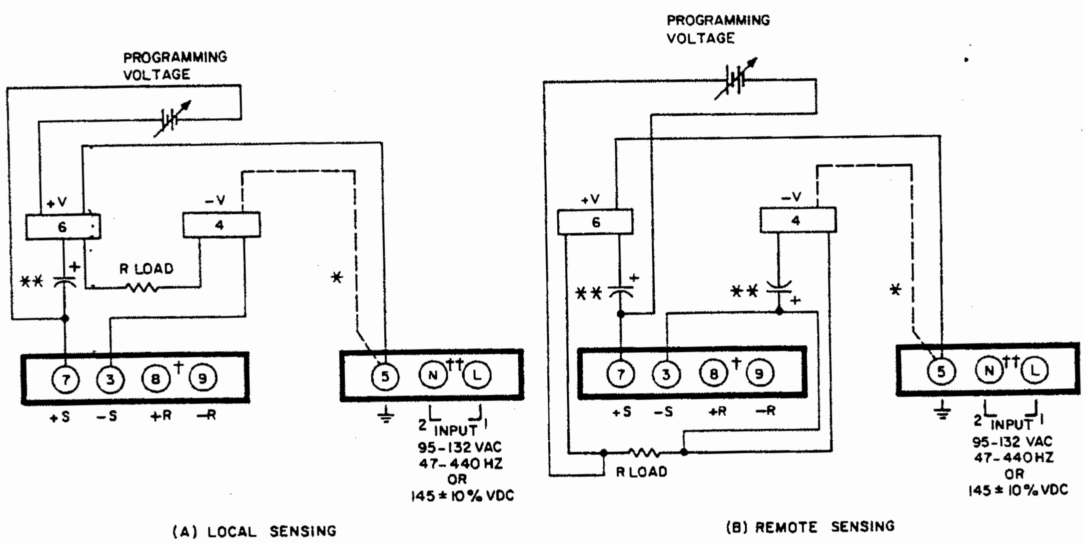
* FOR NEGATIVE GROUND, DISCONNECT JUMPER FROM TERMINALS 5 AND 6 AND RECONNECT TO TERMINALS 5 AND 4.

** A 2.5 MF, ELECT., CAP. MAY BE REQUIRED.

† REMOTE ON/OFF TERMINALS; REFER TO FIGURE 8 FOR PROPER CONNECTION.

†† ALWAYS CONNECT UNGROUNDED (HOT) LEAD TO TERMINAL "L".

Figure 5. Programmed Voltage, With External Resistor.



NOTES:

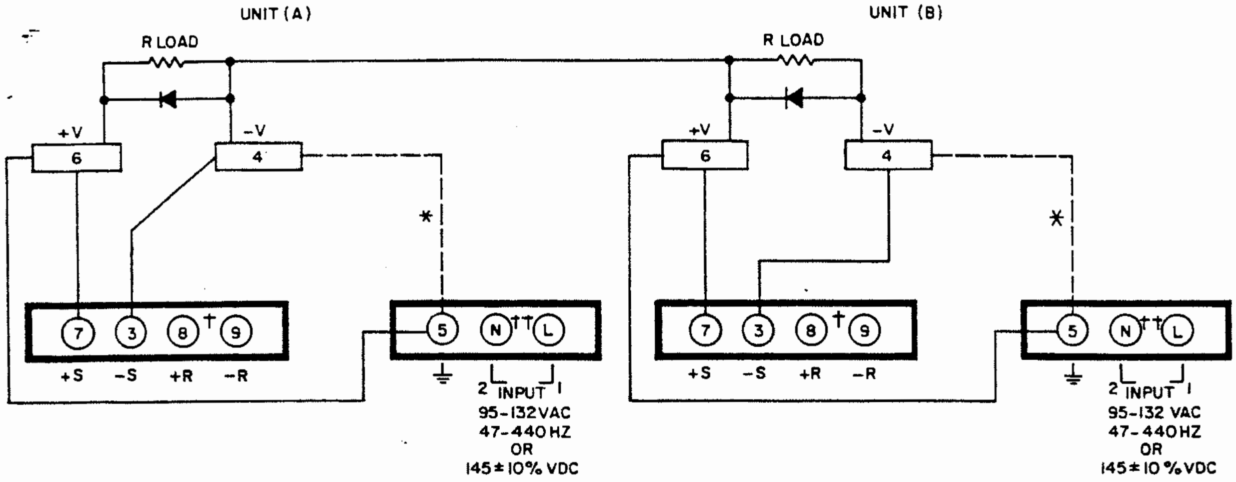
* FOR NEGATIVE GROUND, DISCONNECT JUMPER FROM TERMINALS 5 AND 6 AND RECONNECT TO TERMINALS 5 AND 4.

** A 2.5 MF, ELECT., CAP. MAY BE REQUIRED.

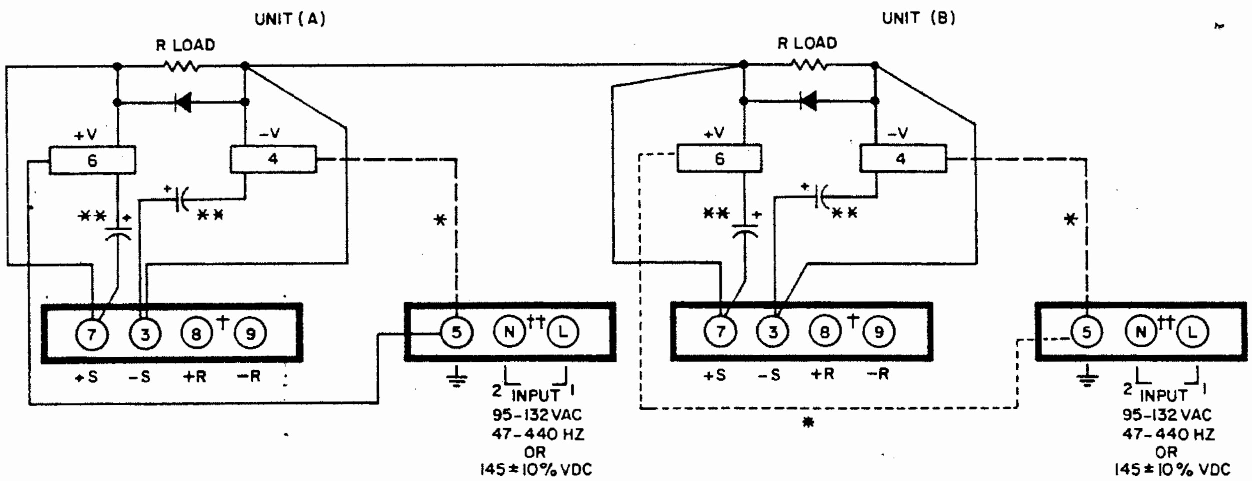
† REMOTE ON/OFF TERMINALS; REFER TO FIGURE 8 FOR PROPER CONNECTION.

†† ALWAYS CONNECT UNGROUNDED (HOT) LEAD TO TERMINAL "L".

Figure 6. Programmed Voltage, With External Programming Voltage Source.



(A) LOCAL SENSING



(B) REMOTE SENSING

NOTES:

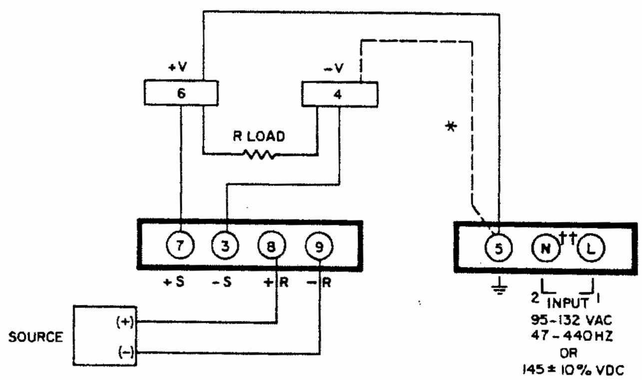
* MAKE ONLY ONE GROUND CONNECTION FOR SERIES COMBINATION. TO CHANGE GROUND AS SHOWN, REMOVE JUMPER FROM TERMINALS 5 AND 6 ON UNIT (A) AND CONNECT ANY ONE OF THE OTHER JUMPERS AS SHOWN IN DOTTED LINE.

** A 2.5 MF, ELECT., CAP. MAY BE REQUIRED.

† REMOTE ON/OFF TERMINALS, REFER TO FIGURE 8 FOR PROPER CONNECTION.

†† ALWAYS CONNECT UNGROUNDED (HOT) LEAD TO TERMINAL "L".

Figure 7. Series Connection.



* FOR NEGATIVE GROUND, DISCONNECT JUMPER FROM TERMINALS 5 AND 6 AND RECONNECT TO TERMINALS 5 AND 4.

†† ALWAYS CONNECT UNGROUNDED (HOT) LEAD TO TERMINAL "L".

Figure 8. Remote Turn-on/Turn-off.

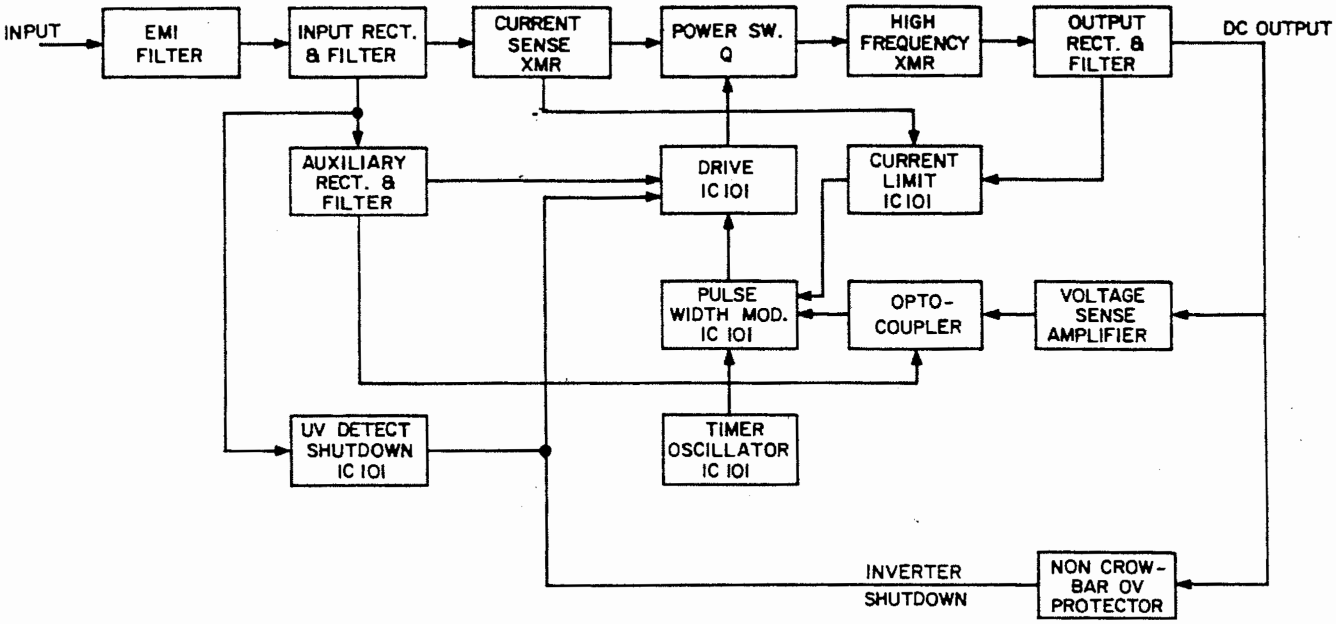
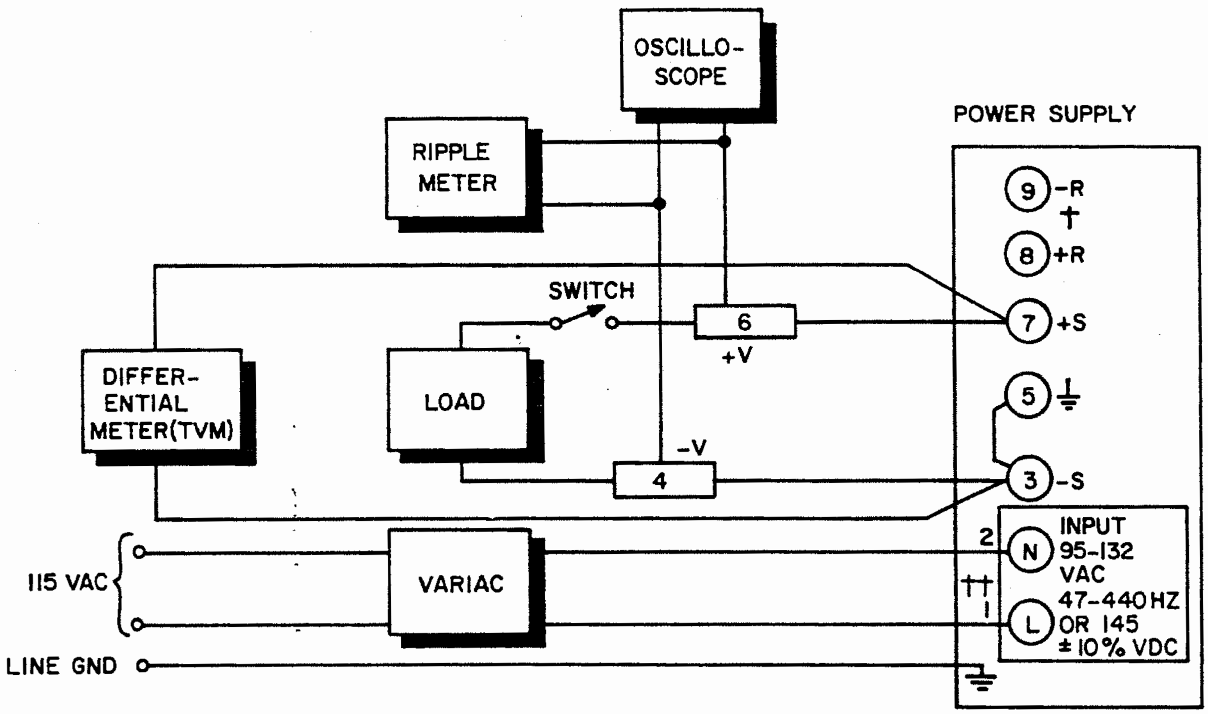


Figure 9. Typical Block Diagram.



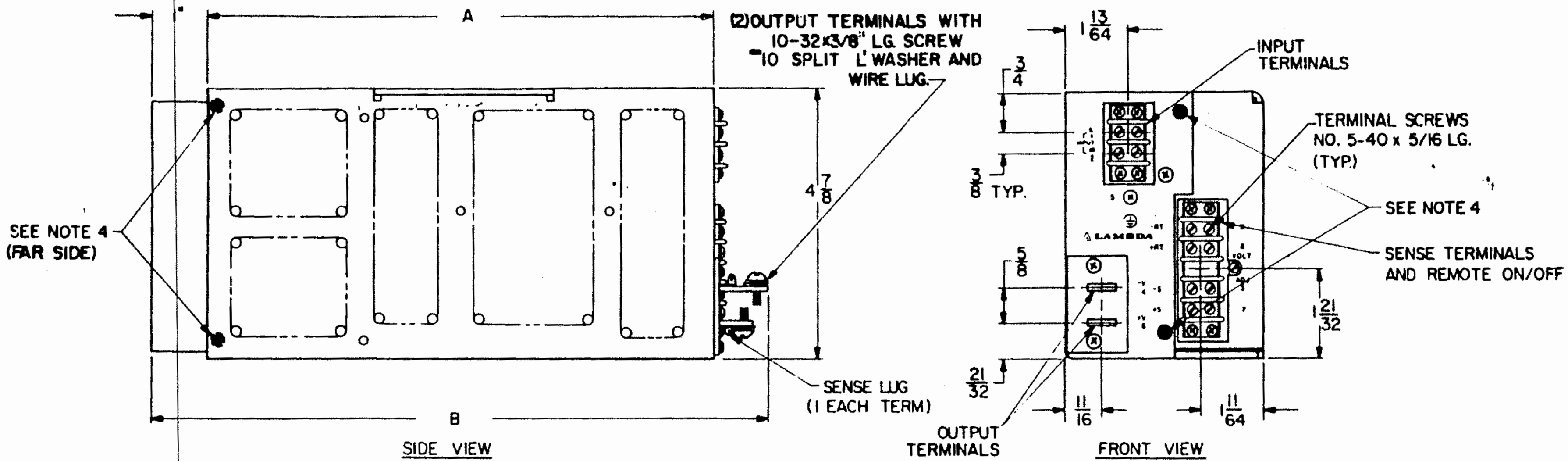
NOTES:

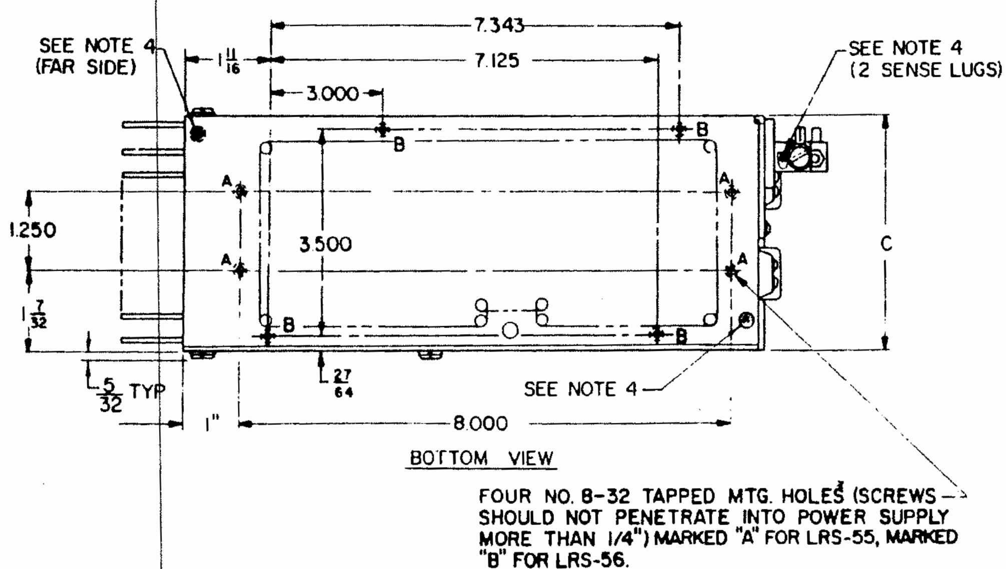
1. REGULATION AND RIPPLE CHECK METERS MUST NOT BE GROUNDED THROUGH THREE WIRE LINE CORD TO GROUND.
2. PERFORM CHECKS WITH LOCAL SENSING CONNECTION ONLY.

† REMOTE ON/OFF TERMINALS; REFER TO FIGURE 8 FOR PROPER CONNECTION.

†† ALWAYS CONNECT UNGROUNDED (HOT) LEAD TO TERMINAL "L".

Figure 10. Test Connections for Performance Checks.





MODEL	A	B	C
LRS-55	9 1/2	11 7/8	3 3/4
LRS-56	10 1/2	12 7/8	4 11/32

NOTES:
1. CUSTOMER MUST PROVIDE CLEARANCE IN HIS MOUNTING SURFACE FOR VENT HOLES TO ALLOW FOR AIR CIRCULATION.
2. UNIT HAS ONE MOUNTING SURFACE.
3. RADIATOR FINS MUST BE MOUNTED IN A VERTICAL POSITION.
4. TO OPEN UNIT FOR SERVICING, REMOVE (8) 6-32 SCREWS SHOWN SHADED.

Figure 11. Outline Drawing.

NOTES

7.5 INFORMATION SYSTEMS IMPORTANT TO SAFETY

7.5.1 DESCRIPTION

7.5.1.1 General

This section describes the instrumentation that provides information that enables the operator to monitor transient reactor plant behavior to verify proper safety system performance following an accident and to perform required safety functions.

The SRDI is listed in Table 7.5-1. This table tabulates equipment illustrated in the various system P&IDs, IEDs, and FCDs referenced in Sections 7.2, 7.3, 7.4, and 7.6.

The instrumentation and ranges shown in Table 7.5-1 are selected based on their ability to provide the reactor operator with the information needed to perform normal plant maneuvers, and their capability to track process variables pertinent to safety during expected operational perturbations.

Table 7.5-3 identifies Regulatory Guide 1.97 variables applicable to LGS. Table 7.5-3 lists the instrumentation from Table 7.5-1 that provides indication of Regulatory Guide 1.97 variables.

Table 7.5-2 summarizes the LGS design and qualification criteria for the Regulatory Guide 1.97 display instrumentation and systems. Conformance with Regulatory Guide 1.97 is discussed in Section 7.5.2.5.1.1.2. Sections 7.5.1.4.2 and 7.5.1.4.3 describe the display instrumentation for indication of Regulatory Guide 1.97 variables. References are made to other sections for descriptions of the instrumentation systems.

The separation of redundant display instrumentation and electrical isolation of redundant sensors and channels is shown in elementary diagrams and in Table 7.1-6. The P&IDs, IEDs, FCDs, and elementary diagrams adequately illustrate the redundancy of monitored variables and component sensors and channels.

The arrangement of the control room and the auxiliary equipment room is shown in drawings M-602 and M-603, respectively.

7.5.1.2 Normal Operation

The information channel ranges and indicators are selected on the basis of giving the reactor operator the information needed to perform all normal plant startup, steady-state maneuvers, and to monitor all the process variables pertinent to safety during expected operational perturbations.

7.5.1.3 Transient Occurrences

The ranges of indicators and recorders provided are capable of covering the extremes of process variables and providing adequate information for all transient events.

7.5.1.4 Accident Conditions

Information readouts are designed to accommodate all credible accidents from the standpoint of operator action, information, and event monitoring requirements.

7.5.1.4.1 Initial Accident Event

The design basis of all ESFs is to mitigate the consequences of an accident without operator action or assistance for the first 10 minutes of the event. This requirement, therefore, makes it mandatory that all protective action necessary in the first 10 minutes be automatic. Therefore, although continuous monitoring of process variables is available, no operator action based on them is required.

7.5.1.4.2 Postaccident Monitoring

No operator action is required for at least 10 minutes following an accident, although the various monitoring devices are continuously tracking and indicating important parameter information and displaying it to the operator as well as recording appropriate data.

The DBA-LOCA serves as the envelope accident sequence event to provide and demonstrate the plant's postaccident monitoring capabilities. All other accidents have less severe requirements.

7.5.1.4.2.1 Reactor Monitoring

The following process instrumentation which monitor reactor conditions provides information to the operator after a design basis LOCA.

7.5.1.4.2.1.1 Reactor Water Level

a. Two wide range water level signals are transmitted from two independent differential pressure transmitters and recorded on two recorders. One channel records the wide range level and the other channel records the reactor pressure as stated in Section 7.5.1.4.2.1.2. The differential pressure transmitters have one side connected to a condensing chamber reference leg and the other side connected directly to a vessel nozzle for the variable leg. The range of the recorded level is from the top of the feedwater control range (just above the high level turbine trip point) to a point near the top of the active fuel. Each independent instrumentation channel is powered from separate Class 1E buses.

b. The fuel zone water level signals are transmitted from two differential pressure transmitters. The level signals are electronically compensated for variation in reactor water and steam density with respect to pressure. One signal goes to a fuel zone water level indicator and the other water level signal goes to a fuel zone water level recorder. The differential pressure transmitters have one side connected to a condensing chamber for the reference leg and the other side connected directly to the bottom tap of a calibrated jet pump for the variable leg. The fuel zone level transmitters are calibrated for saturated conditions for a reactor pressure of 0 psig. The level range is from over the top of the active fuel to near the bottom of the active fuel. The ranges of the wide range level and the fuel zone level overlap. Power is fed from two independent Class 1E power sources.

c. A continuous backfill system is connected to each condensing chamber reference leg. The backfill system provides a continuous flow of water from the Control Rod Drive (CRD) System to the reference leg. This flow of water will continuously purge the reference leg and preclude the build-up of noncondensable gas in the reference leg.

7.5.1.4.2.1.2 Reactor Pressure

Two reactor pressure signals are transmitted from two independent pressure transmitters and are recorded on two recorders. One channel records pressure and the other channel records the wide range level. The range of recorded pressure is from 0-1500 psig. Power is fed from two independent Class 1E power sources.

7.5.1.4.2.1.3 Primary Containment Pressure

Wide range primary containment signals are transmitted from two pressure transmitters. One signal is displayed in a recorder in the control room while the other signal is displayed on an indicator located in the control room. The range of both instruments is from -5 psig to 165 psig. Power is supplied from two independent Class 1E power sources.

One narrow range primary containment signal is transmitted from a pressure transmitter and is indicated in the control room. The range of the indicated pressure is from -5 to +5 psig. Power is supplied from a Class 1E power source.

7.5.1.4.2.1.4 Primary Containment Gas Analyzers

Two redundant analyzer packages, each containing a hydrogen analyzer cell and an oxygen analyzer cell, monitor primary containment hydrogen and oxygen. These analyzers are part of the CAC system (Section 6.2.5.2.2). The hydrogen analyzer has a range of 0% to 30% (by volume) and the oxygen analyzer is a dual range device that measures the ranges of 0% to 10% and 0% to 25% (by volume). The calibrated range of each analyzer is 0% to 7%. The hydrogen and oxygen concentrations are indicated at the sample cabinet and in the control room.

Power is supplied from two independent Class 1E power sources.

7.5.1.4.2.1.5 Primary Containment Radiation Monitors

Four ion chamber sensors measure the gross radioactivity present in the containment atmosphere and transmit their signals to radiation recorders located in the control room. (Section 7.6.1.1.6). The range of recorded radiation is from $1\text{-}10^8$ rads per hour. Power is supplied from two independent Class 1E power sources.

7.5.1.4.2.1.6 Suppression Chamber Pressure

One suppression chamber pressure signal is transmitted from a pressure transmitter and is recorded in the control room. This signal is displayed on a pressure recorder located in the control room. The range of recorded pressure is from -5 psig to 165 psig.

Power is supplied from a Class 1E power source.

7.5.1.4.2.1.7 Suppression Pool Temperature

Two independent divisionalized recorders are located in the control room to monitor temperatures from 16 independent temperature sensors located in the suppression pool. Eight temperature sensors are dedicated to each recorder. Power is supplied from two independent Class 1E power sources. Each recorder has a digital display with which the operator can select the sensor to be displayed. Normally the display indicates the average temperature of the eight temperature sensor inputs. A control room alarm is generated when the average temperature increases to 95°F, 105°F, 110°F, and 120°F. In addition, an alarm will be provided when any temperature loop malfunctions.

The SPTMS is described in Appendix 3A.15.

7.5.1.4.2.1.8 Suppression Pool Water Level

Two suppression pool water level signals are transmitted from two independent level transmitters. Each signal is transmitted to an indicator, located in the control room.

Power is supplied from two independent Class 1E sources (Table 7.5-3).

7.5.1.4.2.2 Reactor Shutdown, Isolation, and Core Cooling Indication

7.5.1.4.2.2.1 Reactor Operator Information and Observations

The following information furnished to the control room operator permits him to assess reactor shutdown, isolation, and availability of emergency core cooling following the postulated accident.

a. Operator verification that reactor shutdown has occurred is made by observing the following indications:

    1. Control rod status lamps indicate each rod to be fully inserted; the power source is one of the instrument ac buses.

    2. Control rod scram pilot valve status lamps indicate open valves; the power source is an instrument ac bus.

    3. Neutron monitoring power range channels and recorders downscale; the power source is a dedicated UPS system.

    4. Annunciators for RPS variables and trip logic in the tripped state. The power source is dc from a plant battery. The function of the annunciators is to supply information to the operator. They are not protective systems as they do not provide trip signals.

    5. The plant process computer (Section 7.7) provides thermal-hydraulic information to the operator that is used to determine plant operating conditions.

b. The reactor operator verifies reactor isolation by observing one or more of the following indications:

    1. Isolation valve position lamps indicate valve closure; the power source is the same as for the associated motor, solenoid, operator and solenoid for AOVs.

2. Annunciators for the containment and reactor vessel isolation system variables, and trip logic in the tripped state. The power source is dc from the plant battery system.

c. Operation of the ECCS following the accident is verified by observing the following indicators:

   1. Annunciators for HPCI, core spray, RHR, and ADS, sensor initiation logic trips. The power source is dc from a plant battery.

   2. There are flow and pressure indications for each ECCS. The power sources are independent and are from the same Class 1E buses as the driven equipment.

   3. Injection valve position lights indicating either open or closed valves. The power source is the same as for the valve motor.

   4. Relief valve position lights indicate valve open or closed status through acoustical sensors located on piping directly downstream of the relief valves. Power is supplied from UPS.

   5. Relief valve discharge pipe temperature monitors. The power source is instrument ac from one of the instrument ac buses.

   6. Operation of the containment systems following the accident is verified by observing the following indications:

      (a) Annunciators for containment high oxygen and high hydrogen. The power source is dc from the plant battery system.

      (b) Indication and recording (by virtue of the non-safety related ERFDS/PMS computer) of containment oxygen and hydrogen concentrations (Table 7.5-1).

   7. Operation of the auxiliary supporting systems following the accident is verified by observing the following indicators:

      (a) Control room emergency ventilation: Recording and annunciation of radiation monitoring of the CREFAS intake and charcoal filter discharge duct (Table 7.5-1).

      (b) Service water systems: RHRSW system flow and temperature (Table 7.5-3).

      (c) SGTS, RERS, and REIS: Indication of flow for SGTS; indication of temperatures for SGTS and RERS charcoal filters; and reactor enclosure and refueling area differential pressure for REIS (Table 7.5-1).

7.5.1.4.2.2.2 <u>System Operation Information Display Equipment</u>

The following information furnished to the control room operator permits the operator to assess the operation of the safety-related systems:

a. HPCI: Three indicators, one displaying HPCI discharge flow rate, one displaying HPCI pump discharge pressure, and one displaying HPCI turbine steam pressure, are located in the control room.

b. CS: Two indicators each displaying CS flow rate for one of the two CS loops are located in the control room.

c. RHR: Four indicators, each displaying RHR flow rate for each loop, two indicators each displaying RHRSW flow rate for each RHR heat exchanger; two indicators each displaying RHR coolant outlet temperature for each RHR heat exchanger and two indicators, each displaying service water inlet temperature for each RHR heat exchanger, located in the control room.

d. RCIC: One indicator displaying RCIC flow rate, located in the control room.

e. ESW: Two indicators, each displaying ESW flow rate for each ESW loop and two indicators, each displaying ESW temperature for each ESW loop, located in the control room.

f. Drywell atmosphere temperature: One 2-pen recorder displaying drywell atmosphere temperature and suppression pool air space temperature, located in the control room.

g. Drywell sump level (floor drain and equipment drain tanks): A level indication for each tank, provided in the control room.

h. Equipment drain collection tank level: One indicator displaying high radioactive equipment drain collection tank level, located in the radwaste control room.

i. Floor drain collection tank level: One indicator displaying high radioactive floor drain collection tank level, located in the radwaste control room.

j. Chemical waste collection tank level: One indicator displaying high radioactive chemical waste tank level, located in the radwaste control room.

k. SLCS storage tank level: A CRT tank level readout for the SLCS storage tank, available in the control room through the ERFDS.

l. PCIG pressure: Two indicators, each displaying PCIG pressure, located in the control room.

m. CRD hydraulic charging water pressure: One indicator displaying CRD hydraulic charging pressure, located in the control room.

n. Reactor recirculation pump flow: One 2-pen recorder displaying reactor recirculation pump flow, located in the control room.

o. Deleted

p. Miscellaneous: In addition to the above displays, the following also provide information to enable the reactor operator in the control room to perform postaccident functions:

1. Control rod status lamps (Section 7.5.1.4.2.2.1.a.1)

2. Scram pilot valve status lamps (Section 7.5.1.4.2.2.1.a.2)

3. Neutron flux level meters (Section 7.5.1.4.2.2.1.a.3)

4. Main feedwater flow: Three indicators, each displaying feedwater flow for each feedwater line, located in the control room. Signals transmitted to these indicators are totalized and displayed as total main feedwater flow on a recorder located in the control room.

5. CST level: One recorder displaying CST level, located in the control room.

6. Condenser hotwell level: One recorder displaying condenser hotwell level, located in the control room.

7. For Unit 1, condenser pressure: condenser vacuum for each condenser shell is displayed on the HMI workstations located in the control room.

   For Unit 2, condenser pressure: Three indicators, each displaying condenser vacuum for each condenser shell, located in the control room.

8. For Unit 1, main steam bypass valve position: Each main steam bypass valve open and close position is displayed on HMI workstations in the control room.

   For Unit 2, main steam bypass valve position: A pair of indicating lights for each main steam bypass valve open and close position, provided in the control room.

9. Circulating water pump discharge pressure: Four indicators, each displaying discharge pressure for each circulating water pump, located in the control room.

7.5.1.4.2.2.3 System Operation Information Display Equipment Qualification

The safety-related reactor shutdown information equipment up to the display is of the same high quality as the safety system's instrumentation.

The postaccident monitoring instrumentation is of a quality that is consistent with minimum maintenance requirements and low failure rates. The safety-related equipment and indication equipment that have seismic and environmental qualifications are discussed in Sections 7.5.2.5.1.3.2 and 7.5.2.5.1.3.3.

7.5.1.4.2.3 Plant Electrical System

Display instrumentation for the status of the 4.16 kV safeguard and battery buses is provided in the control room. Each 4.16 kV safeguard bus is monitored for voltage and frequency, and each battery bus is monitored for voltage and amperage. Eight indicators (two indicators per bus) are provided in the control room to monitor each of the 4.16 kV safeguard buses for voltage and frequency. Eight indicators (two indicators per bus) are provided in the control room to monitor each of the battery buses for voltage and amperage.

7.5.1.4.2.4 Bypass Indication System

There are bypass indications through either administrative control or automatic systems level indications. These bypass capabilities are discussed independently for each safety system (Section 7.3).

7.5.1.4.3 Additional Instrumentation for Regulatory Guide 1.97 Variables

The following instrumentation provides postaccident indication in accordance with Regulatory Guide 1.97.

7.5.1.4.3.1 Radioactivity Concentration in Primary Coolant

Indication of radiation levels in primary coolant is provided by two methods are discussed in Section 7.5.2.5.1.1.2.4.4: (1) by means of the MSL-RMS and the AEO-RMS when the NSSS is not isolated; and (2) by means of the PASS when the NSSS is isolated.

a. MSL-RMS - Four redundant radiation detectors provide signals via rate meters to one 2-pen recorder in the control room. The 2-pen recorder allows the output of any two selected detector channels to be displayed. This system is described in Sections 7.6.1.1.1 and 11.5.2.1.1.

b. AEO-RMS - A radiation monitor designed to sense changes in the off- gas gross fission product concentration transmits signals to a recorder in the control room. This system is described in Section 11.5.2.2.8.

c. PASS - Grab samples of both the primary containment atmosphere and the reactor coolant are provided by the PASS. Analysis of these samples provides information for detection of a breach in the fuel cladding when the NSSS is isolated. This system is described in Section 11.5.5.

7.5.1.4.3.2 Radiation Exposure Rate in Areas Requiring Personnel Access Postaccident

As discussed in Section 7.5.2.5.1.1.2.4.11, area radiation monitors are provided in areas outside the reactor enclosure where access is needed after an accident. Signals from these monitors are available on a CRT display in the control room through the ERFDS or through a multipoint recorder. These monitors are listed in Table 7.5-3. Because these monitors are also used during normal operation, they are described in Section 12.3.4.1.

7.5.1.4.3.3 Airborne Radioactive Materials Released from Plant

In the event of an accident, the secondary containment is isolated and all exhaust from it is processed through the SGTS as described in Sections 6.2.3, 6.5.1.1, and 9.4.2. After a LOCA, any drywell purge from the CAC system is processed through the SGTS (Section 9.4.5.1). Because the SGTS exhausts into the north stack, the wide range accident monitor (Sections 7.6.1.1.8 and 11.5.2.2.1) of the north stack effluent radiation monitoring system is used to provide indication of airborne radioactive material releases postaccident.

Noble gas radioactivity concentration signals are transmitted from the wide range accident monitor to a 3-pen recorder that displays the entire range of the monitor, 10^{-7} □Ci/cc to 10^{+5} □Ci/cc in three overlapping scales. Indication of north stack flow rate is provided from a transmitter and displayed via a digital display/control module in the control room.

The system also provides grab sampling capability for particulates and halogens.

7.5.2 ANALYSIS

7.5.2.1 General

The SRDI provides adequate information to enable the operator to monitor transient reactor plant behavior and to verify proper safety system performance following an accident.

All protective actions required under accident conditions during the first 10 minutes are automatic and redundant so that immediate operator intervention is unnecessary.

The information provided, in addition to the performance of the emergency system, supplies sufficient time for the operator to make reasoned judgments and take action when required.

7.5.2.2 Normal Operation

Instrumentation ranges for normal, transient, or accident, conditions are selected to maintain the accuracy requirements for all conditions. The accuracy of SRDI is shown in Table 7.5-1.

7.5.2.3 Transient Occurrences

These occurrences are not limiting from the point of view of instrument ranges and functional capability (Section 7.5.2.2).

7.5.2.4 Accident Conditions

The DBA LOCA is the most extreme operational event. Information readouts are designed to accommodate this event from the standpoint of operator actions, information, and event tracking requirements.

7.5.2.4.1 Initial Accident Event

The design basis of all ESFs, to mitigate accident event conditions, takes into consideration that no operator action or assistance is required or recommended for the first 10 minutes of the event. This requirement therefore makes it mandatory that all protective action necessary in the first 10 minutes be automatic. Therefore, although continuous tracking of variables is available, no operator action based on them is intended within the first 10 minutes.

7.5.2.4.2 Postaccident Monitoring

The following process instrumentation provides information to the operator after a DBA LOCA for use in monitoring reactor conditions:

a. Reactor Water Level and Pressure:

Vessel water level and pressure sensor instrumentation described in Section 7.5.1.4.2.1 is redundant, electrically independent, and is qualified to be operable during and after a LOCA in conjunction with a SSE. Power is from two independent Class 1E instrument buses supplied from two of the divisional ac buses. This instrumentation complies with the independence and redundancy requirements of IEEE 279 (1971) and provides recorded outputs. The equipment performs its required functions during and after a seismic event, except that the recorders and indicators are not functional during the seismic event.

The reactor water level and pressure transmitters are mounted on two independent local racks. The transmitters and recorders are designed to operate during normal operation and postaccident environmental conditions. The design criteria that the instruments must meet are discussed in Section 7.1.2.1.7. There are two complete and independent channels of wide range reactor water level and reactor vessel pressure with each channel having its readout in the control room on a separate recorder.

Two recorders are furnished, each monitoring one channel of pressure and one channel of reactor water level.

An evaluation was conducted of the effects of high temperature on the reference legs of the water level measuring instruments resulting from the exposure to HELBs.

HELBs are breaks occurring in a fluid system whose normal plant conditions are either in operation or maintained pressurized under conditions where either or both of the following are met:

1. Maximum operating temperature exceeds 200°F.

2. Maximum operating pressure exceeds 275 psig.

These HELBs can be categorized into three sizes: a) large, b) intermediate, and c) small. Each break has its own temperature and pressure effects on the vessel drywell and suppression pool. However, the small HELB imposes the most severe temperature conditions on the drywell and is more likely to affect the vessel level instrumentation reference legs.

The LGS reactor vessel level instrumentation design consists of four divisions of cold reference leg, differential pressure, level indications. The design includes parallel reference and variable legs with an approximate 12 foot vertical drop in the drywell. This design minimizes the error in level indication caused by elevated drywell temperatures. Reference leg temperature and drywell temperature are monitored by the ERFDS. The LGS design also includes redundant, safety-grade, wide range and fuel zone reactor water level indication in the control room.

Under a small break accident condition, the maximum temperature in the drywell could approach 340°F. Given this elevated drywell temperature and the gradual depressurization of the reactor vessel, a potential boil-off condition could occur when the reactor vessel pressure falls below 118 psia (corresponding to a 340°F saturation temperature). However, the EOPs have been written to prevent the boil-off conditions from occurring.

For purposes of illustration, the following sequence of events is assumed to occur. With the reactor and containment at the maximum normal conditions, a small HELB occurs that allows blowdown of reactor steam to the drywell. The resulting pressure increase in the drywell leads to a high drywell pressure signal that scrams the reactor, initiates high pressure ECCS, and activates the containment isolation system. Drywell temperature would increase rapidly and approach 340□F. However, reactor vessel pressure and level would tend to decrease at a slow rate due to the small size of the break. The ADS will not initiate until low RPV level and high drywell pressure signals have been received. In instruments with cold reference legs, the reference leg temperature could increase to the drywell ambient temperature, which will not exceed 340°F. Thus, under the least favorable possible conditions, a cold reference leg will not boil until the system pressure falls to 118 psia (T_{sat} at 340°F). Thus pressure maintained above about 100 psig is an assurance that boiling of the cold reference column has not occurred, even under the worst possible drywell temperature conditions. Given this information, the following scenarios show the accident conditions and operator actions.

The reactor operator is alerted to the incident by the drywell high pressure and temperature alarms and by the reactor scrams. At this point, the operator will start to follow the EOPs. The operator will be aware of the elevated drywell temperature and pressure and is required to monitor and control these parameters as well as suppression pool temperature and pressures. The operator will also be continuously monitoring RPV pressure and reference leg temperature and comparing these parameters against a curve to determine the potential for reference leg boiling.

The EOPs require the operator to control drywell pressure and temperature. The main concern, as it pertains to RPV level instrumentation, is drywell temperature. If the reactor pressure is maintained high enough, the saturation temperature will be higher than the drywell temperature and a boil-off condition will never occur. If, however, the drywell temperature cannot be controlled, the operator is required to shut down the drywell cooling system and initiate drywell sprays in accordance with the restrictions of the drywell spray initiation limit curve. The sprays will lower drywell temperature and prevent the boiling condition from occurring. Drywell sprays are also used to reduce drywell pressure in the event that the pressure cannot be controlled.

As previously mentioned, the operator is sensitive to the reference leg temperature versus reactor pressure curve. If at any time the existing conditions fall into the unsafe area above the curve, the operator will assess the ability to determine RPV water level. If RPV water level cannot be determined, a contingency EOP which addresses the inability to determine RPV level will be performed concurrently with the EOP being executed at that time. The operator would be aware of a boil-off condition by the occurrence of erratic level indications caused by the flashing fluid as it is forced out through the instrument line. The first action is to blowdown the vessel, then, flood the RPV to the elevation of the main steam lines using available low pressure makeup systems until RPV level indication can be determined. The RPV will remain flooded to the elevation of the main steam lines until RPV level indication is available. Flooding the vessel also ensures that adequate core cooling is accomplished.

Therefore, the existing LGS design coupled with the required operator actions as prescribed by the EOPs are sufficient to prevent or mitigate any detrimental effects on the water level indication that may be caused by a small HELB.

b. Suppression Pool Water Level:

This instrumentation complies with the requirements of IEEE 279 (1971) and provides indicated outputs. Power is from two independent Class 1E independent buses. The equipment performs its required function during and after the seismic event, except that the indicators are not functional during the seismic event. The indicators are selected from equipment that meets the design specification in effect at the time of the plant design. This parameter is recorded in the ERFDS data base.

c. Containment Pressure:

This instrumentation is redundant, electrically independent, and is qualified to be operable during and after a LOCA. Power is from two independent Class 1E buses, and the instrumentation complies with the requirements of IEEE 279 (1971) and provides a recorded output. The equipment will perform its required function during and after a seismic event, except that the recorder and indicators are not functional during the seismic event.

d. Emergency Core Cooling Systems:

Performance of ECCS following an accident can be verified by observing redundant and independent indications as described in Section 7.5.1.4.2.2.1.c, and fully satisfies the need for operator verification of the system's operation.

Redundancy of instrumentation within the individual ECCS is not provided; however, redundancy is provided within the combination of ECCS. Each ECCS is provided with system flow-measuring indication and isolation valve status indication, thereby allowing the operator to assess the operating conditions. The indicators and recorders are selected from equipment that meets the design specifications in effect at the time of the plant design.

e. Control Room Habitability:

Performance of the control room habitability is verified by observing the redundant and independent indication listed in Table 7.5-1 which satisfies the need for operator verification of the system operation.

f. Service Water Availability:

The operation of the ESW and the RHRSW systems is verified by observing the indications described in Sections 7.3.1.1.11.12.2 and 7.3.1.1.12.12.2, respectively, and satisfies the need for operator verification of the operation of the system.

g. Containment Atmospheric Control:

The operation of the CAC system is verified by observing the indications described in Sections 7.5.1.4.2.2.1.c.6 and 7.3.1.1.6.1.2.12.2, and satisfies the need for operator verification of the system operation.

h. Primary Containment Post-LOCA Radiation

The rate of buildup/decay of radioactivity inside the primary containment is verified by observing the indications described in Section 7.5.1.4.2.1.5, and satisfies the need for operator verification that safety functions are being accomplished.

i. SGTS, RERS, and REIS:

Performance of the SGTS, RERS, and REIS is verified by observing indication as described in Section 7.5.1.4.2.2, and satisfies the need for operator verification of the operation of the systems.

7.5.2.4.3 Safe Shutdown Displays

The safe shutdown display instrumentation in Section 7.5.1 consists of the control rod status lamps, scram pilot valve status lamps, and neutron monitoring instrumentation. These displays are expected to remain operable following an accident to indicate the occurrence of safe and orderly shutdown.

The displays provide diversity in that they are in three separate systems. The neutron monitoring outputs are recorded. The indicators and recorders are selected from equipment that meets the design specification in effect at the time of the plant design.

Sufficient instrumentation is provided for the operator to verify proper cooling water flow for the various shutdown cooling systems and modes. Proper shutdown cooling mode operation is indicated by RHR pump running indication, position indication for MOVs, cooling water flow indication, and RHRSW flow indication. Annunciators are provided to indicate RHR pump trip, abnormal pump discharge pressure, and loss of RHR system integrity. Proper RCIC operation is indicated by RCIC pump discharge and suction pressure and valve position indication. Annunciators are provided to indicate abnormal turbine operating parameters, low RCIC pump flow, and high turbine steam flow.

The safe shutdown display instrumentation conforms to the power generation design basis in that:

a. Abnormal conditions requiring operator action are indicated or annunciated in the control room.

b. Sufficient instrumentation is provided for the operator to verify proper cooling water flow and cooling system piping integrity.

7.5.2.5 General Functional Requirements Conformance

Conformance of the transmitter/trip unit system, used for safe shutdown display, is discussed in the Licensing Topical Report NEDO-21617, Revision A, "Analog Transmitter/Trip Unit System for Engineered Safeguard Sensor Inputs". Conformance of the other features with the regulatory and industry standards is discussed in the following sections.

7.5.2.5.1 Specific Regulatory Requirements Conformance

7.5.2.5.1.1 Conformance to Regulatory Guides

7.5.2.5.1.1.1 Regulatory Guide 1.47 (1973) - Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

The SRDI is designed to operate continuously, and there is no requirement for bypass provisions. Removal of instrumentation for servicing during plant operation is administratively controlled. Refer to the individual safety system analysis discussions of Regulatory Guide 1.47 contained in Sections 7.2, 7.3, 7.4 and 7.6.

7.5.2.5.1.1.2 Regulatory Guide 1.97 (1980) - Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

LGS conforms with the intent of the regulatory guide, which is to ensure that instrumentation systems be provided to assess equipment and plant conditions during and following an accident as required by GDC 13, GDC 19 and GDC 64. In general, the regulatory guide requirements are implemented except where deviations from the guide are justified technically and can be implemented without disrupting the general intent of the guide. In assessing Regulatory Guide 1.97, LGS has drawn on information in ANSI/ANS 4.5 and on data derived from other analyses and studies.

7.5.2.5.1.1.2.1 Design and Qualification Criteria

LGS conformance with the design and qualification criteria defined in the regulatory positions of Regulatory Guide 1.97 is summarized in Table 7.5-2 and discussed below (the paragraph numbers cited correspond to those in Regulatory Guide 1.97).

a. Paragraph 1.3: Instruments used for accident monitoring to meet the provisions of Regulatory Guide 1.97 have the proper sensitivity, range, transient response, and accuracy to ensure that the control room operator is able to perform the role of bringing the plant to, and maintaining it in, a safe shutdown condition and in assessing actual or possible releases of radioactive material following an accident.

Accident monitoring instruments that are required to be environmentally qualified are qualified to the requirement of NUREG-0588 (Section 3.11.1). The seismic qualification of instruments, where required, is in accordance with Regulatory Guide 1.100 (Section 3.10).

The LGS quality assurance program is used to comply with the quality assurance requirements (Section 8.1.6.1 and Chapter 17).

Periodic checking, testing, calibrating, and calibration verification of accident monitoring instrument channels (Regulatory Guide 1.118) are in accordance with the LGS Technical Specifications.

b. Paragraph 1.4: Instruments designated as Categories 1 and 2 for variable Types A, B, and C are identified in such a manner as to optimize the human factors engineering and presentation of information to the control room operator. This position is taken to clarify the intent of Regulatory Guide 1.97, which specified that these instruments be easily discerned for use during accident conditions.

c. Paragraph 1.6: It is LGS's position that Table 1 of Regulatory Guide 1.97 does not represent the minimum number of variables, correct ranges, or instrumentation categories for accident monitoring at a BWR facility. The LGS list of accident monitoring variables and classification of instrumentation as Category 1, 2 or 3 is in compliance with the intent and method used in Regulatory Guide 1.97. The LGS position on implementation of each variable is presented in Section 7.5.2.5.1.1.2.3.

d. Paragraph 2: Conformance with Paragraph 1.3 described above is applicable to the Type D and E variables of Regulatory Guide 1.97.

7.5.2.5.1.1.2.2 Analysis for Type A Variables

Regulatory Guide 1.97 (Rev 2) designates all Type A variables as plant specific, thereby defining none in particular. Type A variables for LGS have been selected in conformance with the definition in Regulatory Guide 1.97. Variables associated with contingency actions that will be identified in written procedures are excluded. The following is a list of Type A variables specific to LGS. Detailed description of each Type A variable is given in Section 7.5.1.4.2.1. The variables listed here are also included in Section 7.5.2.5.1.1.2.3.

a. Variable A1 - Oxygen and Hydrogen Concentration:

Operator action: If containment atmosphere approaches the combustible limits, initiate CGCS.

Safety function: Prevent combustible concentrations and thus preserve containment integrity.

b. Variable A2 - RPV Pressure:

Operator action: (1) Depressurize RPV and maintain safe cooldown rate by any of several systems, such as main turbine bypass valves, HPCI, RCIC, and RWCU; or

(2) manually open one SRV to reduce pressure to below SRV setpoint if an SRV is cycling.

Safety function: (1) Core cooling; (2) maintain RCS integrity.

c. Variable A3 - RPV Water Level:

Operator action: Restore and maintain RPV water level.

Safety function: Core cooling

d. Variable A4 - Suppression Pool Water Temperature:

Operator action: (1) Operate available suppression pool cooling system when pool temperature exceeds normal operating limits; (2) scram reactor if temperature reaches limit for scram; (3) if suppression pool temperature cannot be maintained below the heat capacity temperature limit, maintain RPV pressure below the corresponding limit; and (4) close any SORV.

Safety function: (1) Maintain containment integrity and (2) maintain RCS integrity.

e. Variable A5 - Suppression Pool Water Level:

Operator action: Maintain suppression pool water level within normal operating limits: (1) transfer RCIC suction from the CST to the suppression pool in the event of high suppression pool level; and (2) if suppression pool water level cannot be maintained below the suppression pool load limit, maintain RPV pressure below corresponding limit.

Safety function: Maintain containment integrity.

f. Variable A6 - Drywell Pressure:

Operator action: Control primary containment pressure by any of several systems, such as containment pressure control systems, SGTS, suppression pool sprays, drywell sprays.

Safety function: (1) Maintain containment integrity and (2) maintain RCS integrity.

7.5.2.5.1.1.2.3 Plant Variables for Accident Monitoring

LGS's implementation of the variables listed in table 1 of Regulatory Guide 1.97 and the fulfillment of design criteria and assignment of qualification categories for the instrumentation proposed for their measurement are summarized in Table 7.5-5.

Measurement of the five variable types provides the following kinds of information to plant operators during and after an accident:

a. Type A - plant pressure, barrier and heat sink information, on the basis of which operators can take specified manual control actions

b. Type B - information about the accomplishment of plant safety functions

c. Type C - plant information about the breaching of barriers to fission product release

d. Type D - information about the operation of individual safety systems

e. Type E - information about the magnitude of the release of radioactive materials.

The categories are also related (in Regulatory Guide 1.97) to "key variables." Key variables are defined differently for the different variable types. For Type B and Type C variables, the key variables are those variables that most directly indicate the accomplishment of a safety function; instrumentation for these key variables is designated Category 1.

Key variables that are Type D variables are defined as those variables that most directly indicate the operation of an emergency safety system; instrumentation for these key variables is usually Category 2. Key variables that are Type E variables are defined as those variables that most directly indicate the release of radioactive material; instrumentation for these key variables is also usually Category 2.

The variables are listed in Table 7.5-5 in the same sequence as in table 1 of Regulatory Guide 1.97; however, for convenience in cross-referencing entries and supporting data, the variables are designated by letter and number. For example, the sixth B-type variable listed in Regulatory Guide 1.97 is denoted in Table 7.5-5 as variable B6.

The LGS position is shown for each variable. In general, there are three kinds of responses, the variable and instrumentation are: (1) implemented to meet the regulatory guide criteria; (2) implemented with qualifying exceptions or (3) not implemented.

As necessary, the positions are elaborated or substantiated in the supplementary analyses in Section 7.5.2.5.1.1.2.4. References to these analyses are made in the tabulation by citing the appropriate UFSAR section.

7.5.2.5.1.1.2.4 Regulatory Guide 1.97 Project Position

The issues used to substantiate deviation of the LGS system from the Regulatory Guide 1.97 criteria are presented below.

7.5.2.5.1.1.2.4.1 Variable B1 - Neutron Flux

Issue Definition

The measurement of neutron flux is specified as the key variable in monitoring the status of reactivity. Neutron flux is classified as a Type B variable, Category 1. The specified range is 10^{-6}% to 100% full power (SRM, APRM). The stated purpose is "Function detection; accomplishment of mitigation."

Discussion

The lower end of the specified range, 10^{-6}% full power, is intended to allow detection of an approach to criticality by some undefined and noncontrolled mechanism after shutdown.

In attempting to analyze the performance of the neutron flux monitoring systems, a scenario was postulated to obtain the required approach to criticality. Basically, it assumes an increase in reactivity from loss of boron in the reactor water after SLCS actuation.

The accident scenario incorporates the following factors:

a. The control rods fail (completely or partially) to insert, and the operator actuates the SLCS.

b. The SLCS shuts down the reactor.

c. A slow leak in the primary system results in an outgo of borated water and its replacement by water that contains no boron.

d. A range of leak rates up to 20 gpm was considered (Table 7.5-4).

Calculations were made to evaluate the rise in neutron population as a function of different leak rates. The calculations were made for a shutdown neutron level of 5×10^{-8}% of full power. The choice of 5×10^{-8}% was based on measurements at two BWR plants. The shutdown level was assumed to have a negative reactivity of 10 dollars, an assumption that is representative of a shutdown with all rods inserted. The results of the calculations are presented in Table 7.5-4. The numbers in the table refer to the time in hours required to increase the flux by 1 decade. For example, with a leak of 5 gpm, it takes 100 hr to increase the power from 5×10^{-8}% to a 5×10^{-7}%, and 10 hr to increase it from 5×10^{-7}% to 5×10^{-6}%.

The reactor is subcritical and the neutron level is given by:

$$\text{Neutron level} = S \cdot M, \qquad \text{(EQ. 7.5-1)}$$

where:

S = the source strength

M = the multiplication, which is given by:

$$M = 1/(1-k) \qquad \text{(EQ. 7.5-2)}$$

For k = 0.9, M is 10; for k = 0.99, M is 100 and so forth. For criticality, the denominator approaches 0, as (k) approaches 1.0. Thus, the above equation was used to calculate relative neutron flux levels for a subcritical reactor until the reactor was near critical; then the critical equation of power with excess reactivity was used. Reactor power is directly proportional to neutron level.

The increase in reactivity toward criticality can be turned around by actuating the SLCS. A second actuation of the SLCS would cause a decrease in reactivity because of the high concentration of boron in the injected SLCS fluid relative to that in the leaking fluid (nominally 400 ppm). The sensitivity of the detector must allow adequate time for the operator to act. For a scaling evaluation, 10 minutes was considered sufficient time for operator action for accident prevention and mitigation.

Table 7.5-4 shows that the detector sensitivity (i.e., lower range) requirement is a function of leak rate and therefore of reactivity addition rate. On the basis of a 20 gpm leak rate, Table 7.5-4 shows that a detector that is one scale (i.e., about 10^{-5}) within 3 decades of the shutdown power (10^{-8}) would allow 0.18 hr (10.8 minutes) for operator action before reactor power increased another decade. A total of 0.36 hr (21.6 minutes) would be available for operator action from the time the indicator comes on scale to the time reactor power reaches 0.5% of full power. An alarm would be provided to warn the operator when the neutron flux reaches some plant specific setpoint.

The 20 gpm leak rate, which was assumed to continue for 27.75 hr, was used to define the sensitivity of the detector. It should be noted that the assumed leak rate, extended over the 27.75 hr period, would result in a loss of inventory so large that it could not in reality go undetected by the operator. Moreover, reactivity addition caused by this gradual boron depletion is unlikely because boron concentration is sampled and measured periodically. Again, the improbable 20 gpm leak rate was used only to obtain a mechanistic and conservative approach for selection of instrument sensitivity.

An absolute criterion for the lower range must include consideration of the neutron source level. The use of the neutron level 100 days after shutdown is conservative. There is high probability that conditions would be stable and controllable 2 days after the emergency shutdown, for the core decay heat is at a low level and the boron monitoring system should be functioning by that time. The actual neutron level will vary with fuel design, fuel history, and shutdown control strength. Measurements of shutdown neutron flux (with all rods inserted) at two BWR reactors show readings of 30-80 counts/sec (1000 counts/sec corresponds to 10^{-8} of full power). Measurements on other BWR reactors and for different fuel histories would show some variation, but those variations would be small compared with a criterion that is concerned with units of decades.

Neutron flux is the key variable for measuring reactivity control. The degree to which this variable is important to safety is another consideration. The large number of detectors (i.e., SRMs and IRMs) that are driven into the core soon after shutdown makes it highly probable that one or more of the existing NMS detectors will be inserted. On the other hand, there is little probability that there would be, simultaneously, a need for this measurement (in terms of operator action to be taken) and an accident environment in which the NMS would be rendered inoperable. Further, the operator can always actuate the SLCS on loss of instrumentation.

Conclusion

The existing NMS is adequate to meet the requirement of neutron flux measurement with some upgrading to improve system reliability. The upgrade of the existing system consists of powering the safety-related portion of the system from an UPS as described in Section 7.6.1.4.5. A rigorous Category 1 requirement is not justified relative to the criterion of "importance to safety." A Category 2 classification of this measurement fully meets the intent of Regulatory Guide 1.97 for neutron flux indication.

7.5.2.5.1.1.2.4.2 Variable B4 - Coolant Level in Reactor

7.5.2.5.1.1.2.4.2.1 Reactor Level Monitoring

LGS used overlapping ranges to monitor coolant level in the reactor as discussed in Section 7.5.1.4.2.1.1.

The reference leg of the wide range water level transmitter is 5 feet lower than the Regulatory Guide 1.97 required tap, i.e., centerline of the main steam lines. It was necessary to use this range to eliminate long runs of exposed sensing line tubing that contribute to erratic indication. The variable leg of the fuel zone water level is below the bottom of the core support plate. These two level monitors cover the range specified by Regulatory 1.97 except as mentioned above.

7.5.2.5.1.1.2.4.2.2 Trend Recording

Issue Definition

The purpose of addressing this issue is to determine which variables set forth in Regulatory Guide 1.97 need trend recording.

Discussion

Regulatory Guide 1.97, paragraph 1.3.2(f), states the general requirements for trend recording as follows: "Where direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available from dedicated recorders." Using the BWROG EPGs as a basis, the only trended variables required for operator action are reactor water level and reactor vessel pressure.

Conclusions

For LGS, only reactor water level (variable B4) and reactor vessel pressure (variable B6) require trend recording.

7.5.2.5.1.1.2.4.3 Variables B8 and C6 - Drywell Sump Level (B8) and Drywell Drain Sumps Level (C6)

Issue Definition

Regulatory Guide 1.97 specifies Category 1 instrumentation to monitor drywell sump level (variable B8) and drywell drain sumps level (variable C6). These designations refer to the drywell equipment drain tank and floor drain tank levels. Category 1 instrumentation indicates that the variable being monitored is a key variable. In Regulatory Guide 1.97, a key variable is defined as "... that single variable (or minimum number of variables) that most directly indicates the accomplishment of a safety function ..." The following discussion supports the BWROG alternative position that drywell sump level and drywell drain sumps levels should be qualified to Category 3 instrumentation requirements.

Discussion

The LGS drywell has two drain sumps. One drain is the equipment drain sump, which collects identified leakage; the other is the floor drain sump, which collects unidentified leakage.

Although the level of the drain sumps can be a direct indication of breach of the RCS pressure boundary, the indication is not unambiguous because there can be water in those sumps during normal operation. Other instrumentation specified in Regulatory Guide 1.97 that would also indicate leakage in the drywell are identified below:

a. Drywell pressure - variable B7, Category 1

b. Drywell temperature - variable D7, Category 2

c. Primary containment area radiation - variable C5, Category 1.

The drywell sump levels signal neither automatically initiates safety-related systems nor alerts the operator to take safety-related actions. Both sumps have level detectors that provide only the following nonsafety indications:

a. Continuous volume indication

b. High level alarm

c. Low level alarm

d. Average sump leak rate

e. Sump leak rate increase alarm

Regulatory Guide 1.97 specifies instrumentation to function during and after an accident. The drywell sump systems are deliberately isolated at the primary containment penetration on receipt of an accident signal to establish containment integrity. This fact renders the drywell sump level signal irrelevant.

Therefore, by design drywell level instrumentation serves no useful accident monitoring function.

The EOPs use the RPV level and the drywell pressure as entry conditions for the RPV Control Procedure. A small line break will cause the drywell pressure to increase before a noticeable increase in the sump level. Therefore, the drywell sumps will provide a "lagging" versus "early" indication of a leak.

Conclusions

Based on the above, LGS believes that Category 3, "high quality off-the-shelf instrumentation" is appropriate for drywell sump level and drywell drain sumps level instrumentation.

7.5.2.5.1.1.2.4.4 Variable C1 - Radioactivity Concentration or Radiation Level in Circulating Primary Coolant

Issue Definition

Regulatory Guide 1.97 specifies that the status of the fuel cladding be monitored. The specified variable is C1 (radioactivity concentration or radiation level in circulating primary coolant). The range is given as "½ technical specification limit to 100 times technical specification limit (R/hr)". Instrumentation for measuring variable C1 is designated as Category 1. The purpose for monitoring this variable is given as "detection of breach", referring, in this case, to breach of fuel cladding.

Discussion

The critical actions that must be taken to prevent and mitigate a gross breach of fuel cladding in a BWR are (1) shut down the reactor and (2) maintain water level. Monitoring variable C1, as directed in Regulatory Guide 1.97, will have no influence on either of these actions. Any usefulness from this monitored variable falls into the category of "information that the barriers to release of radioactive material are being challenged" and "identification of degraded conditions and their magnitude, so the operator can take actions that are available to mitigate the consequences." There are no additional operator actions to mitigate the consequences of fuel barriers being challenged, other than those based on Type A and B variables.

Although the subject of concern in the Regulatory Guide 1.97 requirement is assumed to be an isolated NSSS, LGS has given consideration to events that do not isolate the NSSS. The PASS provides a means of obtaining samples of reactor coolant and primary containment atmosphere. Analyses of these samples provide information on the status of fuel cladding integrity when the plant is isolated. Radiation monitors in the SJAE and main steam lines provide information on the status of fuel cladding when the plant is not isolated.

Conclusion

Instrumentation for measuring variable C1 is implemented as Category 3 because no planned operator actions are identified and no operator actions are anticipated based on this variable serving as the key variable.

7.5.2.5.1.1.2.4.5 Variable C14 - Radiation Exposure Rate

Issue Definition

Variable C14 is defined in table 1 of Regulatory Guide 1.97 as follows: "Radiation exposure rate (inside buildings or areas which are in direct contact with primary containment where penetrations and hatches are located)." The reason for monitoring variable C14 is given as "Indication of breach".

Discussion

The use of local radiation exposure rate monitors to detect breach or leakage through primary containment penetrations is impractical. In general, radiation exposure rate in the secondary containment will be largely a function of radioactivity in primary containment and in the fluids flowing in ECCS piping, which will cause direct radiation shine on the areas of concern where radioactive fluids are piped. Because of the amount of piping and the number of electrical penetrations and hatches and their widely scattered locations, local radiation exposure rate monitors could give ambiguous indications. Breach of containment is more appropriately assessed by using the noble gas effluent monitor provided to monitor variable E4 (Section 7.5.2.1.1.2.4.10).

Conclusion

LGS is not implementing this parameter. Other means of breach detection that are better suited to this function (as described above), are available. Radiation exposure rate monitors as described in Section 12.3.4.1 are provided in these buildings for indication of habitability only.

7.5.2.5.1.1.2.4.6 Variables D3 and D8 - Suppression Spray Flow (D3) and Drywell Spray Flow (D8)

Issue Definition

Regulatory Guide 1.97 specifies flow measurements of suppression chamber spray (variable D3) and drywell spray (variable D8) for monitoring the operation of the primary containment related systems. Instrumentation for measuring these variables is designated Category 2, with a range of 0% to 110% of design flow. These flows relate to spray flow for controlling pressure and temperature of the drywell and suppression chamber.

Discussion

The drywell sprays can be used to control the pressure and temperature of the drywell. The RHR system flow element is used for measuring drywell flow.

The pressure-suppression chamber sprays can be used to control the pressure and temperature in the suppression chamber. From the control room, the operator controls pressure and temperature by adjusting suppression chamber spray flow. The RHR system flow element is used for flow indication. The suppression chamber spray operates in parallel with the drywell spray and is regulated with a throttling valve. The flow is determined by RHR flow indication. The effectiveness of spray flow can be verified by pressure and temperature changes of the drywell and the suppression chamber as indicated in the control room.

Conclusions

The current plant equipment, in conjunction with operating practice, meets performance requirements of accuracy and reliability for measurement of spray flows into the drywell and suppression chamber.

7.5.2.5.1.1.2.4.7 Variables D13 through D17 - RCIC Flow (D13), HPCI Flow (D14), Core Spray System Flow (D15), LPCI System Flow (D16) and SLCS Flow (D17)

Issue Definition

Regulatory Guide 1.97 specifies flow measurements of the following systems: RCIC (variable D13), HPCI (variable D14), core spray (variable D15), LPCI (variable D16), and SLCS (variable D17). The purpose is for monitoring the operation of individual safety systems. Instrumentation for measuring these variables is designated as Category 2; the range is specified as 0% to 110% of design flow. These variables are related to flow into the RPV.

Discussion

The RCIC, HPCI, and CS systems each have one branch line (the test line) downstream of the flow-measuring element. The test line is provided with a MOV that is normally closed (two valves in series in the case of the HPCI). In addition, the valve in the test line closes automatically when the emergency system is actuated, thereby ensuring that indicated flow is not being diverted by the test line. Proper valve position can be verified by a direct indication of valve position.

Although the LPCI has several branch lines located downstream of each flow-measuring element, each of those lines is either normally closed or automatically aligned. On initiation of the LPCI, the valves in the system automatically line up for proper operation and prevent flow diversion by branch lines. Proper valve position can be verified by a direct indication of valve position.

For all of the above systems, there are valid primary indicators other than flow measurement to verify the performance of the emergency system; for example, vessel water level. With respect to SLCS, flow-measuring devices were not provided for this system. The pump discharge header pressure, which is indicated in the control room, will indicate SLCS pump operation. Besides the discharge header pressure observation, the operator can verify the proper functioning of the SLCS by monitoring the following:

a. Decrease in the level of the boric acid storage tank

b. Reactivity change in the reactor as measured by neutron flux and concentration of boron

c. Motor contactor indicating lights (or motor current); the use of these indications is believed to be a valid alternative to SLCS flow indication (some plants have indicators for open/close positions of check valves)

d. Squib valve continuity indicating lights

Conclusion

The flow measurement schemes for the RCIC, HPCI, CS, and LPCI are adequate in that they meet the requirements of Regulatory Guide 1.97. Monitoring the SLCS can be adequately done by measuring variables other than the flow.

7.5.2.5.1.1.2.4.8 Variable D18 - SLCS Storage Tank Level

Issue Definition

Regulatory Guide 1.97 lists SLCS storage tank level as a Type D variable with Category 2 design and qualification criteria.

Discussion

Regarding the instrumentation category requirement for variable D18, Regulatory Guide 1.97 indicates that it is a key variable in monitoring SLCS operation. Regulatory Guide 1.97 also states that, in general, key Type D variables be designed and qualified to Category 2 requirements.

In applying these requirements of the regulatory guide to this instrumentation, the following are noted:

a. The current design basis for the SLCS assumes a need for an alternative method of reactivity control without a concurrent LOCA or HELB. The environment in which the SLCS instrumentation must work is therefore a mild environment for qualification purposes.

b. The current design basis for the SLCS is recognized as considerably less than the importance to safety of the RPS and the engineered safeguards systems. Therefore, in accordance with the graded approach to quality assurance specified in Regulatory Guide 1.97, it is unnecessary to apply a full quality assurance program to this instrumentation.

Conclusion

SLCS storage tank level instrumentation will meet Category 3 design and qualification criteria as required by Regulatory Guide 1.97.

7.5.2.5.1.1.2.4.9 Variables D26 through D30 - Main Steam Bypass Valve Position (D26), Condenser Hotwell Level (D27), Condenser Pressure (D28), Circulating Water Pump Discharge Pressure (D29) and Reactor Recirculation Pump Flow (D30)

Issue Definition

Regulatory Guide 1.97 states that "The plant designer should select variables and information display channels required by his design to enable the control room personnel to ascertain the operating status of each individual safety system and other systems important to safety to that extent necessary to determine if each system is operating or can be placed in operation..." The purpose of this analysis was to determine whether certain other D-type variables should be added to table 1, Regulatory Guide 1.97.

Discussion

Regulatory Guide 1.97 addressed safety systems and systems important to safety to mitigate consequences of an accident. Another list of variables has been compiled for the BWR in NUREG/CR-2100. That report and a companion report, NUREG/CR-1440, address plant systems not important to safety, as well as systems that are important to safety. In particular, these reports consider the potential role of the turbine plant in mitigating certain accidents. These two reports were reviewed in determining whether the listed variables (D26 through D30) should be added to the Regulatory Guide 1.97 list.

The NUREG evaluations used a systematic approach to derive a variables list. The basic approach of the analysis was to focus on those accident conditions under which the operator is most likely to be confronted with "and/or" accident conditions that result in the most serious consequences if the operator should fail to accomplish his required tasks. This is a probabilistic event tree type of study, and the reports used the sequences of the Reactor Safety Study (WASH-1440), and similar studies. The events in each sequence that involved operator action were identified; also, events were added to the event tree to include additional operator actions that could mitigate the accident. The event tree defines a series of key plant states that could evolve as the accident progresses and as the operator attempts to respond. Thus the operator's informational needs are linked to these plant states.

NUREG/CR-2100 is a BWR evaluation undertaken to address appropriate operator actions, the information needed to take those actions, and the instrumentation necessary to provide the required information.

The sequences evaluated were:

a. Anticipated transient followed by loss of decay heat removal

b. Anticipated transients without scram

c. Anticipated transient together with failure of HPCI, RCIC, and low pressure ECCS

d. Large LOCA with failure of ECCS

e. Small LOCA with failure of ECCS

The Regulatory Guide 1.97 list is based on accidents that result in an isolated NSSS. The NUREG documents considered accidents that could be prevented or mitigated by using water inventory and the heat sink in the turbine plant.

Conclusion

Five of the 15 variables identified in the NUREG, but not in Regulatory Guide 1.97, are included as Type D, Category 3 additions to the Regulatory Guide 1.97 list. Four of these variables are in the turbine plant: the main steam bypass valve position, condenser hotwell level, condenser pressure, and circulating water pump discharge pressure. These variables provide a primary measure of the status of a heat sink or water inventory in the turbine plant. The turbine plant systems are not to be classed as "safety systems" or as systems important to safety. The reactor recirculation pump flow is also added to the Regulatory Guide 1.97 list.

7.5.2.5.1.1.2.4.10 Variable E2 - Reactor Building or Secondary Containment Radiation

Issue Definition

Regulatory Guide 1.97 specifies that reactor building or secondary containment area radiation (variable E2) should be monitored over the range of 10^{-1} R/h to 10^{4} R/h for Mark II containments. The classification for Mark II is Category 2.

Discussion

As discussed in the variable C14 analysis (Section 7.5.2.5.1.1.2.4.5), secondary containment area radiation is not an appropriate parameter to use to detect or assess primary containment leakage.

Conclusion

The specified reactor enclosure area radiation monitors are not required for the LGS secondary containment.

7.5.2.5.1.1.2.4.11 Variable E3 - Radiation Exposure Rate

Issue Definition

Regulatory Guide 1.97 specifies in table 1, variable E3, that radiation exposure rate (inside buildings or areas where access is required to service equipment important to safety) be monitored over the range of 10^{-1} R/hr to 10^{4} R/hr for detection of significant releases, for release assessment, and for long-term surveillance.

Discussion

In general, access is not required to any area of the secondary containment to service equipment important to safety in a postaccident situation. If and when accessibility is re-established in the long-term, it will be done by a combination of portable radiation survey instruments and postaccident sampling of the secondary containment atmosphere. Lower range area radiation monitors (typically 3 decades lower than the Regulatory Guide 1.97 range) are provided for use only in those instances in which radiation levels are very mild.

There are areas outside secondary containment where access is needed postaccident for specific sampling, monitoring or analysis tasks. Dose rates greater than 10 R/hr are not expected in these areas. In the event that these areas experience dose rates greater than 10 R/hr, access could result in excessive operator exposure.

Conclusion

The LGS design does not require access to a harsh environment area to service safety-related equipment during an accident; portable radiation monitors will be provided to re-establish accessibility. Lower range area monitors are implemented as Category 3 for areas outside secondary containment where postaccident access is needed.

7.5.2.5.1.1.2.4.12 Variable E8 - Plant and Environs Radiation

Issue Definition

Regulatory Guide 1.97 specifies that plant and environs radiation (variable E8) should be monitored over the range of 10^{-3} R/hr to 10^{4} R/hr, photons and 10^{-3} Rad/hr to 10^{4} Rad/hr, beta radiations and low energy photons. The classification is Category 3.

Discussion

The plant inventory of portable radiation survey instrumentation described in Section 12.5.2.2.3 will be supplemented with additional equipment to enhance postaccident monitoring capabilities. This additional equipment will be comprised of low range, medium range, and high range portable ion chambers (1 mR/hr to 20,000 R/hr gamma and 20,000 Rad/hr beta), open window alpha scintillation probes, pancake GM probes, energy compensated beta/gamma GM probes (for low energy photons), and portable beta/gamma geiger counters. Audio speakers, alarming count rate meters, and extension arms will be provided for attachments to the survey instruments. Airborne radioactivity levels will be determined from laboratory analysis of particulate filters and iodine cartridge samples obtained with high and low volume samplers. Portable instruments and equipment reserved for emergency use will be located at an assembly area remote from the main plant.

Conclusion

A range of monitoring of 10^4 R/hr would not enhance plant and environs radiation monitoring. LGS meets the intention and the purpose of the regulatory guide criteria.

7.5.2.5.1.1.2.4.13 Variable E13 - Primary Coolant and Sump

Issue Definition

Regulatory Guide 1.97 requires installation of the capability for obtaining grab samples (variable E13) of the containment sump, ECCS pump-room sumps, and other similar auxiliary building sumps for the purpose of release assessment, verification, and analysis.

Discussion

The need for sampling a particular sump must take into account its location and the sump design. For all accidents in which radioactive material would be in the primary containment sump, it will be isolated and will overflow to the suppression pool. A suppression pool sample can be obtained through the PASS as described in Section 11.5.5 and this can therefore be used as a valid alternative to a containment sump sample.

The analysis of ECCS pump-room sumps and other similar auxiliary building sump liquid samples can be used for release assessment, as suggested in Regulatory Guide 1.97 only if potentially radioactive water can be pumped out of a controlled area to an area such as radwaste. If the design does not allow sump pump-out on a high radiation signal, a sump sample does not contribute to release assessment. The use of the subject sump samples for verification and analysis is of little value; a sample of the suppression pool and reactor water, as required by other portions of Regulatory Guide 1.97, provides a better measurement for these purposes.

Conclusion

A suppression pool sample will be used as an alternative to a primary containment sump sample. The analysis of ECCS pump-room sumps and other similar auxiliary building sumps is a consideration only if the water is pumped out of the reactor enclosure (e.g., pumped to radwaste). LGS design does not allow sump pump-out on receipt of high radiation signal. The capability for sampling and analysis of ECCS pump-room and auxiliary building sumps is therefore not provided.

7.5.2.5.1.1.2.4.14 Variables C5 and E1 - Primary Containment Post-LOCA Radiation Monitoring System

Issue Definition

Regulatory Guide 1.97 requires that containment radiation after an event be measured to within a factor of two.

Discussion

The primary containment post-LOCA radiation monitoring system installed at Limerick Generating Station meets this requirement except under extreme conditions of high drywell temperature and low radiation levels.

The licensee was notified by the system vendor in February 1987 that under high drywell temperature conditions, Insulation Resistance (IR) leakage current may cause a system error.

Because the instrument signal at low radiation levels is very weak, high temperature IR leakage current significantly affects the accuracy of the indicated readings up to a maximum of 112.5 Rad/hr at the maximum design drywell temperature of 340°F. As a result, the indicated readings below 112.5 Rad/hr may not be within the factor of two accuracy recommendation of Regulatory Guide 1.97 Rev.2. The induced error decreases exponentially with drywell temperature and becomes insignificant below 230°F. This induced error is significant only under low radiation conditions coincident with high drywell temperatures, whereas the system will operate to perform its principal function under normal and varying temperature conditions during and following an accident.

Conclusion

Since this failure is apparent only under low radiation conditions coincident with high drywell temperature, whereas the principal function of the system is to monitor high radiation postaccident conditions, continued operation without replacing the cable is justified. As drywell temperature is reduced with time, the error will also be reduced.

7.5.2.5.1.2 Conformance to 10CFR50, Appendix B

The SRDIs, except the displays, are of the same type, and are subject to the same qualification testing, quality control, and documentation, in accordance with the recommendation of 10CFR50, Appendix B as the safety systems' instrumentation. The displays are of a high quality consistent with the rest of the SRDIs, and are proven through industrial usage. The displays' qualification testing, quality assurance program, and documentation are provided and maintained by the vendor. For further information, refer to Chapter 17.

7.5.2.5.1.3 Conformance to Industry Codes and Standards

7.5.2.5.1.3.1 IEEE 279 (1971) - Criteria for Protection Systems for Nuclear Power Generating Stations

A discussion of the degree of conformance to IEEE 279 (1971) is not appropriate for application to SRDI. Most Regulatory Guide 1.97 Category 1 instrumentation meets the requirements of IEEE 279 (1971). Refer to the individual safety system discussion to find the degree of system conformance, for each parameter with a safety-related display instrumentation output.

7.5.2.5.1.3.2 IEEE 323 (1974) - IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations

Safety-related display instrumentation equipment, where required (Table 7.5-2), meets the requirement of IEEE 323 (1974). Refer to Section 3.11 for a discussion of the degree of conformance of the environment qualified for the SRDI equipment.

7.5.2.5.1.3.3 IEEE 344 (1975) - Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations

SRDI equipment, where required (Table 7.5-2), meets the requirements of IEEE 344 (1975). Refer to Section 3.10 for a discussion of the degree of conformance for the seismic capacity of the qualified SRDI equipment.

Table 7.5-1

SAFETY-RELATED DISPLAY INSTRUMENTATION

PARAMETER MEASURED	INSTRUMENT NO.	NUMBER OF CHANNNELS[5]	RANGE	ACCURACY
Reactor vessel pressure	XR42-1R623A/B	2	0-1500 psig	0.5% FS[1]
Reactor vessel water level	XR42-1R623A/B LR42-1R615 LI42-1R610	2 1 1	-150" to +60"[3] -350" to -100"H_2O[4] -350" to -100"H_2O[4]	0.5% FS
HPCI flow	FI55-1R600-1	1	0-6000 gpm	±2% FS
HPCI discharge pressure	PI55-1R601	1	0-1500 psig	±2% FS
HPCI turbine steam pressure	PI55-1R602	1	0-1500 psig	±2% FS
CS flow	FI52-1R601A/B	2	0-8,800 gpm	±2% FS
RHR flow (LPCI and shutdown cooling)	FI51-1R603A/B/C/D	4	0-12,000 gpm	±2% FS
Containment hydrogen	AI57-151 AI57-188	1 1	0-30% H_2 0-30% H_2	±2% FS ±2% FS
Containment oxygen	AI57-150 AI57-187	1 1	0-10%/0-25% 0_2 0-10%/0-25% 0_2	±2% FS ±2% FS
Containment radiation	RR26-191A/B	2	1-10^8rad/hr	±0.5% FS
Suppression pool temperature	TRS-041-101 TRS-041-103	1 1	30□F - 230°F 30□F - 230°F	±0.3% FS ±0.3% FS

Table 7.5-1 (Cont'd)

PARAMETER MEASURED	INSTRUMENT NO.	NUMBER OF CHANNNELS[5]	RANGE	ACCURACY
Containment pressure	PR57-101	1	-5 to +165 psig	±0.5% FS
	PI42-170-1	1	0-150 psig	±2% FS
	PI42-170-2	1	0-150 psig	±2% FS
	PI42-101	1	-5 to +165 psig	±2% FS
	PI57-121	1	-5 to +5 psig	±2% FS
Containment spray flow (RHR)	FI51-1R603A/B/C/D	4	0-12,000 gpm	±2% FS
Suppression chamber pressure	PR57-101	1	-5 to +165 psig	±0.5 FS
Suppression pool level	LR55-115	1	20-26 ft H_2O	±2% FS
	LI52-140A/B	2	0-50 ft H_2O	±2% FS
	LI55-115-1(Unit 1 only)	1	20-26 ft H_2O	±2% FS
	LI55-115-2	1	20-26 ft H_2O	±2% FS
	LI55-217(Unit 2 only)	1	20-26 ft	±2% FS
	LI55-141 (Unit 1 only)	1	20-30 ft H_2O	±1.5% FS
ESW pump discharge flow	FDI11-012A/B	2	-1000-0-1000 gpm	±2% FS
	FI11-013A/B	2	0-6,000 gpm	±2% FS
RHRSW HX inlet flow	FI51-1R602A/B	2	0-12,000 gpm	±2% FS
SGTS filter heater inlet temperature	TI76-001A/B	2	50°F-200°F	±2%
SGTS filter inlet temperature	TI76-003A/B	2	50°F-200°F	±2%
SGTS filter charcoal temperature	TI76-010A/B	2	200°F-680°F	±2%
Refueling floor/ outside $_{\Delta}$P	PDI76-099A/B	2	-.35 to +.05 in wg	±2%
Reactor enclosure/ outside $_{\Delta}$P	PDI76-198A/B	2	-.35 to + .05 in wg	±2%
Recirculation filter inlet temperature	TI76-182A/B	2	50°F-150°F	±2%

Table 7.5-1 (Cont'd)

PARAMETER MEASURED	INSTRUMENT NO.	NUMBER OF CHANNNELS[5]	RANGE	ACCURACY
Recirculation filter charcoal temperature	TI76-190A/B	2	200°F-680°F	±2%
SGTS flow	FI76-042	1	0-4000 cfm	±2%
Chlorine detector	AI78-016A/B/C/D	4	0-5 PPM	±2%
Control room pressure differential	PDI78-054	1	+0.4 to -0.1 in wg	±2%
Emergency recirculation flow	FI78-015	1	0-3500 cfm	±2%
Emergency fresh air carbon filter temperature	TI78-008A/B	2	200°F-680°F	±2%
Control room return air duct temperature	TI78-024A/B	2	50°F-150°F	±2%
Control enclosure chilled water flow	FI90-034A/B	2	0-800 gpm	±2%
Chilled water supply temperature	TI90-025A/B	2	30°F-80°F	±2%
Chilled water return temperature	TI90-024A/B	2	30°F-80°F	±2%
Control room airborne activity	RIX26-007A/B/C/D (outside air intake)	4	10^{-6} to 10^{-1} Ci/cc	±25%
	RIX26-068C/D (emergency fresh air filter discharge)	2	10^{-6} to 10^{-1} Ci/cc	±25%

(1) FS = Full Scale
(2) Per steam line
(3) Instrument zero equal to 527.5 in above vessel zero
(4) Top active fuel indicated in red
(5) Electrical divisions
(6) The value listed in the column is the manufacturers stated accuracies for the device. These values are "FOR INFORMATION ONLY" the actual calibration requirements for the instrumentation systems are contained within the IISCP Program.

Table 7.5-2

DESIGN AND QUALIFICATION REQUIREMENTS FOR POSTACCIDENT MONITORING INSTRUMENTATION

REGULATORY GUIDE 1.97 CATEGORY/REQUIREMENT	1	2	3
Seismic Qualification (Regulatory Guide 1.100 (Rev 1)	Yes(1)	Yes(2)	No
Single Failure Criterion	Yes	No	No
Environmental Qualification (Regulatory Guide 1.89 (1974) & NUREG-0588 (Rev 1)	Yes(1)	Yes(1)	No
Power Supply	Class 1E	Class 1E or UPS	Instrument ac
Out-of-Service Interval	Continuous(7)	System Technical Specification	None(3)
Display Type	Continuous or On Demand	Continuous or On Demand	On Demand
Display Method	Indication(4)	Indication(5)(6)	Indication(5)(6)

(1) Where the signal is to be displayed by a non-Class 1E computer-based system, qualification includes the sensor and the computer input isolation device.

(2) Seismic qualification is needed, provided that the instrumentation is part of a safety-related system. Where the signal is to be displayed by a non-Class 1E computer-based system, seismic qualification includes the sensor and the computer input isolation device.

(3) Not necessary to include in the safety and environmental technical specifications unless specified by other requirements.

(4) Where direct and immediate trend or transient information is essential for operator information or action, recording is provided on dedicated recorders. Otherwise, it is available on demand.

(5) Dial, digital, CRT, or strip-chart recorder indication.

(6) Recording required for effluent radioactivity monitors, area radiation monitors, and meteorology monitors. Where direct and immediate trend or transient information is essential for operator information or action, the recording is provided on dedicated recorders. Otherwise, it is available on demand.

(7) Continuous service is achieved through redundant instrument channels.

Table 7.5-3

POSTACCIDENT MONITORING INSTRUMENTATION

			INDICATION				
VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Neutron flux	B1	2	Recorder (SRM)	2	0 - 125% rated power (SRM)	XR X-M1-1R602A,B (Non-Div)	Control room
			Recorder (IRM/APRM)	4	0 - 125% rated power (IRM) 0 - 125% rated power (IRM/APRM)	XR X-M1-1R603A,B,C, D (Non-Div)	
Control rod position	B2	3	Indicating lights	1 per control rod	Full-in or not full-in	ZS-047-205-AE-RL (Non-Div)	Control room
Reactor coolant level	A3 B4	1	Recorder	2	-150 to +60 inches [2]	XR42-1R623A (Div I) XR42-1R623A (Div II)	Control room Control room
			Recorder	1	-350 to –100 inches H_2O[3]	LR42-1R615 (Div II)	Control room
			Indicator	1	-350 to –100 inches H_2O[3]	LI42-1R610 (Div II)	Control room
Reactor coolant pressure	A2 B6 C4, 9	1	Recorder	2	0 to 1500 psig	XR42-1R623A (Div I) XR42-1R623B (DIV II)	Control room Control room
Reactor recirculation D30 pump flow	D30	3	Recorder	1	0 – 55,000 gpm/loop	FR43-1R614 (Non-Div) 2-Pens 1/Loop	Control room
RCS soluble boron concentration	B3	3	Grab sample			N/A	(6)

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Primary containment and drywell pressure	A6 B7, 9	1	Recorder	1	-5 to +165 psig	PR57-101 (DIV I)	Control room
	C8, 10		Indicator	1	-5 to +165 psig	PI42-101 (DIV II)	Control room
	D4	2	Indicator	1	-5 psig to +5 psig	PI57-121 (DIV II)	Control room
Suppression pool water level	A5 C7 D5	1	Indicator	2	0 – 50 ft H_2O	LI52-140A (DIV I) LI52-140B (DIV II)	Control room Control room
Suppression pool water temperature	A4 D6	1	Indicator	2	30°F to 230°F	TI41-101 (DIV I) TI41-103 (DIV II)	Control room Control room
Drywell sump level 1. Floor drain sump tank 2. Equipment drain tank	B8 C6	3	CRT	NA	 0-391.1 gal 0-402.7 gal	ERFDS CRT (Non-Div)	Control room
Containment and drywell H_2 Concentration	A1 C11	1	Indicator	2	0% - 30% H_2	AI57-151 (DIV IV) AI57-188 (DIV III)	Control room Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Containment and drywell O_2 concentration	A1 C12	1	Indicator	2	0 – 10% O_2 0 – 25% O_2	AI57-150 (DIV IV) AI57-187 (DIV III)	Control room Control room
Containment Isolation Valve position	B10	1					
HV13-106			Indicating lights	1 pair per valve	open/closed	HS13-106 (DIV III)	Control room
HV13-107			Indicating lights	1 pair per valve	open/closed	HS13-107 (DIV III)	Control room
HV13-108, 111			Indicating lights	1 pair per valve	open/closed	HS13-108 (DIV IV)	Control room
SV26-190A, C			Indicating lights	2 pair per valve	open/closed	HS26-190A (DIV III)	Control room
SV26-190B, D			Indicating lights	2 pair per valve	open/closed	HS26-190B (DIV II)	Control room
HV41-1F022A, B, C, D			Indicating lights	1 pair per valve	open/closed	HS41-122A, B, C, D (DIV I)	Control room
HV41-1F028A, B, C, D			Indicating lights	1 pair per valve	open/closed	HS41-128A, B, C, D (DIV I)	Control room
HV41-1F016			Indicating lights Indicating lights	1 pair per valve 1 pair per valve	open/closed open/closed	HS41-116 (DIV I) HS41-119 (DIV I)	Control room Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV41-109A, B			Indicating lights	I pair per valve	open/closed	HV41-109A, B (DIV I)	Control room
HV41-133A			Indicating lights	I pair per valve	open/closed	HV41-133A (DIV I)	Control room
HV41-133B			Indicating lights	I pair per valve	open/closed	HV41-133B (DIV II)	Control room
HV41-1F084			Indicating lights	I pair per valve	open/closed	HS41-186 (DIV I)	Control room
HV41-1F085			Indicating lights	I pair per valve	open/closed	HS41-187 (DIV II)	Control room
HV41-130A			Indicating lights	I pair per valve	open/closed	HS41-130A (DIV II)	Control room
HV41-130B			Indicating lights	I pair per valve	open/closed	HS41-130B (DIV I)	Control room
HV42-147A			Indicating lights	I pair per valve	open/closed	HS42-147A (DIV I)	Control room
HV42-147B			Indicating lights	I pair per valve	open/closed	HS42-147B (DIV II)	Control room
HV42-147C			Indicating lights	I pair per valve	open/closed	HS42-147C (DIV III)	Control room
HV42-147D			Indicating lights	I pair per valve	open/closed	HS42-147D (DIV IV)	Control room
HV43-1F019			Indicating lights	I pair per valve	open/closed	HS43-120 (DIV I)	Control room
HV43-1F020			Indicating lights	I pair per valve	open/closed	HS43-120 (DIV II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV44-IF001			Indicating lights	l pair per valve	open/closed	HS44-101 (DIV I)	Control room
HV44-1F004			Indicating lights	l pair per valve	open/closed	HS-104 (DIV II)	Control room
HV48-1F006A			Indicating lights	l pair per valve	open/closed	HS48-103A (DIV I)	Control room
HV48-1F006B			Indicating lights	l pair per valve	open/closed	HS48-103B (DIV II)	Control room
HV49-1F013			Indicating lights	l pair per valve	open/closed	HS49-113-1 (DIV I)	Control room
HV49-1F007			Indicating lights	l pair per valve	open/closed	HS49-107-1 (DIV III)	Control room
HV49-1F008			Indicating lights	l pair per valve	open/closed	HS49-108-1 (DIV I)	Control room
HV49-1F076			Indicating lights	l pair per valve	open/closed	HS49-176-1 (DIV I)	Control room
HV49-1F031			Indicating lights	l pair per valve	open/closed	HS49-131-1 (DIV I)	Control room
HV49-1F060			Indicating lights	l pair per valve	open/closed	HS49-118-1 (DIV I)	Control room
HV49-1F019			Indicating lights	l pair per valve	open/closed	HS49-119-1 (DIV I)	Control room
HV49-1F002			Indicating lights	l pair per valve	open/closed	HS49-117-1 (DIV I)	Control room
HV49-1F084			Indicating lights	l pair per valve	open/closed	HS49-184-1 (DIV III)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV49-1F080			Indicating lights	I pair per valve	open/closed	HS49-180-1 (DIV I)	Control room
HV51-1F009			Indicating lights	I pair per valve	open/closed	HS51-109-1 (DIV I)	Control room
HV51-1F008			Indicating lights	I pair per valve	open/closed	HS51-108-1 (DIV II)	Control room
HV51-151A, B			Indicating lights	I pair per valve	open/closed	HS51-150A & B (DIV I)	Control room
HV51-1F015A			Indicating lights	I pair per valve	open/closed	HS51-115-1 (DIV II)	Control room
HV51-1F015B			Indicating lights	I pair per valve	open/closed	HS51-115B (DIV II)	Control room
HV51-1F021A			Indicating lights	I pair per valve	open/closed	HS51-121A (DIV I)	Control room
HV51-1F021B			Indicating lights	I pair per valve	open/closed	HS51-121B (DIV II)	Control room
HV51-1F016A			Indicating lights	I pair per valve	open/closed	HS51-116A-1 (DIV I)	Control room
HV51-1F016B			Indicating lights	I pair per valve	open/closed	HS51-116B (DIV II)	Control room
HV51-142A			Indicating lights	I pair per valve	open/closed	HS51-141A (DIV I)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # [1]	CATEGORY [1]	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV51-142B			Indicating lights	I pair per valve	open/closed	HS51-141B (DIV II)	Control room
HV51-142C			Indicating lights	I pair per valve	open/closed	HS51-141C (DIV III)	Control room
HV51-142D			Indicating lights	I pair per valve	open/closed	HS51-141D (DIV IV)	Control room
HV51-1F017A			Indicating lights	I pair per valve	open/closed	HS51-117A-1 (DIV I)	Control room
HV51-1F017B			Indicating lights	I pair per valve	open/closed	HS51-117B (DIV II)	Control room
HV51-1F017C			Indicating lights	I pair per valve	open/closed	HS51-117C (DIV III)	Control room
HV51-1F017D			Indicating lights	I pair per valve	open/closed	HS51-117D (DIV IV)	Control room
HV51-1F004A			Indicating lights	I pair per valve	open/closed	HS51-104A-1 (DIV I)	Control room
HV51-1F004B			Indicating lights	I pair per valve	open/closed	HS51-104B (DIV II)	Control room
HV51-1F004C			Indicating lights	I pair per valve	open/closed	HS51-104C (DIV III)	Control room
HV51-1F004D			Indicating lights	I pair per valve	open/closed	HS51-104D (DIV IV)	Control room
HV51-125A			Indicating lights	I pair per valve	open/closed	HS51-125A-1 (DIV I)	Control room
HV51-125B			Indicating lights	I pair per valve	open/closed	HS51-125B (DIV II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV51-1F027A			Indicating lights	I pair per valve	open/closed	HS51-127A-1 (DIV I)	Control room
HV51-1F027B			Indicating lights	I pair per valve	open/closed	HS51-127B (DIV II)	Control room
HV51-105A			Indicating lights	I pair per valve	open/closed	HS51-105A (DIV III)	Control room
HV51-105B			Indicating lights	I pair per valve	open/closed	HS51-105B (DIV IV)	Control room
HV-C-51-2F103A (Unit 2 only)			Indicating lights	I pair per valve	open/closed	HS51-213A (DIV I)	Control room
HV-C-51-2F104B (Unit 2 only)			Indicating lights	I pair per valve	open/closed	HS51-234A (DIV II)	Control room
HV52-1F039A			Indicating lights	I pair per valve	open/closed	HS52-106A (DIV I)	Control room
HV52-1F039B			Indicating lights	I pair per valve	open/closed	HS52-106B (DIV II)	Control room
HV52-1F005			Indicating lights	I pair per valve	open/closed	HS52-105 (DIV I)	Control room
HV52-1F001A			Indicating lights	I pair per valve	open/closed	HS52-101A (DIV I)	Control room
HV52-1F001B			Indicating lights	I pair per valve	open/closed	HS52-101B (DIV II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV52-1F001C			Indicating lights	I pair per valve	open/closed	HS52-101C (DIV III)	Control room
HV52-1F001D			Indicating lights	I pair per valve	open/closed	HS52-101D (DIV IV)	Control room
HV52-1F015A			Indicating lights	I pair per valve	open/closed	HS52-115A (DIV I)	Control room
HV52-1F015B			Indicating lights	I pair per valve	open/closed	HS52-115B (DIV II)	Control room
HV52-1F031A			Indicating lights	I pair per valve	open/closed	HS52-131A (DIV I)	Control room
HV52-1F031B			Indicating lights	I pair per valve	open/closed	HS52-131B (DIV II)	Control room
HV52-127			Indicating lights	I pair per valve	open/closed	HS52-127 (DIV I)	Control room
HV52-128			Indicating lights	I pair per valve	open/closed	HS52-128 (DIV II)	Control room
HV52-139, HV55-120,121			Indicating lights	I pair per valve	open/closed	HS55-120 (DIV II)	Control room
HV55-1F002			Indicating lights	I pair per valve	open/closed	HS55-102 (DIV IV)	Control room Panel 10C647
HV55-1F003			Indicating lights	I pair per valve	open/closed	HS55-103 (DIV II)	Control room Panel 10C647
HV55-1F100			Indicating lights	I pair per valve	open/closed	HS55-148 (DIV II)	Control room Panel 10C647
HV55-1F042			Indicating lights	I pair per valve	open/closed	HS55-142 (DIV II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV55-1F072			Indicating lights	I pair per valve	open/closed	HS55-172 (DIV II)	Control room
HV55-1F071			Indicating lights	I pair per valve	open/closed	HS55-171 (DIV II)	Control room
HV55-1F095			Indicating lights	I pair per valve	open/closed	HS55-195 (DIV IV)	Control room
HV55-1F093			Indicating lights	I pair per valve	open/closed	HS55-193 (DIV II)	Control room
HV55-1F012			Indicating lights	I pair per valve	open/closed	HS55-112 (DIV II)	Control room
HV55-1F105			Indicating lights	I pair per valve	open/closed	HS55-105 (DIV II)	Control room
SV52-139, SV57-101			Indicating lights	I pair per valve	open/closed	HS57-101 (DIV I)	Control room
HV57-121			Indicating lights	I pair per valve	open/closed	HS57-121 (DIV I)	Control room
HS57-131			Indicating lights	I pair per valve	open/closed	HS57-131 (DIV I)	Control room
HV57-123			Indicating lights	I pair per valve	open/closed	HS57-123 (DIV I)	Control room
HV57-163			Indicating lights	I pair per valve	open/closed	HS57-163 (DIV IV)	Control room
HV57-111			Indicating lights	I pair per valve	open/closed	HS57-111 (DIV II)	Control room
HV57-114			Indicating lights	I pair per valve	open/closed	HS57-114 (DIV II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV57-161			Indicating lights	2 pair per valve	open/closed	HS57-161 (DIV III)	Control room
SV57-133			Indicating lights	I pair per valve	open/closed	HS55-133 (DIV I)	Control room
HV57-116			Indicating lights	I pair per valve	open/closed	HS57-116 (DIV IV)	Control room
SV57-141, 142, 143, 144, 145, 159			Indicating lights	I pair per valve	open/closed	HS57-153 (DIV IV)	Control room
SV57-132, 134, 150			Indicating lights	I pair per valve	open/closed	HS57-132 (DIV II)	Control room
SV57-184, 185, 186, 190, 195			Indicating lights	I pair per valve	open/closed	HS57-187 (DIV III)	Control room
SV57-183, 191			Indicating lights	I pair per valve	open/closed	HS57-183 (DIV I)	Control room
HV57-124			Indicating lights	I pair per valve	open/closed	HS57-124 (DIV I)	Control room
HV57-164			Indicating lights	2 pair per valve	open/closed	HS57-164 (DIV IV)	Control room
HV57-162			Indicating lights	2 pair per valve	open/closed	HS57-162 (DIV III)	Control room
HV57-105			Indicating lights	2 pair per valve	open/closed	HS57-105 (DIV II)	Control room
FV-D0-101A			Indicating lights	2 pair per valve	open/closed	HS-DO-101A (DIV III)	Control room
FV-DO-101B			Indicating lights	2 pair per valve	open/closed	HS-DO-101B (DIV IV)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV57-104			Indicating lights	l pair per valve	open/closed	HS57-104 (DIV II)	Control room
SV57-181			Indicating lights	l pair per valve	open/closed	HS57-181 (DIV II)	Control room
HV57-109			Indicating lights	l pair per valve	open/closed	HS57-109 (DIV II)	Control room
HV57-135			Indicating lights	l pair per valve	open/closed	HS57-135 (DIV II)	Control room
HV57-117			Indicating lights	l pair per valve	open/closed	HS57-117 (DIV I)	Control room
HV57-147			Indicating lights	l pair per valve	open/closed	HS57-147 (DIV II)	Control room
HV57-118			Indicating lights	l pair per valve	open/closed	HS57-118 (DIV I)	Control room
SV57-139			Indicating lights	l pair per valve	open/closed	HS57-139 (DIV I)	Control room
HV57-112			Indicating lights	l pair per valve	open/closed	HS57-112 (DIV I)	Control room
HV57-166			Indicating lights	2 pair per valve	open/closed	HS57-166 (DIV III)	Control room
HV57-169			Indicating lights	2 pair per valve	open/closed	HS57-169 (DIV IV)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV57-115			Indicating lights	1 pair per valve	open/closed	HS57-115 (DIV I)	Control room
HV59-129A			Indicating lights	1 pair per valve	open/closed	HS59-129A (DIV I)	Control room
HV59-129B			Indicating lights	I pair per valve	open/closed	HS59-129B (DIV II)	Control room
HV59-151A			Indicating lights	I pair per valve	open/closed	HS59-151A (DIV III)	Control room
HV59-151B			Indicating lights	I pair per valve	open/closed	HS59-151B (DIV IV)	Control room
HV59-131			Indicating lights	I pair per valve	open/closed	HS59-131 (DIV II)	Control room
HV59-101			Indicating lights	I pair per valve	open/closed	HS59-101 (DIV I)	Control room
HV59-102			Indicating lights	I pair per valve	open/closed	HS59-102 (DIV II)	Control room
XV59-141A, B, C, D, E			Indicating lights	I pair per valve	open/closed	Valve control monitor (10-C607)	Control room
HV59-135			Indicating lights	I pair per valve	open/closed	HS59-135 (DIV II)	Control room
HV61-102 (deleted on Unit 2) 112, 132			Indicating lights	I pair per valve	open/closed	HS61-112 (DIV I)	Control room
HV61-110			Indicating lights	2 pair per valve	open/closed	HS61-110 (DIV I)	Control room
HV61-130			Indicating lights	2 pair per valve	open/closed	HS61-130 (DIV IV)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV61-111			Indicating lights	2 pair per valve	open/closed	HS61-111 (DIV II)	Control room
HV61-131			Indicating lights	2 pair per valve	open/closed	HS61-131 (DIV II)	Control room
HV87-128, 129			Indicating lights	I pair per valve	open/closed	HS87-128 (DIV II)	Control room
HV87-122, 123			Indicating lights	I pair per valve	open/closed	HS87-122 (DIV II)	Control room
HV87-120A, 121A			Indicating lights	I pair per valve	open/closed	HSS87-121A (DIV I)	Control room
HV87-120B, 121B			Indicating lights	I pair per valve	open/closed	HSS87-121B (DIV I	Control room
Safety Relief Valves Position indication	D10	2					
PSV41-F013A			Indicating light	I per valve	open/closed	HS41-113A (Non-Div)	Control room
PSV41-F013B			Indicating light	I per valve	open/closed	HS41-113B (Non-Div)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
PSV41-F013C			Indicating light	I per valve	open/closed	HS41-113C (Non-Div)	Control room
PSV41-F013D			Indicating light	I per valve	open/closed	HS41-113D (Non-Div)	Control room
PSV41-F013E			Indicating light	I per valve	open/closed	HS41-113E (Non-Div)	Control room
PSV41-F013F			Indicating light	I per valve	open/closed	HS41-113F (Non-Div)	Control room
PSV41-F013G			Indicating light	I per valve	open/closed	HS41-113G (Non-Div)	Control room
PSV41-F013H			Indicating light	I per valve	open/closed	HS41-113H (Non-Div)	Control room
PSV41-F013J			Indicating light	I per valve	open/closed	HS41-113J (Non-Div)	Control room
PSV41-F013K			Indicating light	I per valve	open/closed	HS41-113K (Non-Div)	Control room
PSV41-F013L			Indicating light	I per valve	open/closed	HS41-113L (Non-Div)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
PSV41-F013M			Indicating light	I per valve	open/closed	HS41-113M (Non-Div)	Control room
PSV41-F013N			Indicating light	I per valve	open/closed	HS41-113N (Non-Div)	Control room
PSV41-F013S			Indicating light	I per valve	open/closed	HS41-113S (Non-Div)	Control room
Main steam bypass valve position	D26	3	HMI Unit 1 only		0% - 100% open	XI-031-102 (Non-Div) XI-031-103 (Non-Div)	Control room
			Indicator Unit 2 only	I per valve			
						Z101-205A (Non-Div)	Control room
						Z101-205B (Non-Div)	Control room
						Z101-205C (Non-Div)	Control room
						Z101-205D (Non-Div)	Control room
						Z101-205E (Non-Div)	Control room
						Z101-205F (Non-Div)	Control room
						Z101-205G (Non-Div)	Control room
						Z101-205H (Non-Div)	Control room
						Z101-205J (Non-Div)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Main feedwater flow	D1	3	Indicator	3 (1 per train)	0 – 7x10 ^{6} lbs/hr	FI06-1R604A (Non-Div) FI06-1R604B (Non-Div) FI06-1R604C (Non-Div)	Control room
			Recorder	1 (total flow)	0 – 18x10^6 lbs/hr	FR41-1R607 (Non-Div)	Control room
CST Level	D2	3	Recorder	1	0-45 ft H_2O (Unit 1 only) 0-45 ft H_2O (Unit 2 only)	LR08-102 (Pen 1) (Non-Div) LR08-202 (Pen 1) (Non-Div)	Control room
Condenser Hotwell Level	D27	3	Recorder	1	36 – 56" H_2O	LR05-101 (Pens 1 & 2) (Non-Div)	Control room
Condenser Pressure	D28	3	Indicator	1 per shell	20-30" Hg Vac	PI05-101A (Non-Div) PI05-101B (Non-Div) PI05-101C (Non-Div)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Circulating Water Pump Discharge Pressure	D29	3	Indicator	1 per pump	0 – 100 psig	PI09-117A (Non-Div)	Control room
						PI09-117B (Non-Div)	
						PI09-117C (Non-Div)	
						PI09-117D (Non-Div)	
HPCI Flow	D14	2	Indicator	1	0 – 6000 gpm	FI55-1R600-1 (Div II)	Control room
ESW Flow	D22	2	Indicator	1 per train	0 – 6,000 gpm	FI11-013A (Div I)	Control room
						FI11-013B (Div II)	Control room
ESW Temperature	D22	2	Indicator	1 per train	0 – 200°F	TI11-007A (Div I)	Control room
						TI11-007B (Div II)	Control room
RHRSW System Flow	D22	2	Indicator	1 per train	0 – 12,000 gpm	FI51-1R602A (Div I)	Control room
						FI51-1R602B (Div II)	Control room
RHRSW System Temperature	D21	2	Indicator	1 per train	0 – 200°F	TI51-105A (Div I)	Control room
						TI51-105B (Div II)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
LPCI, RHR System Flow Drywell Spray Flow, Suppression Chamber Flow	D3, 8, 16, 19	2	Indicator	1 per train	0 – 12,000 gpm	FI51-1R603A (Div I)	Control room
						FI51-1R603B (Div II)	Control room
						FI51-1R603C (Div III)	Control room
						FI51-1R603D (Div IV)	Control room
RHR Heat Exchanger outlet temperature	D20	2	Indicator	1 per train	0°F-350°F	TI51-127A (Div I)	Control room
						TI51-127B (Div II)	Control room
RCIC Pump Discharge Flow	D13	2	Indicator	1	0 – 7 gpm	FI49-1R600-1 (Div 1)	Control room
Core Spray Flow	D15	2	Indicator	1 per train	0 – 8, 800 gpm	FI52-1R601A (Div I)	Control room
						FI52-1R601B (Div I)	Control room
Equipment Drain Collection Tank	D23	3	Indicator	1	0-10 ft H_2O	LI62-010 (Non-Div)	Radwaste Control room
Floor Drain Collection Tank	D23	3	Indicator	1	0-15 ft H_2O	LI63-010 (Non-Div)	Radwaste Control room
Chemical Waste Collection Tank	D23	3	Indicator	1	0-12 ft	LI64-001 (Non-Div)	Radwaste Control room
SLCS Storage Tank Level	D18	3	Indicator	1	0 – 5000 gal	LI48-1R601	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Long Term N2 Supply to ADS System	D25	2	Indicator	2	0 – 150 psig	PI59-103A (Div III)	Control room
						PI59-103B (Div IV)	Control room
CRD Hydraulics Charging Water Pressure	D25	2[(4)]	Indicator	1	0 – 1600 psig	PIS46-1N600 (Non-Div)	Control room
Safety-related 4.16 kV Bus Voltage	D25	2	Indicator	1 per bus	0 – 5.25 kV	V/115-2 (Div I)	Control room
						V/116-2 (Div II)	Control room
						V/117-2 (Div III)	Control room
						V/118-2 (Div IV)	Control room
Safety-related 4.16 kV Frequency	D25	2	Indicator	1 per bus	55-60-65 Hz	F/AG501-2 (Div I)	Control room
						F/BG501-2 (Div II)	Control room
						F/CG501-2 (Div III)	Control room
						F/DG501-2 (Div IV)	Control room
125/250 V dc Class IE Power System Voltage	D25	2	Indicator	1 per bus	0-300 V	V/AD101 (Div I)	Control room
						V/BD101 (Div I)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
125/250 V dc Class IE Power System Current	D25	2	Indicator	1 per bus	1200-0-1200 A	A/AD101 (Div I)	Control room
						A/BD101 (Div II)	Control room
125 V dc Class IE Power System Voltage	D25	2	Indicator	1 per bus	0-150 V	V/CD101 (Div III)	Control room
						V/DD101 (Div IV)	Control room
125V dc Class IE Power System Current	D25	2	Indicator	1 per bus	500-0-500 A	A/CD101 (Div III)	Control room
						A/DD101 (Div IV)	Control room
Emergency Ventilation Damper Position	D24	2					
HV76-107			Indicating lights	1 pair per valve	Open/closed	HV76-107 (Div I)	Local Panel
HV76-108			Indicating lights	1 pair per valve	Open/closed	HV76-108 (Div I)	Local Panel
HV76-141 HV76-157			Indicating lights	1 per valve	Open/closed	HV76-141 HV76-157 (Div I)	Local Panel
HV76-142 HV76-158			Indicating lights	1 per valve	Open/closed	HV76-142 HV76-158 (Div II)	Local Panel

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV76-117 HV76-167			Indicating lights	1 per valve	Open/closed	HV76-117 HV76-167 (Div I)	Local Panel
HV76-118 HV76-168			Indicating lights	1 pair per valve	Open/closed	HV76-118 HV76-168 (Div II)	Local Panel
HD78-002A HD78-009A HV78-010A			Indicating lights	1 pair per valve	Open/closed	HS78-010A (Div III)	Control room(5)
HD78-002B HD78-009B HV78-010B			Indicating lights	1 pair per valve	Open/closed	HS78-010B (Div IV)	Control room(5)
HV78-20A			Indicating lights	1 pair per valve	Open/closed	HSS78-017A (Div I)	Control room
HV78-20B			Indicating lights	1 pair per valve	Open/closed	HSS78-017B (Div II)	Control room
HV78-020C HV78-021A			Indicating lights	1 pair per valve	Open/closed	HSS78-017C (Div III)	Control room
HV78-020D HV78-021B			Indicating lights	1 pair per valve	Open/closed	HSS78-017D (Div IV)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV76-019			Indicating lights	1 pair per valve	Open/closed	HV76-019 (Div I)	Local Panel(5)
HV76-020			Indicating lights	1 pair per valve	Open/closed	HV76-020 (Div II)	Local Panel(5)
HV76-151			Indicating lights	1 pair per valve	Open/closed	HV76-151 (Div I)	Local Panel
HV76-152			Indicating lights	1 pair per valve	Open/closed	HV76-152 (Div II)	Local Panel
HV76-159			Indicating lights	1 pair per valve	Open/closed	HV76-159 (Div I)	Local Panel
HV76-160			Indicating lights	1 pair per valve	Open/closed	HV76-160 (Div II)	Local Panel
HD76-183A HD76-193A			Indicating lights	1 pair per valve	Open/closed	HS76-193A (Div I)	Control room(5)
HD76-183B HD76-193B			Indicating lights	1 pair per valve	Open/closed	HS76-193B (Div II)	Control room(5)
HV76-012A HV76-011A			Indicating lights	1 pair per valve	Open/closed	HS76-013A (Div I)	Control room(5)
HV76-012B HV76-011B			Indicating lights	1 pair per valve	Open/closed	HD76-013B (Div II)	Control room(5)

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV76-030			Indicating lights	1 pair per valve	Open/closed	HS76-030 (Div I)	Control room
HV76-031			Indicating lights	1 pair per valve	Open/closed	HS76-031 (Div II)	Control room
HD76-340A PD-C76-341A			Indicating lights	1 pair per valve	Open/closed	HS76-040A (Div I)	Control room(5)
HD76-340B PD-C76-341B			Indicating lights	1 pair per valve	Open/closed	HV76-040B (Div II)	Control room(5)
HV76-196			Indicating lights	1 pair per valve	Open/closed	HV76-196 (Div I)	Local Panel(5)
HV76-197			Indicating lights	1 pair per valve	Open/closed	HV76-197 (Div II)	Local Panel(5)
HD78-026A HD78-027A			Indicating lights	1 pair per valve	Open/closed	HS78-026A (Div III)	Control room
HD78-026B HD78-027B			Indicating lights	1 pair per valve	Open/closed	HS78-026B (Div IV)	Control room
HV78-052A			Indicating lights	1 pair per valve	Open/closed	HV78-052A (Div III)	Control room
HV78-053A			Indicating lights	1 pair per valve	Open/closed	HV78-053A (Div III)	Control room
HV78-053B			Indicating lights	1 pair per valve	Open/closed	HD78-053B (Div IV)	Local panel
HV78-052B			Indicating lights	1 pair per valve	Open/closed	HV78-052B (Div IV)	Local panel
HV78-057A			Indicating lights	1 pair per valve	Open/closed	HV78-057A (Div III)	Local Panel

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
HV78-057B			Indicating lights	1 pair per valve	Open/closed	HV78-057B (Div IV)	Local panel
HV76-109			Indicating lights	1 pair per valve	Open/closed	HV76-109 (Div I)	Local panel
HV76-110			Indicating lights	1 pair per valve	Open/closed	HV76-110 (Div II)	Local panel
HD78-059A			Indicating lights	1 pair per valve	Open/closed	HD78-059A (Div III)	Control room
HD78-059B			Indicating lights	1 pair per valve	Open/closed	HD78-059B (Div IV)	Control room
HD78-060A			Indicating lights	1 pair per valve	Open/closed	HD78-060A (Div III)	Control room
HD78-060B			Indicating lights	1 pair per valve	Open/closed	HD78-060B (Div IV)	Control room
HV78-071A			Indicating lights	1 pair per valve	Open/closed	HV78-071A (Div III)	Local panel
HV78-071B			Indicating lights	1 pair per valve	Open/closed	HV78-071B (Div IV)	Local panel
Primary Containment Area Radiation – High Range	E1, C5	1	Recorder	2	1 to 10^8 R/hr	RR26-191A (Div III)	Control room
						RR26-191B (Div II)	
Radiation Exposure Rate:	E3	3					
o Radwaste Enclosure Hallway, el 217'			Recorder (Multipoint)	1	10^{-2} to 10^4 mR/hr	RR-M1-0R601 (Non-Div)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
o Turbine Area/Operating Floor/OSC, el 269'			Recorder (Multipoint)	1	10^{-2} to 10^{4} mR/hr	RR-M1-0R601 (Non-Div)	Control room
o Main Control Room, el 269'			Recorder (Multipoint)	1	10^{-2} to 10^{4} mR/hr	RR-M1-0R601 (Non-Div)	Control room
o North Stack Instrument Room, Reactor Enclosure el 411'			CRT	NA	10^{-2} to 10^{4} mR/hr	ERFDS CRT (Non-Div)	Control room
o PASS Station Control Structure el 217'			CRT	NA	4.0 to 4 x 10^{3} mR/hr	ERFDS CRT (Non-Div)	Control room
Radiation Level in Circulating Primary Coolant:	C1	3					
o Main Steam Line Radiation			Recorder	1	1 to 10^{6} mR/hr	RR41-1R603 (Non-Div)	Control room
o SJAE Radiation			Recorder	1	1 to 10^{6} mR/hr	RR26-1R601 (Non-Div)	Control room
Noble Gases & Vent Flow Rate:	E4, C13, C15	2					
o North Stack Radioactivity Concentration Wide Range Gas Monitor			Recorder	1	10^{-7} to 10^{+5} μCi/cc	RR26-076 (Div IV)	Control room
o North Stack Flow			Indicator	1	500 - 3000 scfm Accident Range 0 – 664,000 scfm Normal Range	RIX26-076 (Div IV)	Control room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Particulates and Halogens o North Stack Radioactivity Concentration with Onsite Analysis	E5	3	Filter/Sample Cartridge	1/module	Up to $10^2 \mu Ci/cc$	OAF955 OBF955 OCF955	North Stack Rad Monitor
Airborne Radiohalogens and Particulates (Portable Sampling with Onsite Analysis)	E7	3	Portable Air Samples PING Particulate Monitor Spectroscopic System		10^{-12} to $10^{-3} \mu Ci/cc$	-	HP Field Office
Plant and Environs Radiation (Portable Instrumentation)	E8	3	Survey meters		1 mR/hr – 20,000 R/hr	-	-
Plant and Environs Radioactivity (Portable Instrumentation)	E9	3	Portable Air Samples PING Particulate Spectroscopic System		10^{-12} to $10^{-3} \mu Ci/cc$	-	HP Field Office
Wind Direction	E10	3	Recorder	2	0° - 540°	(SX-410 Channel 2/ PMS Point T1DRIHA), (SX-410 Channel 0/ PMS Point T1DRLHA) or (SX-410 Channel 20/ PMS Point T2DRUHA), (SX-410 Channel 15/ PMS Point T2DRIHA)	Control Room

Table 7.5-3 (Cont'd)

VARIABLES	TYPE/ ITEM # (1)	CATEGORY (1)	INDICATION TYPE	QTY	INSTRUMENT RANGE	INSTRUMENT NO. (DIV)	LOCATION
Wind Speed	E11	3	Recorder	2	0 – 100 mph	(SX-410 Channel 3 PMS Point T1SPIHA), (SX-410 Channel 1/ PMS Point T1SPLHA) or (SX-41- Channel 21/ PMS Point T2SPUHA, SX-410 Channel 16/ PMS Point T2SPIHA)	Control room
Estimation of Atmospheric Stability	E12	3	Recorder	2	-10°F to 20°F	XR-MS-091 or XR-MS-092	Control room

(1) For definitions, refer to Section 7.5.2.5.1.1.2.3.

(2) Above instrument zero.

(3) 61 inches above top of active fuel with same instrument zero as the wide range level monitor.

(4) This variable is provided with non-Class 1E power.

(5) For emergency ventilation damper position, ERFDS monitoring is also provided aside from the position indicating lights.

(6) Grab samples of RCS soluble boron can be obtained from the postaccident monitoring system (Section 11.5.5).

Table 7.5-4

RELATIVE NEUTRON FLUX VERSUS TIME[1]

Leakage rate, gpm (ramp rate, c/min)[2]

Percent of power	1(0.03)		5(0.15)		20(0.60)	
	Σ	Δ	Σ	Δ	Σ	Δ
$5x10^{-8}$	-555	500	-111	100	-27.75	25
$5x10^{-7}$	-55	50	-11	10	-2.75	2.5
$5x10^{-6}$	-5	5	-1	1	-0.25	0.25
$5x10^{-5}$	0		0		0	
$5x10^{-4}$	0.8	0.8	0.36	0.36	0.18	0.18
$5x10^{-3}$	1.33	0.53	0.51	0.15	0.25	0.07
$5x10^{-2}$	1.59	0.26	0.62	0.11	0.31	0.06
$5x10^{-1}$	1.80	0.21	0.72	0.10	0.36	0.05
$5x10^{0}$	1.89	0.09	0.80	0.08	0.40	0.04

(1) Shutdown flux = $5x10^{-8}$% of power.

(2) Σ = total number of hours; Δ = hours for neutron flux to increase by one decade.

Table 7.5-5

PLANT VARIABLES FOR ACCIDENT MONITORING

Type A Variables	Position
A1 Oxygen and Hydrogen Concentration	LGS meets regulatory guide criteria for all Type A variables.
A2 RPV Pressure	
A3 RPV Water Level	Criteria met for overlapping range of -350" - +60" (Section 7.5.2.5.1.1.2.4.2)
A4 Suppression Pool Water Temperature	
A5 Suppression Pool Water Level	
A6 Drywell Pressure	
Type B Variables	
B1 Neutron Flux	Classified as Category 2 (Section 7.5.2.5.1.1.2.4.1)
B2 Control Rod Position	Criteria met
B3 RCS soluble boron concentration (Grab sample)	Criteria met
B4 Coolant level in reactor	Criteria met for overlapping range of -350" - +60" (Section 7.5.2.5.1.1.2.4.2)
B5 BWR core thermocouples	Not implemented. Reactor level instrumentation adequate with EPGs
B6 RCS pressure	Criteria met (Section 7.5.2.5.1.1.2.4.2)
B7 Drywell pressure	Criteria met

Table 7.5-5 (Cont'd)

Type B Variables (Cont'd)

B8 Drywell sump level	Criteria for Category 3 met (Section 7.5.2.5.1.1.2.4.3)
B9 Primary Containment Pressure	Criteria met
B10 Primary Containment Isolation Valve Position	Criteria met, redundant indication is not required on redundant isolation valves. Exclude check valves.

Type C Variables

C1 Radioactivity concentration or radiation level in circulating primary coolant	Implemented as Category 3 (Section 7.5.2.5.1.1.2.4.4)
C2 Analysis of primary coolant (gamma spectrum)	Criteria met
C3 BWR core thermocouples	Will not implement (See B5)
C4 RCS pressure	Criteria met
C5 Primary containment area Radiation	Criteria met except at extreme drywell temperature (See E1 and Section 7.5.2.5.1.1.2.4.14)
C6 Drywell drain sumps level	Criteria met for Category 3 (Section 7.5.2.5.1.1.2.4.3)
C7 Suppression pool water level	Criteria met
C8 Drywell pressure	Criteria met
C9 RCS pressure	Criteria met
C10 Primary containment pressure	Criteria met
C11 Containment and drywell H_2 concentration	Criteria met
C12 Containment and drywell O_2 concentration	Criteria met
C13 Containment effluent radioactivity - noble gases (from identified release points including SGTS vent)	Criteria met (See E4)

Table 7.5-5 (Cont'd)

Type C Variables (Cont'd)

C14 Radiation exposure rate (inside buildings or areas in direct contact with primary containment where penetrations are located)	Not implemented (See E2, E3 and Section 7.5.2.5.1.1.2.4.5)
C15 Effluent radioactivity - noble gases (from buildings as indicated above)	Criteria met (See E4)

Type D Variables

D1 Main feedwater flow	Criteria met
D2 CST level	Criteria met
D3 Suppression spray flow	Current design is adequate (Section 7.5.2.5.1.1.2.4.6)
D4 Drywell pressure	Criteria met
D5 Suppression pool water level	Criteria met
D6 Suppression pool water temperature	Criteria met
D7 Drywell atmosphere temperature	Criteria met
D8 Drywell spray flow	Current design is adequate (Section 7.5.2.5.1.1.2.4.6)
D9	Deleted
D10 SRV position	Criteria met
D11 Isolation condenser system shell-side water level	N/A
D12 Isolation condenser system valve position	N/A
D13 RCIC flow	Criteria met (Section 7.5.2.5.1.1.2.4.7)

Table 7.5-5 (Cont'd)

Type D Variables (Cont'd)

D14 HPCI flow	Criteria met (Section 7.5.2.5.1.1.2.4.7)
D15 Core spray system flow	Criteria met (Section 7.5.2.5.1.1.2.4.7)
D16 LPCI system flow	Criteria met (Section 7.5.2.5.1.1.2.4.7)
D17 SLCS flow	Alternate criteria met for Category 3 (Section 7.5.2.5.1.1.2.4.7)
D18 SLCS storage tank level	Criteria met for category 3 (Section 7.5.2.5.1.1.2.4.8)
D19 RHR system flow	Criteria met
D20 RHR heat exchanger outlet temperature	Criteria met
D21 Cooling water temperature to ESF components	Criteria met for main system flow
D22 Cooling water flow to ESF Components	Criteria met for main system flow
D23 High radioactivity liquid tank level	Criteria met
D24 Emergency ventilation damper position	Criteria met for damper actuated under accident conditions whose failure could result in radioactive release.
D25 Status of standby power and other energy sources	Criteria met, onsite sources only
D26 Main steam bypass valve position	Added to Regulatory Guide 1.97 list (Section 7.5.2.5.1.1.2.4.9)
D27 Condenser hotwell level	Added to Regulatory Guide 1.97 list (Section 7.5.2.5.1.1.2.4.9)
D28 Condenser pressure	Added to Regulatory Guide 1.97 list (Section 7.5.2.5.1.1.2.4.9)
D29 Circulating water pump discharge pressure	Added to Regulatory Guide 1.97 list (Section 7.5.2.5.1.1.2.4.9)

Table 7.5-5 (Cont'd)

<u>Type D Variables (Cont'd)</u>	
D30 Reactor recirculation pump Flow	Added to Regulatory Guide 1.97 list (Section 7.5.2.5.1.1.2.4.9)
<u>Type E Variables</u>	
E1 Primary containment area radiation - high range	Criteria met except at extreme drywell temperature (See C5 and Section 7.5.2.5.1.1.2.4.14)
E2 Reactor building or secondary containment area radiation	Not implemented (See C14, E3 and Section 7.5.2.5.1.1.2.4.10)
E3 Radiation exposure rate (inside buildings or areas where access is required to service equipment important to safety)	Implemented as Category 3 (See C14, E2 and Section 7.5.2.5.1.1.2.4.11)
E4 Noble gases and vent flow rate	Criteria met
E5 Particulates and halogens	Criteria met
E6 Radiation exposure meters	Deleted per NRC errata July 1981
E7 Airborne radiohalogens and particulates	Criteria met
E8 Plant and environs radiation	Criteria met (Section 7.5.2.5.1.1.2.4.12)
E9 Plant and environs radioactivity (MCA)	Criteria met
E10 Wind direction	Criteria met
E11 Wind speed	Criteria met
E12 Estimation of atmospheric stability	Criteria met
E13 Primary coolant and sump (grab sample)	Criteria met for primary coolant and suppression pool (Section 7.5.2.5.1.1.2.4.13)
E14 Containment air sample	Criteria met

W/O #: ______________________
ACT COMPLETED BY: ______________
CREM/M&TE by: ______________
TEST DATE/TIME: ______________
GRADE: ______________

1
YELLOW

EXELON GENERATION
LIMERICK GENERATING STATION

ST-6-043-320-1 DAILY JET PUMP OPERABILITY VERIFICATION FOR TWO RECIRCULATION LOOP OPERATION

Test Freq: Daily In OPCON 1 - **OR** - Initiating Events:
Tech Spec: 4.4.1.2.a

A. On Startup prior to thermal power exceeding 25% rated
B. Other
   1. Reason ______________
   2. A/R No. ______________

TEST RESULTS: (Circle **SAT** or **UNSAT** - Below)

SAT - All Asterisk (*) steps completed satisfactorily.

UNSAT - Test Results of one **OR** more Asterisk (*) steps completed unsatisfactorily.

Performed by:..(Sign/Date/Time) ______________ ________ ________

Reviewed by (SSV)...(Sign/Date) ______________ ________

IMMEDIATE NOTIFICATION OF OPERATIONS SHIFT MANAGEMENT (**UNSAT** Results Only)

Shift Supervision:.. (Sign) ______________

(Date/Time) ________ ________

Corrective Action (if required) ....(EDT **or** A/R - Number)______________

ADDITIONAL ACTION/TEST COMMENTS (User may add additional pages, if necessary)

____

____

____

____

____

Person making entry ....(Sign/Date) ______________ ________

PLACEKEEP/INITIALS

1.0 PURPOSE

To verify each of the jet pumps OPERABLE.

2.0 PREREQUISITES

2.1 Correct revision of procedure is being used. [ ]

2.2 Procedure printed on yellow paper. [ ]

2.3 Reactor operating in OPCON 1. [ ]

2.4 **IF** performance of this test has **not** been directed by ON-100
THEN Recirc flow
AND Reactor power must remain constant during this test with Recirc. loop flow mismatch maintained within the limits of Tech Spec 3.4.1.3.
CM-1 [ ]

2.5 **No** other testing
OR plant condition which could interfere with this test is being performed/present. [ ]

2.6 Briefing performed as required. [ ]

PLACEKEEP/INITIALS

3.0 PRECAUTIONS

3.1 IF a procedural step can **not** be completed
OR any other difficulty is encountered during this test,
THEN a comment shall be entered in the Additional Action/Test Comments section. [ ]

3.2 IF a step denoted as a Tech Spec Requirement, marked with an asterisk(*), can **not** be successfully completed,
THEN Shift Supervision (SSV) shall be notified immediately. [ ]

3.3 Turbulence in the jet pump diffuser causes noise in the differential pressure signal which can result in the indication of spiking of individual jet pump readings. Jet pump differential pressure reading less than Tech Spec minimum requirements
OR higher than Tech Spec maximum requirements should be evaluated by confirming values consistent with other jet pumps
AND evaluating the effects of spiking caused by instrument noise. (Ref. 6.5) [ ]

3.4 IF any abnormalities are observed
THEN **PERFORM** the following:

3.4.1 **DOCUMENT** in the Additional Action/Test Comments section. [ ]

3.4.2 **INFORM** SSV. [ ]

3.5 Instruments used in this procedure to satisfy Tech Spec (*)
OR IST (I) requirements shall be free of EDTs
OR the nature of the deficiency shall be understood
AND determined **not** to impact test results. [ ]

PLACEKEEP/INITIALS

NOTE

It is the responsibility of the person
OR persons performing this test to ensure all blanks/data sheets are correctly
AND completely filled in. [ ]

4.0 PROCEDURE

4.1 PREPARATION

4.1.1 **VERIFY** all prerequisites of Section **2.0** are satisfied. ________

4.1.2 **VERIFY** procedure being performed on Unit 1. ________

4.2 SHIFT PERMISSION TO TEST

4.2.1 **OBTAIN** SSV permission to start test. ________

NOTE

Recirc flow
AND Reactor Power shall remain constant during this test. [ ]

4.2.2 **OBTAIN** PRO permission to start test. ________/________
Date/Time

PLACEKEEP/INITIALS

4.3 PUMP TEST

4.3.1 **PERFORM** the following to determine if indicated value of Recirc Loop A flow is within 10% of the loop flow values on the established pump speed - loop flow characteristic curves:

1. **RECORD** Recirc Pp A Flow
   AND CIRCLE device used below
   AND on Attachment 1:

   Device: B037

   Flow = ________ Mlb/hr

   Device: FR-43-1R614, Point 1

   Flow = _____ gpm x $\frac{1\ \text{Mlb/hr}}{2691\ \text{gpm}}$

   Flow =_______ Mlb/hr ________

2. **REFER TO** Attachment 1
   AND PLOT Recirc Pp A Flow vs Speed point. ________

3. **VERIFY** plotted Loop A Recirc Pp Flow vs Recirc Pp Speed point is within box of Attachment 1. ________

PLACEKEEP/INITIALS

4.3.2 **PERFORM** the following to determine if indicated value of Recirc Loop B flow is within 10% of the loop flow values on the established pump speed - loop flow characteristic curves:

1. **RECORD** Recirc Pp B Flow
AND CIRCLE device used below
AND on Attachment 2:

Device: B039

Flow = ______ Mlb/hr

Device: FR-43-1R614, Point 2

Flow = _____ gpm x $\frac{1 \text{ Mlb/hr}}{2691 \text{ gpm}}$

Flow =_______ Mlb/hr ________

2. **REFER TO** Attachment 2
AND PLOT Recirc Pp B Flow vs Speed point. ________

3. **VERIFY** plotted Loop B Recirc Pp Flow vs Recirc Pp Speed point is within box of Attachment 2. ________

PLACEKEEP/INITIALS

4.3.3 **PERFORM** the following to determine if the value of total core flow is within 10% of the established Total Core Flow value derived from Recirc Loop Flow measurements:

1. **CALCULATE** Recirc Total Flow using Recirc Pp Flow:

   RTF = RPFA + RPFB

   RTF = Recirc Total Flow
   RPFA = Recirc Pp Flow A (step 4.3.1.1)
   RPFB = Recirc Pp Flow B (step 4.3.2.1)

   RTF = ________ Mlb/hr + _________ Mlb/hr

   RTF = _____________ Mlb/hr _________

2. **RECORD** indicated Total Core Flow
   AND **CIRCLE** device used on Attachment 3. _________

3. **REFER TO** Attachment 3
   AND **PLOT** Total Core Flow vs Recirc Total Flow point. _________

4. **VERIFY** plotted Total Core Flow vs Recirc Total Flow point is within box of Attachment 3. _________

4.3.4 **PERFORM** the following to determine the indicated Throat-to-Lower Plenum Differential Pressure of any individual Jet Pp is within 10% of the established patterns for Loop A:

1. **RECORD** the indicated Throat-to-Lower Plenum Differential Pressure for each Jet Pp
   AND **CIRCLE** device used on Attachment 4. _________

2. **RECORD** Recirc Pp Flow on Attachment 4. _________

3. **REFER TO** Attachment 4
   PLOT Loop A Jet Pp Differential Pressure vs Recirc Pp A Flow points for each Jet Pp. _________

4. **VERIFY** all plotted Loop A Jet Pp Differential Pressure vs Recirc Pp A Flow are within box of Attachment 4. _________

PLACEKEEP/INITIALS

4.3.5 **PERFORM** the following to determine the indicated Throat-to-Lower Plenum Differential Pressure of any individual Jet Pp is within 10% of the established patterns for Loop B:

1. **RECORD** the indicated Throat-to-Lower Plenum Differential Pressure for each Jet Pp AND **CIRCLE** device used on Attachment 5. ________

2. **RECORD** Recirc Pp Flow on Attachment 5. ________

3. **REFER TO** Attachment 5
   PLOT Loop B Jet Pp Differential Pressure vs Recirc Pp B Flow points for each Jet Pp. ________

4. **VERIFY** all plotted Loop B Jet Pp Differential Pressure vs Recirc Pp B Flow are within box of Attachment 5. ________

4.4 TEST RESULTS EVALUATION

4.4.1 **VERIFY** at least two of the following step combinations are signed off as satisfactory. **CIRCLE** result for each combination. ________(*)

- Step 4.3.1.3
  AND step 4.3.2.3 — SAT / UNSAT
- Step 4.3.3.4 — SAT / UNSAT
- Step 4.3.4.4
  AND step 4.3.5.4 — SAT / UNSAT

4.5 RETURN TO NORMAL

4.5.1 **NOTIFY** SSV
AND PRO of the following:

- Test completion ________
- Test results ________

4.5.2 **ENSURE** cover sheet is correctly AND completely filled in. ________

5.0 ACCEPTANCE CRITERIA

5.1 Jet pumps are demonstrated OPERABLE by satisfying step 4.4.1.

6.0 REFERENCES

6.1 M-42, P&ID-Nuclear Boiler Vessel Instrumentation

6.2 M-43, P&ID-Reactor Recirculation Pump

6.3 SP-RE-009, Jet Pump Data Collection

6.4 **CM-1** LGS LER 1-86-032, T02413

6.5 SIL 330, Attachment A, BWR 3/4 Jet Pump Performance Monitoring

7.0 TECHNICAL SPECIFICATIONS

7.1 4.4.1.2.a

7.2 3.4.1.3

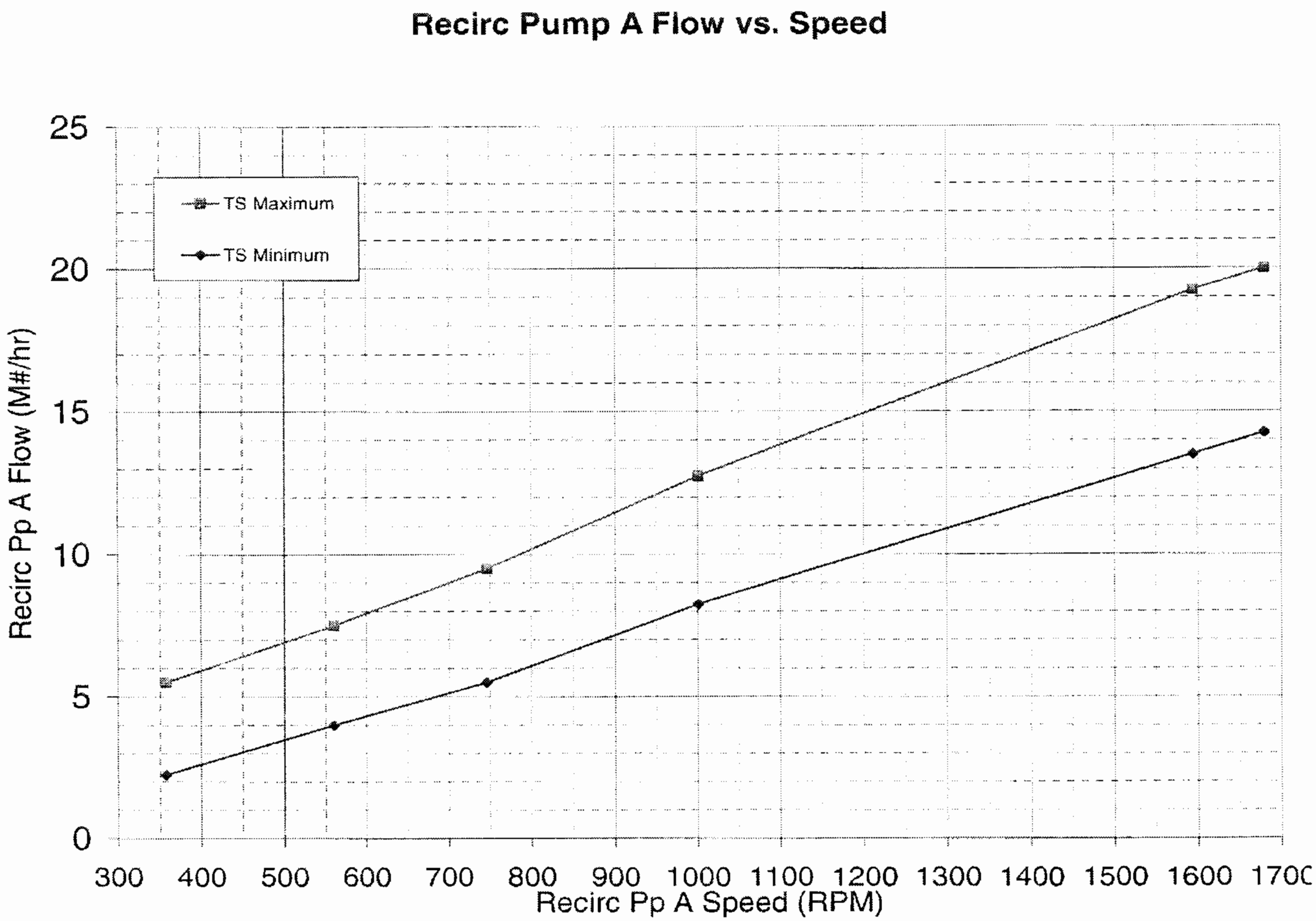
8.0 INTERFACING PROCEDURES

8.1 RT-1-043-232-1, REACTOR RECIRCULATION SYSTEM BASELINE DATA - TWO LOOP OPERATION

1

ATTACHMENT 1
Page 1 of 1

RECIRC PP A SPEED	CIRCLE DEVICE USED
	E1266
	XR-043-101A, Ch 2
	XR-043-101A, Ch 9
RECIRC PP A FLOW	CIRCLE DEVICE USED
	B037
	FR-43-1R614, Pt 1

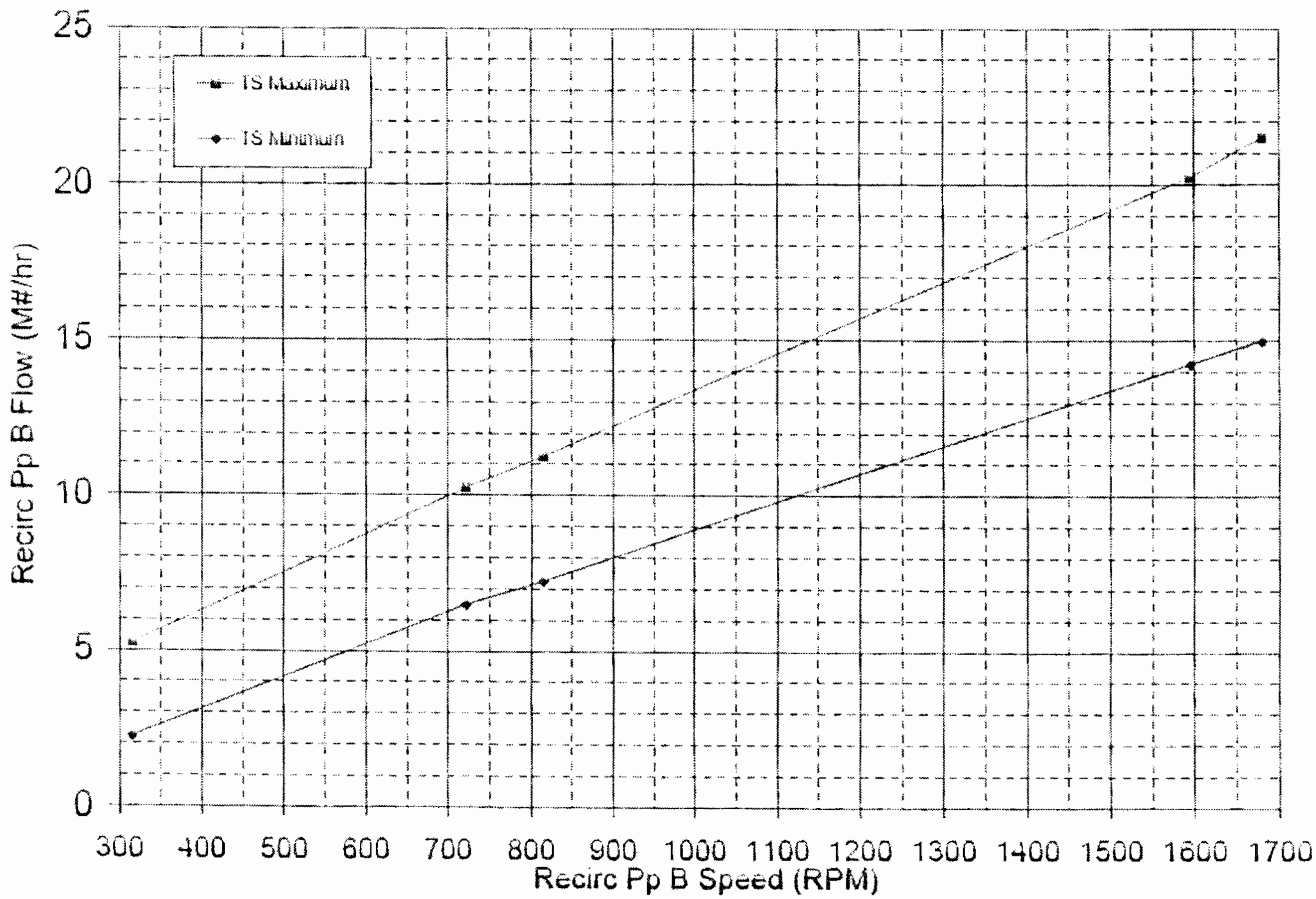


1

ATTACHMENT 2
Page 1 of 1

RECIRC PP B SPEED	CIRCLE DEVICE USED
	E1267
	XR-043-101B, Ch 2
	XR-043-101B, Ch 9
RECIRC PP B FLOW	CIRCLE DEVICE USED
	B039
	FR-43-1R614, Pt 2

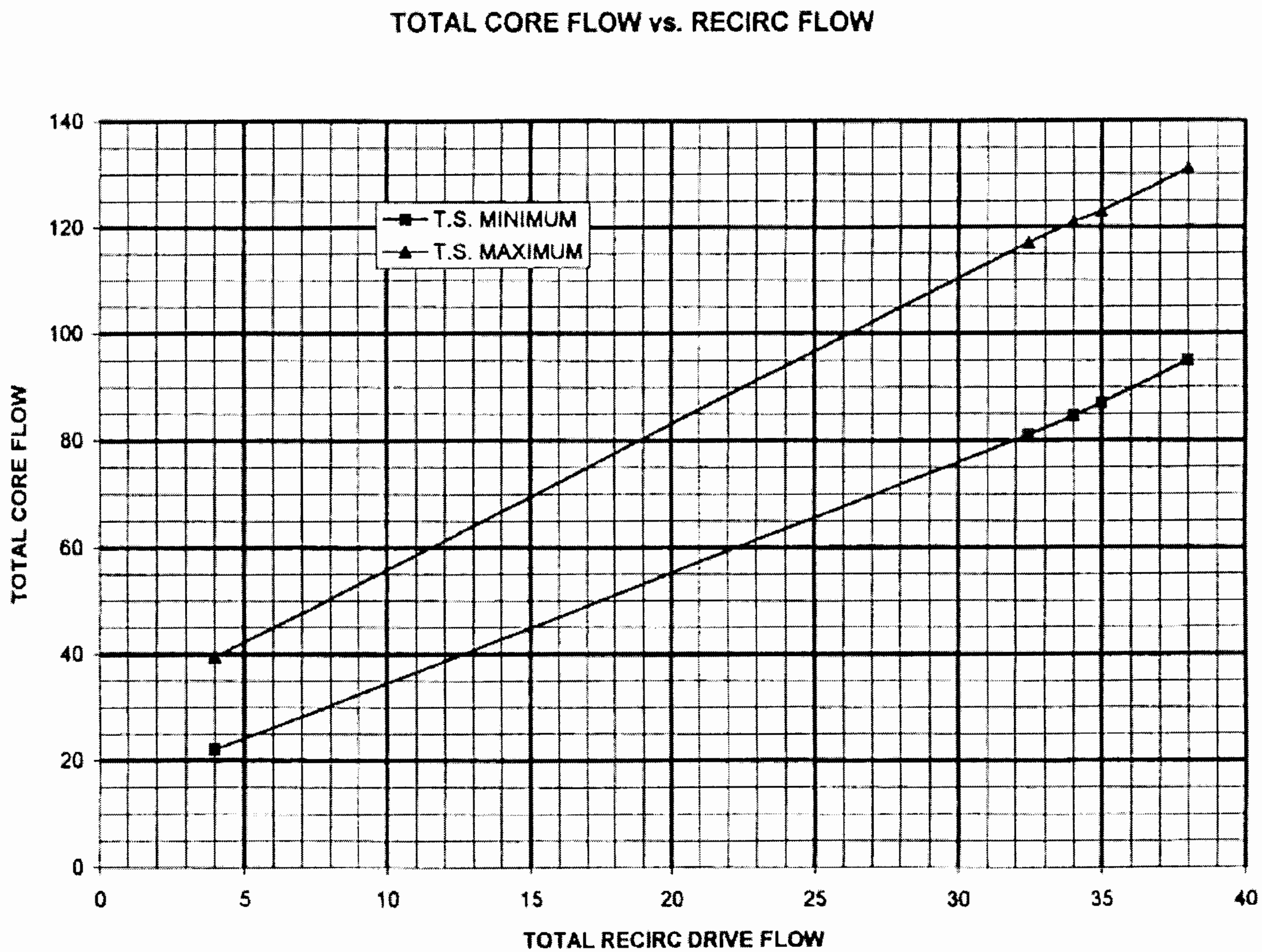
Recirc Pump B Flow vs. Speed



1

ATTACHMENT 3
Page 1 of 1

TOTAL RECIRC FLOW	
	STEP 4.3.3.3
TOTAL CORE FLOW	CIRCLE DEVICE USED
	B018 XR-042-1R613



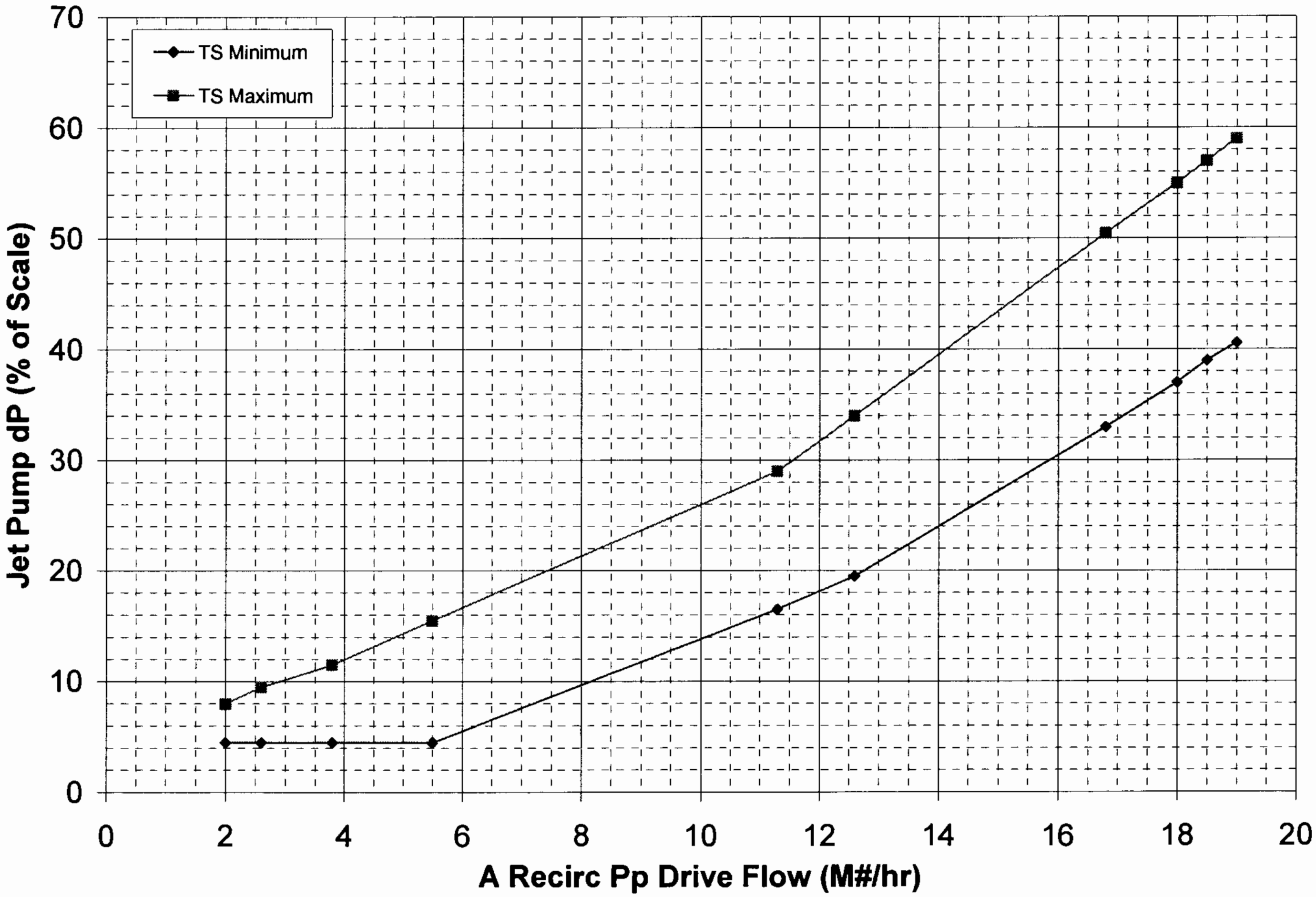
1

ATTACHMENT 4
Page 1 of 1

JET PUMP	PMS	METER	dP %
11	E1037	PDI-42-1R608A	
12	E1097	PDI-42-1R608C	
13	E1104	PDI-42-1R608E	
14	E1235	PDI-42-1R608G	
15	E1254	PDI-42-1R608L	
16	E1248	PDI-42-1R608J	
17	E1256	PDI-42-1R608N	
18	E1258	PDI-42-1R608R	
19	E1260	PDI-42-1R608T	
20	E1264	PDI-42-1R608V	

RECIRC PUMP A FLOW	CIRCLE DEVICE USED
	B037 FR-43-1R614, Pt 1

Jet Pump dP vs. Pump A Flow (Jet Pumps 11 - 20)



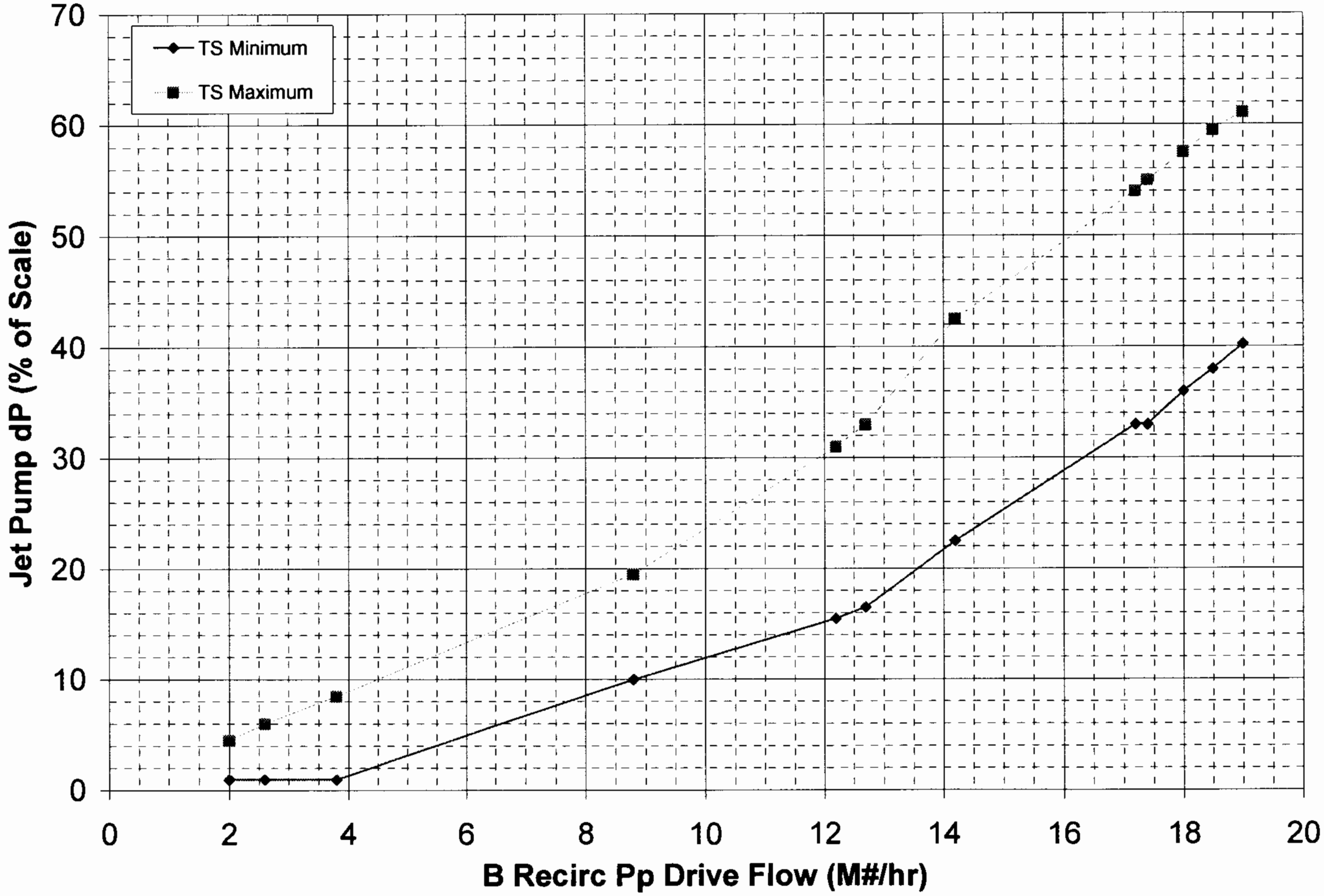
1

ATTACHMENT 5
Page 1 of 1

JET PUMP	PMS	METER	dP %
1	E1084	PDI-42-1R608B	
2	E1103	PDI-42-1R608D	
3	E1105	PDI-42-1R608F	
4	E1247	PDI-42-1R608H	
5	E1249	PDI-42-1R608K	
6	E1255	PDI-42-1R608M	
7	E1257	PDI-42-1R608P	
8	E1259	PDI-42-1R608S	
9	E1263	PDI-42-1R608U	
10	E1265	PDI-42-1R608W	

RECIRC PUMP B FLOW	CIRCLE DEVICE USED
	B039 FR-43-1R614, Pt 2

Jet Pump dP vs. Pump B Flow (Jet Pumps 1 - 10)



PECO ENERGY COMPANY
LIMERICK GENERATING STATION

ON-100 FAILURE OF A JET PUMP

(This is a format change, revision bars are not included)

1.0 **SYMPTOMS**

1.1 Unexplained drop in reactor power.

1.2 Unexplained rise in core flow indication.

1.3 Unexplained rise in recirc drive flow to loop containing defective jet pump.

1.4 Unexplained drop in indicated dP on jet pump sharing riser with defective jet pump.

2.0 **OPERATOR ACTIONS**

2.1 **IF** unexplained drop in Reactor Power.
THEN ENTER OT-104
AND execute concurrently. [ ]

NOTE

1. Depending on the jet pump failure mode, ST-6-043-320-* may pass with a jet pump failure. [ ]
2. Conditional **IF...THEN** steps that are **not** applicable **AND** steps to be skipped per direction of the **IF...THEN** step shall be marked N/A [ ]

2.2 **PERFORM** jet pump operability ST-6-043-320-*, Daily Jet Pump Operability Verification For Two Recirculation Loop Operation, **OR** ST-6-043-321-*, Daily Jet Pump Operability Verification For Single Recirc Loop Operation, as applicable. [ ]

2.2.1 **IF** the acceptance criteria is met for ST-6-043-320-*, **THEN CONTACT** engineering to review plant conditions referencing GE-SIL-330 to confirm jet pump operability. [ ]

CAUTION
HOT SHUTDOWN shall be attained within 12 hours as per Tech. Spec. 3.4.1.2. [ ]

2.3 **IF** jet pump failure is confirmed,
THEN NOTIFY CRS to begin controlled plant shutdown using GP-3. [ ] ⏲

3.0 **REFERENCES**

3.1 TECHNICAL SPECIFICATIONS

3.1.2 3.4.1.2

3.2 INTERFACING PROCEDURES

3.2.1 GP-3, Normal Plant Shutdown

3.2.2 ST-6-043-320-*, Daily Jet Pump Operability Verification For Two Recirculation Loop Operation

3.2.3 ST-6-043-321-*, Daily Jet Pump Operability Verification For Single Recirc Loop Operation

3.3 OTHER

3.3.1 GE SIL No. 330

3.3.2 NRC I.E.Bulletin 80-07

4.0 DISCUSSION

4.1 Core flow indication will rise due to addition of reverse flow. With a displaced jet pump, the flow direction will change from forward to reverse. Since the diffuser pressure tap is still at a lower pressure than the lower plenum, the flow signal is positive.

4.2 Since the 180° bend to nozzle sections of the jet pumps comprise 80-85% of the resistance in the recirc loop, loss of one jet pump removes significant flow resistance. Consequently the recirculation pump will run out on its constant speed characteristic curve, raising its flow by 10% or more with a corresponding reduction in pump head.

4.3 The DP will drop due to the preferential flow of drive water out the displaced jet pump, however, drive flow will still be sufficient to maintain forward flow in the operable pump.

4.4 A section of ST-6-043-320-*, Daily Jet Pump Operability Verification For Two Recirculation Loop Operation, or ST-6-043-321-*, Daily Jet Pump Operability Verification For Single Recirc Loop Operation, is required to be performed following an unexpected change in core flow, recirculation system flow, or established power-core flow relationships by NRC I.E.Bulletin 80-07. The balance of the test should be performed to confirm the jet pump failure and to gather all required information necessary for future analysis of the failure.

4.5 Operation with a displaced jet pump mixer is not part of the licensing basis. Assumptions of a displaced mixer in licensing basic calculations causes a reflood delay and an increase in peak clad temperature, hence, plant shutdown is required.

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Course/Program:	LGS OPERATIONS INITIAL TRAINING	Module/LP ID:	LLOT-0041A
Title:	REACTOR VESSEL INTERNALS ©	Course Code:	Per PIMS Coding
Author:	C. A. FRITZ	Revision/Date:	000/ 07-05-11
Prerequisites:	None	Revision By:	tgf
OPEX Included:	Internal / External / Both / None (circle one)	Est. Duration:	4 /50 Minute Periods

Upon successful completion of this lesson, the trainee shall perform the following using references (as appropriate), and from memory in accordance with the lesson materials.

ENABLING OBJECTIVES

The trainee shall:

Objective #	Objective Description	Pg. #
1.	Identify the relationship between Reactor Vessel Internals and:	
	a. Main Steam System	6
	b. Reactor Recirculation System	6
	c. Reactor Feedwater System	6
	d. High Pressure Coolant Injection (HPCI) System	31
	e. Residual Heat Removal (RHR) System	7,31
	f. Reactor Core Isolation Cooling (RCIC) System	31
	g. Control Rod Drive Hydraulic (CRDH) System	33
	h. Control Rod Drive Mechanism (CRDM)	7
	i. Standby Liquid Control (SBLC) System	31
	j. Reactor Water Cleanup (RWCU) System	7
	k. RPV Instrumentation	5,6,7
	l. Core Spray System	6,31
	m. Automatic Depressurization System (ADS)	36
	n. Nuclear Instrumentation / Traversing Incore Probes (NI/TIP)	7
2.	Identify the effect that a loss or malfunction of Reactor Vessel Internals will have on:	
	a. Reactor water level	23,32,33
	b. Reactor pressure	17

Objective #	Objective Description	Pg. #
	c. Reactor power	17,32
	d. Plant radiation levels	32,33
	e. Offsite radiation levels	32,33
	f. Nuclear Steam Supply Shutoff System (NSSSS)	6
	g. RPV Instrumentation	6
3.	Identify the Reactor Vessel Internal design features which provide for:	
	a. 2/3 core coverage following a DBA LOCA	16
	b. Separation of water flowpaths within the RPV	25
	c. Core orificing	33
	d. Moisture removal from generated steam	13,15
	e. Natural Circulation	14
4.	Identify the operational implications of the following as they apply to Reactor Vessel Internals:	
	a. Brittle fracture	12
	b. Safety limits	36
5.	Identify the effect that a loss or malfunction of the following will have on Reactor Vessel Internals:	34
	a. CRDH	
	b. CRDM	
	c. Reactor Recirculation	
	d. Reactor Feedwater	
	e. RWCU	
	f. RPV Instrumentation	
	g. Core Spray	
	h. HPCI	
	i. RHR	
	j. RCIC	
	k. ADS	
	l. Nuclear Instruments/TIPs	
	m. Main Steam	
6.	Identify the impact of the following on Reactor Vessel Internals and use procedures to correct, control or mitigate the consequences of those conditions:	28
	a. Loss of Coolant Accident (LOCA)	
	b. Overpressure transient	
	c. Control Rod Drop Accident	

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Objective #	Objective Description	Pg. #
	d. Excessive Heatup/Cooldown rate e. Exceeding safety limits	
7.	Identify moisture carryover and steam carryunder, and the problem associated with each.	14
8.	Identify the two types of Fuel Support Pieces, their function, and their relationship to the core structure and control rod assemblies.	28,29
9.	Identify how a central fuel bundle is vertically and horizontally supported.	25-28
10.	Identify the functions of the following major components:	
	a. Core shroud	12
	b. Biological shield	12
	c. Head vent	5
	d. Jet pumps	16
	e. Core differential pressure piping	32
	f. Bottom head drain	7
	g. Incore housing and guide tubes	29
	h. CRD housing support	27
	i. Baffle plate	24
11.	Given a copy of Technical Specifications and various plant conditions: a. Identify any Limiting Conditions for Operation (LCO) relating to the Reactor Vessel Internals. b. For SROs, identify the bases for those LCOs and the actions required for any condition of LCO non-compliance	366, 37

References:

1. LGS UFSAR, Chapter 4, Sections 4.1, 4.4, and 4.6; Chapter 5, Section 5.3
2. Drawings B11-2010-C-003, 004, Reactor Assembly
3. LGS Technical Specifications 3/4.4.6.1
4. GE SIL 644 BWR Steam Dryer Integrity
5. Power Point presentation “LLOT-0041A.ppt".
6. SER 3-09, Unrecognized Reactor Pressure Vessel Head Flange Leak

Materials:

A. Instructor

1. Whiteboard, Markers, and Erasers
2. Power Point Presentation and One-Gun Projector
3. Student Handouts

B. Student

1. Pen/Pencil
2. Notepad
3. Handouts

I. INTRODUCTION

A. Purpose / Terminal Objective

1. This lesson will provide the Licensed Operator candidate with information on Reactor Vessel Internals, as well as the relationships between the Reactor Vessel Internals and other plant systems. This will enable the candidate/trainee to understand the cause/effect relationships between system operations and the workings of the reactor vessel.

Suggested Instructional Methods and Media
Classroom lecture with facilitated discussions, using appropriate questioning techniques. Suggested media includes the visual aids attached to this lesson plan and/or Power-Point presentation.

B. Objectives

1. Identify the relationship between Reactor Vessel Internals and:

   a. Main Steam System
   b. Reactor Recirculation System
   c. Reactor Feedwater System
   d. High Pressure Coolant Injection (HPCI) System
   e. Residual Heat Removal (RHR) System
   f. Reactor Core Isolation Cooling (RCIC) System
   g. Control Rod Drive Hydraulic (CRDH) System
   h. Control Rod Drive Mechanism (CRDM)
   i. Standby Liquid Control (SBLC) System
   j. Reactor Water Cleanup (RWCU) System
   k. RPV Instrumentation
   l. Core Spray System
   m. Automatic Depressurization System (ADS)
   n. Nuclear Instrumentation / Traversing Incore Probes (NI/TIP)

2. Identify the effect that a loss or malfunction of Reactor Vessel Internals will have on:

   a. Reactor water level
   b. Reactor pressure
   c. Reactor power
   d. Plant radiation levels
   e. Offsite radiation levels
   f. Nuclear Steam Supply Shutoff System (NSSSS)
   g. RPV Instrumentation

3. Identify the Reactor Vessel Internal design features which provide for:

a. 2/3 core coverage following a DBA LOCA
b. Separation of water flowpaths within the RPV
c. Core orificing
d. Moisture removal from generated steam
e. Natural Circulation

4. Identify the operational implications of the following as they apply to Reactor Vessel Internals:

   a. Brittle fracture
   b. Safety limits

5. Identify the effect that a loss or malfunction of the following will have on Reactor Vessel Internals:

   a. CRDH
   b. CRDM
   c. Reactor Recirculation
   d. Reactor Feedwater
   e. RWCU
   f. RPV Instrumentation
   g. Core Spray
   h. HPCI
   i. RHR
   j. RCIC
   k. ADS
   l. Nuclear Instruments/TIPs
   m. Main Steam

6. Identify the impact of the following on Reactor Vessel Internals and use procedures to correct, control or mitigate the consequences of those conditions:

   a. Loss of Coolant Accident (LOCA)
   b. Overpressure transient
   c. Control Rod Drop Accident
   d. Excessive Heatup/Cooldown rate
   e. Exceeding safety limits

7. Identify moisture carryover and steam carryunder, and the problem associated with each.

8. Identify the two types of Fuel Support Pieces, their function, and their relationship to the core structure and control rod assemblies.

9. Identify how a central fuel bundle is vertically and horizontally supported.

10. Identify the functions of the following major components:

    a. Core shroud
    b. Biological shield
    c. Head vent
    d. Jet pumps
    e. Core differential pressure piping
    f. Bottom head drain
    g. Incore housing and guide tubes
    h. CRD housing support
    i. Baffle plate

11. Given a copy of Technical Specifications and various plant conditions:

    a. Identify any Limiting Conditions for Operation (LCO) relating to the Reactor Vessel Internals
    b. For SROs, identify the bases for those LCOs and the actions required for any condition of LCO non-compliance.

C. Fundamentals

1. The operations fundamentals covered in this lesson include:

    a. Closely Monitoring Plant Conditions

        1) Operator Rounds

        2) Intolerance for Unexpected Equipment Failure

    b. Conservative Bias to Plant Conditions

        1) Reactor Safety

    c. Solid Understanding of Plant Design and System Interaction

        1) Technical Human Performance

II. PRESENTATION

A. Purpose of Reactor Vessel Internals

1. To contain the reactor core, reactor internals and the reactor coolant-moderator

   a. To serve as a high integrity barrier against the leakage of radioactive materials to the drywell

   b. To provide for production of high quality saturated steam

   c. To provide a floodable volume in which the reactor core can be adequately cooled in the event of a breach in the primary system external to the reactor vessel

Operations Fundamental: Solid Understanding of Plant Design and System Interaction-Technical Human Performance

B. System Description and Flowpath

1. Dimensions and Weights

   a. Overall height 72' – 11 1/8"

   b. Inside diameter 251" (20' – 11")

   c. Wall thickness (Cyl. Sections) 6 1/16" to 7 1/8"

   d. Bottom head thickness 7 1/2"

   e. Weight

      1) Top Head 92 tons

      2) Reactor Vessel 643 tons

2. Reactor Vessel Elevations (from vessel bottom)

   a. Bottom of active fuel 216.5"

   b. Top of active fuel 366.5"

   c. RPV instrument reference "0" 527.5"

C. System Flowpaths

1. The discharge of water from the reactor recirculation pumps is directed through the twenty jet pump nozzles, which creates a low pressure area. This low pressure area draws in water from the surrounding downcomer.

2. The combined flow mixes in the jet pump mixer assembly, and then discharges to the lower plenum, or bottom head region.

3. The water then enters the core region by three paths:

   a. Through holes in the CRD guide tubes and the four orifices in the four-lobed fuel support pieces (for fuel locations which have an associated control rod blade)

   b. Through the fixed, single rod fuel support pieces (for peripheral fuel locations which have no associated control rod blade)

   c. Through the same paths as above, but entering the area outside the fuel channels by planned leakage around the fuel assemblies

4. After being heated by the core, the steam/water mixture exits the core and flows into the steam separators as poor quality steam (approx. 14% quality).

   a. The steam separators improve steam quality to about 90%.

   b. The steam dryers further improve steam quality to about 99.9%.

5. The removed moisture from the steam separator and steam dryer are returned to the downcomer.

6. Steam exits the reactor through four 26 inch main steam nozzles, and is supplied to the main steam system.

D. Component Description – Reactor Vessel

1. Vessel Nozzles (304 overall penetrations)

   a. Level instrumentation and reactor head vent nozzle – Quantity: 1

      Purpose – provide a tap for wide range level instrumentation and a vent for non-condensable gases

      Gamma radiation from the core disassociates water into H_2 and O_2, which must be vented to prevent O_2

buildup near uncladded upper head (vented to MSL 'C' at power, DWEDT when in cold shutdown)

b. Head Spray Nozzle – Quantity: 1

Blanked off due to mod, could be used for testing

c. Spare Head Nozzle – Quantity: 1

Can be used for testing

d. CRD Exhaust Nozzle – Quantity: 1

Blanked off due to mod, no longer used

e. Main Steam Nozzles – Quantity: 4

Purpose is to conduct dry steam out of the reactor

f. Instrument Nozzles – Quantity: 6

Purpose is to provide sensing points for reactor water level and pressure instruments

A loss or malfunction of these nozzles, or the lines attached to them, could result in the loss of indication of these instruments in the MCR or other plant locations. It could also result in various isolations or actuations of systems due to effects on ECCS and NSSSS instrumentation.

g. Feedwater Nozzles – Quantity: 6

Purpose is to conduct feedwater into the reactor vessel as required to replace steam flowing out of the reactor through the main steam lines, and to maintain reactor water level within proper range.

h. Core Spray Nozzles – Quantity: 2

Purpose is to provide for the low pressure spraying of the core in the event of a loss of coolant accident.

i. Recirculation Suction Nozzles – Quantity: 2

Purpose is to provide water from the reactor to the suction of the reactor recirculation pumps.

j. Recirculation Inlet Nozzles – Quantity: 10

Purpose is to route water from the discharge of the reactor recirculation pumps to the driving nozzles of the jet pumps to provide the required core flow.

k. Jet Pump Instrument Nozzles – Quantity: 24

Purpose is to allow measurement of the jet pump differential pressures required for determining individual jet pump flows.

l. Core Differential Pressure Nozzle – Quantity: 1

Purpose is to provide an instrument tap for measurement of reactor core differential pressure. Also provides a pressure reference for determining jet pump flows.

m. Bottom Head Drain – Quantity: 1

Purpose is to provide a means of sending water from the bottom head region to Reactor Water Cleanup. This assures good crud removal, and also is effective at removing cold water from the bottom head region, preventing buildup of cold water during low flow periods.

Also was used to provide for draining of the vessel during construction testing and flushing.

n. Low Pressure Coolant Injection – Quantity: 4

Purpose is to direct LPCI to inside the shroud to cool the core during a LOCA.

o. Other Bottom Penetrations

1) 185 CRD penetrations

2) 55 incore detector penetrations (SRM, IRM, LPRM)

2. Reactor Vessel Construction

a. Materials

1) Base Metal

The shell and head are made from Manganese-Molybdenum low carbon steel alloy

2) Inside Overlay

A 3/16 inch welded overlay of stainless steel cladding is applied to the interior surfaces of the vessel and bottom head.

b. Construction

1) Top head consists of a single piece on top that accommodates the three head nozzles.

2) Vessel head is bolted to the flange by 6-inch studs that are threaded into the flange. The head has holes that allow the studs to pass through, and is then fastened and hydraulically tensioned using large nuts.

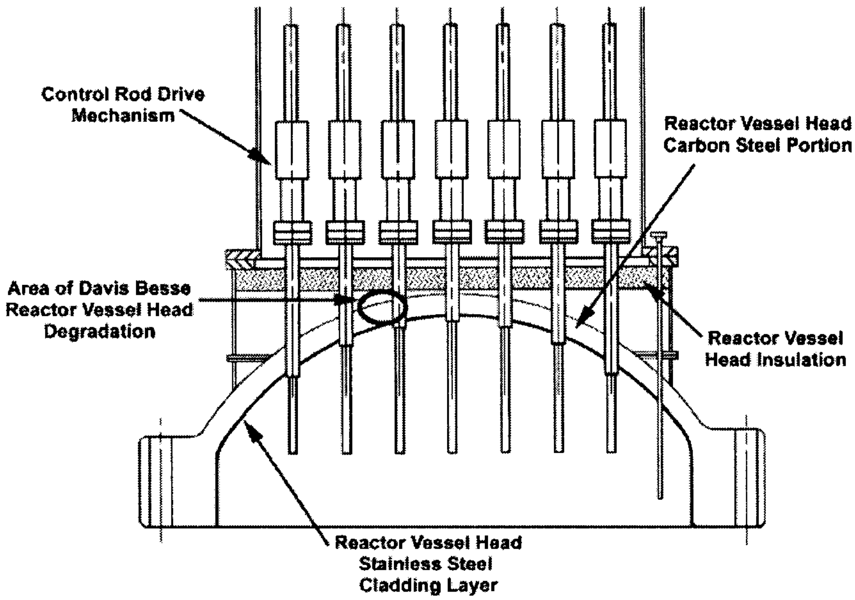
3) The mating surface between the vessel head and the flange contains two o-ring seals that are designed to provide a zero leakage seal. If the inner seal leaks, the outer seal will contain the leakage. Leakage past the first seal will pressurize the area between the seals, resulting in an alarm in the MCR.

4) Operating Experience – Davis-Besse Reactor Vessel Head Degradation

a) On February 16, 2002, the Davis-Besse Nuclear Power Station in Oak Harbor, Ohio, began a refueling outage that included inspecting the nozzles entering the head of the reactor pressure vessel. Of these vessel head penetration nozzles, the licensee's inspections focused on the nozzles associated with the control rod drive mechanisms.

The instructor should also consider discussion of SER 3-09, Unrecognized Reactor Pressure Vessel Head Flange Leak as another example of this type of event.

Reactor Vessel Head Degradation Location



b) In conducting its inspections, the licensee found that three CRDM nozzles had indications of axial cracking, which had resulted in leakage of the reactor's pressure boundary. Specifically, the licensee found these indications in CRDM nozzles 1, 2, and 3, which are located near the center of the RPV head. The licensee reported these findings to the NRC on February 27, 2002, and provided supplemental information on March 5 and March 9, 2002. The licensee also decided to repair the three leaking nozzles, as well as two other nozzles that had indications of leakage but had not resulted in pressure boundary leakage.

c) The repair of these nozzles included roll expanding the CRDM nozzle material into the material of the surrounding RPV head and then machining along the axis of the CRDM nozzle to a point above the indications in the nozzle material. On March 6, 2002, the licensee prematurely terminated the machining process on CRDM nozzle 3 and removed the machining apparatus from the nozzle. During the removal, the nozzle was mechanically agitated and subsequently displaced (or tipped) in the downhill direction (away from the top of the RPV head) until its flange contacted the flange of the adjacent CRDM nozzle.

d) To identify the cause of the displacement, the licensee investigated the condition of the RPV head surrounding CRDM nozzle 3. This investigation included removing the CRDM nozzle from the RPV head, removing boric acid deposits from the top of the RPV head, and ultrasonically measuring the thickness of the RPV head in the vicinity of CRDM nozzles 1, 2, and 3.

e) Upon completing the boric acid removal on March 7, 2002, the licensee conducted a visual examination of the area, which identified a large cavity in the RPV head on the downhill side of CRDM nozzle 3. Follow-up characterization by ultrasonic testing indicated wastage of the low alloy steel RPV head material adjacent to the nozzle. The wastage area was found to extend approximately 5 inches downhill on the RPV head from the penetration for CRDM nozzle 3 and was approximately 4 to 5 inches at its widest part. The minimum remaining thickness of the RPV head in the wastage area was found to be approximately 3/8 inch. This thickness was attributed to the thickness of the stainless steel cladding on the inside surface of the RPV head, which is nominally 3/8 inch thick.

f) The investigation of the causative conditions surrounding the degradation of the RPV head at Davis-Besse is continuing. Boric acid or other contaminants are likely contributing factors. Other factors contributing to the degradation might include the environment of the RPV head during both operating and shutdown conditions (e.g., wet/dry), the duration for which the RPV head is exposed to boric acid, and the source of the boric acid (e.g., leakage from the CRDM nozzle or from sources above the RPV head such as CRDM flanges). Boric acid leaks were known to exist for many years at Davis-Besse, but due to dose and access considerations, the source and severity of the problem was never adequately determined.

g) Corrective Actions

Davis-Besse has obtained a new Reactor Vessel Head from the Midland Plant in Michigan which was never completed as a nuclear power plant. By slightly modifying this head, it can be used on the Davis-Besse reactor. A new head is being produced by Framatome (France) and will be delivered in 2004, but will probably not be used for up to ten years.

On February 11, 2003, the NRC issued Orders to all pressurized water reactors, which modifies their current license to require specific inspections of the reactor pressure vessel head and associated penetration nozzles.

5) The bottom head is welded to the vessel shell and the support skirt (no flange).

6) The bottom head drain line has a thermocouple attached to the outside of the pipe as it leaves the reactor. This provides an accurate indication of bottom head temperature.

c. Biological Shield

1) Purpose is to reduce neutron and gamma radiation from the reactor to:

a) Permit drywell access for maintenance with minimal radiation exposure to personnel.

b) Extend the lifetime of drywell components such as cable insulation to the design life of the plant (prevents gamma radiation degradation of organic components).

c) Prevent neutron activation of components within the drywell and the resultant radiation exposure to personnel.

2) The biological shield is a cylindrical structure surrounding the reactor vessel, built of high density concrete, supported by vertical I-beam support columns, and clad on the inside and outside with an outer steel layer.

d. Limits on Operation

1) The carbon steel reactor vessel is susceptible to brittle fracture if temperature and pressure limitations are not adhered to. For this reason, Technical Specifications limits heatup and cooldown rates to a maximum of 100 F/hr, and provides limits on the minimum reactor vessel metal temperature vs. minimum reactor vessel top head pressure (See T.S. figure 3.4.6.1-1).

E. Component Description – Internals

1. Core Shroud

a. Purpose

1) Divides the downcomer flow from the core flow

2) Provides lateral support for the core plate and top guide, and hence lateral support for the fuel bundles

Vertical support of the top guide, core plate, and all fuel is basically provided by the vessel bottom head and vessel support skirt.

3) Provides a floodable region following a recirculation line break

b. Description

1) 2 inch thick stainless steel cylindrical assembly welded in the field to the upper lip of the baffle plate, extending up beyond the top of the jet pumps and fuel.

2) Closure Surface

   a) Upper surface is machined to provide a leak tight fit with the shroud head (which is the bottom of the steam separator assembly)

   b) Forty-eight (48) sets of lugs are provided for securing the shroud head to the shroud

2. Shroud Head

   a. Purpose

      1) To close off the core outlet so that all water and steam is forced through the steam separators.

   b. Description

      1) The shroud head is a dish shaped stainless steel closure assembly, with 225 penetrations for the steam separators, the bottom of which are welded to the shroud head via standpipes.

      2) The shroud head and steam separators form a single assembly, known simply as the steam separator.

3. Steam Separators (225 total)

   a. Purpose

      1) To increase the steam quality from approximately 14% at the core outlet to approximately 90% at the inlet to the steam dryer.

   b. Description

      1) The steam separators are centrifugal type separators.

      2) They are permanently welded to standpipes, which are welded to the shroud head.

3) Turning vanes at the inlets to the separators impart a rotation to the incoming two phase fluid. The higher density fluid (water) is thrown to the outside by centrifugal force, forming a continuous wall of water against the inner wall of the inner pipe, which runs down into the downcomer. Three stages of separation are used to return the water to the downcomer area.

4) During shutdown cooling operations, RPV level is maintained above the lowest return elevation to ensure a flowpath is available for natural circulation flow in the event forced circulation is lost.

5) The less dense steam exits out the top of the separators to the steam dryer.

c. Moisture Carryover

1) Defined as moisture exiting the steam separator at the top

2) The problem with moisture carryover to the steam dryer is that it will overload it with a resultant decrease in steam quality exiting the reactor vessel.

3) Moisture carryover is minimized in order to decrease turbine blade wear, increase turbine efficiency, and minimize radioactivity carried over to the balance of plant components.

4) Water level effect on moisture carryover

a) If the water level surrounding the separator is too high, the water in the separator tends to backup, resulting in moisture carryover out the top of the separator. A very wide deviation in water level is possible before any significant carryover results.

d. Steam Carryunder

1) Defined as steam exiting the separators as part of the water drained from them.

2) Some steam carryunder is always present. It can become excessive due to running with reactor water level too low.

3) The problems with steam carryunder are that it:

a) decreases NPSH to the reactor recirculation pumps by heating up the downcomer water

b) reduces plant efficiency, since steam escapes to the downcomer is not sent to the turbine

4. Steam Dryer

a. Purpose

1) Remove more of the water from the fluid exiting the steam separator, in order to further increase the quality of the steam sent to the main steam system.

2) Provide a seal between the wet steam area (steam exiting the steam separator), and the dry steam flowing to the main steam system.

b. Description

1) The upper section consists of chevron type steam dryers and moisture collection troughs and drain lines. The upper section has sides "cut away" to permit steam flow to the main steam lines.

2) The lower section consists of the seal skirt.

3) Steam Dryer Panels

a) Chevron type dryers – similar to what is used in the moisture separators in the cross-around piping between the high pressure turbine and low pressure turbines.

b) Operate on the principles of centrifugal force and gravity. The wet steam mixture rises from the steam separators through baffling, and is forced horizontally through the dryer panels.

The wet steam is forced to make a series of rapid changes in direction while traversing the dryer panels. During these traverses, moisture is thrown to the outside where it is caught by the many moisture collection hooks. Removed moisture drops down into collecting troughs, and is routed to the outside of the dryer assembly and into the downcomer annulus by means of drain pipes.

The steam exiting the top of the steam dryer unit will have a quality of approximately 99.9%.

c. Steam Dryer Integrity

1) Quad Cities, Dresden and other BWR's have experienced failures of steam dryer cover plates and generation of loose parts into the vessel and main steam lines. Indications of a failure include increased moisture carryover and possible changes to reactor pressure, level and steamline flow indications. Refer to Attachment 1 for recommendations on monitoring and detection

5. Jet Pumps

a. Purpose

1) Provides for increased flow of coolant through the reactor, greater than could be provided by the reactor recirculation pumps alone, to provide a higher reactor power output.

2) Provides for 2/3 core coverage in the event of a DBA LOCA, since reflood can fill the core, bottom head, and up through the jet pumps to the level of the top of the mixer section, without reaching the break in the recirculation suction piping.

b. Description

1) Ten (10) jet pump assemblies, each consisting of:

a) One inlet riser and thermal sleeve

b) One transition piece welded to the top of the inlet riser

c) Two jet pump mixer assemblies, consisting of the ram's head, nozzle, and mixer

d) One restrainer bracket assembly

e) Two diffuser sections, each welded at the bottom to the baffle plate

c. Assembly

1) Each riser is connected to a transition piece, which is nothing more than a Y connection for the seating of the two mixer assemblies.

2) The mixer assemblies also seat into a slip fit joint with the top of the diffuser. The restrainer bracket aligns the mixer assembly with the slip fit joint, and holds the mixer snug through the use of two set screws and a wedge assembly.

3) Once properly seated at the transition piece and slip fit, each mixer assembly is held firmly in place by a hold-down beam assembly, which fits into a bracket attached to the reactor vessel wall. The beam has a bolt through the center, which when threaded down, will exert downward force on the mixer assembly at the ram's head. The hold-down beam and bolt is then tensioned to maintain adequate downward force on the mixer assembly to hold it in place under all flow conditions.

4) The diffuser section has an expanding diameter from top to bottom, with the narrowest diameter at the slip fit connection, and the broadest diameter at the welded connection to the baffle plate.

d. A loss or malfunction of the jet pumps could result in a reduction in reactor power due to the reduction of core flow, which would also result in a small reduction in reactor pressure.

e. Operating Experience – History of Jet Pump Hold-Down Beam Cracking and Failures

1) Dresden 3

a) On 2/2/80, Dresden Unit 3 experienced a hold-down beam failure with subsequent displacement of the inlet mixer assembly. The condition was indicated by a sudden change in plant electrical output and reactor operating parameters. Subsequent post-shutdown ultrasonic examinations identified crack indications in six additional hold-down beams. The defective beams were replaced with new beams. The cause of the cracking was identified as intergranular stress corrosion cracking (IGSCC) by metallurgical examination of the defective beams, and occurred at the location of the bolt hole at the center of the beam.

b) Evidence of cracks in hold-down beams were revealed by ultrasonic examinations during shutdown periods at Quad Cities 2 (one beam), Pilgrim (three beams), Vermont Yankee (one beam), and Millstone 1 (five beams). The defective beams were replaced and the cracking has been generally attributed to IGSCC. At Vermont Yankee, which is a BWR-4 with some BWR-3 vessel hardware features, the defective BWR-3 design beams were replaced with the higher strength BWR-4 design beams.

c) Later in 1980, GE issued SIL 330, which described the mechanism of the hold-down beam failure at the location of the bolt hole, and explained that while this has only happened with BWR-3 beams so far, BWR-4 beams should also be monitored. It is not apparent from SIL 330 that GE considered failures at locations other than at the bolt hole.

d) In 1981, SER 47-81 was issued. In this SER, guidance was provided to the industry from General Electric regarding hold-down beam failures. The GE investigation of the IGSCC mechanism yielded estimates of approximately 4-1/2 years for crack initiation and an additional 1-1/2 hears (approximately) for crack propagation to beam failure condition. Pre-load hold-down margin over hydraulic lifting force on the jet pump is lost 1 to 2 weeks prior to beam failure, and during this period, indication of abnormal change in reactor operating parameters becomes increasingly recognizable in the control room as beam deflection proceeds to failure at an increasing rate. They added that additional encounters with jet pump hold-down beam cracks due to IGSCC are expected, but the visual inspections and ultrasonic examinations during refueling outages, and daily parameter surveillance procedures during operation that are called for in NRC bulletin 80-07 should ensure that cracks will be identified and defective beams replaced before crack propagation can proceed to the point of causing beam failure.

2) Grand Gulf

a) On September 13, 1993, Grand Gulf power station (a BWR-6 plant) experienced an unplanned high-pressure core spray system initiation due to a reactor low water level signal that resulted in a reactor scram. Initially, the reason for the water level anomalies detected in the "C" and "G" channels of the reactor water level instrumentation could not be determined.

b) On September 28, 1993, during restart from the reactor scram, the plant operations personnel discovered jet pump flow differential pressure anomalies. During the investigation of the problem, the plant experienced oscillating water level indications and jet pump flow readings characteristic of a displaced jet pump mixer section. These indications occurred at high recirculation flows in the reactor core (77 percent core flow). Following reactor shutdown and disassembly, the licensee found the mixer assembly for jet pump 10 had separated from the diffuser and relocated upside-down between jet pump 8 and jet pump 9. The hold-down beam for jet pump 10 had cracked and failed.

c) The jet pump hold-down beam failure at the Grand Gulf power station in September 1993 is unlike previous failures in that the hold-down beam failed at the beam end as shown in Attachment 1. This is believed to be the first failure of a BWR-4 style hold-down beam, and there were no previous reports of failures in this location. One beam end failed completely, causing the beam to come out, removing the restraint on the jet pump elbow and leading to the displacement of jet pump 10. The cracks began in an area where a radius machining cut had been made in the forging and led to failure in a location of the beam with a cross-section smaller than the areas that had been affected in previous cases. GE concluded that the probable cause of failure was an intergranular stress corrosion crack that propagated over 80 percent of the fracture surface.

d) The licensee conducted ultrasonic examinations on the other in-service jet pump beams and found indications on jet pump 8 and jet pump 21 at the bolt hole area in the center of the hold-down beams. This cracking was consistent with that which occurred at Dresden 3 and other sites. The licensee replaced all the jet pump beams with spare beams available on site.

e) In October 1993, ultrasonic test inspection of the hold-down beams at the Clinton power plant (a BWR-6 plant) revealed that one of the beams had crack indications around the center of the bolt hole region and the beam was replaced. On November 22, 1993, Pennsylvania Power and Light Company notified the NRC that it would be replacing all of the jet pump beams at Susquehanna Unit 1 before restarting from their refueling outage. This action was being taken as a precautionary measure given the new failure mode identified at the Grand Gulf station.

f) The water level anomalies that occurred at Grand Gulf on September 13 and at startup on September 28, 1993 were apparently caused by the turbulent flow conditions in the vicinity of jet pump 10.

g) Following the Grand Gulf hold-down beam failure, GE issued RICSIL 065, which provided significant new information based on the Grand Gulf event, including:

- Fracture mechanics evaluations indicated that crack growth to failure could occur in the beam ends in less than one 18-month operating cycle.
- GE recommendation that all jet pumps beams that are in the same population as the failed Grand Gulf beam that have accumulated service of more than eight years at the next refueling outage be replaced with hold-down beams less susceptible to IGSCC (Grand Gulf had been operating for approximately eight years when its beam failed). GE later added that replacement of these beams could be postponed if inspections were performed at the bolt hole and beam ends (transition arm region) each outage.

3) Quad Cities Unit 1

a) On January 9, 2002, Quad Cities Unit 1 experienced indication of a jet pump failure. This was based on the following plant indications:

(1) Gross megawatt electric decrease
(2) Reactor pressure decrease
(3) Total core flow indication increase
(4) Core plate d/p decrease

b) Approximately 30 minutes after the jet pump failure, the 1B recirculation pump tripped.

c) During the resulting forced outage, the licensee inspected the jet pumps for possible damage, and identified that the hold-down beam on jet pump 20 had broken into two pieces, with the associated inlet mixer on jet pump #20 wedged against the restrainer bracket assembly after lifting 10-12 inches, separating at the slip fit joint. The mixer assembly was hanging approximately 20 degrees from vertical.

d) The hold-down beam break occurred in the transition area between the bolt hole and the beam end, in an area that had not previously been considered susceptible to stress corrosion cracking. The previous failures have been in areas of known stress concentration. The transition area was typically not inspected during hold-down beam inspections. The licensee believed the failure mechanism was IGSCC related.

e) The hold-down beams were last inspected in November 2000 with no indications noted, however, the inspectors were not looking for indications in the transition area between the bolt hole and the beam ends. During later review of the previous inspection videotape, indications of the crack could be seen.

f) Jet pumps #5, 6, 7, and 8 had the new style hold-down beams, and the remaining 15 hold-down beams were the originally installed beams. The remaining original beams were cleaned and inspected, and six of the beams had crack-like indications identified. All original hold-down beams were replaced with the revised design hold-down beams prior to restart. These original beams had been in service for approximately 23 years.

4) LaSalle Unit 2

a) On 1/25/03, during a refueling outage of LaSalle Unit 2 (L2R09), ultrasonic (UT) examinations were performed on all jet pump beams per BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41). In addition to the BWRVIP-41 inspections of the beam bolt hole region and the beam transition arm region (beam end), UT examinations were performed on the transition area where the Quad Cities failure occurred. Two recordable indications were detected in jet pump beam number 20 and one recordable indication was detected in jet pump beam numbers 2 and 8. The indications were all at the beam bolt hole area. No other indications were identified. As a result of the examination, the beams were replaced.

b) The beams with indications were the "new design" beams made of Ni-CR-Fe alloy X-750, high temperature annealed, with reduced preload. The beams were installed during original construction. All hold-down beams were examined during L2R07 (Nov. 1996) with no recordable indications detected. Note that LaSalle Unit 2 began commercial operation in October 1984 (about 16 months before LGS Unit 1). The cause of the indications is under investigation.

c) Corrective actions taken were to replace the beams on jet pumps 2, 8, and 20.

5) Significance for LGS:

a) Our hold-down beams on Unit 1 were replaced during 1R05 with a type less susceptible to IGSCC, and Unit 2 has had that type since startup. However, LaSalle has hold-down beams that are also less susceptible to IGSCC and experienced crack indications on several jet pump hold-down beams. Don't assume it can't/won't happen here. Be familiar with the symptoms of ON-100, Failure of a Jet Pump, and immediately report any changes in jet pump or reactor recirculation pump performance to supervision and Reactor Engineering.

f. During a DBA LOCA, a jet pump failure could also affect the ability to achieve 2/3 core coverage.

6. Jet Pump Flow Sensing Lines

   a. Purpose

      1) Provides a mechanism for measuring flow through each jet pump

      2) Every jet pump has an instrument line tapped into the upper end of the diffuser section. This is the low pressure sensing line.

      3) Four jet pumps (5,10,16,20 for Unit 1, 5,10,15,20 for Unit 2) are considered "calibrated jet pumps", and have an instrument line tapped into the lower end of the diffuser section also. This is a high pressure sensing line.

      4) All twenty jet pumps also use the below core plate pressure as a high pressure input.

   b. Flowpaths and Basic Operation

      1) Driving water (from the discharge of the reactor recirculation pumps) enters the inlet riser, flows through the transition piece, and through the ram's head to the nozzle.

      2) The nozzle increases the velocity of the driving water, while lowering the pressure at the discharge of the nozzle.

      3) The low pressure area at the discharge of the nozzle draws in water from the downcomer annulus, equal to about 2-3 times the driving flow, and the combined flows travel through the mixing section, where they are "mixed".

      4) The water then flows into the diffuser section, where the expanding diameter allows the water to slow, resulting in a corresponding increase in pressure.

      5) The higher pressure water exits the lower end of the diffuser into the bottom head of the reactor, where it is then allowed to flow up through the core.

7. Baffle Plate (Diffuser Seal Ring)

   a. Purpose

1) Provides a mounting surface for the jet pump diffuser.

2) Separates recirculation pump suction area (downcomer annulus) from the core inlet plenum area (jet pump discharge)

b. Installation

1) Welded to the reactor vessel wall, it is supported by six columns welded directly to the reactor bottom head.

2) The support columns support the weight of the:

Jet pumps
Shroud and core spray sparger
Core plate
Top guide
Peripheral fuel bundles

8. Top Guide

a. Purpose

1) Provides lateral support for the upper end of all fuel bundles.

2) Provides lateral support for upper end of the neutron monitoring instrument assemblies (SRMs, IRMs, LPRMs)

3) Provided lateral support for the upper end of the neutron sources (no longer installed)

b. Description

1) Box-like structure of stainless steel plate set in a step at the top end of the shroud.

2) Each central box opening accommodates four fuel bundles and one control rod (this is the definition of a fuel cell).

a) Fuel bundles are laterally supported at the top as follows:

Fuel bundle → Top Guide →
Core Shroud → Baffle Plate →
Vessel Wall → Biological Shield →

Containment ⟶ Reactor Enclosure.

3) There are 24 openings near the outside, each of which accommodates one of the 24 peripheral fuel bundles.

4) Each nuclear instrument assembly and source holder is supported by a cutout on the bottom side of the guide plate at the junction of the cross member.

9. Core Plate

a. Purpose

1) Provides vertical and lateral support for the 24 peripheral fuel bundles.

2) Provides lateral support for the control rod guide tubes and hence lateral support for the fuel support castings and fuel bundles.

3) Acts as a partition to force the majority of the coolant up through the fuel bundles rather than outside them.

b. Description

1) Location: Bolted to the lower shoulder of the shroud

a) Contains:

(1) Holes for control rod guide tubes

(2) Holes for neutron source locations

(3) Holes and guide sleeves for incore guide tubes

(4) Holes for peripheral fuel support pieces

(5) Alignment pins for assuring proper control rod guide tube and fuel support piece orientation

(6) Bolts to hold the core plate to the shroud

10. Control Rod Drive Housings (185)

    a. Purpose

        1) An extension of the reactor vessel for external mounting of the control rod drives

        2) Provides both vertical and lateral support for the drives

        3) Transmits the weight of the fuel, fuel support piece, and control rod drive guide tube to the reactor bottom head for support

    b. Description

        1) Hollow stainless steel tube

        2) Flange at the bottom is for permanent attachment of the CRD hydraulic system insert and withdraw lines, and for the bolting of the control rod drive mechanism

        3) Keyway at the flange end is provided for locking the control rod drive thermal sleeve and preventing it from rotating.

11. Control Rod Drive Housing Support (also called "shootout steel")

    a. Purpose

        1) Prevent the rapid ejection of a control rod in the unlikely event of a control rod drive housing failure with the reactor at pressure

    b. Description

        1) Beams are attached at the inside of the steel liner on the inside of the concrete reactor pedestal. Hanging from the beams are hanger rods, which support bars and grid plates. These bars and grid plates are oriented directly under the control rod drive mechanisms, and

will limit the drive travel in the outward direction to 3 inches in the event of a failure.

12. Control Rod Guide Tubes

   a. Purpose

      1) Provides lateral support for the control rod blade velocity limiter

      2) Transmits the weight of the fuel and fuel support piece to the reactor bottom head via the control rod drive housing

   b. Description

      1) Top of the control rod guide tube has four 3" diameter holes to pass core flow from the below core plate area into the fuel bundles

      2) The bottom end is machined to mate with the CRD housing

      3) The guide tube is installed from above, by lowering it through the corresponding hole in the core plate until it rests on the CRD housing (note that the guide tube does not rest on the core plate).

      4) An alignment tab is provided, which fits over the permanently installed alignment pin on the core plate to ensure proper orientation of the flow holes with the holes on the fuel support piece.

13. Fuel Support Piece (four-lobed)

   a. Purpose

      1) Provide lateral alignment for the bottom end of the fuel assemblies

      2) Vertically supports, and transmits the weight of the central fuel bundles to the control rod guide tube and down to the reactor bottom head, then to the vessel support skirt, the pedestal, and finally to the bedrock below

   b. Description

1) A four-lobed stainless steel casting which supports four fuel bundles.

2) Contains four holes, each of which is fitted with an orifice to control the amount of flow which passes to its associated fuel bundle

3) Contains a cruciform shaped hole in the center to allow the control rod to pass up between the fuel bundles

c. Installation

1) The fuel support piece slips into the top of the control rod guide tube (only held down by its own weight and the weight of the fuel bundles which sit on top)

2) It also has alignment fingers that must be engaged with the alignment pin on the core plate in order to ensure the flow holes are aligned with those on the control rod guide tube.

14. Peripheral Fuel Support Piece (24)

a. Purpose

1) Provide vertical and lateral support to the 24 individual peripheral fuel bundles that are not part of complete "four bundle" fuel cells

2) Provides a means to control the amount of flow to a peripheral bundle by use of a removable orifice, which is installed inside the peripheral fuel support piece

b. Description

1) Body of the fuel support piece is permanently welded into a corresponding hole in the core plate

15. Incore Housings/Guide Tubes

a. Purpose

1) These are extensions of the reactor vessel, which provide for mounting of the incore nuclear instrumentation (SRMs, IRMs, and LPRMs), and

convenient bottom leadout of their electrical cables and mechanical drives.

2) Prevent jet pump flow (core flow) impingement on the nuclear instrumentation assemblies in the below core plate area, and thereby eliminate possible vibration damage to these assemblies

b. Description

1) Housings and guide tubes are 2 inch diameter stainless steel pipe

2) The housings have bolted flanges at the bottom end, which extend below the reactor vessel. The housings are welded to the reactor vessel where they penetrate the bottom head, and are precisely aligned directly beneath the location of the respective core plate hole for its nuclear instrument

3) The bottom of the guide tubes are welded to the top of their respective housing, while the top of the guide tubes slip fit into the bottom of the core plate.

   a) There are between 1 and 4 cooling holes surrounding the holes in the core plate for the LPRMs. These holes were designed to provide cooling for the LPRMs. However, it was later determined that the LPRMs had adequate cooling without the holes, and flow through these holes was causing excessive vibration of the instruments. Therefore, the holes have been plugged.

4) These housings and guide tubes are located between the fuel cells

5) 55 housings/guide tubes total (43 for LPRMs, 8 for IRMs, 4 for SRMs)

6) NIs are loaded into the guide tubes from above vessel, and once seated, are secured at the top of the instrument by a plunger which fits into a notch under an intersection of the top guide

16. Feedwater Spargers

a. Purpose

1) To evenly distribute feedwater to the jet pumps and recirculation pump suctions in a manner such that:

a) Cold water does not impinge upon the reactor vessel walls, and

b) The flow up through the core is properly mixed and of consistent temperature

2) The "A" feedwater line spargers also provide an entry point into the reactor for High Pressure Coolant Injection (HPCI) flow

3) The "B" feedwater line spargers also provide an entry point into the reactor for Reactor Core Isolation Cooling (RCIC) flow.

b. Description

1) Six 12" spargers are located 60 degrees apart around the inside of the reactor vessel, at a height of 658.5 inches above vessel bottom

2) Each sparger is welded to a thermal sleeve, which slip fits into a reactor vessel nozzle (penetration)

3) Two rows of holes in each sparger provide for even distribution of the relatively cooler feedwater throughout the downcomer annulus.

17. Core Spray Spargers

a. Purpose

1) Provides two redundant spray networks that will yield a spray pattern covering the entire top of the core in the event of a loss of coolant accident

2) The "B" Core Spray line also provides an entry point into the reactor for High Pressure Coolant Injection (HPCI) flow, and for Standby Liquid Control (SLC) system flow.

b. Description

1) Consists of two 10" loops that penetrate the reactor vessel wall at 484.5 inches from the vessel bottom.

2) These piping in these loops then runs down to the elevation of the shroud, and penetrates the top of the shroud

3) Once inside the shroud, each loop distributes its water through two sets of spargers, each of which is oriented 180 degrees around the top of the core. This ensures that even with only one loop of core spray in operation, the spray pattern will provide a full 360 coverage of the core.

4) The spray nozzles on the spargers were adjusted prior to initial startup to ensure the proper spray pattern is achieved, then were tack welded in place.

c. A loss or malfunction of the core spray spargers could result in the inability to maintain RPV level, and failure to adequately cool the core in the case of a Loss of Coolant Accident (LOCA). This could also lead to increased levels of plant radiation and offsite radiation. A core spray line break outside of the shroud could also affect the ability of SLC to inject to the core during an ATWS, thus minimizing the ability of SLC to reduce power.

18. Core Differential Piping

a. Purpose

1) Provides for measurement of above and below core plate pressures, and hence, core differential pressures

2) Provides for measurements of below core plate pressure for use in jet pump flow measurements

b. Description

1) Penetration from the reactor vessel is a pipe within a pipe

a) The outer pipe serves as the LP tap (senses pressure above the core plate)

b) The above core plate pressure line also is used by the Control Rod Drive Hydraulics (CRDH) system. It provides input for the drive water differential pressure indications used by the operators for adjusting the drive water pressure control valve.

c) The inner pipe serves as the HP tap (senses pressure below the core plate)

19. Low Pressure Coolant Injection Lines

a. Purpose

1) Provide a source of makeup and cooling water for the core following a LOCA

b. Description

1) Four 12" lines penetrate the reactor vessel and shroud

a) This arrangement allows water to be discharged directly onto the core inside the shroud

2) Baffle plates direct the flow downward toward the core

c. A loss or malfunction of the LPCI nozzles could result in the inability to maintain RPV level, and failure to adequately cool the core in the case of a Loss of Coolant Accident (LOCA). This could also lead to increased levels of plant radiation and offsite radiation.

F. Orificing of Flow Through Fuel Bundles

1. Why orifice?

a. Assume a core that has no flow orificing...

b. In a BWR, as power is increased, the amount of boiling (two phase flow) within a fuel bundle increases. More cooling is necessary for a higher power bundle (one with more boiling) than a lower power bundle.

c. Peripheral bundles run at approximately half the power of the central region bundles. As two phase flow increases within the higher power bundles, increased resistance to flow occurs. This would tend to reroute flow to the lower power fuel bundles, thereby tending to starve the higher power bundles. This is precisely what we don't want to happen.

d. By providing flow orificing in the fuel support pieces, the majority of the pressure drop across the core is taken at the orifice. The pressure drop across the orifice is large in comparison to the pressure drop in the fuel bundle itself, and consequently, any changes in two phase flow within individual fuel bundles causes insignificant changes in flow patterns between high and lower power fuel bundles. Also, smaller orifices are generally used for the peripheral bundles to provide a smaller amount of flow to those bundles, and a larger amount to the central, higher power bundles.

2. Flow Orificing

   a. The core is divided into two orificing zones:

      1) Central region

      2) Peripheral region

   b. The peripheral region consists of bundles near the outside of the core that are supported on four-lobed fuel support pieces, as well as those supported on the peripheral fuel support pieces.

G. System Interrelations – How a Loss of the Following Systems Will Affect Reactor Vessel Internals

1. CRDH – a loss of CRDH flow will result in the loss of the ability to move control rods using normal control rod drive mechanism operation. Control rod movement via scram is not affected, so long as scram discharge volume is adequately drained, and either accumulator pressure is adequate, or RPV pressure is adequate (>900 psig).

2. CRDM – a loss or failure of a Control Rod Drive Mechanism will affect the ability to move the associated control rod.

3. Reactor Recirculation – a loss or malfunction of the reactor recirculation system could result in:

a. Loss of ability to achieve 2/3 core coverage if the failure is associated with a jet pump

b. Excessive thermal stress on CRD Housing stub tube welds or reactor vessel support skirt welds if the malfunction involves an increase in flow when bottom head temperatures are too cold relative to recirc. loop temperatures or steam dome temperatures.

c. Potential damage to fuel bundles if the malfunction involves an increase in flow which raises power above allowed limits

4. Reactor Feedwater – a loss or malfunction of the reactor feedwater system could result in excessive moisture carryover in the steam separators if feedwater injection into the RPV was excessive and caused a high level condition in the reactor.

5. Reactor Water Cleanup (RWCU) – a loss or malfunction of the Reactor Water Cleanup system (loss of bottom head drain flow, for example) could result in excessive cooling in the bottom head region, which could result in excessive thermal stress of the CRD Housing stub tube welds and reactor vessel support skirt welds if flow were increased.

6. RPV Instrumentation – a loss or malfunction of RPV instrumentation could result in high water level due to excessive feedwater injection (and moisture carryover concerns), inadvertent initiation of ECCS systems, low water level (and core cooling concerns)

7. Core Spray – a loss or malfunction of the Core Spray system could result in failing to keep the core adequately cooled during a LOCA event, potentially resulting in fuel damage. It could also result in the inability to inject with HPCI or SLC if the “B” Core Spray system discharge piping failed.

8. High Pressure Coolant Injection (HPCI) – a loss or malfunction of HPCI could result in a power excursion if it injected at power. If not terminated, this could potentially result in fuel damage if the reactor could not be properly shutdown.

9. Residual Heat Removal (RHR) – a loss or malfunction of RHR could result in the inability to keep the core

adequately cooled during a LOCA event, potentially resulting in fuel damage.

10. Reactor Core Isolation Cooling (RCIC) – a loss or malfunction of the RCIC system could result in increased power if it injected while operating the reactor at power.

11. Automatic Depressurization System (ADS) – a loss or malfunction of the ADS system could result in a cooldown rate in excess of 100 deg. F per hour if ADS were to inadvertently actuate. This could cause excessive thermal stresses on the reactor vessel and other reactor components.

12. Nuclear Instruments / TIPs – a loss or malfunction of NI's or TIPs could result in operating the reactor at a higher than allowed power level if the malfunction was that the NI's were indicating lower than actual power. This could cause localized fuel damage.

13. Main Steam – a loss or malfunction of the main steam system could result in a RPV cooldown rate in excess of 100 deg. F per hour in the event of a main steam line break or SRV opening. This could cause excessive thermal stresses on the reactor vessel and other reactor components.

H. Other System Interrelations

1. ADS – The ADS system receives inputs from reactor vessel instrumentation for RPV level, which use the RPV instrumentation nozzles mentioned earlier. The ADS also uses 5 safety relief valves, which are located off the main steam piping upstream of the MSIVs. The Main Steam nozzles mentioned earlier provide the steam to those SRVs.

I. Technical Specifications

1. 2.1, Safety Limits

   a. THERMAL POWER, Low Pressure or Low Flow

      THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

b. THERMAL POWER, High Pressure and High Flow

The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.10 (1.07 U/2) for two recirculation loop operation and shall not be less than 1.11 (1.09 U/2) for single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

c. REACTOR COOLANT SYSTEM PRESSURE

The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

d. REACTOR VESSEL WATER LEVEL

The reactor vessel water level shall be above the top of the active irradiated fuel.

2. 3.4.6.1, Reactor Coolant System, Pressure/Temperature Limits

   a. Review temperature/pressure limits, including heatup and cooldown limitations.

   b. Surveillance 4.4.6.1.3 requires examination of reactor vessel material surveillance specimens at intervals specified in Table 4.4.6.1.3-1.

   c. The specimens are located in specimen holders, which are hanging on the reactor vessel wall at azimuths 30°, 120°, and 300°.

3. 3.1.3.8, CRD Housing Support

III. SUMMARY

A. Review Objectives

1. Reviewing the objectives will ensure the important and testable aspects of this lesson are understood. This will enable the RO/SRO candidate to properly operate systems associated with the LGS reactor vessel.

B. Review Fundamentals

1. The fundamentals covered in this lesson were:

   a. Closely Monitoring Plant Conditions

      1) Operator Rounds

      2) Intolerance for Unexpected Equipment Failure

   b. Conservative Bias to Plant Conditions

      1) Reactor Safety

   c. Solid Understanding of Plant Design and System Interaction

      1) Technical Human Performance

IV. EVALUATION

A. Students must pass test with a score of 80% or greater per the applicable Course Plan

B. Practice Questions

1. The moisture content of the water vapor entering the steam separators is __________ and the quality of the water vapor exiting the steam dryer is __________.

   a. 14%, >99%

   b. 86%, <1%

   c. 14%, <1%

   d. 86%, >99%

2. Which component provides a floodable region following a recirculation line break as well as supporting the fuel bundles laterally?

   a. Bottom core plate

   b. Core shroud

   c. Upper Fuel support grid

   d. Steam separator

C. Answers to Practice Questions

1. D

2. B

V. ATTACHMENTS

A. GE SIL 644 Steam Dryer Integrity

B. LLOT-0041A Power Point

C. SER 3-09, Unrecognized Reactor Pressure Vessel Head Flange Leak

EXELON NUCLEAR
LIMERICK GENERATING STATION

T-102 PRIMARY CONTAINMENT CONTROL - BASES

1.0 PURPOSE

The purpose of T-102, Primary Containment Control, relates directly to four basic functions performed by the primary containment:

1. Providing a barrier to the uncontrolled release of fission products.
2. Containing and condensing steam discharged through the SRVs and primary system breaks that occur inside the primary containment.
3. Shielding personnel from radiation emitted by the reactor.
4. Providing a protected environment for key equipment important to safety.

The purpose of T-102 is accomplished through the parallel control of five key primary containment parameters:

- Suppression pool temperature
- Drywell temperature
- Primary containment pressure
- Suppression pool water level
- Combustible gas concentration

T-102 is entered when one or more of the following conditions exists:

- Suppression pool temperature above 95°F
- Suppression pool water level outside the band of 22 ft. to 24 ft. 3 inches
- Drywell pressure above 1.68 psig
- Drywell temperature above 145°F.
- Primary containment hydrogen (H_2) concentration above 4%

These conditions are symptomatic of an emergency or conditions which, if not corrected, could degrade into an emergency. This set of entry conditions is sufficient to assure that the appropriate flowpath(s) of the procedure will be entered for transients and accidents which are within the design and licensing basis for LGS, and for additional events which have been evaluated as significant with respect to emergency response actions.

Similar to the rationale which formed the basis for selecting the entry conditions for T-101, RPV Control, setpoints for T-102 entry conditions are simple, unambiguous, operationally significant, readily identifiable, and familiar to plant operators as follows.

- High D/W pressure scram setpoint of 1.68 psig as defined in Tech Spec Table 2.2.1-1 RPS Instrumentation Setpoints
- DW
  OR Suppression Pool High H_2 annunciator alarm setpoint of 4% per ARC-MCR-114
- Technical Specification limits for the following parmeters:
- Suppression pool water level outside limits of 22'0" to 24'3" per Tech Spec 3/4.6.2.1.a.1
  - Suppression pool temperature greater than 95°F per Tech Spec. 3/4.6.2.1.a.2
  - Drywell temperature greater than 145°F per Tech Spec 3/4.6.1.7

The intent of this condition is to enter T-102 if drywell temperature exceeds the Tech. Spec 3/4.6.1.7 temperature of 145°F which is based on a volumetric average temperature. T-102 should be entered when drywell temperature exceeds 145°F as determined by the volumetric average per ST-6-107-590-*.

The entry condition setpoints are specified so as to provide advance warning of potential emergency conditions, allowing action to be taken sufficiently early to prevent more severe consequences.

If while executing T-102 another T-102 entry condition is reached, operators must re-enter T-102 at the beginning.

Operators are directed to continue at Step PCC-1.

2.0 PROCEDURE

PCC-1 **IF** SAMP procedures have been entered,
THEN exit all TRIP procedures

DISCUSSION

LGS TRIP Step PCC-1 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action.

If primary containment flooding is required, all TRIP procedures are exited and all SAMP procedures are entered. The SAMP procedures then remain in effect until an emergency no longer exists. If additional TRIP procedure entry conditions occur, the TRIP procedures should not be re-entered. If operators do re-enter the TRIP procedures, however, the wording of this continue re-checking step ("IF ... SAMP PROCEDURES HAVE BEEN ENTERED ...") returns control of primary containment gas, suppression pool temperature, suppression pool level, primary containment pressure, and drywell temperature to the SAMP procedures. Criteria for entry into the SAMP procedures are established in T-111, Level Restoration/Steam Cooling; T-116, RPV Flooding; and T-117, Level/Power Control.

PCC-2 **IF** a Primary Containment emergency no longer exists,
THEN exit this procedure

DISCUSSION

LGS TRIP Step PCC-2 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified actions.

Step PCC-2 directs actions to exit T-102, Primary Containment Control, when a Primary Containment emergency no longer exists.

When a Primary Containment emergency no longer exists, no further actions need be taken in T-102. The determination that a Primary Containment emergency no longer exists is to be made by Shift Management, after having considered current plant conditions and the relative value of continuing execution of T-102. In addition, if the Emergency Response Organization is manned, it is expected that Shift Management will consult with the Technical Support Center regarding the determination, with respect to continued performance of T-102. If T-102 is exited prematurely, it is possible that plant conditions could continue to degrade to the point of meeting another entry condition and therefore require re-entry into T-102.

PCC-3 Execute sections PC/G, SP/T, SP/L, PC/P, AND DW/T concurrently

DISCUSSION

LGS TRIP Step PCC-3 directs the concurrent execution of the five flowpaths of T-102, Primary Containment Control.

T-102 has been structured along five parallel flowpaths because actions taken to control any one of the key primary containment parameters may directly affect control of the other parameters. This approach reflects the fact that the primary containment functions as a closed thermodynamic system and the transient responses of all primary containment parameters are directly interrelated. For example:

- Changes in suppression pool temperature can directly effect changes in primary containment pressure.
- Changes in drywell temperature can directly effect changes in primary containment pressure.
- Changes in suppression pool level can directly effect changes in suppression pool pressure.

Parallel execution is also required because the symptomatic approach to emergency response upon which the TRIP procedures are based precludes advance prioritization of the flowpaths. Rather, parameter values and trends, and the status of plant systems and equipment, dictate the relative importance of individual T-102 steps and the order in which they should be accomplished.

PC/G PRIMARY CONTAINMENT GAS CONTROL

The primary containment gas (PC/G) control flowpath of T-102, Primary Containment Control, contains instructions for controlling combustible gas concentrations.

Hydrogen and oxygen must both be present and in sufficient concentration for combustion to occur. Excessive hydrogen concentration, mixed with high oxygen concentration and ignited in the confined space of the primary containment generates peak pressures which may exceed the structural capability of the drywell, suppression pool, or drywell-to-suppression pool boundary. Such an event may result in the uncontrolled release of radioactivity to the environment. In addition, the temperature and pressure shock waves created during the rapid burning of these gases may damage equipment important to the safe shutdown of the plant. Concentrations of both gases are therefore monitored and controlled to prevent the development of a flammable condition.

Measurable levels of hydrogen could appear in the primary containment from the following sources:

- The high temperature reaction of metal (typically zirconium) with water to produce hydrogen gas and metal oxide
- Radiolysis of water to produce hydrogen and oxygen
- Feedwater injection of hydrogen to control reactor chemistry

Entry into the PC/G flowpath requires activation of the H_2/O_2 Analyzers, if not already in operation.

Elevated concentrations of oxygen are not expected during normal power operations except during brief periods at startup and shutdown of the plant when the primary containment atmosphere is being inerted and de-inerted. However, oxygen may be generated due to the radiolysis of water, and oxygen could enter the primary containment from leaks in the Instrument Air System. Oxygen concentration is routinely monitored and controlled during reactor operation in accordance with Technical Specification requirements.

PC/G-1 **IF** H_2/O_2 analyzers isolate,

THEN bypass Grp VI isol logic per GP-8

DISCUSSION

LGS TRIP Step PC/G-1 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action.

Subsequent steps in the primary containment gas (PC/G) control flowpath require knowledge of hydrogen and oxygen concentrations in both the drywell and suppression pool. If a Group VI Primary Containment Isolation System (PCIS) isolation has occurred, and it can be determined that a primary containment leak from Group VI systems is not occurring, it is appropriate to bypass the necessary isolation interlocks to restore the Hydrogen/Oxygen (H_2/O_2) Analyzers.

PC/G-2 **IF** H_2/O_2 analyzers become unavailable,

THEN sample DW AND Supp Pool for H_2 AND O_2 per CY-LG-120-910

DISCUSSION

LGS TRIP Step PC/G-2 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action.

Should the H_2/O_2 Analyzers become unavailable, the concentration of these gases must be determined by sample and analysis.

At the time this continue re-checking step is reached, it may not be possible to determine if the H_2/O_2 Analyzers are presently unavailable because of the time required to complete such prerequisites as sample path alignment and detector warm-up. However, the H_2/O_2 Analyzers should not be considered unavailable until ample opportunity has been taken for placing them in-service. During the time it takes to place the H_2/O_2 Analyzers in-service, it may be learned that they will not become available (e.g., due to equipment malfunction, etc.). In this case, it would be appropriate to initiate sample and analysis activities as directed by this continue re-checking step so that primary containment gas concentrations can be determined by alternate means.

CY-LG-120-910, Operation Of The Post Accident Sampling System (PASS), directs sampling and analysis of the primary containment for hydrogen and oxygen concentrations when the H_2/O_2 Analyzers are not available.

PC/G-3 **WHEN** DW OR Supp Pool H_2 concentration reaches 1%,
THEN continue

DISCUSSION

LGS TRIP Step PC/G-3 is a hold/wait step, and should not be exited until one of the two conditions specified in the "WHEN" statement exists.

Subsequent actions in the primary containment gas (PC/G) control flowpath are directed based on the status of hydrogen and oxygen concentrations in the drywell and suppression pool. No actions are directed, however, until either drywell or suppression pool hydrogen concentration rises above 1%.

When either drywell or suppression pool hydrogen concentration has risen above 1%, operators are directed to continue at Step PC/G-4.

PC/G-4 Enter T-102 Sheet 2 PC/G-5

DISCUSSION

LGS TRIP Step PC/G-4 is a continuation arrow, that directs entry into and concurrent execution of, Sheet 2 of the T-102, Primary Containment Control, flowchart, beginning at Step PC/G-5.

PC/G-5 Monitor AND control DW AND Supp Pool H_2 AND O_2 per Tables PC/G-1 AND PC/G-2

TABLE PC/G-1 DW H_2 AND O_2 CONTROL

<table>
<tr><td colspan="2" rowspan="4"></td><td colspan="4">DW O_2</td></tr>
<tr><td rowspan="3">< 5%</td><td colspan="3">≥ 5% OR UNKNOWN</td></tr>
<tr><td colspan="3">SUPP POOL H_2</td></tr>
<tr><td>< 1%</td><td>1% - 5.99%</td><td>≥ 6% OR UNKNOWN</td></tr>
<tr><td rowspan="3">DW H_2</td><td>< 1%</td><td>NO ACTION REQUIRED</td><td>NO ACTION REQUIRED</td><td rowspan="2">DW/G-2</td><td rowspan="3">DW/G-3</td></tr>
<tr><td>1% - 5.99%</td><td rowspan="2">DW/G-1</td><td></td></tr>
<tr><td>≥ 6% OR UNKNOWN</td><td colspan="2"></td></tr>
</table>

TABLE PC/G-2 SUPP POOL H_2 AND O_2 CONTROL

<table>
<tr><td colspan="2" rowspan="4"></td><td colspan="4">SUPP POOL O_2</td></tr>
<tr><td rowspan="3">< 5%</td><td colspan="3">≥ 5% OR UNKNOWN</td></tr>
<tr><td colspan="3">DW H_2</td></tr>
<tr><td>< 1%</td><td>1% - 5.99%</td><td>≥ 6% OR UNKNOWN</td></tr>
<tr><td rowspan="3">SUPP POOL H_2</td><td>< 1%</td><td>NO ACTION REQUIRED</td><td>NO ACTION REQUIRED</td><td rowspan="2">SP/G-2</td><td rowspan="3">SP/G-3</td></tr>
<tr><td>1% - 5.99%</td><td rowspan="2">SP/G-1</td><td></td></tr>
<tr><td>≥ 6% OR UNKNOWN</td><td colspan="2"></td></tr>
</table>

DISCUSSION

LGS TRIP Step PC/G-5 is a continue re-checking step, and as such, should be referred to frequently. If at any time during the performance of subsequent steps in the primary containment gas (PC/G) control flowpath the conditions specified in TABLE PC/G-1 or PC/G-2 for a drywell gas (DW/G) and/or suppression pool gas (SP/G) control flowpath exist, the respective flowpath should be executed.

Hydrogen and oxygen concentrations are monitored and controlled in the drywell by the actions specified in the DW/G-1, DW/G-2, or DW/G-3 flowpaths. Hydrogen and oxygen concentrations are monitored and controlled in the suppression pool by the actions specified in the SP/G-1, SP/G-2, or SP/G-3 flowpaths. Combustible gas concentrations may vary between the two areas. For example, a hydrogen generation event in conjunction with a small steam break in the drywell may raise drywell hydrogen concentration above detectable levels without a rise in suppression pool hydrogen concentration. SRV discharge in conjunction with a hydrogen generation event may raise suppression pool hydrogen concentration above detectable levels without a rise in drywell hydrogen concentration. For this reason, the drywell and suppression pool gas control flowpaths are performed in parallel.

TABLE PC/G-1 dictates the appropriate steps to be performed to control drywell combustible gas concentrations. TABLE PC/G-2 dictates the appropriate steps to be performed to control suppression pool combustible gas concentrations. The hydrogen concentration in the area of concern is compared to the oxygen concentration in the area of concern and either no action or the action in one of the three identified flowpaths is selected. Note that when oxygen concentration in the area of concern is at or above 5% or cannot be determined to be below 5%, the hydrogen concentration in the other area becomes a factor in the selection of the appropriate flowpath. When the area of concern is not inerted, hydrogen from the other area could migrate to the de-inerted area, creating a deflagrable mixture before the H_2/O_2 Analyzers sense a rise in hydrogen concentration.

Only one DW/G flowpath and one SP/G flowpath (i.e., one from each table) may be performed concurrently. TABLES PC/G-1 and PC/G-2 should be continuously evaluated to ensure that necessary preventative and mitigative actions are performed as the concentrations of combustible gases change.

The specified value of 6% hydrogen concentration is the minimum which can support a deflagration. Likewise, the minimum concentration of oxygen required to support a deflagration is 5%. Combustion of hydrogen in the deflagration concentration range creates a traveling flame front, heating the primary containment atmosphere and causing a rapid rise in primary containment pressure. A deflagration may result in a peak primary containment pressure high enough to rupture the primary containment or damage the drywell-to-suppression pool boundary.

If hydrogen and oxygen concentrations in the drywell or suppression pool cannot be determined by any means, it must be assumed that levels in excess of those required to support combustion are present. The tables therefore specify the appropriate flowpaths to perform when either or both of these gases cannot be determined with respect to their combustible limits.

Failure or unavailability of the H_2/O_2 Analyzers does not necessarily mean that the concentration of these gases cannot be determined. Rather, the decision requires a judgment based on related plant conditions. Since hydrogen concentration may not be continually monitored, the exact level at any specific time may not be known, even during normal plant operations. However, if there is reasonable assurance that RPV level has not dropped below the top of active fuel (TAF, -161 inches), it is likely that significant hydrogen has not been produced. On the other hand, if an event has occurred and RPV level cannot be determined or did in fact drop below TAF, then operators have reason to be concerned that significant amounts of hydrogen may have been generated unless alternate methods of determination indicate otherwise. Alternate methods include sampling, which may involve a long time delay, or deduction based on actual plant status. Deduction, which may be inconclusive, may be based on, for example, plant conditions with the potential for radiolysis of water, the status and operation of the Hydrogen Water Chemistry Injection System, or from the combination of safety relief valve (SRV) actuations and drywell pressure indications.

Based on the current status of drywell and suppression pool H_2 and O_2 concentrations, operators are directed to continue in the appropriate flowpath identified in TABLE PC/G-1 and/or PC/G-2.

Status TABLES PC/G-3 and PC/G-4 have been provided next to TABLES PC/G-1 and PC/G-2 for tracking the current status of hydrogen and oxygen concentration in the drywell and suppression pool.

PC/G-6 **BEFORE** DW press drops to 0 psig,
THEN terminate DW sprays

DISCUSSION

LGS TRIP Step PC/G-6 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action. The logic term "BEFORE" means drywell spray operation should be terminated before drywell pressure drops to 0 psig. If drywell pressure has already dropped to 0 psig when this step is entered, operators should also terminate operation of drywell sprays.

The operation of drywell sprays must be terminated by the time drywell pressure drops to 0 psig to ensure drywell pressure is not reduced below atmospheric. Maintaining a positive drywell pressure provides a positive margin to the negative design pressure of the primary containment.

Terminating drywell sprays "before drywell pressure drops to 0 psig" permits use of drywell sprays for fission product scrubbing at low primary containment pressures or if the primary containment has failed, yet still avoids negative primary containment pressures.

Consistent with the definition of the logic term "before", the actual pressure value at which drywell sprays should be secured is event specific:

- Reducing primary containment pressure below the high drywell pressure scram setpoint will clear the scram logic and maximize the margin to primary containment pressure limits.
- If the primary containment has failed or if primary containment venting is anticipated, it may be advisable to continue drywell spray operation at low primary containment pressures to scrub the primary containment atmosphere.
- If hydrogen and oxygen are above deflagration limits, drywell spray operation is prescribed in the DW/G-3 flowpath to reduce the flammability of combustible gases and to mitigate the effects of a possible deflagration.
- Reducing primary containment pressure will reduce the net positive suction head (NPSH) available for pumps drawing suction from the suppression pool. As stated in the discussion for the CAUTION preceding Step DW/G-3.7, NPSH limits should be observed if possible, but may be exceeded if warranted by event specific conditions. If there is no need for continued drywell spray operation, however, drywell sprays may be terminated at higher pressures if NPSH limits are approached.

Initiation and operation of drywell sprays is addressed in Step DW/G-3.8. Note that while operation of drywell sprays is permitted down to pressures approaching 0 psig, the initiation of drywell spray is prohibited when on the unsafe side of the Drywell Spray Initiation Limit curve (DSIL, CURVE DW/G-3-2).

PC/G-7 **BEFORE** Supp Pool press drops to 0 psig,

THEN terminate Supp Pool sprays

DISCUSSION

LGS TRIP Step PC/G-7 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action. The logic term "BEFORE" means suppression pool spray operation should be terminated before suppression pool pressure drops to 0 psig. If suppression pool pressure has already dropped to 0 psig when this step is entered, operators should also terminate operation of suppression pool sprays.

The operation of suppression pool sprays must be terminated by the time suppression pool pressure drops to 0 psig to ensure suppression pool pressure is not reduced below atmospheric. Maintaining a positive suppression pool pressure provides a positive margin to the negative design pressure of the primary containment.

Terminating suppression pool sprays "before suppression pool pressure drops to 0 psig" permits use of suppression pool sprays for fission product scrubbing at low primary containment pressures or if the primary containment has failed, yet still avoids negative primary containment pressures.

Consistent with the definition of the logic term "before", the actual pressure value at which suppression pool sprays should be secured is event specific:

- Reducing primary containment pressure below the high drywell pressure scram setpoint will clear the scram logic and maximize the margin to primary containment pressure limits.

- If the primary containment has failed or if primary containment venting is anticipated, it may be advisable to continue suppression pool spray operation at low primary containment pressures to scrub the primary containment atmosphere.

- If hydrogen and oxygen are above deflagration limits, suppression pool spray operation is prescribed in the SP/G-3 flowpath to reduce the flammability of combustible gases and to mitigate the effects of a possible deflagration.

- Reducing primary containment pressure will also reduce the net positive suction head (NPSH) available for pumps drawing suction from the suppression pool. As stated in the discussion for the CAUTION preceding Step SP/G-3.3, NPSH limits should be observed if possible, but may be exceeded if warranted by event specific conditions. If there is no need for continued suppression pool spray operation, however, suppression pool sprays may be terminated at higher pressures if NPSH limits are approached.

Initiation and operation of suppression pool sprays is addressed in Step SP/G-3.4.

PC/G-8 **IF** DW AND Supp Pool $H_2 < 6\%$

OR

DW AND Supp Pool $O_2 < 5\%$,

THEN Pri Cont sprays permitted using ONLY those pumps NOT required for core cooling

DISCUSSION

LGS TRIP Step PC/G-8 is a continue re-checking step, and as such, should be referred to frequently to determine if one of the conditions listed exists, and if so, to carry out the specified action.

At both LGS units, primary containment spray flow is supplied by pumps which are also capable of injecting into the RPV. Primary containment spray operation is permitted, however, only if there is no immediate threat of hydrogen deflagration, and only if the pumps to be used do not have to be operated continuously in an RPV injection mode to assure adequate core cooling. Maintaining adequate core cooling takes precedence over initiating primary containment sprays when deflagration concentrations of hydrogen and oxygen do not exist in the primary containment, since catastrophic failure of the primary containment is not expected under these conditions.

If drywell and suppression pool hydrogen concentration cannot be determined to be below 6% or drywell and suppression pool oxygen concentration cannot be determined to be below 5%, a potential for a deflagration exists. Only then may pumps be operated in the primary containment spray mode regardless of whether adequate core cooling can be assured. This priority is established because a deflagration could result in a loss of primary containment integrity leading, in turn, to a loss of core cooling capability.

DW/G-1

The DW/G-1 flowpath is entered when either of the following sets of conditions exist:

- Drywell H_2 concentration is between 1% and 5.99% AND drywell O_2 concentration is below 5%
- Drywell H_2 concentration is at or above 6% or is unknown AND drywell O_2 concentration is below 5%

The existence of a detectable amount of hydrogen in the drywell warrants corrective action regardless of the amount of oxygen present or the condition which required entry into T-102, Primary Containment Control. Inerting/Venting and hydrogen recombiner operation are the two methods normally used to control primary containment atmospheric conditions. Although continued rises in hydrogen concentration above the deflagration limit in the presence of oxygen will potentially threaten primary containment integrity, the presence of hydrogen in the absence of oxygen is not by itself primary containment threatening. Inerting/Venting is only permitted in the DW/G-1 flowpath if it can be done within the limits prescribed for normal (non-emergency) plant operation.

DW/G-1.1 Execute both sections concurrently

DISCUSSION

LGS TRIP Step DW/G-1.1 directs the concurrent performance of actions to lower drywell combustible gas concentrations.

Here the DW/G-1 flowpath divides into two parallel flowpaths. One flowpath deals with recombiner and drywell unit cooler fan operation, and the other deals with primary containment inerting/venting. Both flowpaths should be executed concurrently to lower combustible gas concentrations in the drywell.

Operators are directed to continue at Steps DW/G-1.2 and DW/G-1.6 concurrently.

DW/G-1.2 **WHEN** DW OR Supp Pool $H_2 > 1\%$,

THEN continue

DISCUSSION

LGS TRIP Step DW/G-1.2 is a hold/wait step, and should not be exited until one of the two conditions specified in the "WHEN" statement exist.

The hydrogen concentration limit specified in Step DW/G-1.2 is the minimum hydrogen concentration required for recombiner operation. Starting recombiners below this limit would serve no useful purpose because there would be an insufficient supply of hydrogen to support recombination. (NOTE: there is no lower oxygen concentration limit for recombiner operation).

When it has been determined that either drywell or suppression pool hydrogen concentration is above 1%, operators are directed to continue at Step DW/G-1.3.

DW/G-1.3 Operate Post LOCA recombiners per S58.1.B AND DW unit clr fans as necessary

DISCUSSION

LGS TRIP Step DW/G-1.3 directs operation of the recombiners and the drywell unit cooler fans in an attempt to lower drywell hydrogen concentration and/or to re-distribute hydrogen within the drywell.

Recombiner operation is conditioned on hydrogen and oxygen concentrations being below deflagration limits, conditions which exist when actions in the DW/G-1 flowpath are being executed. Starting recombiners and/or allowing continued operation of the recombiners above these limits could either create the ignition source which would cause a deflagration to occur, or damage the recombiners and auxiliary system components due to operation at reaction temperatures above equipment design values. (Recombiners generate intense heat even under normal operating conditions.)

S58.1.B, Startup Of Containment Hydrogen Recombiner From Standby Condition Or Following A Trip, directs the performance of actions necessary to startup and operate the hydrogen recombiners.

Operation of the drywell unit cooler fans is specified in an attempt to re-distribute the hydrogen throughout the drywell, thereby diluting localized regions of high hydrogen concentrations.

Operators are directed to continue at Step DW/G-1.4.

DW/G-1.4 **WHEN** DW AND Supp Pool H_2 **<** 1%,

THEN continue

DISCUSSION

LGS TRIP Step DW/G-1.4 is a hold/wait step, and should not be exited until both of the conditions specified in the "WHEN" statement exist.

When hydrogen concentration has been reduced below the minimum required for recombiner operation, recombiner operation is no longer required. Step DW/G-1.4 ensures these conditions have been met before permitting the recombiners to be secured.

When it has been determined that both drywell and suppression pool hydrogen concentrations are less than 1%, operators are directed to continue at Step DW/G-1.5.

DW/G-1.5 Secure Post LOCA recombiners per S58.2.A

DISCUSSION

LGS TRIP Step DW/G-1.5 directs actions to shutdown the recombiners.

When hydrogen concentrations have been reduced below the minimum required for recombiner operation, recombiner operation is no longer required. Step DW/G-1.5, therefore, directs the shutdown of the recombiners.

S58.2.A, Shutdown Of Containment Hydrogen Recombiner To Ready Mode, directs the performance of actions necessary to shutdown the hydrogen recombiners.

DW/G-1.6 Execute CY-LG-130-009

DISCUSSION

LGS TRIP Step DW/G-1.6 directs the execution of CY-LG-130-009, Determination Of Gaseous Effluent Radmonitor Setpoints.

As mentioned previously, inerting/venting of the primary containment is only permitted in the DW/G-1 flowpath if it can be done within the limits prescribed for normal (non-emergency) plant operation (i.e., within Technical Specification and/or Offsite Dose Calculation Manual (ODCM) limits). Primary containment inerting/venting lineups at LGS discharge to either: (1) the North Vent Stack, (2) the South Vent Stack, or (3) directly to the environment (via the truck bay access doors or diesel generator corridor). Only those lineups which discharge to the North and/or South Vent Stacks are used while performing actions in the DW/G-1 flowpath.

Actions specified in CY-LG-130-009 provide the following:

- Reset of the Wide Range Accident Monitor (WRAM) high-high alarm and isolation setpoint to the Technical Specification limit. During normal plant operations, the WRAM (which monitors radioactive releases from the North Vent Stack) is set to alarm (and isolate primary containment inerting/venting discharge paths) at an offsite release less than Technical Specification limits. For the WRAM to provide accurate indication as to whether Technical Specification offsite radioactivity releases are being exceeded, its alarm and isolation setpoint must be reset.

  NOTE: The South Vent Stack high-high radiation alarm setpoint is not reset by actions specified in CY-LG-130-009. The setpoint will remain at the ODCM limit.

- Calculation of the inerting/purge flowrate, for both the North and Stack Vent Stacks, which ensures Technical Specification (North Vent Stack) and ODCM (South Vent Stack) offsite radioactive release limits will not be exceeded. Chemistry provides the Control Room with this inerting/purge flowrate value.

Operators are directed to continue at Step DW/G-1.7.

DW/G-1.7 **WHEN** it has been determined that the CY-LG-130-009 HI-HI alarm setpoint will NOT be exceeded,
THEN continue

DISCUSSION

LGS TRIP Step DW/G-1.7 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

Consistent with LGS procedures, it is appropriate to wait for the results of the analysis of the primary containment air sample specified in CY-LG-130-009, Determination Of Gaseous Effluent Radmonitor Setpoints, before commencing drywell inerting/venting operations. If existing plant conditions and the most recently obtained air sample results indicate the radioactive release to areas outside of primary containment will remain within allowable Technical Specification/Offsite Dose Calculation Manual (ODCM) limits, drywell inerting/venting may be initiated.

Actions specified in CY-LG-130-009 provide the following:

- Reset of the Wide Range Accident Monitor (WRAM) high-high alarm and isolation setpoint to the Technical Specification limit. During normal plant operations, the WRAM (which monitors radioactive releases from the North Vent Stack) is set to alarm (and isolate primary containment inerting/venting discharge paths) at an offsite release less than Technical Specification limits. For the WRAM to provide accurate indication as to whether Technical Specification offsite radioactivity releases are being exceeded, its alarm and isolation setpoint must be reset.

  NOTE: The South Vent Stack high-high radiation alarm setpoint is not reset by actions specified in CY-LG-130-009. The setpoint will remain at the ODCM limit.

- Calculation of the highest inerting/purge flowrate (for both the North and Stack Vent Stacks) which ensures Technical Specification (North Vent Stack) and ODCM (South Vent Stack) offsite radioactive release limits will not be exceeded. Chemistry provides the Control Room with this inerting/purge flowrate value.

When it has been determined that the CY-LG-130-009 high-high alarm setpoint will not be exceeded (i.e., Chemistry has provided the Control Room with the highest permissible inerting/purge flowrate value for the North and South Vent Stacks), operators are directed to continue at Step DW/G-1.8.

DW/G-1.8 **IF** CY-LG-130-009 HI-HI alarm setpoint exceeded,

THEN ensure isolation of DW vent flowpath UNLESS required by PC/P OR PC/G

DISCUSSION

LGS TRIP Step DW/G-1.8 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action.

Primary containment inerting/venting must be isolated if offsite radioactivity release reaches allowable Technical Specification/ Offsite Dose Calculation Manual (ODCM) radioactivity release limits. Unrestricted inerting/venting to reduce combustible gas concentrations is only appropriate when deflagration concentrations are reached. Therefore, Step DW/G-1.8 directs actions to ensure isolation of drywell vent flowpaths (which includes effluent flowpaths which have been established in accordance with T-228, Inerting/Purging Primary Containment, and vent flowpaths established in accordance with T-200, Primary Containment Emergency Vent Procedure) if the CY-LG-130-009 high-high alarm setpoint (for the inerting/vent lineup established) has been exceeded. Alarm levels correspond to the new, North Vent Stack high-high alarm (which has been reset to alarm at the Technical Specification limit) and the South Stack high-high alarm (which is set to alarm at the ODCM limit).

The release pathway from the North Vent Stack should automatically isolate when the North Vent Stack high-high alarm setpoint (i.e., Wide Range Accident Monitor (WRAM) high-high alarm setpoint) is reached. If the WRAM high-high alarm is received, operators should verify that the release pathway from the North Vent Stack has automatically isolated. If not, manual action shall be taken to isolate this pathway.

No automatic isolation occurs for the release pathway from the South Vent Stack. Therefore, if a drywell inerting/venting lineup has been established to the South Vent Stack, upon receipt of the South Vent Stack high-high alarm, manual action shall be taken to isolate this pathway.

It should be noted that if an indirect suppression pool inerting/ venting lineup has been established as directed by the SP/G-1, SP/G-2, or SP/G-3 flowpaths (i.e., the drywell is being inerted/vented to lower suppression pool combustible gas concentrations), or if a drywell vent lineup has been established as required by the primary containment pressure (PC/P) control flowpath, the lineup need not be secured at this time. Inerting/venting may be continued until the respective flowpath directs the lineup to be secured.

DW/G-1.9 Inert DW per T-228 using $\mathbf{N_2}$

DISCUSSION

LGS TRIP Step DW/G-1.9 directs actions to inert the drywell using nitrogen in an attempt to lower drywell hydrogen and oxygen concentrations.

Inerting is established to provide the driving force for "pushing" hydrogen and oxygen out of the drywell. Since the drywell must be inerted (i.e., oxygen concentration below 5%) in the DW/G-1 flowpath, inerting by use of nitrogen instead of air purge is used so that the inerted atmosphere can be maintained.

T-228, Inerting/Purging Primary Containment, directs the performance of actions necessary to inert the drywell with nitrogen as specified in this step.

Operators are directed to continue at Step DW/G-1.10.

DW/G-1.10 DW inerting established

DISCUSSION

LGS TRIP Step DW/G-1.10 is a decision diamond that has operators evaluate the status of drywell inerting.

A "YES" response indicates that the drywell inerting lineup has been established. Operators are directed to continue at Step DW/G-1.12, where actions to secure the inerting lineup will be directed when combustible gas concentrations have been sufficiently lowered.

A "NO" response indicates that for some reason the drywell inerting lineup could not be established. Operators are directed to continue at Step DW/G-1.11, where actions to establish a drywell vent lineup are addressed.

DW/G-1.11 Vent DW per T-200

DISCUSSION

LGS TRIP Step DW/G-1.11 directs actions to vent the drywell in an attempt to lower drywell hydrogen and oxygen concentrations.

Step DW/G-1.11 is entered as the result of a "NO" response to Step DW/G-1.10 (i.e., drywell inerting could not be established). If drywell inerting could not be established, it is appropriate to attempt establishment of a drywell vent flowpath in an attempt to lower drywell hydrogen and oxygen concentrations.

T-200, Primary Containment Emergency Vent Procedure, directs the performance of actions necessary to vent the drywell as specified in this step.

Operators are directed to continue at Step DW/G-1.12.

DW/G-1.12 **WHEN** DW $H_2 < 1\%$

THEN continue

DISCUSSION

LGS TRIP Step DW/G-1.12 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exist.

When hydrogen concentration in the drywell is reduced below the minimum detectable level, further operation of drywell inerting/venting is unnecessary. Step DW/G-1.12 ensures this condition has been met before allowing the inerting/venting lineup to be secured.

When it has been determined that drywell hydrogen concentration is less than 1%, operators are directed to continue at Step DW/G-1.13.

DW/G-1.13 Secure DW inerting UNLESS required by PC/G

OR

Secure DW vent UNLESS required by PC/G OR PC/P

DISCUSSION

LGS TRIP Step DW/G-1.13 directs actions to secure the drywell inerting/venting lineup.

When hydrogen concentration in the drywell has been reduced below the minimum detectable level and the drywell is inerted, further inerting/venting of the drywell is unnecessary.

It should be noted that if an indirect suppression pool inerting/ venting lineup has been established as directed by the SP/G-1, SP/G-2, or SP/G-3 flowpaths (i.e., the drywell is being inerted/vented to lower suppression pool combustible gas concentrations), or if a drywell vent lineup has been established as required by the primary containment pressure (PC/P) control flowpath, the lineup need not be secured at this time. Inerting/venting may be continued until the respective flowpath directs the lineup to be secured.

TABLE PC/G-1 should be consulted to determine if further actions for controlling combustible gas concentrations in the drywell are required.

DW/G-2

The DW/G-2 flowpath is entered when any of the following sets of conditions exist:

- Drywell H_2 concentration is below 1% AND drywell O_2 concentration is at or above 5% or is unknown AND suppression pool H_2 concentration is between 1% and 5.99%
- Drywell H_2 concentration is between 1% and 5.99% AND drywell O_2 concentration is at or above 5% or is unknown AND suppression pool H_2 concentration is below 1%
- Drywell H_2 concentration is between 1% and 5.99% AND drywell O_2 concentration is at or above 5% or is unknown AND suppression pool H_2 concentration is between 1% and 5.99%

Actions in the DW/G-2 flowpath for reducing drywell hydrogen and oxygen concentrations include inerting/venting of the drywell and operation of the hydrogen recombiners.

DW/G-2.1 Execute both sections concurrently

DISCUSSION

LGS TRIP Step DW/G-2.1 directs the concurrent performance of actions to lower drywell combustible gas concentrations.

Here the DW/G-2 flowpath divides into two parallel flowpaths. One flowpath deals with recombiner and drywell unit cooler fan operation, and the other deals with primary containment inerting/venting. Both flowpaths should be executed concurrently to lower combustible gas concentrations in the drywell.

Operators are directed to continue at Steps DW/G-2.2 and DW/G-2.5 concurrently.

DW/G-2.2 Operate Post LOCA recombiners per S58.1.B AND DW unit clr fans

DISCUSSION

LGS TRIP Step DW/G-2.2 directs operation of the recombiners and drywell unit cooler fans in an attempt to lower drywell hydrogen concentration and/or to re-distribute hydrogen within the drywell.

Recombiner operation is conditioned on hydrogen and oxygen concentrations being below deflagration limits, conditions which exist when actions in the DW/G-2 flowpath are being executed. Starting recombiners and/or allowing continued operation of the recombiners above these limits could either create the ignition source which would cause a deflagration to occur, or damage the recombiners and auxiliary system components due to operation at reaction temperatures above equipment design values. (Recombiners generate intense heat even under normal operating conditions.)

S58.1.B, Startup Of Containment Hydrogen Recombiner From Standby Condition Or Following A Trip, directs the performance of actions necessary to startup and operate the hydrogen recombiners.

Operation of the drywell unit cooler fans is specified in an attempt to redistribute the hydrogen throughout the drywell, thereby diluting localized regions of high hydrogen concentrations.

Operators are directed to continue at Step DW/G-2.3.

DW/G-2.3 **WHEN** DW AND Supp Pool H_2 **<** 1%,
THEN continue

DISCUSSION

LGS TRIP Step DW/G-2.3 is a hold/wait step, and should not be exited until both of the conditions specified in the "WHEN" statement exist.

When hydrogen concentration has been reduced below the minimum required for recombiner operation, recombiner operation is no longer required. Step DW/G-2.3 ensures these conditions have been met before permitting the recombiners to be secured.

When it has been determined that both drywell and suppression pool hydrogen concentrations are less than 1%, operators are directed to continue at Step DW/G-2.4.

DW/G-2.4 Secure Post LOCA recombiners per S58.2.A

DISCUSSION

LGS TRIP Step DW/G-2.4 directs actions to shutdown the recombiners.

When hydrogen concentrations have been reduced below the minimum required for recombiner operation, recombiner operation is no longer required. Step DW/G-2.4, therefore, directs the shutdown of the recombiners.

S58.2.A, Shutdown Of Containment Hydrogen Recombiner To Ready Mode, directs the performance of actions necessary to shutdown the hydrogen recombiners.

DW/G-2.5 **WHEN** core cooling NOT assured

OR

offsite release will remain below GENERAL EMERGENCY level per LGS Emergency Plan Annex, Table 3-1,

THEN continue

DISCUSSION

LGS TRIP Step DW/G-2.5 is a hold/wait step, and should not be exited until one of the two conditions specified in the "WHEN" statement exists.

Subsequent steps in this section of the DW/G-2 flowpath direct drywell inerting/venting operations. These actions are appropriate, however, only when it has been determined that either adequate core cooling is not assured or offsite release will remain below General Emergency action levels. Step DW/G-2.5 ensures that one of these conditions has been met before permitting a drywell inerting/venting lineup to be established.

If adequate core cooling cannot be established and maintained, significant amounts of hydrogen may be generated. With drywell oxygen concentration above 5%, a potential for deflagration would then exist. Nitrogen inerting or drywell venting is established in subsequent steps of the DW/G-2 flowpath to reduce oxygen concentration below the deflagration limit, thereby precluding deflagration. The potential risk to the primary containment from the presence of a deflagrable mixture of hydrogen and oxygen warrants this action even if offsite radioactivity release may exceed the offsite radioactivity release which requires a General Emergency.

Adequate core cooling is defined in the LGS TRIP procedures as heat removal from the reactor sufficient to prevent rupturing the fuel clad. Two viable mechanisms for establishing adequate core cooling exist - core submergence and steam cooling.

Submergence is the preferred method for cooling the core. The core is adequately cooled by submergence when it can be determined that RPV level is at or above the top of active fuel (TAF). All fuel nodes are then assumed to be covered with water and heat is removed by boiling heat transfer. This method is addressed in the RPV level (RC/L) control flowpath of T-101, RPV Control or when it has been determined that the RPV is flooded to the Main Steam Lines in T-116, RPV Flooding.

The core is adequately cooled by steam if the steam flow across the uncovered length of each fuel bundle is sufficient to maintain the hottest peak clad temperature (PCT) below the appropriate limiting value - 1500°F if makeup can be injected; 1800°F if makeup cannot be injected. The covered portion of the core remains cooled by boiling heat transfer and generates the steam which cools the uncovered portion.

Steam cooling with makeup capability is employed in: (1) the Level Restoration section of T-111, Level Restoration/Steam Cooling, if RPV level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level (MSCRWL, -186 inches); (2) in T-116, RPV Flooding, during an RPV flooding evolution where the reactor may not be shutdown; and (3) in T-117, Level/Power Control, if RPV level drops below TAF or if emergency RPV depressurization is required. When RPV level cannot be restored and maintained above the MSCRWL (-186 inches) in the Level Restoration section of T-111, and when RPV level drops below TAF in T-117, adequate steam flow is established by maintaining RPV level above the MSCRWL (-186 inches). When the reactor may not be shutdown during an RPV flooding evolution in T-116 and when emergency RPV depressurization is required under failure-to-scram conditions in T-117, adequate steam flow exists as long as RPV pressure is above the Minimum Steam Cooling Pressure (MSCP, 250 psig with 5 safety relief valves (SRVs) open). In all cases, PCT is limited to 1500°F, the threshold for fuel rod perforation.

Steam cooling without makeup capability is employed in the Steam Cooling section of T-111. With no makeup to the RPV, adequate steam flow exists as long as RPV level remains above the Minimum Zero-Injection RPV Water Level (MZIRWL, -198 inches). When RPV level drops below this value, emergency RPV depressurization must be performed. The PCT is permitted to rise to 1800°F, the threshold for significant metal-water reaction, to maximize the heat transfer to the steam and to delay the RPV depressurization for as long as possible.

The minimum RPV level at which adequate steam flow exists is higher when makeup capability exists because:

- The limiting fuel temperature is lower (1500°F). The higher limit of 1800°F is used only when cladding perforation cannot be avoided.
- With injection, water at the core inlet is subcooled. Some of the energy produced by the core must then be expended in raising the temperature of the liquid to saturation and less steam will be produced to cool the uncovered portions of the core.

Operators must direct Dose Assessment personnel to determine whether or not General Emergency Offsite radioactive release limits will be reached if a drywell inerting/venting lineup is established.

If offsite radioactivity release is expected to remain below the offsite radioactivity release which requires a General Emergency, nitrogen inerting or drywell venting is prescribed in subsequent steps of the DW/G-2 flowpath even if significant hydrogen generation is not anticipated. As long as offsite radioactivity release remains below that associated with a General Emergency, exposure to the public from the release will be well below that specified in the LGS licensing bases under accident conditions (10CFR100 limits). An offsite radioactivity release above the General Emergency action level, however, represents a substantial challenge to plant emergency response and, accordingly, presents a more immediate threat to the continued health and safety of the public.

LGS Emergency Plan Annex, Table 3-1 lists the offsite radioactive releases which correspond to General Emergency action levels.

When it has been determined that core cooling is not assured <u>or</u> that the offsite release will remain below General Emergency action levels as specified in LGS Emergency Plan Annex, Table 3-1, operators are directed to continue at Step DW/G-2.6.

DW/G-2.6 **IF** core cooling assured

AND

offsite release reaches GENERAL EMERGENCY level per LGS Emergency Plan Annex, Table 3-1,

THEN secure DW vent flowpath UNLESS required by PC/P OR PC/G

DISCUSSION

LGS TRIP Step DW/G-2.6 is a continue re-checking step, and as such, should be referred to frequently to determine if both of the conditions listed exist, and if so, to carry out the specified action.

Drywell inerting/venting is allowed to proceed in the DW/G-2 flowpath as long as the potential for generation of a significant amount of hydrogen exists or the offsite radioactivity release remains below that associated with a General Emergency. If neither condition exists, continued inerting/venting is inappropriate and the drywell inerting/venting lineup must be secured.

Adequate core cooling is defined in the LGS TRIP procedures as heat removal from the reactor sufficient to prevent rupturing the fuel clad. Two viable mechanisms for establishing adequate core cooling exist - core submergence and steam cooling.

Submergence is the preferred method for cooling the core. The core is adequately cooled by submergence when it can be determined that RPV level is at or above the top of active fuel (TAF). All fuel nodes are then assumed to be covered with water and heat is removed by boiling heat transfer. This method is addressed in the RPV level (RC/L) control flowpath of T-101, RPV Control or when it has been determined that the RPV is flooded to the Main Steam Lines in T-116, RPV Flooding.

The core is adequately cooled by steam if the steam flow across the uncovered length of each fuel bundle is sufficient to maintain the hottest peak clad temperature (PCT) below the appropriate limiting value - 1500°F if makeup can be injected; 1800°F if makeup cannot be injected. The covered portion of the core remains cooled by boiling heat transfer and generates the steam which cools the uncovered portion.

Steam cooling with makeup capability is employed in: (1) the Level Restoration section of T-111, Level Restoration/Steam Cooling, if RPV level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level (MSCRWL, -186 inches); (2) in T-116, RPV Flooding, during an RPV flooding evolution where the reactor may not be shutdown; and (3) in T-117, Level/Power Control, if RPV level is drops below TAF or if emergency RPV depressurization is required. When RPV level cannot be restored and maintained above the MSCRWL (-186 inches) in the Level Restoration section of T-111, and when RPV level drops below TAF in T-117, adequate steam flow is established by maintaining RPV level above the MSCRWL (-186 inches). When the reactor may not be shutdown during an RPV flooding evolution in T-116 and when emergency RPV depressurization is required under failure-to-scram conditions in T-117, adequate steam flow exists as long as RPV pressure is above the Minimum Steam Cooling Pressure (MSCP, 250 psig with 5 safety relief valves (SRVs) open). In all cases, PCT is limited to 1500°F, the threshold for fuel rod perforation.

Steam cooling without makeup capability is employed in the Steam Cooling section of T-111. With no makeup to the RPV, adequate steam flow exists as long as RPV level remains above the Minimum Zero-Injection RPV Water Level (MZIRWL, -198 inches). When RPV level drops below this value, emergency RPV depressurization must be performed. The PCT is permitted to rise to 1800°F, the threshold for significant metal-water reaction, to maximize the heat transfer to the steam and to delay the RPV depressurization for as long as possible.

The minimum RPV level at which adequate steam flow exists is higher when makeup capability exists because:

- The limiting fuel temperature is lower (1500°F). The higher limit of 1800°F is used only when cladding perforation cannot be avoided.
- With injection, water at the core inlet is subcooled. Some of the energy produced by the core must then be expended in raising the temperature of the liquid to saturation and less steam will be produced to cool the uncovered portions of the core.

Dose Assessment personnel provide the Control Room with information relative to whether offsite radioactive releases have reached and/or exceeded General Emergency action levels.

LGS Emergency Plan Annex, Table 3-1 lists the offsite releases which correspond to General Emergency action levels.

DW/G-2.7 Inert DW per T-228 using max N_2 flowrate

DISCUSSION

LGS TRIP Step DW/G-2.7 directs actions to inert the drywell using nitrogen in an attempt to lower drywell hydrogen and oxygen concentrations.

The intent of Step DW/G-2.7 is to return drywell oxygen concentration to below its deflagration limit as quickly as possible without raising primary containment pressure significantly above atmospheric pressure. Inerting with nitrogen is established, at the maximum possible rate, to provide the driving force for "pushing" hydrogen and oxygen out of the drywell as quickly as possible.

T-228, Inerting/Purging Primary Containment, directs the performance of actions necessary to inert the drywell with nitrogen as specified in this step, and also to bypass/defeat isolation interlocks which may prevent establishment of the drywell inerting lineup. For example, T-228 directs actions to defeat the Wide Range Accident Monitor (WRAM) high-high isolation of the North Vent Stack release pathway when inerting the drywell as specified in the DW/G-2 flowpath.

Operators are directed to continue at Step DW/G-2.8.

DW/G-2.8 DW inerting established

DISCUSSION

LGS TRIP Step DW/G-2.8 is a decision diamond that has operators evaluate the status of drywell inerting.

A "YES" response indicates that the drywell inerting lineup has been established. Operators are directed to continue at Step DW/G-2.10, where actions to secure the inerting lineup will be directed when combustible gas concentrations have been sufficiently lowered.

A "NO" response indicates that for some reason the drywell inerting lineup could not be established. Operators are directed to continue at Step DW/G-2.9, where actions to establish a drywell vent lineup are addressed.

DW/G-2.9 Vent DW per T-200

DISCUSSION

LGS TRIP Step DW/G-2.9 directs actions to vent the drywell in an attempt to lower drywell hydrogen and oxygen concentrations.

Step DW/G-2.9 is entered as the result of a "NO" response to Step DW/G-2.8 (i.e., drywell inerting could not be established). If drywell inerting could not be established, it is appropriate to attempt establishment of a drywell vent flowpath in an attempt to lower drywell hydrogen and oxygen concentrations.

T-200, Primary Containment Emergency Vent Procedure, directs the performance of actions necessary to vent the drywell as specified in this step, and also to bypass/defeat isolation interlocks which may prevent establishment of the drywell vent lineup. For example, T-200 directs actions to defeat the Wide Range Accident Monitor (WRAM) high-high isolation of the North Vent Stack release pathway when venting the drywell as specified in the DW/G-2 flowpath.

Operators are directed to continue at Step DW/G-2.10.

DW/G-2.10 **WHEN**
- DW H_2 **<** 1%

  AND

  EITHER
- DW O_2 **<** 5%

  OR
- Supp Pool H_2 < 1%

THEN continue

DISCUSSION

LGS TRIP Step DW/G-2.10 is a hold/wait step, and should not be exited until both of the conditions specified in the "WHEN" statement exist.

When hydrogen concentration in the drywell is reduced below the minimum detectable level and either drywell oxygen concentration is below 5% or suppression pool hydrogen is below 1%, further operation of drywell inerting/venting is unnecessary. Step DW/G-2.10 ensures these conditions have been met before allowing the inerting/venting lineup to be secured.

When it has been determined that drywell hydrogen concentration is less than 1% and either drywell oxygen concentration is below 5% or suppression pool hydrogen is below 1% operators are directed to continue at Step DW/G-2.11.

DW/G-2.11 Secure DW inerting UNLESS required by PC/G

OR

Secure DW vent UNLESS required by PC/G OR PC/P

DISCUSSION

LGS TRIP Step DW/G-2.11 directs actions to secure drywell inerting or venting.

When hydrogen concentration in the drywell has been reduced below the minimum detectable level and the drywell is inerted, further inerting or venting of the drywell is unnecessary.

It should be noted that if an indirect suppression pool inerting/venting lineup has been established as directed by the SP/G-1, SP/G-2, or SP/G-3 flowpaths (i.e., the drywell is being inerted/vented to lower suppression pool combustible gas concentrations), or if a drywell vent lineup has been established as required by the primary containment pressure (PC/P) control flowpath, the lineup need not be secured at this time. Inerting/venting may be continued until the respective flowpath directs the lineup to be secured.

TABLE PC/G-1 should be consulted to determine if further actions for controlling combustible gas concentrations in the drywell are required.

DW/G-3

The DW/G-3 flowpath is entered when either of the following sets of conditions exist:

- Drywell H_2 concentration is at or below 5.99%, AND drywell O_2 concentration is at or above 5% or is unknown, AND suppression pool H_2 concentration is at or above 6% or is unknown
- Drywell H_2 concentration is at or above 6% or is unknown AND drywell O_2 concentration is at or above 5% or is unknown

When either of the above sets of conditions exist, a deflagration could occur. The DW/G-3 flowpath directs actions to place the reactor in the safest possible state (i.e., reactor scrammed and RPV depressurized), to secure possible ignition sources, to spray the drywell, and to inert/purge/vent the drywell in an attempt to lower hydrogen and oxygen concentrations in the drywell and/or to mitigate the effects of a deflagration should one occur.

DW/G-3.1 Enter T-101 AND execute concurrently

DISCUSSION

LGS TRIP Step DW/G-3.1 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-101, RPV Control.

When deflagration concentrations exist in the primary containment, the reactor must be placed in the safest possible state (i.e., reactor scrammed and RPV depressurized). Entry into T-101 is directed to ensure the initiation of a reactor scram if one has not yet been initiated, and to ensure operators are provided with appropriate RPV level, RPV pressure, and reactor power control actions following the initiation of the reactor scram.

Operators are directed to continue at Step DW/G-3.2.

DW/G-3.2 Enter T-112 AND execute concurrently

DISCUSSION

LGS TRIP Step DW/G-3.2 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-112, Emergency Blowdown.

When deflagration concentrations exist in the primary containment, the reactor must be placed in the safest possible state (i.e., reactor scrammed and RPV depressurized). Entry into T-112 is directed because this procedure directs the performance of actions necessary to rapidly depressurize the RPV.

Operators are directed to continue at Step DW/G-3.3.

DW/G-3.3 Secure DW unit clr fans

AND

Post LOCA recombiners per S58.2.A

DISCUSSION

LGS TRIP Step DW/G-3.3 directs actions to secure the drywell unit cooler fans and recombiners.

Drywell unit cooler fans and recombiners are secured to eliminate potential ignition sources.

S58.2.A, Shutdown Of Containment Hydrogen Recombiner To Ready Mode, directs the performance of actions necessary to shutdown the hydrogen recombiners.

Operators are directed to continue at the Step DW/G-3.4.

DW/G-3.4 Inert/Purge DW per T-228 using max flowrate **EXCEEDING** offsite release rate limits if necessary

DISCUSSION

LGS TRIP Step DW/G-3.4 directs actions to inert/purge the drywell, using the maximum flowrate, in an attempt to lower hydrogen and oxygen concentrations in the drywell.

The intent of Step DW/G-3.4 is to return drywell combustible gas concentrations below deflagration limits as quickly as possible without raising primary containment pressure significantly above atmospheric pressure. In order to push hydrogen and oxygen out of the drywell as quickly as possible, either inerting with nitrogen at the maximum flowrate, or purging with air at the maximum flowrate, is established, whichever will more quickly lower drywell hydrogen and oxygen concentrations.

The potential for hydrogen deflagration warrants exceeding normal radioactivity release rate limits if necessary and defeating any isolation interlocks that are associated with the required valve lineup. The consequences of not inerting/purging could include Primary Containment damage resulting in larger, uncontrolled releases of radioactivity. Inerting/purging should not be performed indiscriminately, however. The anticipated benefits of the action should be balanced against the possible radiological consequences. Controlled releases, if necessary, should be performed in a manner that minimizes the total dose to the public while accomplishing the actions to lower combustible gas concentrations in the DW/G-3 flowpath.

The reference to LGS TRIP NOTE #17 reminds operators that exceeding offsite release rate limits in Tech Spec/ODCM is authorized if necessary. LGS TRIP NOTE #17 also provides guidance on additional actions when the Tech Spec/ODCM radioactivity release rate limits are exceeded.

T-228, Inerting/Purging Primary Containment, directs the performance of actions necessary to inert/purge the drywell as specified in this step, and also to bypass/defeat isolation interlocks which may prevent establishment of the drywell inerting/purge lineup. For example, T-228 directs actions to defeat the Wide Range Accident Monitor (WRAM) high-high isolation of the North Vent Stack release pathway when inerting/purging the drywell as specified in the DW/G-3 flowpath.

Operators are directed to continue at Step DW/G-3.5.

DW/G-3.5 DW inerting/purge established

DISCUSSION

LGS TRIP Step DW/G-3.5 is a decision diamond that has operators evaluate the status of drywell inerting/purge.

A "YES" response indicates that the drywell inerting/purge lineup has been established. Operators are directed to continue at the CAUTION preceding Step DW/G-3.7.

A "NO" response indicates that for some reason the drywell inerting/purge lineup could not be established. Operators are directed to continue at Step DW/G-3.6, where actions to establish a drywell vent lineup are addressed.

DW/G-3.6 Vent DW per T-200 **EXCEEDING** offsite release rate limits if necessary

DISCUSSION

LGS TRIP Step DW/G-3.6 directs actions to vent the drywell, in an attempt to lower hydrogen and oxygen concentrations in the drywell.

The potential for hydrogen deflagration warrants exceeding normal radioactivity release rate limits if necessary and defeating any isolation interlocks that are associated with the required valve lineup. The consequences of not inerting/purging could include Primary Containment damage resulting in larger, uncontrolled releases of radioactivity. Inerting/purging should not be performed indiscriminately, however. The anticipated benefits of the action should be balanced against the possible radiological consequences. Controlled releases, if necessary, should be performed in a manner that minimizes the total dose to the public while accomplishing the actions to lower combustible gas concentrations in the DW/G-3 flowpath.

The reference to LGS TRIP NOTE #17 reminds operators that exceeding offsite release rate limits in Tech Spec/ODCM is authorized if necessary. LGS TRIP NOTE #17 also provides guidance on additional actions when the Tech Spec/ODCM radioactivity release rate limits are exceeded.

Step DW/G-3.6 is entered as the result of a "NO" response to Step DW/G-3.5 (i.e., drywell inerting/purging lineup could not be established). If the drywell inerting/purging lineup could not be established, it is appropriate to attempt establishment of a drywell vent lineup in an attempt to lower drywell hydrogen and oxygen concentrations.

T-200, Primary Containment Emergency Vent Procedure, directs the performance of actions necessary to vent the drywell as specified in this step, and also to bypass/defeat isolation interlocks which may prevent establishment of the drywell vent lineup. For example, T-200 directs actions to defeat the Wide Range Accident Monitor (WRAM) high-high isolation of the North Vent Stack release pathway when venting the drywell as specified in the DW/G-3 flowpath.

Operators are directed to continue at the CAUTION preceding Step DW/G-3.7.

CAUTION

RHR flow above NPSH OR vortex limits may result in pump damage

DISCUSSION

This CAUTION reminds operators to remain aware of the net positive suction head (NPSH) and vortex limits for Residual Heat Removal (RHR) System pumps when they are taking a suction on the suppression pool.

NPSH and vortex limits are addressed within this CAUTION for the following reasons:

- It is difficult to define in advance exactly when the limits should be observed and when pumps should be operated irrespective of the limits.
- Pumps to which the limits apply are used in more than one flowpath. For example, RHR pumps are used in flowpaths in both T-101, RPV Control, and T-102, Primary Containment Control. Authorizing operation of the pumps "regardless of NPSH and vortex limits" in one flowpath may conflict with instructions in another flowpath where flow would normally be controlled below the limits.
- Pump characteristics, and the shape of NPSH and vortex limit curves, vary widely. If a limit is relatively flat, throttling pump flow will be of little benefit; operators can only choose whether or not to operate the pump.

Step DW/G-3.8 directs initiation of drywell sprays. If RHR pumps are used to spray the drywell, they should be operated within NPSH and vortex limits if possible. If the situation warrants, however, the limits may be exceeded. A judgment as to whether a pump should be operated beyond its limits in a particular event should consider such factors as:

- The availability of other systems
- The current trend of plant parameters
- The anticipated time such operation will be required
- The degree to which the limit will be exceeded
- The sensitivity of the pump to operation beyond the limit
- The consequences of not operating the pump beyond the limit

Immediate and catastrophic failure is not expected if a pump is operated beyond its respective NPSH or vortex limit.

NPSH limits are defined to be the highest suppression pool temperature values which provide adequate NPSH for the pumps which take a suction on the suppression pool. The NPSH limits are functions of pump flow and suppression pool overpressure (airspace pressure plus the hydrostatic head of water over the pump suction), and are utilized to preclude pump damage from cavitation.

Vortex limits are defined to be the lowest suppression pool level above which air entrainment is not expected to occur in pumps that take a suction on the suppression pool. These suppression pool levels are functions of Emergency Core Cooling System (ECCS) flow. Exceeding the limits can lead to air entrainment at the pump suction strainers.

The reference to LGS TRIP NOTE #3 reminds operators that the suppression pool level restriction of 13.5 ft. is based on RHR, Core Spray, and Reactor Core Isolation Cooling (RCIC) pump NPSH and vortex limits.

Operators are directed to continue at Step DW/G-3.7.

DW/G-3.7 Determine DW spray suct source per Table DW/G-3-1

TABLE DW/G-3-1

DW SPRAY SUCT SOURCE

CONDITION	**SUCT SOURCE**
Safe side of Curve DW/G-3-1 can be restored <u>AND</u> maintained	• Internal (Supp Pool) preferred • External (RHRSW <u>OR</u> Fire Water)
Safe side of Curve DW/G-3-1 <u>CANNOT</u> be restored <u>AND</u> maintained	Internal (Supp Pool) <u>ONLY</u>

<u>DISCUSSION</u>

LGS TRIP Step DW/G-3.7 is a continue re-checking step, and as such, should be referred to frequently during the performance of the remainder of the DW/G-3 flowpath.

At both LGS units, drywell spray flow can be supplied from sources both internal (i.e., suppression pool) and external to the primary containment (i.e., Residual Heat Removal Service Water (RHRSW) and Fire Water Systems). Both external sources will add water to the primary containment, and may be used only if primary containment water level and suppression pool pressure can be restored and maintained on the safe side of the Primary Containment Pressure Limit (PCPL, CURVE DW/G-3-1). Prior to directing initiation of drywell sprays, and following initiation of drywell sprays, operators are required to evaluate appropriate drywell spray suction sources.

The PCPL is defined to be the lesser of:

- The pressure capability of the primary containment.
- The maximum primary containment pressure at which vent valves sized to reject all decay heat from the primary containment can be opened and closed.
- The maximum primary containment pressure at which safety relief valves (SRVs) can be opened and will remain open.
- The maximum primary containment pressure at which RPV vent valves can be opened and closed.

The PCPL is a function of primary containment water level. Exceeding the limit may challenge primary containment vent valve operability, SRV operability, RPV vent valve operability, or the structural integrity of the primary containment.

If primary containment water level and suppression pool pressure can be restored and maintained on the safe side of the PCPL (CURVE DW/G-3-1), all three drywell spray suction sources may be used, as required. However, internal sources (i.e., systems which take suction on the suppression pool) are the preferred drywell spray suction source, because they will not cause primary containment water level to rise.

If primary containment water level and suppression pool pressure cannot be restored and maintained on the safe side of the PCPL (CURVE DW/G-3-1), only drywell spray sources which take a suction from an internal source may be used.

DW/G-3.8 **IF** on safe side of Curve DW/G-3-2

AND

Supp Pool level is below 38.7 ft,

THEN spray DW per T-225 **REGARDLESS** of core cooling

DISCUSSION

LGS TRIP Step DW/G-3.7 is a continue re-checking step, and as such, should be referred to frequently to determine if both conditions listed exist, and if so, to carry out the specified action.

Step DW/G-3.8 directs the initiation of drywell sprays, regardless of whether adequate core cooling is assured, provided drywell spray initiation restrictions can be met.

Drywell sprays are initiated to:

- Reduce the flammability of combustible gases through the addition of water vapor to the gas mixture.
- Suppress the temperature and pressure increase following combustion if a deflagration does occur.
- Scrub the primary containment atmosphere in anticipation of radioactivity release.

Operation of drywell sprays mitigates the consequences of a deflagration should one occur. Spraying water inside primary containment aids in:

- Minimizing the temperature and pressure rise caused by a deflagration.
- Scrubbing radioactivity from the primary containment atmosphere during inerting/purging/venting.
- Thoroughly mixing the primary containment atmosphere to reduce localized buildup of combustible gases.

A drywell inerting/purge/vent flowpath is established in Step DW/G-3.4 or DW/G-3.6, prior to the initiation of drywell sprays. When drywell sprays are initiated, energized components inside the drywell could spark and ignite the combustible atmosphere. The pressure and temperature pulse from the deflagration could render the primary containment vent valves inoperable. Opening the vent valves first ensures that a vent path will be available following drywell spray initiation.

Considering the pressure drop concerns which occur when drywell sprays are initiated, drywell spray initiation is not permitted if drywell temperature and pressure are on the unsafe side of the Drywell Spray Initiation Limit curve (DSIL, CURVE DW/G-3-2).

The restriction on suppression pool level being below 38.7 ft. is concerned with covering the suppression pool-to-drywell vacuum breakers. These vacuum breakers will not function as designed if any portion of the valve is covered with water. The specified suppression pool level assures that no portion of the drywell side of the valve is submerged for any drywell-below-suppression pool differential pressure less than or equal to the valve opening differential pressure. Drywell spray operation with vacuum breakers inoperable (i.e., with no drywell vacuum relief capability) may cause the primary containment differential pressure capability to be exceeded and therefore is not permitted.

Instructions to shut down recirculation pumps and drywell cooling fans prior to initiating drywell sprays is provided in T-225, Startup And Operation Of Suppression Pool And Drywell Spray Operation. These actions are appropriate to prevent damage to the recirculation pumps and drywell cooling fans, because they are not designed to be operated in a spray environment. Additionally, direction to secure the drywell cooling fans was previously given in Step DW/G-3.3, because they are possible ignition sources.

Authorization is provided in T-225 to defeat interlocks, as required, which may be preventing initiation of drywell sprays. Interlocks, such as those from Emergency Core Cooling System (ECCS) initiation logic channels, may preclude drywell spray operation even when adequate core cooling is assured. Since Step PC/G-8 requires an evaluation as to whether the pumps to be used for drywell spray are needed to assure adequate core cooling, the automatic logic is unnecessary and may be defeated.

Authorization is also provided in T-225 to defeat interlocks which may prevent drywell spray operation at low primary containment pressures. Defeating these interlocks permits use of drywell sprays for fission product scrubbing if the primary containment has failed or has been vented to below the interlock setpoints. Step PC/G-6, and the restrictions upon drywell spray initiation in Step DW/G-3.8, avert negative primary containment pressures, thus obviating the need for the automatic logic.

DW/G-3.9	**IF**	DW spray suct source must be swapped per step DW/G-3.7,
	THEN	secure current DW spray lineup per T-225
		AND
		return to step DW/G-3.7

DISCUSSION

LGS TRIP Step DW/G-3.9 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified actions.

Step DW/G-3.7 requires operators to continuously evaluate appropriate drywell spray suction sources. Following drywell spray initiation, if plant conditions change such that Step DW/G-3.7 requires the drywell spray suction source to be swapped (from an external source to an internal source), direction is provided to secure the current drywell spray lineup per T-225, and to return to Step DW/G-3.7. The return to Step DW/G-3.7 ensures that all restrictions on drywell spray initiation can be met before re-establishing drywell spray flow with the new suction source.

T-225, Startup And Operation Of Suppression Pool And Drywell Spray Operation, directs the performance of actions necessary to secure the drywell spray lineup.

SP/G-1

The SP/G-1 flowpath is entered when either of the following sets of conditions exist:

- Suppression pool H_2 concentration is between 1% and 5.99% AND suppression pool O_2 concentration is below 5%
- Suppression pool H_2 concentration is at or above 6% or is unknown AND suppression pool O_2 concentration is below 5%

The existence of a detectable amount of hydrogen in the suppression pool warrants corrective action regardless of the amount of oxygen or the condition which required entry to this guideline.
Inerting/Venting and hydrogen recombiner operation are the two methods normally used to control primary containment atmospheric conditions.
Although continued rises in hydrogen concentration above the deflagration limit in the presence of oxygen will potentially threaten primary containment integrity, the presence of hydrogen in the absence of oxygen is not by itself primary containment threatening.
Inerting/Venting is therefore permitted in the SP/G-1 flowpath only if it can be done within the limits prescribed for normal (non-emergency) plant operation.

SP/G-1.1 Execute both sections concurrently

DISCUSSION

LGS TRIP Step SP/G-1.1 directs the concurrent performance of actions to lower suppression pool combustible gas concentrations.

Here the SP/G-1 flowpath divides into two parallel flowpaths. One flowpath deals with recombiner operation, and the other deals with primary containment inerting/venting. Both flowpaths should be executed concurrently to lower combustible gas concentrations in the suppression pool.

Operators are directed to continue at Steps SP/G-1.2 and SP/G-1.6 concurrently.

SP/G-1.2 **WHEN** Supp Pool OR DW $H_2 > 1\%$,

THEN continue

DISCUSSION

LGS TRIP Step SP/G-1.2 is a hold/wait step, and should not be exited until one of the two conditions specified in the "WHEN" statement exists.

The hydrogen concentration limit specified in Step SP/G-1.2 is the minimum hydrogen concentration required for recombiner operation. Starting recombiners below this limit would serve no useful purpose because there would be an insufficient supply of hydrogen to support recombination. (NOTE: there is no lower oxygen concentration limit for recombiner operation).

When it has been determined that either suppression pool or drywell hydrogen concentration is above 1%, operators are directed to continue at Step SP/G-1.3.

SP/G-1.3 Operate Post LOCA recombiners per S58.1.B as necessary

DISCUSSION

LGS TRIP Step SP/G-1.3 directs operation of the recombiners in an attempt to lower suppression pool hydrogen and oxygen concentrations.

Recombiner operation is conditioned on hydrogen and oxygen concentrations being below deflagration limits, conditions which exist when actions in the SP/G-1 flowpath are being executed. Starting recombiners and/or allowing continued operation of the recombiners above these limits could either create the ignition source which would cause a deflagration to occur, or damage the recombiners and auxiliary system components due to operation at reaction temperatures above equipment design values. (Recombiners generate intense heat even under normal operating conditions.)

S58.1.B, Startup Of Containment Hydrogen Recombiner From Standby Condition Or Following A Trip, directs the performance of actions necessary to startup and operate the hydrogen recombiners.

Operators are directed to continue at Step SP/G-1.4.

SP/G-1.4 **WHEN** Supp Pool AND DW H_2 **<** 1%,
THEN continue

DISCUSSION

LGS TRIP Step SP/G-1.4 is a hold/wait step, and should not be exited until both of the conditions specified in the "WHEN" statement exist.

When hydrogen concentration has been reduced below the minimum required for recombiner operation, recombiner operation is no longer required. Step SP/G-1.4 ensures these conditions have been met before permitting the recombiners to be secured.

When it has been determined that both drywell and suppression pool hydrogen concentrations are less than 1%, operators are directed to continue at Step SP/G-1.5.

SP/G-1.5 Secure Post LOCA recombiners per S58.2.A

DISCUSSION

LGS TRIP Step SP/G-1.5 directs actions to shutdown the recombiners.

When hydrogen concentrations have been reduced below the minimum required for recombiner operation, recombiner operation is no longer required. Step SP/G-1.5, therefore, directs the shutdown of the recombiners.

S58.2.A, Shutdown Of Containment Hydrogen Recombiner To Ready Mode, directs the performance of actions necessary to shutdown the hydrogen recombiners.

SP/G-1.6 Execute CY-LG-130-009

DISCUSSION

LGS TRIP Step SP/G-1.6 directs the execution of CY-LG-130-009, Determination Of Gaseous Effluent Radmonitor Setpoints.

As mentioned previously, inerting/venting of the primary containment is only permitted in the SP/G-1 flowpath if it can be done within the limits prescribed for normal (non-emergency) plant operation (i.e., within Technical Specification and/or Offsite Dose Calculation Manual (ODCM) limits). Primary containment inerting/venting lineups at LGS discharge to either: (1) the North Vent Stack, (2) the South Vent Stack, or (3) directly to the environment (via the truck bay access doors or diesel generator corridor). Only those lineups which discharge to the North and/or South Vent Stacks are used while performing actions in the SP/G-1 flowpath.

Actions specified in CY-LG-130-009 provide the following:

- Reset of the Wide Range Accident Monitor (WRAM) high-high alarm and isolation setpoint to the Technical Specification limit. During normal plant operations, the WRAM (which monitors radioactive releases from the North Vent Stack) is set to alarm (and isolate primary containment inerting/venting discharge paths) at an offsite release less than Technical Specification limits. For the WRAM to provide accurate indication as to whether Technical Specification offsite radioactivity releases are being exceeded, its alarm and isolation setpoint must be reset.

  NOTE: The South Vent Stack high-high radiation alarm setpoint is not reset by actions specified in CY-LG-130-009. The setpoint will remain at the ODCM limit.

- Calculation of the inerting/purge flowrate, for both the North and Stack Vent Stacks, which ensures Technical Specification (North Vent Stack) and ODCM (South Vent Stack) offsite radioactive release limits will not be exceeded. Chemistry provides the Control Room with this inerting/purge flowrate value.

Operators are directed to continue at Step SP/G-1.7.

SP/G-1.7 **WHEN** it has been determined that the CY-LG-130-009 HI-HI alarm setpoint will NOT be exceeded,

THEN continue

DISCUSSION

LGS TRIP Step SP/G-1.7 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

Consistent with LGS procedures, it is appropriate to wait for the results of the analysis of the primary containment air sample specified in CY-LG-130-009, Determination Of Gaseous Effluent Radmonitor Setpoints, before commencing suppression pool inerting/venting operations. If existing plant conditions and the most recently obtained air sample results indicate the radioactive release to areas outside of primary containment will remain within allowable Technical Specification/Offsite Dose Calculation Manual (ODCM) limits, suppression pool inerting/venting may be initiated.

Actions specified in CY-LG-130-009 provide the following:

- Reset of the Wide Range Accident Monitor (WRAM) high-high alarm and isolation setpoint to the Technical Specification limit. During normal plant operations, the WRAM (which monitors radioactive releases from the North Vent Stack) is set to alarm (and isolate primary containment inerting/venting discharge paths) at an offsite release less than Technical Specification limits. For the WRAM to provide accurate indication as to whether Technical Specification offsite radioactivity releases are being exceeded, its alarm and isolation setpoint must be reset.

  NOTE: The South Vent Stack high-high radiation alarm setpoint is not reset by actions specified in CY-LG-130-009. The setpoint will remain at the ODCM limit.

- Calculation of the highest inerting/purge flowrate (for both the North and Stack Vent Stacks) which ensures Technical Specification (North Vent Stack) and ODCM (South Vent Stack) offsite radioactive release limits will not be exceeded. Chemistry provides the Control Room with this inerting/purge flowrate value.

When it has been determined that the CY-LG-130-009 high-high alarm setpoint will not be exceeded (i.e., Chemistry has provided the Control Room with the highest permissible inerting/purge flowrate value for the North and South Vent Stacks), operators are directed to continue at Step SP/G-1.8.

SP/G-1.8 **IF** CY-LG-130-009 HI-HI alarm setpoint exceeded,

THEN ensure isolation of Supp Pool vent flowpath UNLESS required by PC/P OR PC/G

DISCUSSION

LGS TRIP Step SP/G-1.8 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action.

Primary containment inerting/venting must be isolated if offsite radioactivity release reaches allowable Technical Specification/ Offsite Dose Calculation Manual (ODCM) radioactivity release limits. Unrestricted inerting/venting to reduce combustible gas concentrations is only appropriate when deflagration concentrations are reached. Therefore, Step SP/G-1.8 directs actions to ensure isolation of suppression pool vent flowpaths (which includes effluent flowpaths which have been established in accordance with T-228, Inerting/Purging Primary Containment, and vent flowpaths established in accordance with T-200, Primary Containment Emergency Vent Procedure) if the CY-LG-130-009 high-high alarm setpoint (for the inerting/vent lineup established) has been exceeded. Alarm levels correspond to the new, North Vent Stack high-high alarm (which has been reset to alarm at the Technical Specification limit) and the South Stack high-high alarm (which is set to alarm at the ODCM limit).

The release pathway from the North Vent Stack should automatically isolate when the North Vent Stack high-high alarm setpoint (i.e., Wide Range Accident Monitor (WRAM) high-high alarm setpoint) is reached. If the WRAM high-high alarm is received, operators should verify that the release pathway from the North Vent Stack has automatically isolated. If not, manual action shall be taken to isolate this pathway.

No automatic isolation occurs for the release pathway from the South Vent Stack. Therefore, if a suppression pool inerting/venting lineup has been established to the South Vent Stack, upon receipt of the South Vent Stack high-high alarm, manual action shall be taken to isolate this pathway.

It should be noted that if an indirect drywell inerting/venting lineup has been established as directed by the DW/G-1, DW/G-2, or DW/G-3 flowpaths (i.e., the suppression pool is being inerted/vented to lower drywell combustible gas concentrations), or if a suppression pool vent lineup has been established as required by the primary containment pressure (PC/P) control flowpath, the lineup need not be secured at this time. Inerting/venting may be continued until the respective flowpath directs the lineup to be secured.

SP/G-1.9 Inert Supp Pool per T-228 using N_2

DISCUSSION

LGS TRIP Step SP/G-1.9 directs actions to inert the suppression pool using nitrogen in an attempt to lower suppression pool hydrogen and oxygen concentrations.

Inerting is established to provide the driving force for "pushing" hydrogen and oxygen out of the suppression pool. Since the suppression pool must be inerted in the SP/G-1 flowpath (i.e., oxygen concentration below 5%), inerting by use of nitrogen instead of air purge is used so that the inerted atmosphere can be maintained.

T-228, Inerting/Purging Primary Containment, directs the performance of actions necessary to inert the suppression pool with nitrogen as specified in this step.

Operators are directed to continue at Step SP/G-1.10.

SP/G-1.10 Supp Pool inerting established

DISCUSSION

LGS TRIP Step SP/G-1.10 is a decision diamond that has operators evaluate the status of suppression pool inerting.

A "YES" response indicates that the suppression pool inerting lineup has been established. Operators are directed to continue at Step SP/G-1.12, where actions to secure the inerting lineup will be directed when combustible gas concentrations have been sufficiently lowered.

A "NO" response indicates that for some reason the suppression pool inerting lineup could not be established. Operators are directed to continue at Step SP/G-1.11, where actions to establish a suppression pool vent lineup are addressed.

SP/G-1.11 Vent Supp Pool per T-200

DISCUSSION

LGS TRIP Step SP/G-1.11 directs actions to vent the suppression pool in an attempt to lower suppression pool hydrogen and oxygen concentrations.

Step SP/G-1.11 is entered as the result of a "NO" response to Step SP/G-1.10 (i.e., suppression pool inerting could not be established). If suppression pool inerting could not be established, it is appropriate to attempt establishment of a suppression pool vent lineup in an attempt to lower suppression pool hydrogen and oxygen concentrations.

T-200, Primary Containment Emergency Vent Procedure, directs the performance of actions necessary to vent the suppression pool as specified in this step.

Operators are directed to continue at Step SP/G-1.12.

SP/G-1.12 **WHEN** Supp Pool $H_2 < 1\%$

THEN continue

DISCUSSION

LGS TRIP Step SP/G-1.12 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exist.

When hydrogen concentration in the suppression pool is reduced below the minimum detectable level, further operation of suppression pool inerting/venting is unnecessary. Step SP/G-1.12 ensures this condition has been met before allowing the inerting/venting lineup to be secured.

When it has been determined that suppression pool hydrogen concentration is less than 1%, operators are directed to continue at Step SP/G-1.13.

SP/G-1.13 Secure Supp Pool inerting UNLESS required by PC/G

OR

Secure Supp Pool vent UNLESS required by PC/G OR PC/P

DISCUSSION

LGS TRIP Step SP/G-1.13 directs actions to secure the suppression pool inerting/venting lineup.

When hydrogen concentration in the suppression pool has been reduced below the minimum detectable level and the suppression pool is inerted, further inerting/venting of the suppression pool is unnecessary.

It should be noted that if an indirect drywell inerting/venting lineup has been established as directed by the DW/G-1, DW/G-2, or DW/G-3 flowpaths (i.e., the suppression pool is being inerted/vented to lower drywell combustible gas concentrations), or if a suppression pool vent lineup has been established as required by the primary containment pressure (PC/P) control flowpath, the lineup need not be secured at this time. Inerting/venting may be continued until the respective flowpath directs the lineup to be secured.

TABLE PC/G-2 should be consulted to determine if further actions for controlling combustible gas concentrations in the suppression pool are required.

SP/G-2

The SP/G-2 flowpath is entered when any of the following set of conditions exist:

- Suppression pool H_2 concentration is below 1% AND suppression pool O_2 concentration is at or above 5% or is unknown AND drywell H_2 concentration is between 1% and 5.99%
- Suppression pool H_2 concentration is between 1% and 5.99% AND suppression pool O_2 concentration is at or above 5% or is unknown AND drywell H_2 concentration is below 1%
- Suppression pool H_2 concentration is between 1% and 5.99% AND suppression pool O_2 concentration is at or above 5% or is unknown AND drywell H_2 concentration is between 1% and 5.99%

Actions in the SP/G-2 flowpath for reducing suppression pool hydrogen and oxygen concentrations include inerting/venting of the suppression pool and operation of the hydrogen recombiners.

SP/G-2.1 Execute both sections concurrently

DISCUSSION

LGS TRIP Step SP/G-2.1 directs the concurrent performance of actions to lower suppression combustible gas concentrations.

Here the SP/G-2 flowpath divides into two parallel flowpaths. One flowpath deals with recombiner operation, and the other deals with primary containment inerting/venting. Both flowpaths should be executed concurrently to lower combustible gas concentrations in the suppression pool.

Operators are directed to continue at Steps SP/G-2.2 and SP/G-2.5 concurrently.

SP/G-2.2 Operate Post LOCA recombiners per S58.1.B

DISCUSSION

LGS TRIP Step SP/G-2.2 directs operation of the recombiners in an attempt to lower suppression pool hydrogen concentration.

Recombiner operation is conditioned on hydrogen and oxygen concentrations being below deflagration limits, conditions which exist when actions in the SP/G-2 flowpath are being executed. Starting recombiners and/or allowing continued operation of the recombiners above these limits could either create the ignition source which would cause a deflagration to occur, or damage the recombiners and auxiliary system components due to operation at reaction temperatures above equipment design values. (Recombiners generate intense heat even under normal operating conditions.)

S58.1.B, Startup Of Containment Hydrogen Recombiner From Standby Condition Or Following A Trip, directs the actions necessary to startup and operate the hydrogen recombiners.

Operators are directed to continue at Step SP/G-2.3.

SP/G-2.3 **WHEN** Supp Pool AND DW H_2 **<** 1%,

THEN continue

DISCUSSION

LGS TRIP Step SP/G-2.3 is a hold/wait step, and should not be exited until both of the conditions specified in the "WHEN" statement exist.

When hydrogen concentrations have been reduced below the minimum required for recombiner operation, recombiner operation is no longer required. Step SP/G-2.3 ensures these conditions have been met before permitting the recombiners to be secured.

When it has been determined that both drywell and suppression pool hydrogen concentrations are less than 1%, operators are directed to continue at Step SP/G-2.4.

SP/G-2.4 Secure Post LOCA recombiners per S58.2.A

DISCUSSION

LGS TRIP Step SP/G-2.4 directs actions to shutdown the recombiners.

When hydrogen concentrations have been reduced below the minimum required for recombiner operation, recombiner operation is no longer required. Step SP/G-2.4, therefore, directs the shutdown of the recombiners.

S58.2.A, Shutdown Of Containment Hydrogen Recombiner To Ready Mode, directs the performance of actions necessary to shutdown the hydrogen recombiners.

SP/G-2.5 **WHEN** core cooling NOT assured

OR

offsite release will remain below GENERAL EMERGENCY level per LGS Emergency Plan Annex, Table 3-1,

THEN continue

DISCUSSION

LGS TRIP Step SP/G-2.5 is a hold/wait step, and should not be exited until one of the two conditions specified in the "WHEN" statement exists.

Subsequent steps in this section of the SP/G-2 flowpath direct suppression pool inerting/venting operations. These actions are appropriate, however, only when it has been determined that either adequate core cooling is not assured or offsite release will remain below General Emergency action levels. Step SP/G-2.5 ensures that one of these conditions has been met before permitting a suppression pool inerting/venting lineup to be established.

If adequate core cooling cannot be established and maintained, significant amounts of hydrogen may be generated. With suppression pool oxygen concentration above 5%, a potential for deflagration would then exist. Nitrogen inerting or suppression pool venting is established in subsequent steps of the SP/G-2 flowpath to reduce oxygen concentration below the deflagration limit, thereby precluding deflagration. The potential risk to the primary containment from the presence of a deflagrable mixture of hydrogen and oxygen warrants this action even if offsite radioactivity release may exceed the offsite radioactivity release which requires a General Emergency.

Adequate core cooling is defined in the LGS TRIP procedures as heat removal from the reactor sufficient to prevent rupturing the fuel clad. Two viable mechanisms for establishing adequate core cooling exist - core submergence and steam cooling.

Submergence is the preferred method for cooling the core. The core is adequately cooled by submergence when it can be determined that RPV level is at or above the top of active fuel (TAF). All fuel nodes are then assumed to be covered with water and heat is removed by boiling heat transfer. This method is addressed in the RPV level (RC/L) control flowpath of T-101, RPV Control or when it has been determined that the RPV has been flooded to the Main Steam Lines in T-116, RPV Flooding.

The core is adequately cooled by steam if the steam flow across the uncovered length of each fuel bundle is sufficient to maintain the hottest peak clad temperature (PCT) below the appropriate limiting value - 1500°F if makeup can be injected; 1800°F if makeup cannot be injected. The covered portion of the core remains cooled by boiling heat transfer and generates the steam which cools the uncovered portion.

Steam cooling with makeup capability is employed in: (1) the Level Restoration section of T-111, Level Restoration/Steam Cooling, if RPV level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level (MSCRWL, -186 inches); (2) in T-116, RPV Flooding, during an RPV flooding evolution where the reactor may not be shutdown; and (3) in T-117, Level/Power Control, if RPV level is drops below TAF or if emergency RPV depressurization is required. When RPV level cannot be restored and maintained above the MSCRWL (-186 inches) in the Level Restoration section of T-111, and when RPV level drops below TAF in T-117, adequate steam flow is established by maintaining RPV level above the MSCRWL (-186 inches). When the reactor may not be shutdown during an RPV flooding evolution in T-116 and when emergency RPV depressurization is required under failure-to-scram conditions in T-117, adequate steam flow exists as long as RPV pressure is above the Minimum Steam Cooling Pressure (MSCP, 250 psig with 5 safety relief valves (SRVs) open). In all cases, PCT is limited to 1500°F, the threshold for fuel rod perforation.

Steam cooling without makeup capability is employed in the Steam Cooling section of T-111. With no makeup to the RPV, adequate steam flow exists as long as RPV level remains above the Minimum Zero-Injection RPV Water Level (MZIRWL, -198 inches). When RPV level drops below this value, emergency RPV depressurization must be performed. The PCT is permitted to rise to 1800°F, the threshold for significant metal-water reaction, to maximize the heat transfer to the steam and to delay the RPV depressurization for as long as possible.

The minimum RPV level at which adequate steam flow exists is higher when makeup capability exists because:

- The limiting fuel temperature is lower (1500°F). The higher limit of 1800°F is used only when cladding perforation cannot be avoided.
- With injection, water at the core inlet is subcooled. Some of the energy produced by the core must then be expended in raising the temperature of the liquid to saturation and less steam will be produced to cool the uncovered portions of the core.

Dose Assessment personnel provide the Control Room with information relative to whether offsite radioactive releases have reached and/or exceeded General Emergency action levels.

When it has been determined that core cooling is not assured or that offsite release will remain below General Emergency levels as specified in LGS Emergency Plan Annex, Table 3-1, operators are directed to continue at Step SP/G-2.6.

SP/G-2.6	**IF**	core cooling assured
		AND
		offsite release reaches GENERAL EMERGENCY level per LGS Emergency Plan Annex, Table 3-1,
	THEN	secure Supp Pool vent flowpath UNLESS required by PC/P OR PC/G

DISCUSSION

LGS TRIP Step SP/G-2.6 is a continue re-checking step, and as such, should be referred to frequently to determine if both conditions listed exist, and if so, to carry out the specified action.

Suppression pool inerting/venting is allowed to proceed in the SP/G-2 flowpath as long as the potential for generation of a significant amount of hydrogen exists or the offsite radioactivity release remains below that associated with a General Emergency. If neither condition exists, continued inerting/venting is inappropriate and the drywell inerting/venting lineup must be secured.

Adequate core cooling is defined in the LGS TRIP procedures as heat removal from the reactor sufficient to prevent rupturing the fuel clad. Two viable mechanisms for establishing adequate core cooling exist - core submergence and steam cooling.

Submergence is the preferred method for cooling the core. The core is adequately cooled by submergence when it can be determined that RPV level is at or above the top of active fuel (TAF). All fuel nodes are then assumed to be covered with water and heat is removed by boiling heat transfer. This method is addressed in the RPV level (RC/L) control flowpath of T-101, RPV Control or when it has been determined that the RPV is flooded to the Main Steam Lines in T-116, RPV Flooding.

The core is adequately cooled by steam if the steam flow across the uncovered length of each fuel bundle is sufficient to maintain the hottest peak clad temperature (PCT) below the appropriate limiting value - 1500°F if makeup can be injected; 1800°F if makeup cannot be injected. The covered portion of the core remains cooled by boiling heat transfer and generates the steam which cools the uncovered portion.

Steam cooling with makeup capability is employed in: (1) the Level Restoration section of T-111, Level Restoration/Steam Cooling, if RPV level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level (MSCRWL, -186 inches); (2) in T-116, RPV Flooding, during an RPV flooding evolution where the reactor may not be shutdown; and (3) in T-117, Level/Power Control, if RPV level is drops below TAF or if emergency RPV depressurization is required. When RPV level cannot be restored and maintained above the MSCRWL (-186 inches) in the Level Restoration section of T-111, and when RPV level drops below TAF in T-117, adequate steam flow is established by maintaining RPV level above the MSCRWL (-186 inches). When the reactor may not be shutdown during an RPV flooding evolution in T-116 and when emergency RPV depressurization is required under failure-to-scram conditions in T-117, adequate steam flow exists as long as RPV pressure is above the Minimum Steam Cooling Pressure (MSCP, 250 psig with 5 safety relief valves (SRVs) open). In all cases, PCT is limited to 1500°F, the threshold for fuel rod perforation.

Steam cooling without makeup capability is employed in the Steam Cooling section of T-111. With no makeup to the RPV, adequate steam flow exists as long as RPV level remains above the Minimum Zero-Injection RPV Water Level (MZIRWL, -198 inches). When RPV level drops below this value, emergency RPV depressurization must be performed. The PCT is permitted to rise to 1800°F, the threshold for significant metal-water reaction, to maximize the heat transfer to the steam and to delay the RPV depressurization for as long as possible.

The minimum RPV level at which adequate steam flow exists is higher when makeup capability exists because:

- The limiting fuel temperature is lower (1500°F). The higher limit of 1800°F is used only when cladding perforation cannot be avoided.
- With injection, water at the core inlet is subcooled. Some of the energy produced by the core must then be expended in raising the temperature of the liquid to saturation and less steam will be produced to cool the uncovered portions of the core.

Dose Assessment personnel provide the Control Room with information relative to whether offsite radioactive releases have reached and/or exceeded General Emergency action levels.

LGS Emergency Plan Annex, Table 3-1 lists the offsite releases which correspond to General Emergency action levels.

It should be noted that if an indirect drywell inerting/venting lineup has been established as directed by the DW/G-1, DW/G-2 or DW/G-3 flowpaths (i.e., the suppression pool is being inerted/vented to lower drywell combustible gas concentrations), or if a suppression pool vent lineup has been established as required by the PC/P flowpath, the lineup need not be secured at this time. Inerting/venting may be continued until the respective flowpath directs the lineup to be secured.

SP/G-2.7 Inert Supp Pool per T-228 using max N_2 flowrate

DISCUSSION

LGS TRIP Step SP/G-2.7 directs actions to inert the suppression pool using nitrogen in an attempt to lower suppression pool hydrogen and oxygen concentrations.

The intent of Step SP/G-2.7 is to return suppression pool oxygen concentration to below its deflagration limit as quickly as possible without raising primary containment pressure significantly above atmospheric pressure. Nitrogen inerting is established, at the maximum possible rate, to provide the driving force for "pushing" hydrogen and oxygen out of the suppression pool as quickly as possible.

T-228, Inerting/Purging Primary Containment, directs the performance of actions necessary to inert the suppression pool with nitrogen as specified in this step, and also to bypass/defeat isolation interlocks which may prevent establishment of the suppression pool inerting lineup. For example, T-228 directs actions to defeat the Wide Range Accident Monitor (WRAM) high-high isolation of the North Vent Stack release pathway when inerting the suppression pool as specified in the SP/G-2 flowpath.

Operators are directed to continue at Step SP/G-2.8.

SP/G-2.8 Supp Pool inerting established

DISCUSSION

LGS TRIP Step SP/G-2.8 is a decision diamond that has operators evaluate the status of suppression pool inerting.

A "YES" response indicates that the suppression pool inerting lineup has been established. Operators are directed to continue at Step SP/G-2.10, where actions to secure the inerting lineup will be directed when combustible gas concentrations have been sufficiently lowered.

A "NO" response indicates that for some reason the suppression pool inerting lineup could not be established. Operators are directed to continue at Step SP/G-2.9, where actions to establish a suppression pool vent lineup are addressed.

SP/G-2.9 Vent Supp Pool per T-200

DISCUSSION

LGS TRIP Step SP/G-2.9 directs actions to vent the suppression pool in an attempt to lower suppression pool hydrogen and oxygen concentrations.

Step SP/G-2.9 is entered as the result of a "NO" response to Step SP/G-2.8 (i.e., suppression pool inerting could not be established). If suppression pool inerting could not be established, it is appropriate to attempt establishment of a suppression pool vent lineup in an attempt to lower suppression pool hydrogen and oxygen concentrations.

T-200, Primary Containment Emergency Vent Procedure, directs the performance of actions necessary to vent the suppression pool as specified in this step, and also to bypass/defeat isolation interlocks which may prevent establishment of the suppression pool vent lineup. For example, T-200 directs actions to defeat the Wide Range Accident Monitor (WRAM) high-high isolation of the North Vent Stack release pathway when venting the suppression pool as specified in the SP/G-2 flowpath.

Operators are directed to continue at Step SP/G-2.10.

SP/G-2.10 **WHEN**

- Supp Pool H_2 less than 1%

AND

EITHER

- Supp Pool O_2 < 5%

OR

- DW H_2 **<** 1%

THEN continue

DISCUSSION

LGS TRIP Step SP/G-2.10 is a hold/wait step, and should not be exited until both of the conditions specified in the "WHEN" statement exist.

When hydrogen concentration in the suppression pool is reduced below the minimum detectable level and the suppression pool is inerted (i.e., oxygen concentration below 5%), further inerting/venting of the suppression pool is unnecessary. Step SP/G-2.10 ensures these conditions have been met before allowing the inerting/venting lineup to be secured.

When it has been determined that suppression pool hydrogen concentration is less than 1% and either suppression pool oxygen concentration is below 5% or suppression pool hydrogen is below 1%, operators are directed to continue at Step SP/G-2.11.

SP/G-2.11 Secure Supp Pool inerting UNLESS required by PC/G

OR

Secure Supp Pool vent UNLESS required by PC/G OR PC/P

DISCUSSION

LGS TRIP Step SP/G-2.11 directs actions to secure the suppression pool inerting/venting lineup.

When hydrogen concentration in the suppression pool has been reduced below the minimum detectable level and the suppression pool is inerted, further inerting/venting of the suppression pool is unnecessary.

It should be noted that if an indirect drywell inerting/venting lineup has been established as directed by the DW/G-1, DW/G-2 or DW/G-3 flowpaths (i.e., the suppression pool is being inerted/vented to lower drywell combustible gas concentrations), or if a suppression pool vent lineup has been established as required by the PC/P flowpath, the lineup need not be secured at this time. Inerting/venting may be continued until the respective flowpath directs the lineup to be secured.

TABLE PC/G-2 should be consulted to determine if further actions for controlling combustible gas concentrations in the suppression pool are required.

SP/G-3

The SP/G-3 flowpath is entered when either of the following sets of conditions exist:

- Suppression pool H_2 concentration is at or below 5.99% AND suppression pool O_2 concentration is at or above 5% or is unknown AND drywell H_2 concentration is at or above 6% or is unknown
- Suppression pool H_2 concentration is at or above 6% or is unknown AND suppression pool O_2 concentration is at or above 5% or is unknown

When either of the above sets of conditions exist, a deflagration could occur. The SP/G-3 flowpath directs actions to place the reactor in the safest possible state (i.e., reactor scrammed and RPV depressurized), to secure possible ignition sources, to spray the suppression pool, and to inert/purge/vent the suppression pool in an attempt to lower hydrogen and oxygen concentrations in the suppression pool and/or to mitigate the effects of a deflagration should one occur.

SP/G-3.1 Enter T-101 AND execute concurrently

DISCUSSION

LGS TRIP Step SP/G-3.1 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-101, RPV Control.

When deflagration concentrations exist in the primary containment, the reactor must be placed in the safest possible state (i.e., reactor scrammed and RPV depressurized). Entry into T-101 is directed to ensure the initiation of a reactor scram if one has not yet been initiated, and to ensure operators are provided with appropriate RPV level, RPV pressure, and reactor power control actions following the initiation of the reactor scram.

Operators are directed to continue at Step SP/G-3.2.

SP/G-3.2 Enter T-112 AND execute concurrently

DISCUSSION

LGS TRIP Step SP/G-3.2 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-112, Emergency Blowdown.

When deflagration concentrations exist in the primary containment, the reactor must be placed in the safest possible state (i.e., reactor scrammed and RPV depressurized). Entry into T-112 is directed because this procedure directs the actions required to rapidly depressurize the RPV.

Operators are directed to continue at Step SP/G-3.3.

CAUTION

RHR flow above NPSH OR vortex limits may result in pump damage

DISCUSSION

This CAUTION reminds operators to remain aware of the net positive suction head (NPSH) and vortex limits for Residual Heat Removal (RHR) System pumps when they are taking a suction on the suppression pool.

NPSH and vortex limits are addressed within this CAUTION for the following reasons:

- It is difficult to define in advance exactly when the limits should be observed and when pumps should be operated irrespective of the limits.
- Pumps to which the limits apply are used in more than one flowpath. For example, RHR pumps are used in flowpaths in both T-101, RPV Control, and T-102, Primary Containment Control. Authorizing operation of the pumps "regardless of NPSH and vortex limits" in one flowpath may conflict with instructions in another flowpath where flow would normally be controlled below the limits.
- Pump characteristics, and the shape of NPSH and vortex limit curves, vary widely. If a limit is relatively flat, throttling pump flow will be of little benefit; operators can only choose whether or not to operate the pump.

Step SP/G-3.4 directs initiation of suppression pool sprays. If RHR pumps are used to spray the suppression pool, they should be operated within NPSH and vortex limits if possible. If the situation warrants, however, the limits may be exceeded. A judgment as to whether a pump should be operated beyond its limits in a particular event should consider such factors as:

- The availability of other systems
- The current trend of plant parameters
- The anticipated time such operation will be required
- The degree to which the limit will be exceeded
- The sensitivity of the pump to operation beyond the limit
- The consequences of not operating the pump beyond the limit

Immediate and catastrophic failure is not expected if a pump is operated beyond its respective NPSH or vortex limit.

NPSH Limits are defined to be the highest suppression pool temperature values which provide adequate NPSH for the pumps which take a suction on the suppression pool. The NPSH limits are functions of pump flow and suppression pool overpressure (airspace pressure plus the hydrostatic head of water over the pump suction), and are utilized to preclude pump damage from cavitation.

Vortex limits are defined to be the lowest suppression pool level above which air entrainment is not expected to occur in pumps that take a suction on the suppression pool. These suppression pool levels are functions of Emergency Core Cooling System (ECCS) flow. Exceeding the limits can lead to air entrainment at the pump suction strainers.

The reference to LGS TRIP NOTE #3 reminds operators that the suppression pool level restriction of 13.5 ft. is based on RHR, Core Spray, and Reactor Core Isolation Cooling (RCIC) pump NPSH and vortex limits.

Operators are directed to continue at Step SP/G-3.3.

SP/G-3.3 Determine Supp Pool spray suct source per Table SP/G-3-1

TABLE SP/G-3-1

SUPP POOL SPRAY SUCT SOURCE

CONDITION	SUCT SOURCE
Safe side of Curve SP/G-3-1 can be restored AND maintained	• Internal (Supp Pool) preferred • External (RHRSW OR Fire Water)
Safe side of Curve SP/G-3-1 CANNOT be restored AND maintained	Internal (Supp Pool) ONLY

DISCUSSION

LGS TRIP Step SP/G-3.3 is a continue re-checking step, and as such, should be referred to frequently during the performance of the remainder of the SP/G-3 flowpath.

At both LGS units, suppression pool spray flow can be supplied from sources both internal (i.e., suppression pool) and external to the primary containment (i.e., Residual Heat Removal Service Water (RHRSW) and Fire Water Systems). Both external sources will add water to the primary containment, and may be used only if primary containment water level and suppression pool pressure can be restored and maintained on the safe side of the Primary Containment Pressure Limit (PCPL, CURVE SP/G-3-1). Prior to directing initiation of suppression pool sprays, and following initiation of suppression pool sprays, operators are required to evaluate appropriate suppression pool spray suction sources.

The PCPL is defined to be the lesser of:

- The pressure capability of the primary containment.
- The maximum primary containment pressure at which vent valves sized to reject all decay heat from the primary containment can be opened and closed.
- The maximum primary containment pressure at which safety relief valves (SRVs) can be opened and will remain open.
- The maximum primary containment pressure at which RPV vent valves can be opened and closed.

The PCPL is a function of primary containment water level. Exceeding the limit may challenge primary containment vent valve operability, SRV operability, RPV vent valve operability, or the structural integrity of the primary containment.

If primary containment water level and suppression pool pressure can be restored and maintained on the safe side of the PCPL (CURVE SP/G-3-1), all three suppression pool spray suction sources may be used, as required. However, internal sources (i.e., systems which take suction on the suppression pool) are the preferred suppression pool spray suction source, because they will not cause primary containment water level to rise.

If primary containment water level and suppression pool pressure cannot be restored and maintained on the safe side of the PCPL (CURVE SP/G-3-1), only suppression pool spray sources which take a suction from an internal source may be used.

SP/G-3.4 **IF** Supp Pool level is below 48 ft,

THEN spray Supp Pool per T-225 **REGARDLESS** of core cooling

DISCUSSION

LGS TRIP Step SP/G-3.4 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action.

Step SP/G-3.4 directs initiation of suppression pool sprays, regardless of whether adequate core cooling is assured, provided suppression pool level is below the elevation of the suppression pool spray nozzles (48 ft).

Suppression pool sprays are initiated to:

- Reduce the flammability of combustible gases through the addition of water vapor to the gas mixture.
- Suppress the temperature and pressure rise following combustion if a deflagration does occur.
- Scrub the primary containment atmosphere in anticipation of radioactivity release.

Suppression pool spray is required before the direction is given to inert/purge/vent the suppression pool because:

- The scrubbing of suppression pool sprays may reduce the radioactivity released.
- Suppression pool vent isolation valves are located outside the primary containment and are therefore unaffected by suppression pool spray operation inside the primary containment.
- Unlike the drywell, there is a minimal amount of electrical equipment in the suppression pool. It is therefore unlikely that suppression pool spray operation will cause an electrical short and ignition source.

Authorization is provided in T-225 to defeat interlocks, as required, which may be preventing initiation of suppression pool sprays. Interlocks, such as those from Emergency Core Cooling System (ECCS) initiation logic channels, may preclude suppression pool spray operation even when adequate core cooling is assured. Since Step PC/G-8 requires an evaluation as to whether the pumps to be used for drywell spray are needed to assure adequate core cooling, the automatic logic is unnecessary and may be defeated.

SP/G-3.5 Inert/Purge Supp Pool per T-228 using max flowrate **EXCEEDING** offsite release rate limits if necessary

DISCUSSION

LGS TRIP Step SP/G-3.5 directs actions to inert/purge the suppression pool, using the maximum flowrate, in an attempt to lower hydrogen and oxygen concentrations in the suppression pool.

The intent of Step SP/G-3.5 is to return suppression pool combustible gas concentrations below deflagration limits as quickly as possible without raising primary containment pressure significantly above atmospheric pressure. In order to push hydrogen and oxygen out of the suppression pool as quickly as possible, either inerting with nitrogen at the maximum flowrate, or purging with air at the maximum flowrate, is established, whichever will more quickly lower suppression pool hydrogen and oxygen concentrations.

The potential for hydrogen deflagration warrants exceeding normal radioactivity release rate limits if necessary and defeating any isolation interlocks that are associated with the required valve lineup. The consequences of not inerting/purging could include Primary Containment damage resulting in larger, uncontrolled releases of radioactivity. Inerting/purging should not be performed indiscriminately, however. The anticipated benefits of the action should be balanced against the possible radiological consequences. Controlled releases, if necessary, should be performed in a manner that minimizes the total dose to the public while accomplishing the actions to lower combustible gas concentrations in the DW/G-3 flowpath.

The reference to LGS TRIP NOTE #17 reminds operators that exceeding offsite release rate limits in Tech Spec/ODCM is authorized if necessary. LGS TRIP NOTE #17 also provides guidance on additional actions when the Tech Spec/ODCM radioactivity release rate limits are exceeded.

T-228, Inerting/Purging Primary Containment, directs the performance of actions necessary to inert/purge the suppression pool as specified in this step, and also to bypass/defeat isolation interlocks which may prevent establishment of the suppression pool inerting/purge lineup. For example, T-228 directs actions to defeat the Wide Range Accident Monitor (WRAM) high-high isolation of the North Vent Stack release pathway when inerting/purging the suppression pool as specified in the SP/G-3 flowpath.

Operators are directed to continue at Step SP/G-3.6.

SP/G-3.6 **IF** Supp Pool inerting/purge CANNOT be established,

THEN vent Supp Pool per T-200 **EXCEEDING** offsite release rate limits if necessary

DISCUSSION

LGS TRIP Step SP/G-3.6 directs actions to vent the suppression pool, in an attempt to lower suppression pool hydrogen and oxygen concentrations.

Step SP/G-3.6 is performed only if the suppression pool inerting/purging lineup could not be established. If the suppression pool inerting/purging lineup could not be established, it is appropriate to attempt establishment of a suppression pool vent lineup in an attempt to lower suppression pool hydrogen and oxygen concentrations.

The potential for hydrogen deflagration warrants exceeding normal radioactivity release rate limits if necessary and defeating any isolation interlocks that are associated with the required valve lineup. The consequences of not inerting/purging could include Primary Containment damage resulting in larger, uncontrolled releases of radioactivity. Inerting/purging should not be performed indiscriminately, however. The anticipated benefits of the action should be balanced against the possible radiological consequences. Controlled releases, if necessary, should be performed in a manner that minimizes the total dose to the public while accomplishing the lowering combustible gas concentrations in the DW/G-3 flowpath.

The reference to LGS TRIP NOTE #17 reminds operators that exceeding offsite release rate limits in Tech Spec/ODCM is authorized if necessary. LGS TRIP NOTE #17 also provides guidance on additional actions when the Tech Spec/ODCM radioactivity release rate limits are exceeded.

T-200, Primary Containment Emergency Vent Procedure, directs the performance of actions necessary to vent the suppression pool as specified in this step, and also to bypass/defeat isolation interlocks which may prevent establishment of the suppression pool vent lineup. For example, T-200 directs actions to defeat the Wide Range Accident Monitor (WRAM) high-high isolation of the North Vent Stack release pathway when venting the suppression pool as specified in the SP/G-3 flowpath.

SP/G-3.7 **IF** Supp Pool spray suct source must be swapped per step SP/G-3.3,

THEN secure current Supp Pool spray lineup per T-225

AND

return to step SP/G-3.3

DISCUSSION

LGS TRIP Step SP/G-3.7 directs actions to secure the current suppression pool spray lineup, and to return to Step SP/G-3.3, when conditions in SP/G-3.3 require the suppression pool spray suction source to be swapped.

Step SP/G-3.3 requires operators to continuously evaluate appropriate suppression pool spray suction sources. Following suppression pool spray initiation, if plant conditions change such that Step SP/G-3.3 requires the suppression pool spray suction source to be swapped (from an external source to an internal source), direction is provided to secure the current suppression pool spray lineup per T-225, and to return to Step SP/G-3.3. The return to Step SP/G-3.3 ensures that all restrictions on suppression pool spray initiation can be met before re-establishing suppression pool spray flow with the new suction source.

T-225, Startup And Operation Of Suppression Pool And Drywell Spray Operation, directs the performance of actions necessary to secure the suppression pool spray lineup.

SP/T SUPPRESSION POOL TEMPERATURE CONTROL

CAUTION

RHR flow above NPSH OR vortex limits may result in pump damage

DISCUSSION

This CAUTION reminds operators to remain aware of the net positive suction head (NPSH) and vortex limits for Residual Heat Removal (RHR) pumps when they are taking a suction on the suppression pool.

NPSH and vortex limits are addressed within this CAUTION for the following reasons:

- It is difficult to define in advance exactly when the limits should be observed and when pumps should be operated irrespective of the limits.
- Pumps to which the limits apply are used in more than one flowpath. For example, RHR pumps are used in flowpaths in both T-101, RPV Control, and T-102, Primary Containment Control. Authorizing operation of the pumps "regardless of NPSH and vortex limits" in one flowpath may conflict with instructions in another flowpath where flow would normally be controlled below the limits.
- Pump characteristics, and the shape of NPSH and vortex limit curves, vary widely. If a limit is relatively flat, throttling pump flow will be of little benefit; operators can only choose whether or not to operate the pump.

Where this CAUTION is referenced, the identified pumps should be operated within the NPSH and vortex limits if possible. If the situation warrants, however, the limits may be exceeded. A judgment as to whether a pump should be operated beyond its limits in a particular event should consider such factors as:

- The availability of other systems
- The current trend of plant parameters
- The anticipated time such operation will be required
- The degree to which the limit will be exceeded
- The sensitivity of the pump to operation beyond the limit
- The consequences of not operating the pump beyond the limit

Immediate and catastrophic failure is not expected if a pump is operated beyond its respective NPSH or vortex limit.

NPSH limits are defined to be the highest suppression pool temperature values which provide adequate NPSH for the pumps which take a suction on the suppression pool. The NPSH limits are functions of pump flow and suppression pool overpressure (airspace pressure plus the hydrostatic head of water over the pump suction), and are utilized to preclude pump damage from cavitation.

Vortex limits are defined to be the lowest suppression pool level above which air entrainment is not expected to occur in pumps that take a suction on the suppression pool. These suppression pool levels are functions of Emergency Core Cooling System (ECCS) flow. Exceeding the limits can lead to air entrainment at the pump suction strainers.

The reference to LGS TRIP NOTE #3 reminds operators that the suppression pool level restriction of 13.5 ft. is based on RHR, Core Spray, and Reactor Core Isolation Cooling (RCIC) pump NPSH and vortex limits.

Operators are directed to continue at Step SP/T-1.

SP/T-1 Read SPOTMOS for Supp Pool temp indication

DISCUSSION

LGS TRIP Step SP/T-1 is a continue re-checking step, and as such, should be referred to frequently. At all times during the performance of subsequent actions in the suppression pool temperature (SP/T) control flowpath, operators should use the Suppression Pool Temperature Monitoring System (SPOTMOS) to obtain suppression pool temperature indications.

SPOTMOS is the preferred suppression pool temperature monitoring system because it is environmentally qualified in a post loss of coolant accident (post-LOCA) environment, and provides bulk temperature indication, as opposed to localized temperature indication.

Monitoring of suppression pool temperature as specified in T-102, Primary Containment Control, satisfies the requirements of Technical Specification Surveillance Requirement 4.6.2.1, which requires monitoring of suppression pool temperature once it has exceeded 95°F.

The reference to LGS TRIP NOTE #2 reminds operators that the SPOTMOS probes are located in the suppression pool at an elevation which corresponds to an indicated suppression pool level of 17.8 ft. If indicated suppression pool level drops below 17.8 ft., Residual Heat Removal (RHR) pump suction temperature can be used as a valid alternate method for determining suppression pool temperature provided an RHR pump is running.

SP/T-2 Operate available Supp Pool cooling

DISCUSSION

LGS TRIP Step SP/T-2 directs actions to operate available suppression pool cooling to control suppression pool temperature.

The initial action taken to control suppression pool temperature employs the same method typically used during normal plant operations: monitoring its status and placing available suppression pool cooling in-service as required to maintain suppression pool temperature within Technical Specification limits. Step SP/T-2 thus provides a smooth transition from normal plant procedures to the LGS TRIP procedures, and assures that the normal method of suppression pool temperature control is attempted in advance of initiating more complex actions to terminate an increasing suppression pool temperature trend.

As used in this step, the term "available" means that the pumps and support systems necessary to supply suppression pool cooling are capable of performing their identified function and can be placed in service to cool the suppression pool water volume. If a pump cannot be operated due to plant conditions or physical restrictions, it is not considered "available." For example, if an electrical loading concern were to limit operation of Residual Heat Removal Service Water (RHRSW) System pumps, the pumps would not be considered available within the context of this step.

As long as suppression pool temperature remains below the value of the most limiting suppression pool temperature Technical Specification Limiting Condition for Operation (LCO, 95°F), no further operator action is required in the suppression pool temperature (SP/T) control flowpath other than continuing to monitor and control suppression pool temperature using available suppression pool cooling systems.

Operators are directed to continue at Step SP/T-3.

SP/T-3 **WHEN** Supp Pool temp CANNOT BE maintained below 95°F,
THEN continue

DISCUSSION

LGS TRIP Step SP/T-3 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

When it has been determined that suppression pool temperature cannot be maintained below 95°F, a conclusion that may be reached in advance of suppression pool temperature actually reaching this value, further actions to control suppression pool temperature must be taken.

When it has been determined that suppression pool temperature cannot be maintained below 95°F, operators are directed to continue at Step SP/T-4.

SP/T-4 Log Supp Pool temp on Table SP/T-1 every 15 minutes

OR

Perform ST-6-060-390-* as required

DISCUSSION

ST-6-060-390-*, "Suppression Pool Temperature Check" is performed to verify, that while above the normal operating limit of 95°F, or while during testing which adds heat to the suppression pool that suppression pool temperature is under positive control, and corrective actions as defined in Tech Spec 3.6.2.1 are implemented in a timely manner. Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. "Surveillance Test procedure ST-6-060-390-*, "Suppression Pool Temperature Check" is performed when the suppression pool average temperature exceeds 95° as prescribed by LGS Technical Specification Section 4.6.2.1.b.(1), (2) and (3). When the suppression pool average temperature exceeds 95° F the operator will continue in the SP/T control leg as directed by LGS TRIP Step SP/T-3 and will commence suppression pool monitoring to implement the requirements of the LGS Technical Specifications. ST-6-060-390-* prescribes the necessary time interval for monitoring suppression pool average temperature as determined by plant conditions and prescribed in Tech Spec sections 4.6.2.1.b.1, 4.6.2.1.b.2 or 4.6.2.1.b.3. Table SP/T-1 "Suppression Pool Temperature Check" is added to LGS TRIP step SP/T-4 to provide a means for the operator to record suppression pool average temperature while efforts to control primary containment parameters in T-102 continue.

Operators are directed to continue at step SP/T-5

SP/T-5	**IF**	NOT required for core cooling,
	THEN	operate 2 loops of Supp Pool cooling

DISCUSSION

LGS TRIP Step SP/T-5 is a conditional IF/THEN step. The action specified in the "THEN" statement should only be performed if the condition specified in the "IF" statement has been met.

Step SP/T-5 directs actions to operate two loops of suppression pool cooling, but only if operation of the Residual Heat Removal (RHR) pumps is not required to assure adequate core cooling.

Maintaining adequate core cooling takes precedence over maintaining suppression pool temperature below 95°F since catastrophic failure of the primary containment is not expected to occur at this temperature. In addition, further actions in the suppression pool temperature (SP/T) control flowpath still remain available for reversing a rising suppression pool temperature trend. Therefore, only if continuous operation of an RHR pump in an RPV injection mode is not required to assure adequate core cooling is it permissible to use that pump for suppression pool cooling. Step SP/T-5 does, however, permit alternating the use of RHR pumps between the RPV injection mode and the suppression pool cooling mode as the need for each occurs and so long as adequate core cooling is maintained.

Operators are directed to continue at Step SP/T-6.

SP/T-6 **BEFORE** Supp Pool temp reaches 110°F,

THEN

1. Transfer house loads
2. Runback Recirc to minimum
3. Manually SCRAM at 60% core flow

DISCUSSION

LGS TRIP Step SP/T-6 is a before step, and should be performed, if possible, in advance of the specified condition. The timing of the required actions is event-dependent. No particular margin to the identified action level is intended. If suppression pool temperature is above 110°F when this step is reached, the specified actions should still be performed, unless expressly prohibited.

Step SP/T-6 directs actions to rapidly shutdown the reactor before suppression pool temperature reaches 110°F, the LGS specific value for the Boron Injection Initiation Temperature (BIIT).

The BIIT specifies the suppression pool temperature before which boron injection must be started. It is defined to be the greater of either:

- The highest suppression pool temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight (HSBW) of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit (HCTL), or
- The suppression pool temperature at which a reactor scram is required by plant Technical Specifications.

The BIIT is a function of reactor power, and is utilized to establish one of the two requirements for boron injection following a failure to scram. If boron injection is initiated before suppression pool temperature reaches the BIIT, emergency RPV depressurization may be precluded at lower reactor power levels. At higher reactor power levels, however, the suppression pool heatup rate may become so high that the HSBW cannot be injected before suppression pool temperature reaches the HCTL even if boron injection is initiated early in the event.

The Technical Specification suppression pool temperature value at which a reactor scram is required is 110°F. This value, rather than the BIIT curve, was conservatively chosen as the plant condition at which injection of boron into the RPV must be commenced.

The conditions defining the requirement for boron injection are located in the reactor power (RC/Q) control flowpath of T-101, RPV Control:

- When sustained power oscillations exceed 25% peak-to-peak, or
- Before suppression pool temperature reaches 110°F

Operators are directed to continue at Step SP/T-7.

SP/T-7 Enter T-101 AND execute concurrently

DISCUSSION

LGS TRIP Step SP/T-7 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-101, RPV Control.

Entry or re-entry into T-101 requires initiation of a redundant reactor scram signal if the reactor scram signal initiated in Step SP/T-6 was unsuccessful. Entry into T-101 is also required to ensure operators are provided with appropriate RPV level, RPV pressure, and reactor power control actions following the initiation of the reactor scram.

Operators are directed to continue at Step SP/T-8.

SP/T-8 **IF** Supp Pool temp CANNOT be maintained on safe side of Curve SP/T-1,
THEN maintain RPV press on safe side of Curve SP/T-1

DISCUSSION

LGS TRIP Step SP/T-8 is a conditional IF/THEN step. The action specified in the "THEN" statement should only be performed if the condition specified in the "IF" statement has been met.

LGS TRIP Step SP/T-8 directs actions to control RPV pressure below the Heat Capacity Temperature Limit (HCTL, CURVE SP/T-1) when the actions to control suppression pool temperature below the curve have proven unsuccessful.

If the actions being taken to limit the suppression pool temperature rise are inadequate or not effective, RPV pressure must be reduced in order to remain on the safe side of the HCTL. RPV pressure control actions have, therefore, been added to the suppression pool temperature (SP/T) control flowpath to accommodate these requirements. Failure to do so could lead to a failure of the primary containment or a loss of equipment necessary for the safe shutdown of the plant.

The HCTL is defined to be the highest suppression pool temperature at which initiation of RPV depressurization will not result in exceeding either:

- The maximum temperature capability of the suppression pool and equipment within the suppression pool which may be required to operate when the RPV is pressurized, or
- The Primary Containment Pressure Limit (PCPL)

before the rate of energy transfer from the RPV to the primary containment is within the capacity of the containment vent.

The HCTL curve contains six separate curves, each one associated with a different RPV pressure band. To use this curve, operators must first determine the current value of RPV pressure. Once this data is obtained, the current HCTL should be identified. If RPV pressure is reduced, as specified in Step SP/T-8, the HCTL curve will change (i.e., will move up), provided RPV pressure can be reduced to the next lowest RPV pressure band. This process may be continued until RPV pressure has been reduced to the Decay Heat Removal Pressure (DHRP, the 0-51 psig curve).

It should be noted that if during the initial evaluation of the HCTL curve, the operating point is on the unsafe side of the HCTL, no action may be taken to restore and maintain the safe side of the HCTL. The heat capacity of the suppression pool has been lost and emergency RPV depressurization is required.

The reference to LGS TRIP NOTE #5 reminds operators that exceeding the normal RPV cooldown rate is permitted if necessary when controlling RPV pressure as directed in this step, because remaining on the safe side of the HCTL takes precedence over normal RPV cooldown rate limits. If RPV pressure cannot be maintained on the safe side of the HCTL, emergency RPV depressurization will be required. LGS TRIP NOTE #5 also provides guidance on additional actions which may be required when the normal RPV cooldown rate must be exceeded.

Operators are directed to continue at Step SP/T-9.

SP/T-9 **WHEN** safe side of Curve SP/T-1 CANNOT be maintained,

THEN continue

DISCUSSION

LGS TRIP Step SP/T-9 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

Subsequent actions in the suppression pool temperature (SP/T) control flowpath direct actions to emergency depressurize the RPV. These actions are appropriate, however, only if suppression pool temperature and RPV pressure cannot be maintained on the safe side of the Heat Capacity Temperature Limit (HCTL, CURVE SP/T-1).

The HCTL is defined to be the highest suppression pool temperature at which initiation of RPV depressurization will not result in exceeding either:

- The maximum temperature capability of the suppression pool and equipment within the suppression pool which may be required to operate when the RPV is pressurized, or
- The Primary Containment Pressure Limit (PCPL)

before the rate of energy transfer from the RPV to the primary containment is within the capacity of the containment vent.

The HCTL curve contains six separate curves, each one associated with a different RPV pressure band. To use this curve, operators must first determine the current value of RPV pressure. Once this data has been obtained, the current HCTL should be identified. If RPV pressure is reduced, as specified in Step SP/T-8, the HCTL curve will change (i.e., will move up), provided RPV pressure can be reduced to the next lowest RPV pressure band. This process may be continued until RPV pressure has been reduced to the Decay Heat Removal Pressure (DHRP, the 0-51 psig curve). It should be noted that if during the initial evaluation of the HCTL curve, the operating point is on the unsafe side of the HCTL, no action may be taken to restore and maintain the safe side of the HCTL. The heat capacity of the suppression pool has been lost and emergency RPV depressurization is required.

When it has been determined that suppression pool temperature and RPV pressure cannot be maintained below the HCTL, operators are directed to continue at Step SP/T-10.

SP/T-10 Enter T-112 AND execute concurrently

DISCUSSION

LGS TRIP Step SP/T-10 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-112, Emergency Blowdown.

The RPV must be depressurized when suppression pool temperature and RPV pressure cannot be maintained on the safe side of the HCTL, to avoid the failure of the primary containment and equipment necessary for the safe shutdown of the plant.

SP/L SUPPRESSSION POOL LEVEL CONTROL

SP/L-1 Control Supp Pool level:

- Supp Pool Clg/Lev Control (S51.8.A) ↓
- Supp Pool Cleanup (S52.1.B) ↑↓
- Alt Supp Pool Cleanup (S52.1.D) ↑

DISCUSSION

LGS TRIP Step SP/L-1 directs actions to control suppression pool level using the same methods used during normal plant operations.

The initial actions taken to control suppression pool level employ the same methods used during normal plant operations: monitoring the status of suppression pool level, and filling or draining the suppression pool as required to maintain level within Technical Specification limits. Step SP/L-1 thus provides a smooth transition from normal plant procedures to the LGS TRIP procedures, and assures that the normal methods for controlling suppression pool level are attempted in advance of initiating more complex actions to correct an out of limit suppression pool level condition.

As long as suppression pool level remains within Technical Specification maximum and minimum limitations, no further operator action is required in the suppression pool level (SP/L) control flowpath other than continuing to monitor and control suppression pool level using normal methods.

Operators are directed to continue at Step SP/L-2.

SP/L-2 **WHEN** Supp Pool level CANNOT be maintained between 22′ AND 24′-3",

THEN continue

DISCUSSION

LGS TRIP Step SP/L-2 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

When it has been determined that suppression pool level cannot be maintained between the minimum and maximum Technical Specification limitations, a conclusion that may be reached in advance of actually exiting the control band, further actions to control suppression pool level must be taken.

When it has been determined that suppression pool level cannot be maintained between the minimum and maximum Technical Specification limitations, operators are directed to continue at Step SP/L-3.

SP/L-3 Supp Pool level high OR low

DISCUSSION

LGS TRIP Step SP/L-3 is a decision diamond that has operators evaluate the status of suppression pool level (i.e., whether it is below the minimum Technical Specification limit or above the maximum Technical Specification Limit).

There are two major concerns with suppression pool level. The suppression pool provides a heat sink to absorb energy from the RPV. Primarily the energy to be absorbed is latent energy associated with depressurization. The suppression pool is also sized to absorb decay heat for some period including the period required to depressurize. If suppression pool level is too low, there may be an insufficient mass of water to absorb this energy resulting in primary containment pressurization. Beyond this, if suppression pool level drops low enough, insufficient net positive suction head (NPSH) to the Emergency Core Cooling System (ECCS) and Reactor Core Isolation Cooling (RCIC) pumps will result. Furthermore, if suppression pool level gets too low, the downcomers will become uncovered and the primary containment will no longer function as a pressure suppression containment. Finally, it will be possible to uncover the Safety Relief Valve (SRV) discharge devices. Then, not only would the pressure suppression function be compromised, but also the ability to depressurize the RPV within the primary containment. Under these conditions, an inadvertent opening of a SRV would cause a rapid rise in primary containment pressure.

The problems associated with a high suppression pool level are entirely different. The first concern is as suppression pool level rises, so does the pressure on the suppression pool walls with a resultant rise in stress. If this raised stress were combined with the stress associated with SRV operation, the design limits of the suppression pool could be exceeded. As suppression pool level continues to rise, the suppression pool-to-drywell vacuum breakers and suppression pool vent lines would eventually become covered. Finally, as suppression pool level continues to rise, it would result in water actually entering the drywell and filling the drywell.

A "LOW" response indicates that suppression pool level is below the minimum Technical Specification limit. Operators are directed to continue at Step SP/L-4, where additional actions to raise suppression pool level are addressed.

A "HIGH" response indicates that suppression pool level is above the maximum Technical Specification limit. Operators are directed to continue at Step SP/L-11, where additional actions to lower suppression pool level are addressed.

SP/L-4 **IF** Supp Pool level CANNOT be maintained above 18 ft,

THEN secure HPCI **REGARDLESS** of core cooling

DISCUSSION

LGS TRIP Step SP/L-4 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action.

Step SP/L-4 directs actions to secure the High Pressure Coolant Injection (HPCI) System when it has been determined that suppression pool level will not be able to be maintained above the elevation of the HPCI turbine discharge device (18 ft.).

Operation of the HPCI System with its exhaust discharge device not submerged will directly pressurize the suppression pool. HPCI System operation is therefore secured as required to preclude the occurrence of this condition. The consequences of not doing so may extend to failure of the primary containment from overpressurization, and thus the HPCI System must be secured regardless of adequate core cooling concerns.

No comparable instruction regarding Reactor Core Isolation Cooling (RCIC) System operation is provided because:

- The exhaust flowrate of the RCIC turbine is no greater than the steam generated by decay heat after reactor shutdown. The basis for determining the Primary Containment Pressure Limit (PCPL) assumes the operability of a primary containment vent capable of removing decay heat ten minutes after reactor shutdown. Thus, any steam discharged by the RCIC turbine into the suppression pool airspace can be removed through the primary containment vent and will not cause suppression pool pressure to exceed the PCPL even if the RCIC turbine exhaust is not submerged.
- Elevated suppression pool pressure will cause the RCIC turbine to trip much sooner than the HPCI turbine.

The reference to LGS TRIP NOTE #2 reminds operators that the SPOTMOS probes are located in the suppression pool at an elevation which corresponds to an indicated suppression pool level of 17.8 ft. If indicated suppression pool level drops below 17.8 ft., Residual Heat Removal (RHR) pump suction temperature can be used as a valid alternate method for determining suppression pool temperature provided an RHR pump is running.

SP/L-5 **IF** Supp Pool level drops below 13.5 ft

AND

core cooling is assured,

THEN align pumps to CST

OR

shutdown ECCS/RCIC pumps

DISCUSSION

LGS TRIP Step SP/L-5 is a continue re-checking step, and as such, should be referred to frequently to determine if both of the conditions listed exist, and if so, to carry out the specified action.

Step SP/L-5 provides directions for Residual Heat Removal (RHR), Core Spray, and Reactor Core Isolation Cooling (RCIC) pump operation relative to their respective net positive suction head (NPSH) and vortex limits.

At both LGS units, the limiting suppression pool level condition for RHR pump vortex limits is 13.42 ft., for Core Spray pump vortex limits it is 12.4 ft., and for the RCIC pump vortex limits it is 12.31 ft.

RHR, Core Spray, and RCIC pump NPSH limits have been evaluated with respect to suppression pool level and temperature and it has been concluded that as long as suppression pool level is maintained above 13.5 ft., that NPSH limits would not be a concern.

Therefore, 13.5 ft. has been conservatively chosen as the single limiting value which addresses both NPSH and vortex limit concerns for the RHR, Core Spray, and RCIC pumps.

High Pressure Coolant Injection (HPCI) System pump NPSH and vortex limits are not addressed because the limiting suppression pool level condition for the HPCI pump is 15.42 ft. As directed by Step SP/L-4, the HPCI System must be shutdown when suppression pool level drops below 18 ft. (elevation of the HPCI turbine exhaust).

The reference to LGS TRIP NOTE #3 reminds operators that the suppression pool level restriction of 13.5 ft. is based on RHR, Core Spray, and RCIC pump NPSH and vortex limits.

SP/L-6 Maintain Supp Pool level above 12 ft:

- CST to Core Spray (T-234)
- Cond Trans (T-235)
- RHRSW to Supp Pool (T-231)
- Refuel Water Trans (T-235)

DISCUSSION

LGS TRIP Step SP/L-6 directs actions to maintain suppression pool level above the elevation of the downcomer openings (12 ft.).

When suppression pool level drops to 12 ft., any further reduction in level could result in direct exposure of the drywell atmosphere to the suppression pool airspace, thus compromising the pressure suppression function of the primary containment. Suppression pool level should, therefore, be maintained above 12 ft.

T-200 series procedures for each suppression pool fill method have been identified. These procedures provide the necessary guidance to establish the lineup required to raise suppression pool level.

Operators are directed to continue at Step SP/L-7.

SP/L-7 **WHEN** Supp Pool level CANNOT be maintained above 12 ft,

THEN continue

DISCUSSION

LGS TRIP Step SP/L-7 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

If suppression pool level is approaching the elevation of the downcomer openings (12 ft.), a direct threat to continued primary containment integrity exists. Subsequent steps in the "LOW" suppression pool level (SP/L) control flowpath direct actions to rapidly shutdown of the reactor and depressurize the RPV. These actions are appropriate, however, only when it has been determined that suppression pool level cannot be maintained above 12 ft. Step SP/L-7 ensures this condition has been met before allowing these actions to be performed.

When it has been determined that suppression pool level cannot be maintained above 12 ft, operators are directed to continue at Step SP/L-8.

SP/L-8

1. Transfer house loads
2. Runback Recirc to minimum
3. Manually SCRAM at 60% core flow

DISCUSSION

LGS TRIP Step SP/L-8 directs actions to rapidly shutdown the reactor when it has been determined that suppression pool level cannot be maintained above the elevation of the downcomer openings (12 ft.).

If suppression pool level is approaching the elevation of the downcomer openings, a direct threat to continued primary containment integrity exists. Performing a rapid shutdown of the reactor will reduce to decay heat levels the energy that a primary system may be discharging into the primary containment.

The actions specified in Step SP/L-8 are consistent with the guidance provided in GP-4, Rapid Plant Shutdown To Hot Shutdown. The phrase "Runback Recirc to minimum" means Reactor Recirculation Pump speed should be runback to the low speed stop.

Operators are directed to continue at Step SP/L-9 following the rapid shutdown of the reactor.

SP/L-9 Enter T-101 AND execute concurrently

DISCUSSION

LGS TRIP Step SP/L-9 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-101, RPV Control.

Entry or re-entry into T-101 requires initiation of a redundant reactor scram signal if the reactor scram signal initiated in Step SP/L-8 was unsuccessful. Entry into T-101 is also required to ensure operators are provided with appropriate RPV level, RPV pressure, and reactor power control actions following the initiation of the reactor scram.

Operators are directed to continue at Step SP/L-10.

SP/L-10 Enter T-112 AND execute concurrently

DISCUSSION

LGS TRIP Step SP/L-10 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-112, Emergency Blowdown.

The RPV is not permitted to remain at pressure if the suppression of steam discharged from the RPV into the drywell cannot be assured. When the downcomer openings are not adequately submerged, any steam discharged from the RPV into the drywell may not condense in the suppression pool before suppression pool pressure reaches unacceptable levels. Emergency RPV depressurization is required at or before the point at which this low suppression pool level condition occurs.

SP/L-11 Reduce Supp Pool level:

- Supp Pool Cleanup Pump (T-232)
- RHR to Radwaste (T-233)
- HPCI/RCIC To CST (T-230)

DISCUSSION

LGS TRIP Step SP/L-11 directs actions to reduce suppression pool level.

When suppression pool level rises above 24 ft. 3 inches, it is appropriate to begin actions to reduce level. Step SP/L-11 directs this reduction in suppression pool level and identifies the systems and procedures to be used to accomplish the level reduction.

T-200 series procedures for each suppression pool level reduction method have been identified. These procedures provide the necessary guidance to establish the lineup required to lower suppression pool level.

Operators are directed to continue at Step SP/L-12.

SP/L-12 Execute the following sections concurrently

DISCUSSION

LGS TRIP Step SP/L-12 directs the concurrent performance of actions to control a high suppression pool level condition.

Here the "HIGH" suppression pool level (SP/L) control flowpath divides into three parallel flowpaths. The first flowpath deals with controlling suppression pool level below the SRV Tail Pipe Level Limit (STPLL, CURVE SP/L-1). The second flowpath deals with appropriate actions to be taken when suppression pool level cannot be maintained below the elevation of the internal suppression pool-to-drywell vacuum breaker (less vacuum breaker opening pressure in feet of water). The third flowpath deals with appropriate actions to be taken when suppression pool level cannot be maintained on the safe side of the Primary Containment Pressure Limit (PCPL, CURVE SP/L-2).

Operators are directed to continue at Steps SP/L-13, SP/L-21, and SP/L-24 concurrently.

SP/L-13 **WHEN** safe side of Curve SP/L-1 CANNOT be maintained,
THEN continue

DISCUSSION

LGS TRIP Step SP/L-13 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

When it has been determined that suppression pool level cannot be maintained on the safe side of the SRV Tail Pipe Level Limit (STPLL, CURVE SP/L-1) curve, a conclusion that may be reached in advance of actually exceeding the STPLL, further actions to control suppression pool level must be taken.

The STPLL is the lesser of:

- The Maximum Pressure Suppression Primary Containment Water Level (MPSPCWL, 38.7 ft.), or
- The highest suppression pool level at which opening a safety relief valve (SRV) will not result in exceeding the code allowable stresses in the SRV tail pipe, tail pipe supports, quencher, or quencher supports.

The STPLL is a function of suppression pool level and RPV pressure. SRV operation with suppression pool level above the STPLL could damage the SRV discharge lines. This, in turn, could lead to primary containment failure from direct pressurization and damage to equipment inside the primary containment (Emergency Core Cooling System (ECCS) piping, RPV level instrument runs, suppression pool-to-drywell vacuum breakers, etc.) from pipe-whip and jet-impingement loads.

When it has been determined that suppression pool level cannot be maintained on the safe side of the STPLL, operators are directed to continue at Step SP/L-14.

SP/L-14
1. Transfer house loads
2. Runback Recirc to minimum
3. Manually SCRAM At 60% core flow

DISCUSSION

LGS TRIP Step SP/L-14 directs actions to rapidly shutdown the reactor when it has been determined that suppression pool level cannot be maintained on the safe side of the SRV Tail Pipe Level Limit (STPLL, CURVE SP/L-1).

If suppression pool level is approaching the STPLL, a direct threat to continued primary containment integrity may exist. Performing a rapid shutdown of the reactor reduces to decay heat levels the energy that a primary system may be discharging into the primary containment.

The actions specified in Step SP/L-14 are consistent with the guidance provided in GP-4, Rapid Plant Shutdown To Hot Shutdown. The phrase "Runback Recirc to minimum" means Reactor Recirculation Pump speed should be runback to the low speed stop.

Operators are directed to continue at Step SP/L-15 following the rapid shutdown of the reactor.

SP/L-15 Enter T-101 AND execute concurrently

DISCUSSION

LGS TRIP Step SP/L-15 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-101, RPV Control.

Entry or re-entry into T-101 requires initiation of a redundant reactor scram signal if the reactor scram signal initiated in Step SP/L-14 was unsuccessful. Entry into T-101 is also required to ensure operators are provided with appropriate RPV level, RPV pressure, and reactor power control actions following the initiation of the reactor scram.

Operators are directed to continue at Step SP/L-16.

SP/L-16 Reduce RPV press to stay on safe side of Curve SP/L-1

DISCUSSION

LGS TRIP Step SP/L-16 directs actions to lower RPV pressure to remain on the safe side of the SRV Tail Pipe Level Limit (STPLL, CURVE SP/L-1).

Control of suppression pool level relative to Technical Specification maximum limitations is directed in the Step SP/L-11. If the actions currently being taken to limit the suppression pool level rise are inadequate or ineffective, RPV pressure must be reduced in order to stay on the safe side of the STPLL. Failure to do so may lead to safety relief valve (SRV) system damage and primary containment failure.

The STPLL is the lesser of:

- The Maximum Pressure Suppression Primary Containment Water Level (MPSPCWL, 38.7 ft.), or
- The highest suppression pool level at which opening an SRV will not result in exceeding the code allowable stresses in the SRV tail pipe, tail pipe supports, quencher, or quencher supports.

The STPLL is a function of suppression pool level and RPV pressure. SRV operation with suppression pool level above the STPLL could damage the SRV discharge lines. This, in turn, could lead to primary containment failure from direct pressurization and damage to equipment inside the primary containment (Emergency Core Cooling System (ECCS) piping, RPV level instrument runs, suppression pool-to-drywell vacuum breakers, etc.) from pipe-whip and jet-impingement loads.

The reference to LGS TRIP NOTE #5 reminds operators that exceeding the normal RPV cooldown rate during pressure reduction directed in this step is permitted if necessary , since remaining below the STPLL takes precedence over normal RPV cooldown rate limits. If suppression pool level and RPV pressure cannot be restored and maintained below the STPLL, emergency RPV depressurization will be required.

Operators are directed to continue at Step SP/L-17.

SP/L-17 **WHEN** safe side of Curve SP/L-1 CANNOT be maintained,

THEN continue

DISCUSSION

LGS TRIP Step SP/L-17 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

When it has been determined that the combination of suppression pool level and RPV pressure cannot be maintained on the safe side of the SRV Tail Pipe Level Limit (STPLL, CURVE SP/L-1), a conclusion that may be reached in advance of actually exceeding the STPLL, further actions to control suppression pool level must be taken.

The STPLL is the lesser of:

- The Maximum Pressure Suppression Primary Containment Water Level (MPSPCWL, 38.7 ft.), or
- The highest suppression pool level at which opening a safety relief valve (SRV) will not result in exceeding the code allowable stresses in the SRV tail pipe, tail pipe supports, quencher, or quencher supports.

The STPLL is a function of suppression pool level and RPV pressure. SRV operation with suppression pool level above the STPLL could damage the SRV discharge lines. This, in turn, could lead to primary containment failure from direct pressurization and damage to equipment inside the primary containment (Emergency Core Cooling System (ECCS) piping, RPV level instrument runs, suppression pool-to-drywell vacuum breakers, etc.) from pipe-whip and jet-impingement loads.

When it has been determined that the combination of suppression pool level and RPV pressure cannot be maintained on the safe side of the STPLL, operators are directed to continue at Step SP/L-18.

SP/L-18 **IF** core cooling is assured,

THEN terminate injection into RPV from sources external to Pri Cont except from:

- Boron Injection Systems
- CRD

DISCUSSION

LGS TRIP Step SP/L-18 is a conditional IF/THEN step. The actions specified in the "THEN" statement should only be performed if the condition specified in the "IF" statement has been met.

Step SP/L-18 directs actions to limit the continued rise in suppression pool level by terminating injection into the RPV from systems which take a suction external to the primary containment.

A break in the RPV may be contributing to the high suppression pool level condition; water being injected into the RPV may be spilling out a break and accumulating in the suppression pool. Accordingly, injection from sources outside the primary containment is terminated to prevent any further rise in suppression pool level that may occur through this mechanism. Assuring adequate core cooling takes precedence over terminating injection into the RPV from external sources since additional action can still be taken to prevent safety relief valve (SRV) system damage and primary containment failure. Operation of systems used to inject boron or insert control rods need not be terminated if the systems are being used to shut down the reactor.

Operators are directed to continue at Step SP/L-19.

SP/L-19 **WHEN** safe side of Curve SP/L-1 CANNOT be restored AND maintained,

THEN continue

DISCUSSION

LGS TRIP Step SP/L-19 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

When it has been determined that the combination of suppression pool level and RPV pressure cannot be restored and maintained on the safe side of the SRV Tail Pipe Level Limit (STPLL, CURVE SP/L-1), actions must be taken to ensure the continued integrity of the primary containment (i.e., emergency RPV depressurization).

Consistent with the definition of the term "restore," emergency RPV depressurization is not required until it has been determined that actions to restore suppression pool level and/or RPV pressure on the safe side of the STPLL will not be effective.

The STPLL is the lesser of:

- The Maximum Pressure Suppression Primary Containment Water Level (MPSPCWL, 38.7 ft.), or
- The highest suppression pool level at which opening a safety relief valve (SRV) will not result in exceeding the code allowable stresses in the SRV tail pipe, tail pipe supports, quencher, or quencher supports.

The STPLL is a function of suppression pool level and RPV pressure. SRV operation with suppression pool level above the STPLL could damage the SRV discharge lines. This, in turn, could lead to primary containment failure from direct pressurization and damage to equipment inside the primary containment (Emergency Core Cooling System (ECCS) piping, RPV level instrument runs, suppression pool-to-drywell vacuum breakers, etc.) from pipe-whip and jet-impingement loads.

When it has been determined that the combination of suppression pool level and RPV pressure cannot be restored and maintained on the safe side of the STPLL, operators are directed to continue at Step SP/L-20.

SP/L-20 Enter T-112 AND execute concurrently

DISCUSSION

LGS TRIP Step SP/L-20 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-112, Emergency Blowdown.

SRV operation with suppression pool level above the SRV Tail Pipe Level Limit (STPLL, CURVE SP/L-1) could damage the safety relief valve (SRV) discharge lines. This, in turn, could lead to primary containment failure from direct pressurization and damage to equipment inside the primary containment (Emergency Core Cooling System (ECCS) piping, RPV level instrument runs, suppression pool-to-drywell vacuum breakers, etc.) from pipe-whip and jet-impingement loads. The RPV is, therefore, not permitted to remain at pressure if the combination of suppression pool level and RPV pressure cannot be restored and maintained on the safe side of the STPLL.

SP/L-21	**WHEN**	Supp Pool level CANNOT BE maintained below 38.7 ft,
	THEN	continue

DISCUSSION

LGS TRIP Step SP/L-21 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

If suppression pool level is at or above the elevation of the bottom of the internal suppression pool-to-drywell vacuum breakers (less vacuum breaker opening pressure in feet of water), there is no assurance that the vacuum breakers will function as designed. The specified suppression pool level (38.7 ft) assures that no portion of the drywell side of the valve is submerged for any drywell-below-suppression pool differential pressure less than or equal to the valve opening differential pressure.

When it has been determined that suppression pool level cannot be maintained below 38.7 ft, actions must be taken to ensure the continued integrity of the primary containment (i.e., termination of drywell sprays and termination of injection into the RPV from sources external to the primary containment).

When it has been determined that suppression pool level cannot be maintained below 38.7 ft, operators are directed to continue at Step SP/L-22.

SP/L-22 Terminate DW sprays

DISCUSSION

LGS TRIP Step SP/L-22 directs actions to terminate the operation of drywell sprays.

When it has been determined that suppression pool level cannot be maintained below 38.7 ft (elevation of the bottom of the internal suppression pool-to-drywell vacuum breakers less vacuum breaker opening pressure in feet of water), operation of drywell sprays must be terminated because post spray drywell vacuum relief cannot be assured.

Operators are directed to continue at Step SP/L-23.

SP/L-23 **IF** core cooling is assured,

THEN terminate injection into RPV from sources external to Pri Cont except from:

- Boron Injection Systems
- CRD

DISCUSSION

LGS TRIP Step SP/L-23 is a conditional IF/THEN step. The actions specified in the "THEN" statement should only be performed if the condition specified in the "IF" statement has been met.

Step SP/L-23 directs actions to limit the continued rise in suppression pool level by terminating injection into the RPV from systems which take a suction external to the primary containment.

A break in the RPV may be contributing to the high suppression pool level condition; water being injected into the RPV may be spilling out a break and accumulating in the suppression pool. Accordingly, injection from sources outside the primary containment is terminated to prevent any further rise in suppression pool level that may occur through this mechanism. Assuring adequate core cooling takes precedence over terminating injection into the RPV from external sources since terminating drywell sprays minimizes the potential for primary containment damage with submerged vacuum breakers. Operation of systems used to inject boron or insert control rods need not be terminated if the systems are being used to shut down the reactor.

SP/L-24 **WHEN** safe side of Curve SP/L-2 CANNOT be maintained,

THEN continue

DISCUSSION

LGS TRIP Step SP/L-24 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

If an unisolable break exists inside the drywell, continued RPV injection from sources external to the primary containment will raise primary containment water level after RPV level reaches the elevation of the break. The rising primary containment water level will, in turn, raise the hydrostatic pressure over submerged components and compress the primary containment airspace, thereby raising the atmospheric pressure. The combination of these effects tends to lower the margin to the Primary Containment Pressure Limit (PCPL) as water level goes up. If primary containment water level and suppression pool pressure cannot be maintained on the safe side of the PCPL, the next step in this section of the "HIGH" suppression pool level (SP/L) control flowpath direct actions to terminate injection into the RPV from sources external to the primary containment, provided the injection is not needed to assured adequate cooling.

The Primary Containment Pressure Limit (PCPL) is the lesser of:

- The pressure capability of the primary containment.
- The maximum primary containment pressure at which vent valves sized to reject all decay heat from the primary containment can be opened and closed.
- The maximum primary containment pressure at which SRVs can be opened and will remain open.
- The maximum primary containment pressure at which RPV vent valves can be opened and closed.

The PCPL is a function of primary containment water level. Exceeding the PCPL may challenge primary containment vent valve operability, SRV operability, RPV vent valve operability, or the structural integrity of the primary containment.

When it has been determined that the safe side of the PCPL (CURVE SP/L-2) cannot be maintained, operators are directed to continue at Step SP/L-25.

SP/L-25 **IF** core cooling is assured,

THEN terminate all injection into RPV from sources external to Pri Cont

DISCUSSION

LGS TRIP Step SP/L-25 is a conditional IF/THEN step. The actions specified in the "THEN" statement should only be performed if the condition specified in the "IF" statement has been met.

Step SP/L-25 directs actions to terminate injection into the RPV from systems which are taking a suction from a source external to the primary containment, provided adequate core cooling is assured. Step PC/P-13 directs actions to vent the primary containment to control suppression pool pressure on the safe side of the Primary Containment Pressure Limit (PCPL). At primary containment water levels below the elevation at which the suppression pool airspace pressure limit is 0 psig, suppression pool pressure can exceed the PCPL only if the rate of heat removal provided by available vent paths is less than decay heat. Terminating injection under these conditions cannot preclude primary containment failure. Thus, if adequate core cooling cannot be assured, injection from sources external to the primary containment is continued to protect the core.

It should be noted that this step directs actions to terminate injection into the RPV from external sources only. Injection from systems taking suction from the suppression pool need not be terminated.

PC/P PRIMARY CONTAINMENT PRESSURE CONTROL

PC/P-1 Control Pri Cont press:

- Pri Cont press control <u>AND</u> N_2 makeup (S57.3.B)
- High DW press (OT-101)

DISCUSSION

LGS TRIP Step PC/P-1 directs actions to control primary containment pressure using the same methods used during normal plant operations.

The initial actions taken to control primary containment pressure employ the same methods used during normal plant operations: monitoring the status of primary containment pressure and using primary containment and drywell pressure control systems as required to maintain primary containment pressure below the high drywell pressure scram setpoint (1.68 psig). Step PC/P-1 provides a smooth transition from normal operating procedures to the LGS TRIP procedures, and assures that the normal methods of primary containment pressure control are employed in advance of initiating more complex control actions to terminate a rising primary containment pressure trend.

As long as primary containment pressure remains below 1.68 psig, no further action is required in the primary containment pressure (PC/P) control flowpath other than continuing to monitor and control primary containment pressure using primary containment and drywell pressure control systems as required.

Operators are directed to continue at Step PC/P-2.

PC/P-2 **WHEN** Pri Cont press CANNOT be maintained below 1.68 psig,

THEN continue

DISCUSSION

LGS TRIP Step PC/P-2 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

When it has been determined that primary containment pressure cannot be maintained below the high drywell pressure scram setpoint (1.68 psig), a conclusion that may be reached in advance of primary containment pressure actually reaching this value, further actions to control primary containment pressure must be taken.

The reference to LGS TRIP NOTE #9 reminds operators that emergency diesel generators (EDGs) may be returned to their normal standby lineup when offsite power, RPV, and primary containment conditions have stabilized and it has been determined that EDG operation is no longer required.

The reference to LGS TRIP NOTE #10 reminds operators that SE-10, LOCA, directs the performance of additional actions during loss of coolant accident (LOCA) conditions.

When it has been determined that primary containment pressure cannot be maintained below 1.68 psig, operators are directed to continue at Step PC/P-3.

PC/P-3 **BEFORE** Supp Pool press drops to 0 psig,

THEN terminate Supp Pool spray

DISCUSSION

LGS TRIP Step PC/P-3 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action. The logic term "BEFORE" means suppression pool spray operation should be terminated before suppression pool pressure drop to 0 psig. If suppression pool pressure has already dropped to 0 psig when this step is entered, operators should also terminate operation of suppression pool sprays.

The operation of suppression pool sprays must be terminated by the time suppression pool pressure drops to 0 psig to ensure suppression pool pressure is not reduced below atmospheric. Maintaining a positive suppression pool pressure provides a positive margin to the negative design pressure of the primary containment.

Terminating suppression pool sprays "before suppression pool pressure drops to 0 psig" permits use of suppression pool sprays for fission product scrubbing at low primary containment pressures or if the primary containment has failed, yet still avoids negative primary containment pressures.

Consistent with the definition of the logic term "before", the actual pressure value at which suppression pool sprays should be secured is event specific:

- Reducing primary containment pressure below the high drywell pressure scram setpoint will clear the scram logic and maximize the margin to primary containment pressure limits.
- If the primary containment has failed or if primary containment venting is anticipated, it may be advisable to continue suppression pool spray operation at low primary containment pressures to scrub the primary containment atmosphere.
- If hydrogen and oxygen are above deflagration limits, suppression pool spray operation is prescribed in the SP/G-3 flowpath to reduce the flammability of combustible gases and to mitigate the effects of a possible deflagration.
- Reducing primary containment pressure will also reduce the net positive suction head (NPSH) available for pumps drawing suction from the suppression pool. As stated in the discussion for the CAUTION preceding Step PC/P-4, NPSH limits should be observed if possible, but may be exceeded if warranted by event specific conditions. If there is no need for continued suppression pool spray operation, however, suppression pool sprays may be terminated at higher pressures if NPSH limits are approached.

Initiation and operation of suppression pool sprays is addressed in Step PC/P-5.

CAUTION

RHR flow above NPSH OR vortex limits may result in pump damage

DISCUSSION

This CAUTION reminds operators to remain aware of the net positive suction head (NPSH) and vortex limits for Residual Heat Removal (RHR) pumps when they are taking a suction on the suppression pool.

NPSH and vortex limits are addressed within this CAUTION for the following reasons:

- It is difficult to define in advance exactly when the limits should be observed and when pumps should be operated regardless of the limits.
- Pumps to which the limits apply are used in more than one flowpath. For example, RHR pumps are used in flowpaths in both T-101, RPV Control, and T-102, Primary Containment Control. Authorizing operation of the pumps "regardless of NPSH and vortex limits" in one flowpath may conflict with instructions in another flowpath where flow would normally be controlled below the limits.
- Pump characteristics, and the shape of NPSH and vortex limit curves, vary widely. If a limit is relatively flat, throttling pump flow will be of little benefit; operators can only choose whether or not to operate the pump.

Where this CAUTION is referenced, the identified pumps should be operated within the NPSH and vortex limits if possible. If the situation warrants, however, the limits may be exceeded. A judgment as to whether a pump should be operated beyond its limits in a particular event should consider such factors as:

- The availability of other systems
- The current trend of plant parameters
- The anticipated time such operation will be required
- The degree to which the limit will be exceeded
- The sensitivity of the pump to operation beyond the limit
- The consequences of not operating the pump beyond the limit

Immediate and catastrophic failure is not expected if a pump is operated beyond its respective NPSH or vortex limit.

NPSH limits are defined to be the highest suppression pool temperature values which provide adequate NPSH for the pumps which take a suction on the suppression pool. The NPSH limits are functions of pump flow and suppression pool overpressure (airspace pressure plus the hydrostatic head of water over the pump suction), and are utilized to preclude pump damage from cavitation.

Vortex limits are defined to be the lowest suppression pool level above which air entrainment is not expected to occur in pumps that take a suction on the suppression pool. These suppression pool levels are functions of Emergency Core Cooling System (ECCS) flow. Exceeding the limits can lead to air entrainment at the pump suction strainers.

The reference to LGS TRIP NOTE #3 reminds operators that the suppression pool level restriction of 13.5 ft. is based on RHR, Core Spray, and Reactor Core Isolation Cooling (RCIC) pump NPSH and vortex limits.

Operators are directed to continue at Step PC/P-4.

PC/P-4 Determine Supp Pool/DW spray suct source per Table PC/P-1

TABLE PC/P-1

SUPP POOL/DW SPRAY SUCT SOURCE

CONDITION	SUCT SOURCE
Safe side of Curve PC/P-1 can be restored AND maintained	• Internal (Supp Pool) preferred • External (RHRSW OR Fire Water)
Safe side of Curve PC/P-1 CANNOT be restored AND maintained	Internal (Supp Pool) ONLY

DISCUSSION

LGS TRIP Step PC/P-4 is a continue re-checking step, and as such, should be referred to frequently during the performance of the remainder of the primary containment pressure (PC/P) control flowpath.

At both LGS units, suppression pool and drywell spray flow can be supplied from sources both internal (i.e., suppression pool) and external to the primary containment (i.e., Residual Heat Removal Service Water (RHRSW) and Fire Water Systems). Both external sources will add water to the primary containment, and may be used only if primary containment water level and suppression pool pressure can be restored and maintained on the safe side of the Primary Containment Pressure Limit (PCPL, CURVE PC/P-1). Prior to directing initiation of suppression pool/drywell sprays, and following initiation of suppression pool/drywell sprays, operators are required to evaluate appropriate suppression pool and drywell spray suction sources.

The PCPL is defined to be the lesser of:

- The pressure capability of the primary containment.
- The maximum primary containment pressure at which vent valves sized to reject all decay heat from the primary containment can be opened and closed.
- The maximum primary containment pressure at which SRVs can be opened and will remain open.
- The maximum primary containment pressure at which RPV vent valves can be opened and closed.

The PCPL is a function of primary containment water level. Exceeding the limit may challenge primary containment vent valve operability, SRV operability, RPV vent valve operability, or the structural integrity of the primary containment.

If primary containment water level and suppression pool pressure can be restored and maintained on the safe side of the PCPL (CURVE PC/P-1), all three suppression pool/drywell spray suction sources may be used, as required. However, internal sources (i.e., systems which take suction on the suppression pool) are the preferred suppression pool/drywell spray suction source, because they will not cause primary containment water level to rise.

If primary containment water level and suppression pool pressure cannot be restored and maintained on the safe side of the PCPL (CURVE PC/P-1), only suppression pool and drywell spray sources which take suction from an internal source may be used.

PC/P-5	**BEFORE**	Supp Pool press reaches 7.5 psig
	IF	Supp Pool level < 48 ft,
	THEN	spray Supp Pool per T-225 UNLESS required for core cooling

DISCUSSION

LGS TRIP Step PC/P-5 is a continue re-checking step, and as such, should be referred to frequently. If at any time during the performance of subsequent actions in the primary containment pressure (PC/P) control flowpath both of the listed conditions exist, the action specified in the "THEN" statement must be taken. Note that the logic term "BEFORE" is one of the conditions specified in this step. This indicates that suppression pool sprays should be initiated, other conditions permitting, before suppression pool pressure reaches the Suppression Chamber Spray Initiation Pressure (SCSIP, 7.5 psig). If suppression pool pressure is already at or above 7.5 psig when this step is reached, suppression pool sprays should also be initiated.

Operation of suppression pool sprays reduces primary containment pressure by condensing any steam that may be present and by absorbing heat from the suppression pool atmosphere through the combined effects of evaporative and convective cooling.

The SCSIP is defined to be the lowest suppression pool pressure which can occur when 95% of the non-condensible gases in the drywell have been transferred to the suppression pool. The SCSIP is utilized to preclude chugging - the cyclic condensation of steam at the downcomer openings of the drywell vents.

Chugging is the cyclic condensation of steam at the downcomer openings. When a steam bubble collapses at the exit of the downcomers, the rush of water drawn into the downcomers to fill the void induces stresses at the junction of the downcomers and drywell floor. Repeated application of such stresses could cause fatigue failure of these joints, thereby creating a direct path between the drywell and suppression pool. Steam discharged through the downcomers could then bypass the suppression pool and directly pressurize the primary containment. Scale model tests have demonstrated that chugging will not occur if the drywell atmosphere contains at least 1% non-condensible gases. Chugging can thus be prevented by maintaining the drywell non-condensible gas fraction greater than 1%.

Although spraying the suppression pool will not prevent chugging, it can reduce primary containment pressure through steam condensation and convective cooling. If steam is bypassing the suppression pool and entering the suppression pool directly, initiation of suppression pool sprays may thus obviate the need for drywell sprays. If the pressure rise is due to the transfer of non-condensible gases from the drywell to the suppression pool, however, suppression pool pressure will be relatively unaffected by the operation of suppression pool sprays and use of drywell sprays may be required.

T-225, Startup And Operation Of Suppression Pool And Drywell Spray Operation, directs the performance of actions necessary to initiate suppression pool sprays.

PC/P-6	**IF**	Supp Pool spray suct source must be swapped per step PC/P-4,
	THEN	secure current Supp Pool spray lineup per T-225
		AND
		return to step PC/P-4

DISCUSSION

LGS TRIP Step PC/P-6 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified actions.

Step PC/P-4 requires operators to continuously evaluate appropriate suppression pool spray suction sources. Following suppression pool spray initiation, if plant conditions change such that step PC/P-4 requires the suppression pool spray suction source to be swapped (from an external source to an internal source), direction is provided to secure the current suppression pool spray lineup per T-225, and to return to Step PC/P-4. The return to Step PC/P-4 ensures that all restrictions on suppression pool spray initiation can be met before re-establishing suppression pool spray flow with the new suction source.

T-225, Startup And Operation Of Suppression Pool And Drywell Spray Operation, directs the performance of actions necessary to secure the suppression pool spray lineup.

PC/P-7 **WHEN** Supp Pool press exceeds 7.5 psig,
THEN continue

DISCUSSION

LGS TRIP Step PC/P-7 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

When it has been determined that suppression pool pressure has exceeded the Suppression Chamber Spray Initiation Pressure (SCSIP, 7.5 psig), further actions to control primary containment pressure must be taken.
When primary containment pressure exceeds 7.5 psig, operators are directed to continue at Step PC/P-8.

PC/P-8 **BEFORE** DW press drops to 0 psig,
THEN terminate DW sprays

DISCUSSION

LGS TRIP Step PC/P-8 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action. The logic term "BEFORE" means drywell spray operation should be terminated before drywell pressure drops to 0 psig. If drywell pressure has already dropped to 0 psig when this step is entered, operators should also terminate operation of drywell sprays.

The operation of drywell sprays must be terminated by the time drywell pressure decrease to 0 psig to ensure drywell pressure is not reduced below atmospheric. Maintaining a positive drywell pressure provides a positive margin to the negative design pressure of the primary containment.

Terminating drywell sprays "before drywell pressure drops to 0 psig" permits use of drywell sprays for fission product scrubbing at low primary containment pressures or if the primary containment has failed, yet still avoids negative primary containment pressures.

Consistent with the definition of the logic term "before", the actual pressure value at which drywell sprays should be secured is event specific:

- Reducing primary containment pressure below the high drywell pressure scram setpoint will clear the scram logic and maximize the margin to primary containment pressure limits.
- If the primary containment has failed or if primary containment venting is anticipated, it may be advisable to continue drywell spray operation at low primary containment pressures to scrub the primary containment atmosphere.
- If hydrogen and oxygen are above deflagration limits, drywell spray operation is prescribed in DW/G-3 flowpath to reduce the flammability of combustible gases and to mitigate the effects of a possible deflagration.
- Reducing primary containment pressure will reduce the NPSH available for pumps drawing suction from the suppression pool. As stated in the discussion for the CAUTION preceding Step PC/P-4, NPSH limits should be observed if possible, but may be exceeded if warranted by event specific conditions. If there is no need for continued drywell spray operation, however, drywell sprays may be terminated at higher pressures if NPSH limits are approached.

Initiation and operation of drywell sprays is addressed in Step PC/P-9. Note that while operation of drywell sprays is permitted down to pressures approaching 0 psig, the initiation of drywell spray is prohibited when on the unsafe side of the Drywell Spray Initiation Limit Curve (DSIL, CURVE PC/P-2).

PC/P-9	**IF**	on safe side of Curve PC/P-2
		AND
		Supp Pool level is below 38.7 ft,
	THEN	spray DW per T-225 UNLESS required for cooling

DISCUSSION

LGS TRIP Step PC/P-9 is a continue re-checking step, and as such, should be referred to frequently to determine if both of the conditions listed exist, and if so, to carry out the specified action.

Drywell spray operation effects a drywell pressure and temperature reduction through the combined effects of evaporative cooling and convective cooling. In evaporative cooling the water spray undergoes a change of state, liquid to vapor, whereas convective cooling involves no change of state.

Evaporative cooling refers to spray droplet heat and mass transfer which occurs when water is sprayed into a superheated atmosphere. The water in each droplet is assumed to instantaneously heat and flash to steam until the surrounding atmosphere saturates, absorbing heat energy from the atmosphere. For bounding calculations with typical drywell spray flowrates, this cooling process results in an immediate, rapid, and large reduction in drywell pressure at a rate much faster than can be compensated for by the primary containment vacuum relief system. Unrestricted operation of drywell sprays could thus result in a negative drywell-to-suppression pool differential pressure large enough to cause a loss of primary containment integrity.

Convective cooling refers to spray droplet heat transfer which occurs when water is sprayed into a saturated atmosphere. The sprayed water droplets absorb heat from the surrounding atmosphere through convective heat transfer (sensible heat from the drywell atmosphere is transferred to the water droplets), reducing drywell ambient temperature and pressure until equilibrium conditions are established. This process proceeds at a rate much slower than the evaporative cooling process; the drywell temperature/pressure reduction caused by convective cooling can be controlled by terminating sprays.

Considering the pressure drop concerns described above, the Drywell Spray Initiation Limit (DSIL, CURVE PC/P-2) is defined to be the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below either:

- The drywell-below-suppression pool differential pressure capability, or
- The high drywell pressure scram setpoint.

The DSIL is a function of drywell pressure, and is utilized to preclude primary containment failure following initiation of drywell sprays.

The restriction on suppression pool level being below 38.7 ft. is concerned with covering the suppression pool-to-drywell vacuum breakers. These vacuum breakers will not function as designed if any portion of the valve is covered with water. The specified suppression pool level assures that no portion of the drywell side of the valve is submerged for any drywell-below-suppression pool differential pressure less than or equal to the valve opening differential pressure. Drywell spray operation with vacuum breakers inoperable (i.e., with no drywell vacuum relief capability) may cause the primary containment differential pressure capability to be exceeded and therefore is not permitted.

Instructions to shut down recirculation pumps and drywell cooling fans prior to drywell spray initiation is provided in T-225, Startup And Operation Of Suppression Pool And Drywell Spray Operation. These actions are appropriate to prevent damage to the recirculation pumps and drywell cooling fans, because they are not designed to be operated in a spray environment.

Authorization is provided in T-225 to defeat interlocks, as required, which may be preventing initiation of drywell sprays. Interlocks, such as those from Emergency Core Cooling System (ECCS) initiation logic channels, may preclude drywell spray operation even when adequate core cooling is assured. Since this step requires an evaluation as to whether the pumps to be used for drywell spray are needed to assure adequate core cooling, the automatic logic is unnecessary and may be defeated.

Authorization is also provided in T-225 to defeat interlocks which may prevent drywell spray operation at low primary containment pressures. Defeating these interlocks permits use of drywell sprays for fission product scrubbing if the primary containment has failed or has been vented to below the interlock setpoints. Step PC/P-8 and the restrictions upon drywell spray initiation in Step PC/P-9 avert negative primary containment pressures, thus obviating the need for the automatic logic.

PC/P-10 **IF** DW spray suct source must be swapped per step PC/P-4,

THEN secure current DW spray lineup per T-225

AND

return to step PC/P-4

DISCUSSION

LGS TRIP Step PC/P-10 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified actions.

Step PC/P-4 requires operators to continuously evaluate appropriate drywell spray suction sources. Following drywell spray initiation, if plant conditions change such that Step PC/P-4 requires the drywell spray suction source to be swapped (from an external source to an internal source), direction is provided to secure the current drywell spray lineup per T-225, and to return to Step PC/P-4. The return to Step PC/P-4 ensures that all restrictions on drywell spray initiation can be met before re-establishing drywell spray flow with the new suction source.

T-225, Startup And Operation Of Suppression Pool And Drywell Spray Operation, directs the performance of actions necessary to secure the drywell spray lineup.

PC/P-11	**WHEN**	safe side of Curve PC/P-3 CANNOT be maintained,
	THEN	continue

DISCUSSION

LGS TRIP Step PC/P-11 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

When it has been determined that the combination of suppression pool pressure and suppression pool level cannot be maintained on the safe side of the Pressure Suppression Pressure (PSP, CURVE PC/P-3), a conclusion that may be reached in advance of actually reaching the unsafe side of the curve, further actions to control primary containment pressure must be taken.

The PSP is the lesser of:

- The highest suppression pool pressure which can occur without steam in the suppression pool airspace.
- The highest suppression pool pressure at which initiation of RPV depressurization will not result in exceeding the Primary Containment Pressure Limit (PCPL) before RPV pressure drops to the Decay Heat Removal Pressure (DHRP, 51 psig).
- The highest suppression pool pressure which can be maintained without exceeding the suppression pool boundary design load if safety relief valves (SRVs) are opened.

The PSP is a function of suppression pool pressure and suppression pool level, and is utilized to assure the pressure suppression function of the primary containment is maintained while either the RPV is at pressure or primary containment flooding is required.

When it has been determined that the combination of suppression pool pressure and suppression pool level cannot be maintained on the safe side of the PSP (CURVE PC/P-3), operators are directed to continue at Step PC/P-12.

PC/P-12 Enter T-112 AND execute concurrently

DISCUSSION

LGS TRIP Step PC/P-12 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-112, Emergency Blowdown.

If suppression pool and/or drywell sprays cannot be initiated or are ineffective in reversing the rising trend in primary containment pressure, as evidenced by not being able to maintain the combination of suppression pool pressure and suppression pool level below the Pressure Suppression Pressure (PSP, CURVE PC/P-3), the RPV is depressurized to minimize the further release of energy from the RPV to the primary containment. This action serves to terminate, or reduce as much as possible, any continued rise in primary containment pressure.

No explicit direction to enter T-101, RPV Control, is included in the primary containment pressure (PC/P) control flowpath since drywell pressure above the scram setpoint (1.68 psig) is an entry condition into T-101.

Operators are directed to continue at Step PC/P-13.

PC/P-13 **BEFORE** entering unsafe side of Curve PC/P-1,

THEN vent Pri Cont per T-200 **EXCEEDING** offsite release rate limits if necessary to maintain Supp Pool press on safe side of Curve PC/P-1

DISCUSSION

LGS TRIP Step PC/P-13 directs actions to vent the primary containment, regardless of offsite radioactivity release. The logic term "BEFORE" means primary containment venting should be initiated before the unsafe side of the Primary Containment Pressure Limit (PCPL, CURVE PC/P-1) is reached. The timing of the required actions is event-dependent. No particular margin to the identified action level is intended. If the combination of primary containment water level and suppression pool pressure has already gone outside the limits of the PCPL when this step is reached, the specified actions should still be performed, unless expressly prohibited.

The PCPL is defined to be the lesser of:

- The pressure capability of the primary containment, or
- The maximum primary containment pressure at which vent valves sized to reject all decay heat from the primary containment can be opened and closed, or
- The maximum primary containment pressure at which safety relief valves (SRVs) can be opened and will remain open, or
- The maximum primary containment pressure at which RPV vent valves can be opened and closed.

The PCPL is a function of primary containment water level and suppression pool pressure. Exceeding the limit may challenge primary containment vent valve operability, SRV operability, RPV vent valve operability, or the structural integrity of the primary containment.

The directions to vent "BEFORE entering unsafe side of Curve PC/P-1" and to "... maintain Supp Pool press on safe side of Curve PC/P-1" allow, but do not require, venting at significantly lower pressures. Early or extended venting can permit primary containment pressure reductions before significant fuel damage occurs, thereby raising the capacity of the primary containment to retain fission products and reducing the radioactivity released to the environment. If the primary containment has failed, venting may also reduce the offsite dose by directing fission products through an elevated release point.

The importance of maintaining primary containment pressure below the PCPL warrants exceeding normal radioactivity release rate limits if necessary and defeating any isolation interlocks that interfere with the vent lineup. The consequences of not venting could include primary containment damage resulting in larger, uncontrolled releases of radioactivity. Venting should not be performed indiscriminately, however. The anticipated benefits of venting should be balanced against the possible radiological consequences. Controlled releases, if necessary, should be performed in a manner that minimizes the total dose to the public while preventing primary containment damage.

The reference to LGS TRIP NOTE #17 reminds operators that exceeding offsite release rate limits in Tech Spec/ODCM is authorized if necessary. LGS TRIP NOTE #17 also provides guidance on additional actions when the Tech Spec/ODCM radioactivity release rate limits are exceeded.

T-200, Primary Containment Emergency Vent Procedure, directs the performance of actions necessary to vent the primary containment, and also to bypass/defeat isolation interlocks which may interfere with establishing the primary containment vent lineup.

DW/T DRYWELL TEMPERATURE CONTROL

CAUTION
Elevated DW temp affects RPV level indication

DISCUSSION

This CAUTION warns operators of the possible effects of elevated drywell temperatures on RPV level instrument indications. This CAUTION, along with Step DW/T-1, define the conditions under which RPV level indications must be considered invalid due to the effects of RPV pressure, primary containment temperatures, and/or Reactor Enclosure temperatures.

RPV level instruments are calibrated to provide accurate indication under expected operating conditions. Indicated RPV level will be inaccurate if primary containment temperature, Reactor Enclosure temperature, or RPV pressure vary from the calibrated conditions of the associated instrument.

Note that the information provided in both this CAUTION and T-291, Temperature Effects On Reactor Level Instrumentation, do not simply correct for instrument inaccuracies due to variances from calibrated conditions. Rather, they define conditions under which neither the displayed value nor the indicated trend of RPV level can be relied upon.

Operators are directed to continue at Step DW/T-1.

DW/T-1 Use T-291 for RPV level effects

DISCUSSION

LGS TRIP Step DW/T-1 is a continue re-checking step, and as such, should be referred to frequently. At all times during the performance of subsequent actions in the DW/T flowpath, T-291, Temperature Effects On Reactor Level Instrumentation, should be used to verify that the RPV level and level trend information provided by RPV level instrumentation is valid.

RPV level instruments are calibrated to provide accurate indication under expected operating conditions. Indicated RPV level will be inaccurate if primary containment temperature, Reactor Enclosure temperature, or RPV pressure varies from the calibrated conditions of the associated instrument.

T-291 contains RPV level instrument reference and variable leg location information, as well as guidance on the possible RPV level indication errors due to temperatures in the vicinity of the RPV level instrument reference and variable legs.

Additionally, T-291 and the previous CAUTION define the conditions under which neither the displayed value nor the indicated trend of RPV level can be relied upon.

DW/T-2 Operate available DW cooling

DISCUSSION

LGS TRIP Step DW/T-2 directs actions to operate available drywell cooling, the normal mechanism by which drywell temperature is controlled.

The initial action taken to control drywell temperature employs the same method typically used during normal plant operations: monitoring its status and placing available drywell cooling in operation as required to maintain temperature within specified normal operating limits (below the drywell temperature Limiting Condition for Operation (LCO)). Thus, Step DW/T-2 provides a smooth transition from normal operating procedures to the LGS TRIP procedures, and assures that the normal methods of drywell temperature control are employed in advance of initiating more complex control actions to terminate a rising drywell temperature trend.

As long as drywell temperature remains below normal operating limits no further operator action is required in the drywell temperature (DW/T) control flowpath other than continuing to monitor and control drywell temperature using available drywell cooling.

As used in this step, the term "available" means that the pumps, fans, and support systems necessary to supply drywell cooling are capable of performing their identified function and can be placed in service to provide cooling. If a component cannot be operated due to plant conditions or physical restrictions, it is not considered "available." For example, if an electrical loading concern were to limit operation of the drywell cooling fans, the fans would not be considered "available" within the context of this step.

Operators are directed to continue at Step DW/T-3.

DW/T-3 **IF** DW press exceeds 1.68 psig,

THEN ONLY read TR-57-*22 for DW temp

DISCUSSION

LGS TRIP Step DW/T-3 is a continue re-checking step, and as such, should be referred to frequently to determine if both of the conditions listed exist, and if so, to carry out the specified action

The actions specified in Step DW/T-3 ensure that only environmentally qualified instruments are used when harsh environmental conditions exist in the drywell. Should drywell pressure exceed 1.68 psig, operators would have positive indication of the development of an adverse environmental condition in the drywell.

If TR-57-*22 is, or becomes unavailable when required to be used to monitor drywell temperature, then T-291 provides direction for temporary connection of a thermocouple calibrator which should be used to monitor drywell temperatures.

DW/T-4 **WHEN** DW temp CANNOT be maintained below 145°F, **THEN** continue

DISCUSSION

LGS TRIP Step DW/T-4 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

When it has been determined that drywell temperature cannot be maintained below the drywell temperature Limiting Condition for Operation (LCO, 145°F), a conclusion that may be reached prior to drywell temperature actually reaching this value, further actions to control drywell temperature must be taken.

When it has been determined that drywell temperature cannot be maintained below 145°F, operators are directed to continue at Step DW/T-5.

DW/T-5 Maximize DW cooling bypassing isol per GP-8 as necessary.

DISCUSSION

LGS TRIP Step DW/T-5 directs actions to maximize drywell cooling in an effort to reduce the rising trend in drywell temperature. All available drywell cooling should be used, utilizing whatever coolers and cooling water (i.e., Drywell Chilled Water, Reactor Enclosure Cooling Water, if in OPCON 4 or 5) may be available.

If the availability of the Drywell Chilled Water (DWCW) System is in question based on plant conditions, it may be determined by the DWCW Head Tank High/Low Level Alarm (computer point G532) not being in alarm OR DWCW pump suction pressure greater than 35 psig as read on PI-87-*09A(B).

If the availability of the Reactor Enclosure Cooling Water (RECW) System is in question based on plant conditions, it may be determined by the RECW Head Tank High/Low Level Alarm (annunciator H-5 SERVICES PANEL *18) not being in alarm OR RECW pump suction pressure greater than 80 psig as read on PI-013-*05A(B).

The phrase "Maximize DW cooling" relates to the operation of the drywell unit coolers, and is defined for each unit cooler as operation of one fan, two loops of cooling water, and the use of both DWCW circulating water pumps. If the RECW System is providing cooling water to the drywell unit coolers, the maximum amount of cooling water flow available should be sent to each unit cooler.

Authorization is provided in this step to bypass isolation interlocks as necessary to effect the use of drywell cooling mechanisms. This action recognizes that concurrent actions directed by other LGS TRIP procedures (e.g., lowering RPV level as specified in T-117, Level/Power Control) may otherwise preclude drywell cooler operation.

Operators are directed to continue at Step DW/T-6.

DW/T-6	**BEFORE**	DW press drops to 0 psig,
	THEN	terminate DW sprays

DISCUSSION

LGS TRIP Step DW/T-6 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified action. The logic term "BEFORE" means drywell spray operation should be terminated before drywell pressure drops to 0 psig. If drywell pressure has already dropped to 0 psig when this step is entered, operators should also terminate operation of drywell sprays.

The operation of drywell sprays must be terminated by the time drywell pressure decrease to 0 psig to ensure drywell pressure is not reduced below atmospheric. Maintaining a positive drywell pressure provides a positive margin to the negative design pressure of the primary containment.

Terminating drywell sprays "before drywell pressure drops to 0 psig" permits use of drywell sprays for fission product scrubbing at low primary containment pressures or if the primary containment has failed, yet still avoids negative primary containment pressures.

Consistent with the definition of the logic term "before", the actual pressure value at which drywell sprays should be secured is event specific:

- Reducing primary containment pressure below the high drywell pressure scram setpoint will clear the scram logic and maximize the margin to primary containment pressure limits.
- If the primary containment has failed or if primary containment venting is anticipated, it may be advisable to continue drywell spray operation at low primary containment pressures to scrub the primary containment atmosphere.
- If hydrogen and oxygen are above deflagration limits, drywell spray operation is prescribed in DW/G-3 flowpath to reduce the flammability of combustible gases and to mitigate the effects of a possible deflagration.
- Reducing primary containment pressure will reduce the net positive suction head (NPSH) available for pumps drawing suction from the suppression pool. As stated in the discussion for the CAUTION preceding Step DW/T-7, NPSH limits should be observed if possible, but may be exceeded if warranted by event specific conditions. If there is no need for continued drywell spray operation, however, drywell sprays may be terminated at higher pressures if NPSH limits are approached.

Initiation and operation of drywell sprays is addressed in Step DW/T-8. Note that while operation of drywell sprays is permitted down to pressures approaching 0 psig, initiation of drywell spray is prohibited when on the unsafe side of the Drywell Spray Initiation Limit Curve (DSIL, CURVE PC/P-2).

CAUTION

RHR flow above NPSH OR vortex limits may result in pump damage

DISCUSSION

This CAUTION reminds operators to remain aware of the net positive suction head (NPSH) and vortex limits for Residual Heat Removal (RHR) pumps when they are taking a suction on the suppression pool.

NPSH and vortex limits are addressed within this CAUTION for the following reasons:

- It is difficult to define in advance exactly when the limits should be observed and when pumps should be operated irrespective of the limits.
- Pumps to which the limits apply are used in more than one flowpath. For example, RHR pumps are used in flowpaths in both T-101, RPV Control, and T-102, Primary Containment Control. Authorizing operation of the pumps "regardless of NPSH and vortex limits" in one flowpath may conflict with instructions in another flowpath where flow would normally be controlled below the limits.
- Pump characteristics, and the shape of NPSH and vortex limit curves, vary widely. If a limit is relatively flat, throttling pump flow will be of little benefit; operators can only choose whether or not to operate the pump.

Where this CAUTION is referenced, the identified pumps should be operated within the NPSH and vortex limits if possible. If the situation warrants, however, the limits may be exceeded. A judgment as to whether a pump should be operated beyond its limits in a particular event should consider such factors as:

- The availability of other systems
- The current trend of plant parameters
- The anticipated time such operation will be required
- The degree to which the limit will be exceeded
- The sensitivity of the pump to operation beyond the limit
- The consequences of not operating the pump beyond the limit

Immediate and catastrophic failure is not expected if a pump is operated beyond its respective NPSH or vortex limit.

NPSH limits are defined to be the highest suppression pool temperature values which provide adequate NPSH for the pumps which take a suction on the suppression pool. The NPSH limits are functions of pump flow and suppression pool overpressure (airspace pressure plus the hydrostatic head of water over the pump suction), and are utilized to preclude pump damage from cavitation.

Vortex limits are defined to be the lowest suppression pool level above which air entrainment is not expected to occur in pumps that take a suction on the suppression pool. These suppression pool levels are functions of Emergency Core Cooling System (ECCS) flow. Exceeding the limits can lead to air entrainment at the pump suction strainers.

The reference to LGS TRIP NOTE #3 reminds operators that the suppression pool level restriction of 13.5 ft. is based on RHR, Core Spray, and Reactor Core Isolation Cooling (RCIC) pump NPSH and vortex limits.

Operators are directed to continue at Step DW/T-7.

DW/T-7 Determine DW spray suct source per Table DW/T-1

TABLE DW/T-1

DW SPRAY SUCT SOURCE

CONDITION	**SUCT SOURCE**
Safe side of Curve DW/T-2 can be restored AND maintained	• Internal (Supp Pool) preferred • External (RHRSW OR Fire Water)
Safe side of Curve DW/T-2 CANNOT be restored AND maintained	Internal (Supp Pool) ONLY

DISCUSSION

LGS TRIP Step DW/T-7 is a continue re-checking step, and as such, should be referred to frequently during the performance of the remainder of the drywell temperature (DW/T) control flowpath.

At both LGS units, drywell spray flow can be supplied from sources both internal (i.e., suppression pool) and external to the primary containment (i.e., Residual Heat Removal Service Water (RHRSW) and Fire Water Systems). Both external sources will add water to the primary containment, and may be used only if primary containment water level and suppression pool pressure can be restored and maintained on the safe side of the Primary Containment Pressure Limit (PCPL, CURVE DW/T-2). Prior to directing initiation of drywell sprays, and following initiation of drywell sprays, operators are required to evaluate appropriate drywell spray suction sources.

The PCPL is defined to be the lesser of:

- The pressure capability of the primary containment.
- The maximum primary containment pressure at which vent valves sized to reject all decay heat from the primary containment can be opened and closed.
- The maximum primary containment pressure at which SRVs can be opened and will remain open.
- The maximum primary containment pressure at which RPV vent valves can be opened and closed.

The PCPL is a function of primary containment water level. Exceeding the limit may challenge primary containment vent valve operability, SRV operability, RPV vent valve operability, or the structural integrity of the primary containment.

If primary containment water level and suppression pool pressure can be restored and maintained on the safe side of the PCPL (CURVE DW/T-2), all three drywell spray suction sources may be used, if required. However, internal sources (i.e., systems which take suction on the suppression pool) are the preferred drywell spray suction source, because they will not cause primary containment water level to rise. If primary containment water level and suppression pool pressure cannot be restored and maintained on the safe side of the PCPL (CURVE DW/T-1), only drywell spray sources which take suction from an internal source may be used.

DW/T-8 **BEFORE** DW temp reaches 340°F:

IF on safe side of Curve DW/T-3

AND

Supp Pool level is below 38.7 ft,

THEN spray DW per T-225 UNLESS required for core cooling

DISCUSSION

LGS TRIP Step DW/T-8 is a continue re-checking step, and as such, should be referred to frequently. Note that the logic term "BEFORE" is one of the conditions specified in this continue re-checking step. This indicates that drywell sprays should be initiated, other conditions permitting, before drywell temperature reaches 340°F. If drywell temperature is already at or above 340°F when this step is reached, drywell sprays should also be initiated, unless expressly prohibited.

Drywell spray operation may be initiated in the LGS TRIP procedures to effect a reduction in drywell temperature and pressure, to control hydrogen and oxygen concentrations in the drywell, and/or to mitigate the effects of a deflagration should one occur.

Drywell spray operation effects a drywell pressure and temperature reduction through the combined effects of evaporative cooling and convective cooling. In evaporative cooling the water spray undergoes a change of state, liquid to vapor, whereas convective cooling involves no change of state.

Evaporative cooling refers to spray droplet heat and mass transfer which occurs when water is sprayed into a superheated atmosphere. The water in each droplet is assumed to instantaneously heat and flash to steam until the surrounding atmosphere saturates, absorbing heat energy from the atmosphere. For bounding calculations with typical drywell spray flowrates, this cooling process results in an immediate, rapid, and large reduction in drywell pressure at a rate much faster than can be compensated for by the primary containment vacuum relief system. Unrestricted operation of drywell sprays could thus result in a negative drywell-to-suppression pool differential pressure large enough to cause a loss of primary containment integrity.

Convective cooling refers to spray droplet heat transfer which occurs when water is sprayed into a saturated atmosphere. The sprayed water droplets absorb heat from the surrounding atmosphere through convective heat transfer (sensible heat from the drywell atmosphere is transferred to the water droplets), reducing drywell ambient temperature and pressure until equilibrium conditions are established. This process proceeds at a rate much slower than the evaporative cooling process; the drywell temperature/pressure reduction caused by convective cooling can be controlled by terminating sprays.

Considering the pressure drop concerns described above the Drywell Spray Initiation Limit (DSIL, CURVE DW/T-3) is defined to be the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below either:

- The drywell-below-suppression pool differential pressure capability, or
- The high drywell pressure scram setpoint.

The DSIL is a function of drywell pressure, and is utilized to preclude primary containment failure following initiation of drywell sprays.

The restriction on suppression pool level being below 38.7 ft. is concerned with covering the suppression pool-to-drywell vacuum breakers. These vacuum breakers will not function as designed if any portion of the valve is covered with water. The specified suppression pool level assures that no portion of the drywell side of the valve is submerged for any drywell-below-suppression pool differential pressure less than or equal to the valve opening differential pressure. Drywell spray operation with vacuum breakers inoperable (i.e., with no drywell vacuum relief capability) may cause the primary containment differential pressure capability to be exceeded and therefore is not permitted.

Instructions to shut down recirculation pumps and drywell cooling fans prior to drywell spray initiation is provided in T-225, Startup And Operation Of Suppression Pool And Drywell Spray Operation. These actions are appropriate to prevent damage to the recirculation pumps and drywell cooling fans, because they are not designed to be operated in a spray environment.

Authorization is provided in T-225 to defeat interlocks, as required, which may be preventing initiation of drywell sprays. Interlocks, such as those from Emergency Core Cooling System (ECCS) initiation logic channels, may preclude drywell spray operation even when adequate core cooling is assured. Since this step requires an evaluation as to whether the pumps to be used for drywell spray are needed to assure adequate core cooling, the automatic logic is unnecessary and may be defeated.

Authorization is also provided in T-225 to defeat interlocks which may prevent drywell spray operation at low primary containment pressures. Defeating these interlocks permits use of drywell sprays for fission product scrubbing if the primary containment has failed or has been vented to below the interlock setpoints. Step DW/T-6 and the restrictions upon drywell spray initiation in Step DW/T-8 avert negative primary containment pressures, thus obviating the need for the automatic logic.

DW/T-9 **IF** DW spray suct source must be swapped per step DW/T-7,

THEN secure current DW spray lineup per T-225

AND

return to step PC/P-4

DISCUSSION

LGS TRIP Step DW/T-9 is a continue re-checking step, and as such, should be referred to frequently to determine if the condition listed exists, and if so, to carry out the specified actions.

Step DW/T-7 requires operators to continuously evaluate appropriate drywell spray suction sources. Following drywell spray initiation, if plant conditions change such that step DW/T-7 requires the drywell spray suction source to be swapped (from an external source to an internal source), direction is provided to secure the current drywell spray lineup per T-225, and to return to Step DW/T-7. The return to Step DW/T-7 ensures that all restrictions on drywell spray initiation can be met before re-establishing drywell spray flow with the new suction source.

T-225, Startup And Operation Of Suppression Pool And Drywell Spray Operation, directs the performance of actions necessary to secure the drywell spray lineup.

DW/T-10 **WHEN** DW temp CANNOT be restored AND maintained below 340°F,

THEN continue

DISCUSSION

LGS TRIP Step DW/T-10 is a hold/wait step, and should not be exited until the condition specified in the "WHEN" statement exists.

Emergency RPV depressurization is not required until it has been determined that drywell sprays and available drywell cooling are ineffective in reducing drywell temperature. It is not expected that either containment integrity or SRV operability will be immediately challenged when the respective temperature limits are reached. If drywell temperature is already above 340°F when Step DW/T-8 is reached, drywell sprays may still be used, if available, in preference to emergency depressurization. If sprays are effective in reducing drywell temperature, emergency depressurization need not be performed. Extended operation above the specified temperature is not permitted, however.

A determination that drywell temperature cannot be restored and maintained below 340°F may be made when, before, or after temperature actually reaches the value.

Subsequent steps in the drywell temperature (DW/T) control flowpath direct actions to rapidly shutdown the reactor and to emergency depressurize the RPV. These actions are appropriate, however, only if it has been determined that drywell temperature cannot be restored and maintained below 340°F (i.e., the maximum temperature at which ADS is qualified and drywell design temperature). Step DW/T-9 ensures this condition has been met before allowing emergency RPV depressurization.

When it has been determined that drywell temperature cannot be restored and maintained below 340°F, operators are directed to continue at Step DW/T-11.

DW/T-11

1. Transfer house loads
2. Runback Recirc to minimum
3. Manually SCRAM at 60% core flow

DISCUSSION

LGS TRIP Step DW/T-11 directs actions to rapidly shutdown the reactor.

When drywell temperature cannot otherwise be maintained below 340°F (i.e., the maximum temperature at which ADS is qualified and drywell design temperature), further release of energy from the RPV to the drywell will be directed in the drywell temperature (DW/T) control flowpath to terminate, or reduce as much as possible, any continued rise in drywell temperature by rapidly depressurizing the RPV.

This action should be taken, however, only after the reactor has been shutdown. This ensures that, if possible, the reactor is scrammed and shutdown by control rod insertion before RPV depressurization is initiated. Therefore, a rapid plant shutdown is directed in Step DW/T-11.

The actions specified in Step DW/T-11 are consistent with the guidance provided in GP-4, Rapid Plant Shutdown To Hot Shutdown. The phrase "Runback Recirc to minimum" means Reactor Recirculation Pump speed should be runback to the low speed stop.

Operators are directed to continue at LGS TRIP Step DW/T-12.

DW/T-12 Enter T-101 AND execute concurrently

DISCUSSION

LGS TRIP Step DW/T-12 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-101, RPV Control.

Entry or re-entry into T-101 requires initiation of a redundant reactor scram signal if the reactor scram signal initiated in Step DW/T-11 was unsuccessful. Entry into T-101 is also required to ensure operators are provided with appropriate RPV level, RPV pressure, and reactor power control actions following the initiation of the reactor scram.

Operators are directed to continue at Step DW/T-13.

DW/T-13 Enter T-112 AND execute concurrently

DISCUSSION

LGS TRIP Step DW/T-13 is an execute concurrently arrow that directs entry into, and concurrent execution of, T-112, Emergency Blowdown.

When drywell temperature cannot otherwise be maintained below applicable component qualification and structural design limits, further release of energy from the RPV to the drywell is minimized by rapidly depressurizing the RPV.

Exelon Generation

Course/Program:	LGS Operations Initial Training	Module/LP ID:	**LGSOPS0077** LLOT0140 LEOT0077
Title:	DRYWELL VENTILATION	Course Code:	Per LMS
Author	D. J. WEIKSNER	Revision/Date:	001/ 7/ 01 /2013
Prerequisites:	NONE	Revision By:	Ray Bohner
OPEX Included:	Internal / External / Both / None (circle one)	Est. Teach Time:	2/50 Minute Periods

TCO #	Terminal Objectives
2220010101	Startup and monitor The Drywell Recirculation System
2221040401	Actions To Drywell Air Cooler Condensate Drain Flow Monitor System Alarm
2221060401	Actions For A Group 8a (DWCW) Isolation, bypass isolation

Upon successful completion of this lesson, the trainee shall perform the following using references (as appropriate), and from memory in accordance with the lesson materials

Objective #	EO Enabling Objective Description	Pg. #
EO1.	Identify the purpose of the Drywell Ventilation System.	1
EO2.	Determine the basic arrangement of components for a typical unit cooler.	1
EO3.	Identify how each of the following support the operation of the Drywell Ventilation System:	
	a. Drywell Chilled Water System	2
	b. RECW	2
	c. AC Electrical Distribution	3
EO4.	Identify the control functions associated with the drywell ventilation fans.	6

OBJECTIVES

Objective #	EO Enabling Objective Description	Pg. #
EO5.	Given the applicable system procedures, P&ID's, and operating conditions, identify the appropriate system response. (Continuing training only)	8
EO6.	Given the applicable system procedures, P&ID's, and operating conditions, identify the appropriate operator actions. (Continuing training only)	8
EO7.	Given P&ID M-77, determine the flowpath of the Drywell Ventilation System identifying major components and interconnecting systems	N/A
Objective #	**LO Enabling Objective Description**	**Pg. #**
IL01.	State the purpose of the Drywell Ventilation System	1
IL02.	Describe the basic arrangement of components for a typical unit cooler	1
IL03.	Identify which fans are designated as LOCA fans and explain why they are needed	6
IL04.	Summarize how each of the following support the operation of the Drywell Ventilation system. a. Drywell Chilled Water System b. RECW c. AC Electrical Distribution	 2 2 3
IL05.	Given P&ID M-77, trace the Drywell Ventilation System flowpaths identifying major components, control room indications and controls, and interconnecting systems.	N/A
IL06	Summarize the control functions associated with the drywell ventilation fans.	6
IL07	Given a drawing of controls and indications located in the control room: a. Determine the status of the system b. State the effect on the system due to the manipulation of the controls.	N/A
IL08	Provided with a copy of Technical Specifications and various plant conditions: a. Recognize any conditions which does not comply with the applicable Limiting Condition for Operation. In addition, the SRO candidate In addition, the SRO candidate should be able to: b. Determine the action required for any condition of non-compliance. c. State the basis for applicable LCO's	10

Exelon Generation

Evaluation Method & Passing Criteria:

Trainee will be evaluated as part of a comprehensive end-of-week examination with a passing grade on the examination of $\geq$80%

References:

]

1. LGS UFSAR, Sections 6.2, 9.4
2. LGS Piping and Instrumentation Diagrams: M-77, M-87
3. Electrical Drawings: E-465, E-476
4. LGS Technical Specifications
5. LGS System Procedures:
   S77.1.A Startup of Drywell Recirculation System
6. LGS Modification (Unit 1); 87-5621

Materials:

A. Instructor:

1. Whiteboard, Markers, Eraser
2. Student Handout(s)
3. PowerPoint Presentation LGSOPS0077

B. Student:

1. Pen/Pencil
2. Student Handout(s)

Content/Skills	Activities/Notes
I. INTRODUCTION	
A. Present Objectives	**LGSOPS0077 (3 – 7)**
B. Non-safety related purpose	**Objective IL1** **Objective EO1**
1. Remove heat from drywell during normal operation	
C. Safety related purpose	
1. Maintain drywell circulation during accident conditions	
2. Review all Terminal and Enabling Objectives.	**Instructor Note:** Throughout this lesson, note that maintaining a Thinking and Engaged Organization could help to prevent events (See SOER 10-2 for examples of discussion points, if needed).
II. PRESENTATION	
A. SYSTEM DESCRIPTION	
1. Eight (8) unit coolers	
a. Circulate air within the drywell	Throughout this lesson, facilitate discussion of Operations Fundamentals per OP-AA-101-113
b. Cooling is provided by	
1) DW chilled water	**Objective EO3a, b**
2) RECW (backup)	RECW valves are closed and their feeds are maintained open.
2. The coolers are arranged in pairs. Each pair services an area of the drywell.	
3. Areas serviced include	**LGSOPS0077 - 8**
a. Main Steam Relief area	
b. CRD area inside reactor pedestal	
c. Drywell head area	
d. Refueling Bellows	
e. Biological Shield	
B. COMPONENT DESCRIPTION	
1. Unit Coolers (8)	**Objective EO2** **Objective IL2** **LGSOPS0077 - 9**
a. Provide a forced convection heat transfer mechanism to cool the drywell atmosphere	

b. Consists of

1) Coolers (Heat Exchangers)

2) Fans

Coolers A,B,E,F,G,H
Located at elevation 243 ft.

Coolers C and D
Located at elevation 265 ft.

c. Coolers (Heat Exchangers)

1) Two (2) coolers per unit (100% capacity each)

2) Arranged in series with respect to air flow

3) Cooling water supplied by

a) Drywell chilled water (Normal supply)

Objective IL4a, b

b) RECW (Backup Supply)

Objective EO3a, b

d. Fans

1) Two (2) fans per unit (Arranged in parallel)

2) Each fan

a) 100% capacity

b) 7000 cfm

c) Direct coupled

d) Vane axial fan

e) Fan Motor

(1) 30 HP

(2) 440 VAC

Objective IL4c

Power supplies:

D*14-R-G
*A1V212
*C1V212
*E1V212
*G1V212

D*24-R-G
*B1V212
*D1V212
*F1V212
*H1V212

D*34-R-E
*A2V212
*E2V212

D*34-R-H
*C2V212
*G2V212

D*44-R-H
*D2V212
*F2V212

D*44-R-E
*B2V212
*H2V212

3) Powered from Class IE buses. Fans of each unit coolers are powered from separate divisions

Objective EO3C
Objective IL4C

a) Each fan's power supply is via two (2) "in series" 480VAC breakers with each breaker having its own overcurrent protective device.

This provides for Primary Containment penetration protection

2. Ventilation Ducting

LGSOPS0077 (10 – 12)

a. Provides the necessary flow path for fan suction and discharge

b. Discharge Duct

Content/Skills	Activities/Notes
1) All units have discharge ducting	
2) Directs air flow to the serviced area	
3) Each unit cooler fan utilizes a back flow damper that is opened by the force of flow upon fan start. The damper remains closed when the associated fan is not running, to prevent recirculation of cooled air through it.	
c. Suction Duct	
1) Only for units C and D	
2) Direct flow from DW head area to cooler inlet	
3) All other units receive air from surrounding areas	
d. Duct Relief Valves	**LGSOPS0077 (13 – 14)**
1) Relieve external pressure into duct	
2) Prevents duct collapse during initial stages of LOCA	
3. Areas Unit Coolers Service	
a. Units *AV-212 and *BV-212	
1) Cool air supplied to:	
a) Main steam relief valve area	
b) CRD area inside pedestal	
2) Air enters pedestal area:	
a) Via two (2) penetrations	
b) 180° apart	
3) Air exhaust from CRD piping penetrations	
b. Units *CV-212 and *DV-212	
1) Cool air supplied to:	
a) Area below refuel bellows	
b) DW head above seal plate	

2) Return air from head area ducted to inlet of coolers

c. Units *EV-212 and *FV-212

1) Air supplied to:

a) Refueling bellows area

b) Top of shield wall

c) Annulus between vessel and shield wall

2) Ring Supply Duct

a) 360° ring

b) Surrounds biological shield

c) Supplied by both units (180° apart)

d) Twelve (12) evenly spaced penetrations direct flow from ring to annulus

3) Air flows

a) From bottom of annulus

b) Up between reactor and wall

c) Out the top to general area

Content/Skills	Activities/Notes
d. Units *GV-212 and *HV-212	
1) Air supplied to:	
a) Refueling bellows area	
b) Top of shield wall	
c) Drywell head area	
2) Air from head area returns via ducting to Units C and D	
4. LOCA Units	**Objective IL3**
a. Units A, B, G and H are LOCA Units	Required by Tech Specs. (LCO 3.6.6.2)
1) These units were chosen due to the locations of air discharges	
b. Must be run following LOCA to maintain drywell atmosphere thoroughly mixed thereby preventing localized accumulations of hydrogen and oxygen in excess of the lower flammability limits.	Covers all containment area.
c. Oxygen is prevented from accumulating in concentrations in excess of 5% (volume).	
C. CONTROL FUNCTIONS AND INTERLOCKS	**Objective IL6** **Objective EO 4** Panel *0C681
1. Fans	
a. Control Switch - three (3) position maintained contacts: "OFF/AUTO/RUN".	**LGSOPS0077 (15 – 17)**
b. Standby fan automatically starts if:	
1) Switch in AUTO and	
2) Low discharge air flow	
(Sensed by flow switch in common discharge line for 55 seconds)	
c. Fan Trips	If low flow trip occurs, can restart fan by going to OFF then RUN
1) Overcurrent	
2) Low flow (45 second time delay)	

Content/Skills	Activities/Notes
d. Flow Switches are heated thermocouple type:	After flow is established a loss of flow may not be sensed for 2-3 minutes depending on device setpoint.
1) Senses flow by cooling down element	
2) Loss of flow allows element to heat up	
2. Coolers	**LGSOPS0077 (18 – 22)**
a. Two (2) chilled water supplies	
1) Loop A	
2) Loop B	
b. RECW can backup either loop	
D. INSTRUMENTATION AND ALARMS	
1. Instrumentation	(Located on C681)
a. Unit cooler inlet temperature	**LGSOPS0077 (23 – 24)**
1) Normal full power value - ≈120°F	TI-77-*01-A thru H (50-220°F)
b. Unit Cooler Outlet Temperature	TI-77-*04 A thru H (50-220°F)
1) Normal full power value - ≈60°F.	
2. Alarms	
a. No control room annunciators	
b. Computer Alarms	
1) Fan Trouble	
a) Low flow (45 second time delay)	KS-77-*07 A thru H and KS-77-*08 A thru H
b) Fan in OFF	
2) Unit cooler high temperature	TITSH-77-*04 A thru H
a) Outlet air temperature if greater than 95°F	
E. OPERATION	
1. Normal Operation	
a. One (1) fan per cooler running, the other in standby (AUTO)	

Content/Skills	Activities/Notes
b. Two (2) cooling coils supplied by DWCW	
c. Operate as many fans as necessary to maintain drywell temperature and pressure	
2. Startup of Drywell Recirculation System	**Objective EO5, EO6** S77.1.A **Conservative Bias to Plant Conditions**
a. Prerequisite	Regulatory Compliance • Understand Tech Specs, TRM, ODCM, and other licenses and permits. Tech. Spec. 3.6.1.7:Drywell Ave. air temp shall not exceed 145°F
1) DWCW System in service	S87.1.A
b. Align Drywell HVAC IAW S77.1.A COL	
1) Allow at least five (5) minutes from energizing MCC's before starting fan	Allows flow-sensing elements to equalize (heat up) HS-77-*05A Panel C681
c. Start Drywell Cooler, place switch in "RUN" unit *A1V212	**Closely Monitoring Plant Conditions**
d. Verify cooler start by	Control Board Awareness – OP-AA-103-102 • Monitor indications frequently and closely with emphases on trending
1) Observing red indicating light on	
2) Monitor for 45 seconds to insure low flow condition does not exist	**Precisely Controlling Plant Evolutions**
e. Proper operation verified by	Procedural Adherence • Adhere to procedures as written • Place keep. Designate using initials check marks, or recording data.
1) Observing differential temperature	≈60°ΔT
a) Drywell cooler inlet	TI-77-*01A Panel C681
b) Drywell cooler outlet	TI-77-*04A Panel C681
f. Place drywell cooler *A2V212 in Auto	
NOTE: Flow switches located upstream of fans transmit signals on low flow to computer alarms and standby fan actuation	55 second time delay for standby fan actuation
g. Place as many drywell unit coolers on line as required to maintain drywell temperature less than 145°F	**Objective EO5**

Content/Skills	Activities/Notes
3. Loss of Off Site Power	E-10/20
a. DW chilled water is not available b. RECW manually lined up to supply coolers	Circulating pumps non-safeguard powered RECW backed up with ESW (OPCON 4 or 5 only)
4. Loss of Fan Power (480 VAC Load Center)	
a. If fans lose power, automatically restart when power returns	
1) No time delay for fans in “RUN” (Unit 2)	
2) 55 second time delay for fans in AUTO following low flow (Unit 2)	
3) 30 second time delay for any A2, B2, C2, D2, E2, F2, G2 or H2 fan in "RUN" (Unit 1 only). 4) No time delay for A1, B1, C1, D1, E1, F1, G1 or H1 if in RUN (Unit 1 only)	Mod 87-5621 incorporated to enable acceptable voltage regulation at the 480 V AC buses when the Load Centers are reenergized.
5) 30 second time delay, in addition to existing 55 second time delay, low flow auto start circuit for any A1, B1, C1, D1, E1, F1, G1, or H1 fan in Auto (Unit 1 only).	
6) Normal 55 second time delay, on low flow AUTO START, for any A2, B2, C2, D2, E2, F2, G2 or H2 fan in AUTO (Unit 1 only)	
5. LOCA	
a. DW chilled water isolates (Hi Drywell press 1.68 psig or Low low low level -129")	
b. Fans lose power momentarily (3 seconds) due to Load Center load shed sequence. Fans restart as stated in "D" above.	
c. LOCA signal may be bypassed to allow restoration of DW cooling (as directed by T-102).	
d. DW cooling is maximized with one (1) chiller, two (2) pumps, and eight (8) fans in operation.	1 fan per cooler. 2 fans per cooler only adds heat to the containment, not additional cooling.

F. TECHNICAL SPECIFICATIONS

1. Drywell Hydrogen Mixing (LCO 3.6.6.2)

   a. Four (4) unit coolers must be operable

      1) These must include

         a) Unit coolers A, B, G, and H

      2) Each unit will include at least one (1)

         a) Operable fan (Cooling water flow is not required).

   b. Bases - The primary containment atmospheric mixing system is provided to prevent localized accumulations of hydrogen and oxygen from exceeding the lower flammability limit.

2. Drywell Average Air Temperature (LCO 3.6.1.7)

   a. Drywell Average air temperature shall not exceed 145°F.

   b. Average air temperature requires at least one reading from each elevation (4.6.1.7).

Objective IL8

Instructor will insure following LCO's and bases are covered with current copy of Tech. Specs.

*A1V212/*A2V212
*B1V212/*B2V212
*G1V212/*G2V212
*H1V212/*H2V212

G. PROCEDURE REVIEW

1. Group Discussion/Briefing

2. Hand out procedures throughout the class, have students review in groups

3. Report out/discuss led by the various students

4. Ensure thorough understanding of the Purpose of the procedure along with, Prerequisites, Precautions, Notes and Cautions.

Content/Skills	Activities/Notes
H.	
III. SUMMARY	
A. Purpose	
B. System Description	
C. Component Description	
1. Unit Coolers	
a. Coolers	
b. Fans	
2. Ventilation Ducting	
3. Areas Unit Coolers Service	
4. LOCA Units	
D. Control Functions and Interlocks	
E. Instrumentation and Alarms	
F. Operation	
G. Technical Specifications	
H. Operating Experience	
1. Brunswick Unit One, May 23, 1998 Brunswick unit one shutdown for refueling and unit two at 100 percent power, a review of drywell cooling fan motor failures determined the three failures within the last year shared a common failure mode. All three fans had been in service in excess of 20 years. Insufficient varnish coating within the windings and externally on the endturns had allowed excessive movement due to normal vibration, causing insulation breakdown and motor failure. In addition, moisture intrusion through the motor drain plug had caused deterioration in the motor windings. The failure mode was also common with motor failures observed at other nuclear plants using the same cooling fans. Root cause of the failures was end of life for the winding insulation. Corrective action will include replacing all five remaining original equipment motors during the next refueling outage.	**Closely Monitoring Plant Conditions** Intolerance for Unexpected Equipment Failure – ER-AA-10 • Proactively identify degraded equipment conditions and system performance • Perform PMs as scheduled. Document feedback to improve PM program effectiveness and failure history. **SOER 10-02** **Operating Experience** Learn from others mistakes

Preventive maintenance tasks will be implemented to periodically replace the fan motors rather than run to failure.

2. PBAPS, July 12, 1999

   Most of the bulbs on PBAPS's MCR panels are a long, slender two piece bulb which consists of the glass bulb and a metal clip which slides into the socket. When replacing the bulb on 11 DW Cooling Fan, breaker, the glass bulb separated from the metal clip. The metal clip stayed in the socket while the bulb pulled out. Small wires sticking out of the glass bulb then shorted to the socket which resulted in the blown control power fuse and the trip of the DW Cooling Fan. Given that 11 Drywell Cooling Fan tripped due to a blown control power fuse, the fan was no longer capable of performing its intended function. The loss of one drywell cooling fan is relatively insignificant as there are 5 other fans and drywell temperature only increased by 4 or 5 degrees. A component inquiry was conducted on WC Mosse and no other similar failures were identified. The apparent cause for the trip of 11 DW Cooling Fan, during bulb replacement by Unit 1 Operations was that a "degraded subcomponent contributed to failure" of the control power fuse for the fan. The breaker cubicle was inspected, the control power fuse replaced and the broken portion of the bulb was removed from the socket. No further corrective/preventive actions necessary.

IV. ATTACHMENTS

A. Power Point

1. LGSOPS0077

SAMPLE QUESTIONS

1 **WHICH ONE** of the following describes the basic arrangement of components for a drywell unit cooler?

a. Two parallel cooling coils, in series with two parallel fans

b. Two series cooling coils, in series with two series fans

c. Two parallel cooling coils, in series with two series fans

d. Two series cooling coils, in series with two parallel fans

2. Given the following:

- Unit 1 Drywell Vent Fan A1 in RUN
- Unit 1 Drywell Vent Fan A2 in AUTO

WHICH ONE of the following identifies when Fan A2 would start should Fan A1 trip?

a. immediately

b. 30 seconds

c. 55 seconds

d. 85 seconds

ANSWERS TO SAMPLE QUESTIONS

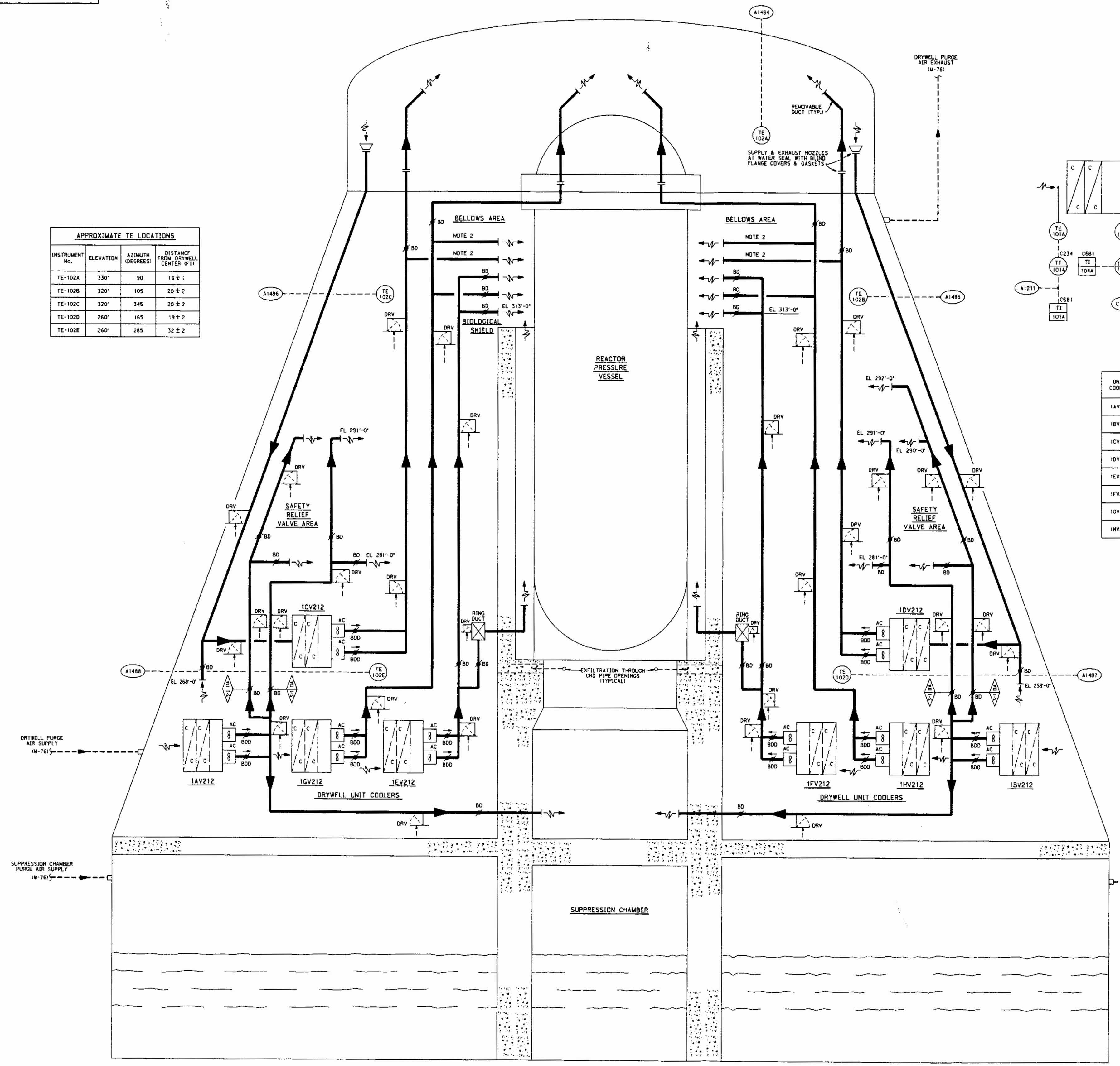
1. d

Cooling coils are in series; fans are in parallel.

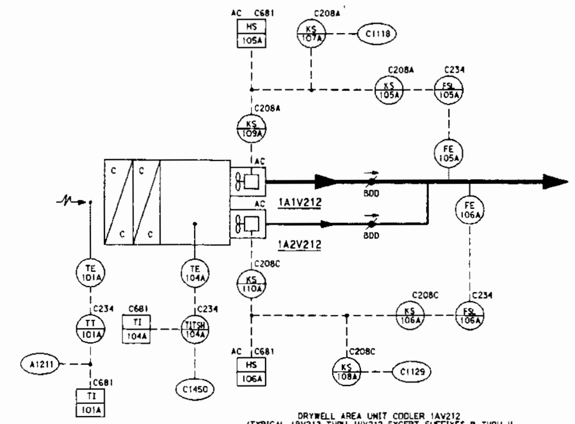
2. c

The 30 second time delay added to Unit 1 #1 fans in AUTO and Unit 1 #2 fans in RUN applies following a loss of power. The "NORMAL" 55 second time delay for AUTO start of a fan applies.

8031-M-77 SH. 1



APPROXIMATE TE LOCATIONS			
INSTRUMENT No.	ELEVATION	AZIMUTH (DEGREES)	DISTANCE FROM DRYWELL CENTER (FT)
TE-102A	330'	90	16 ± 1
TE-102B	320'	105	20 ± 2
TE-102C	320'	345	20 ± 2
TE-102D	260'	165	19 ± 2
TE-102E	260'	285	32 ± 2



DRYWELL AREA UNIT COOLER 1AV212
(TYPICAL 1BV212 THRU 1HV212 EXCEPT SUFFIXES B THRU H FOR ALL INSTRUMENTS & CONTROLS SHOWN)

UNIT COOLER	FAN	FAN TROUBLE ALARM	FAN TEMPERATURE ALARM	PANEL	DRYWELL TEMPERATURE
1AV212	1A1V212	C1118	C1450	1AC208	A1211
	1A2V212	C1129		1CC208	
1BV212	1B1V212	C1125	C1452	1BC208	A1212
	1B2V212	C1136		1DC208	
1CV212	1C1V212	C1120	C1454	1AC208	A1213
	1C2V212	C1148		1CC208	
1DV212	1D1V212	C1126	C1082	1BC208	A1214
	1D2V212	C1168		1DC208	
1EV212	1E1V212	C1121	C1451	1AC208	A1215
	1E2V212	C1132		1CC208	
1FV212	1F1V212	C1127	C1453	1BC208	A1216
	1F2V212	C1169		1DC208	
1GV212	1G1V212	C1123	C1455	1AC208	A1217
	1G2V212	C1149		1CC208	
1HV212	1H1V212	C1128	C1090	1BC208	A1218
	1H2V212	C1139		1DC208	

NOTES:

1. UNIT COOLERS 1AV212, 1BV212, 1GV212 & 1HV212 AND THE DUCTWORK CONNECTED TO THE COOLERS ARE SEISMIC CLASS I. UNIT COOLERS 1CV212, 1DV212, 1EV212 & 1FV212 AND THE DUCTWORK CONNECTED TO THE COOLERS AND PORTION OF DUCTWORK CONNECTED TO COOLERS 1AV212 & 1BV212 ARE SEISMIC CLASS IIA.
2. SUPPLY AIR REGISTERS ARE PROVIDED WITH BALANCING DAMPERS NOT SHOWN ON P&ID.
3. TE-101C AND TE-101D ARE LOCATED IN THE RETURN DUCT FROM THE DRYWELL HEAD AREA.

Best Copy Available

NO	DATE	DESCRIPTION	REV'D	CHK'D	APP'D
12	7/20/95	ADDED NOTATION "CATEGORY A1" BELOW TITLE BLOCK.	DJD		

TABLE OF ADDITIONS & CHANGES
USE ONLY PRINTS SHOWING LATEST DATE

MECHANICAL

P&ID
DRYWELL HVAC
(UNIT 1)

LIMERICK GENERATING STATION
PECO ENERGY CO.

SCALE	DRAWN	IGDS	CHECKED	INSPECTED	APPROVED		DATE
NONE	BECHTEL	BECHTEL	BECHTEL	BECHTEL	BECH	BECH	5/24/89

APPROVED BECHTEL APPROVED

SHEET 01 OF 8031-M-77 REV 12

I CATEGORY A1

Levels on 1-52,54,56,58-62

M-77 (SHT. 1)