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FEB 6 1980

REGULATORY DOCKET FILE COPY

MEMORANDUM FOR: R. Baer, Chief, Light Water Reactors Branch No. 2, DPM

FROM: R. M. Satterfield, Chief, Instrumentation and Control Systems Branch, DSS

SUBJECT: ADDITIONAL ICSB QUESTIONS - GRAND GULF

Plant Name: Grand Gulf Units 1 & 2
Docket Number: 50-416/417
Licensing Stage: OL
Milestone Number: 8
Responsible Branch: LWR #2
Project Manager: T. Houghton
ICSB Reviewer: G. McManaway (Savannah River Plant)
Review Status: Incomplete

Enclosed are additional questions that were generated by the Savannah River Plant ICSB reviewer following his assessment of Section 7.2 and 7.3 of the Grand Gulf Final Safety Analysis Report. We anticipate that additional questions will be forthcoming as the review continues. We suggest that these questions be sent to the applicant as soon as possible. The applicant should be informed that there is a need to respond as promptly as possible to insure that the reviewer completes his review on schedule.

The numbers that appear with the enclosed questions were generated by Savannah River. Please revise the numbers to be consistent with the previous round one questions.

Also enclosed are one set of informal editorial comments on Section 7.3 of the Grand Gulf FSAR. These comments should be forwarded informally to the applicant. A formal response to these comments is not required; however, the applicant should be prepared to discuss these items in the future when a meeting is arranged between the SRL reviewer and the applicant.

ORIGINAL SIGNED BY
RODNEY M. SATTERFIELD

R. M. Satterfield, Chief
Instrumentation & Control Systems Branch
Division of Systems Safety

8002220316 9

Enclosures:
As stated

cc: ~~T. Dunning~~
T. Houghton
A. Hadden (SRL)
Contact:

OFFICE	ICSB	ICSB			
SURNAME	TDunning:cc	RSatterfield			
DATE	2/ 1 /80	2/ 3 /80			

Q030. The following discrepancies between the actual plant drawings and
(7.2) the FSAR discussion and figures were noted in the review of
(F7.2-2) Section 7.2:
(F7.2-3) 1. The channel Test Switch shown in Figure 7.2-5 does not appear
(F7.2-5) on the Reactor Protection System Elementary Diagram C71-1050
(F7.2-8) (Revision 7). This test switch is mentioned a number of times
(Documents in Section 7.2.2 and is offerred as a "backup to the manual
C71-1010, scram." The switch is also identified as the first contact in
1050, 1070, the trip channel logic.
and 4010) 2. Section 7.2 and Figures 7.2-5, 8 indicate that there are two
GG isolation valves in each steam line and that they are
9 connected into the scram logic so that valves in at least
three systems must be closing to generate a scram. The trip is key bypassable in all operating modes but "Run". Drawing C71-1050 (sheets 5 thru 9) confirm this trip circuit but also shows a separate unbypassable trip (not mentioned in Section 7.2) connected in 1 out of 2 twice logic that is derived from a third isolation valve in each steam line. The presence of a third valve in each steam line is verified by the P & I Diagrams in Section 5.2.

3. Section 7.2.1.2.8 indicates that there are no RPS pressure transmitters inside the drywell; however, Drawing C71-1050 (sheet 17) locates the transmitters for reactor pressure, drywell pressure, reactor level and scram discharge volume level inside the drywell. Drawing C71-1070 (Revision 1) agrees with Section 7.2.1.2.8.

4. The FSAR states that both the turbine stop valve and the

turbine control valve protective system trips utilize pressure transmitters that are sensing hydraulic pressure for the valve operating mechanism. The trip point for the two systems are indicated to be approximately 40 psig with normal system pressures of 42-70 psig (control valve) and 165 psig (stop valve). The system drawings and specifications you provided indicate that the trip signal for the stop valve is generated by position switches mounted on the valve (some of the FSAR discussion appears to corroborate this design). The drawings indicate that the control valve trip is initiated by a pressure switch. The design specification data sheet indicates that normal hydraulic pressure is 1100-1500 psig and that the trip point is 850 psig.

5. The FSAR states that identification of the specific channels that tripped is obtained from computer typeout or by visual observation of the relay contacts (7.2.2.1.2.3.1.19). The contact positions could not be observed on the relays in the trip unit cabinets that we were shown. Verify that the contact status of relays associated with RPS trip units can be readily observed.

In addition, Drawing C71-1050 indicates that arming of the manual scram is an alarm condition and implies, therefore, that the manual scram is normally disarmed. The FSAR, Section 7.2.1.1.4.2, indicates that the manual scram switches in each group are "located close enough to permit one hand motion to initiate a scram", implying that switches are normally armed.

Resolve the discrepancies noted above and correct the appropriate document. Verify that the instruments and controls described in all of Chapter 7 are the systems that are being installed. In the case of the manual scram pushbuttons, justify the use of armed pushbuttons and the logic of operating with a disarmed system. Include in Section 7.2.1 a complete description of the steps in the actuation of a manual scram, in each reactor mode if any differences exist between modes.

Q030.
(7.0)
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Section 7.1.3 states that pressure and level transmitters were provided so that the improved reliability would not require testing of sensors except at the end of cycles. Sections 7.2 and 7.4 discuss testability and testing of transmitters for specific protective functions and state that transmitters can be and/or are valved out and tested during operation. Clarify your position on transmitter testing frequency and revise the FSAR as necessary to provide consistency. It is the Staff's position that the potential of valving errors disabling a protective channel outweighs any advantage of simulating a trip level signal to the transmitter input, provided that the normal transmitter input can be perturbed sufficiently during normal reactor operation to demonstrate that the transmitter is reading and responding correctly. Demonstrating the operability of switches and trip units still requires simulating the trip level input to these devices.

Q030.
(7.3.1.1.4.1)

Justify the claim that the containment spray cooling can be manually actuated when the drawings indicate that the manual

(7.3.1.1.4.3) pushbutton must be held depressed continuously for 90 seconds to
MPL E12-1050 initiate spray B. Identify any other manual pushbuttons that must
GG be held depressed for more than a few seconds to initiate the
11 desired action.

Q030. Justify the claim for diversity in the control circuit for
(7.3.1.1.4.7) Containment Spray since both drywell and containment pressure are
GG required and low water level can neither initiate or prevent
12 system initiation.

Q030. Section 7.3.1.2 identifies and defines "Operational Limits" and
(7.3.1.2) implies that they are the level at which the trip unit initiates
GG the ESF function. "Levels Requiring Protective Action" is not
13 defined but is identified as one of the parameters tabulated in the 7.3 tables. "Margin" is defined as the difference between the "Operational Limit" and undefined "limiting conditions". Both "Operational Limits" and "Levels Requiring Protective Action" are said to be tabulated in the 7.3 tables but only one value is tabulated in some tables and none are identified by either of the two defined titles. Amend your FSAR to fully define the terms used in specifying the design basis and to utilize consistent terminology throughout the discussion and tabulations. Indicate the method used to include the effect of the rate of change of the variable initiating the trip and the transient overshoot as a result of the incident. For reactor water level transmitters confirm that the setpoints, limits and margins include worst case affects of drywell and/or containment temperature on the sensed

reactor level. For each case in which the trip setpoint is 10% or less from end of scale, provide the actual margin between the trip point and the worst case response limits of the measuring circuit (For example, what would be the highest signal level that could exist with the water level below the measuring side pressure tap for a low water level trip circuit).

Q030. Section 7.3.1.1.2 identifies CRVICS as being comprised of 12
(7.3.1.1.2) subsystems and discusses the design bases for each subsystem
(7.3.2.2) individually. This grouping of subsystems is essentially the
(T7.1-3) same as the breakdown used in Table 7.1-5 to identify specific
(T7.1-5) requirements for the system. In the analysis for compliance with
GG system requirements the system is divided into 5 groupings, (1)
14 "CRVICS", (2) MSIV, (3) other Isolation Valves, (4) MSL High Radiation, and (5) PRM Subsystems. No definition of terms is given and it is not clear whether categories 2 thru 5 completely covers the CRVICS since some compliance statements address "CRVICS" and one or more of 2 thru 5 while others only address one or more of groups 2 thru 5.

Amend your FSAR to identify the breakdown of the CRVICS into the various analysis groupings. Use either Table 7.1-5 or Section 7.3.1.1.2 to define the subsystems that comprise the CRVICS, but state which. The complete CRVICS should be addressed in each step of the analysis for compliance.

Q030. In Section 7.3.2.3.1, it is stated the MSIV-LCS will be able to
(7.3.2.3) maintain its functional capability assuming a single <u>active</u>

GG failure. Appendix A of 10 CFR 50 defines a single failure as a
15 failure of an active component assuming all passive components function properly or a passive component fails assuming all active components function properly. Section 7.3.2.3.2.3.1.2 states that the MSIV-LCS meets the single failure criterion. Resolve this inconsistency and confirm your acceptance of the single failure definition of 10 CFR 50.

Q030. The details presented under paragraph 4.1 and paragraph 4.16
(7.3.2.3. correctly identify the operation of the MSIV-LCS; however, the
2.3.1) system does not and is not intended to conform to the requirements
GG of IEEE 279 - paragraph 4.1 and paragraph 4.16. Amend your FSAR
16 to indicate non-compliance and the design basis that supports it. Similar changes are required for the IEEE 279 analysis for other manually actuated ESF's.

Q030. Throughout the discussions of single failure criteria the terms
(7.3) "credible" and "credible aspects of" are occasionally used to
GG modify "single failure". Define these terms and state the
17 specific aspects of the single failure criterion that are not credible.

Q030. In the discussion of channel independence for the Suppression
(7.3.2.9.2) Pool Makeup System and for the CRACIS, it is stated that physical
(7.3.2.10.2) separation is maintained where it adds to the reliability of
GG operation. Identify the particular places where physical
18 separation doesn't add to the reliability and indicate the

distances over which physical separation is not maintained.

Q030. The following inconsistencies and deficiencies have been noted in
(7.3) Section 7.3:
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1. Tables identifying the sensor type, instrument range, accuracy and trip setpoint are provided for 7 of the 10 ESF systems discussed in 7.3. The three systems without tables are the MSIV-LCS, the CSCS, and the SSW. The FSAR states the SSW is initiated by other systems and has no ESF instrumentation. No reason is given for omitting the other two systems.
2. Failure mode and effects analysis are provided for 6 of the 10 systems. The systems omitted are the ECCS, CRVICS, MSIV-LCS, and CSCS.
3. References to plant drawings range from an actual drawing number to a general reference to Section 1.7. A general reference to Section 1.7 is inadequate in those cases where the system nomenclature used in the FSAR is not used in the drawing titles.
4. The analysis for conformance generally covers the criteria identified in Table 7.1-3 but occasionally omits some. The worst case is the Containment Spray Cooling System which only addresses IEEE 279.

Amend your FSAR to provide complete and consistent information for each ESF system.

Q030. The accident analysis in Chapter 15 takes credit for the pressure
(7.3) relief available through the automatic sequencing and operation of

GG	the safety relief valves. Justify the exclusion of the
20	instrumentation and controls for the relief valves from Chapter 7
	in general and Section 7.3 in particular.
Q030.	The figures in Section 7.1 and 7.3 are inconsistent in the
(7.3.1.1.2.4)	separation and logic used in the ESF. The Elementary Diagrams
(F7.1-3)	agree with F7.3-4 rather than F7.1-3 for Division 1 and Division 2
(F7.1-4)	but do not agree with either for Division 3 and RCIC. Reference
(F7.1-5)	B21-1090 agrees with F7.3-5 and disagrees with F7.1-5; however,
(F7.3-4)	the discription of the logic in 7.3.1.1.2.4 describes both systems
(F7.3-5)	in successive paragraphs. Figures 7.1-4 and 7.3-6 are
(F7.3-6)	functionally the same but disagree on the assignment of logic
MPL B21-1090	between inboard and outboard valves. Revise the appropriate
MPL E22-1050	documents to achieve consistency and correctness.
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21	
Q030.	The response to Question 211.11 states that the High Differential
(7.0)	Temperature isolation circuit will not be connected (to RCIC and
(Q & R 7.6-8)	RHR Steam isolation circuits) until a setpoint can be established
GG	which will minimize inadvertent isolation. Verify that every
22	protective circuit and interlock described in the FSAR will be
	inplemented before the reactor is started up and will thereafter
	be used as described in the FSAR.

INFORMAL EDITORIAL COMMENTS ON SECTION 7.3 OF GRAND GULF FSAR

Page	Comment
7.3-55, 58	7.3.1.1.4.4 says containment pressure is measured with absolute P.T.'s while 7.3.1.1.4.12.3 says the permissive setpoint of the P.T. is 9 PSIG. Also, 2nd and 3rd paragraphs on page 55 are redundant.
7.3-90 & 7.3-119	Question the statement that sensors are not subject to saturation when overranged. Although the "sensors" may not be saturated, transmitter output will have to saturate at some point.
7.3-91	Paragraph d. First sentence does not specifically state the system meets single failure criterion.
7.3-103 & 7.3-148	Last paragraph, 7.3.2.4.3.1.14. Question the use of the word "may" rather than will.
7.3-105	ADS paragraph 1 refers to "AC interlocks" in the last sentence. These interlocks are not explained in the section.
7.3-106	First paragraph. The first two sentences don't seem to be pertinent to the subject and these points are not made for any other ESF system.
7.3-107	Paragraph a. HPCS. The first three sentences are vague on the HPCS readout.
7.3-117	Item 8. It is not clear whether the phrase "without malfunction of either subsystem" is a result or a qualifying condition required to make the system "accident tolerant".
7.3-133	7.3.2.2.2.3.4. This is the first indication that the CRVICS and NSSSS are the same thing. Also, the reference is wrong, it should be 7.1.2.4 or better yet, 3.11.2 since referring to 3.11.2 is all that 7.1.2.4 does.
7.3-120 & 7.3-133	Conformance to single failure criteria of IEEE 279 is discussed under IEEE 379 for MSL high radiation while other subsystems are discussed under IEEE 279.
7.3-134	Last paragraph. Reference is incorrect; should be 7.5.1.3.
7.3-137	First paragraph. Run on sentence; replace with first paragraph from page 140.

Page	Comment
7.3-137 Cont'd.	Paragraph 7.3.2.3.2.3.1.7 indicates the MSIV-LCS is part of the control system.
7.3-138	Paragraph 7.3.2.3.2.3.1.12. It is not clear what the statement says about "operating bypasses".
7.3-106, 7.3-130 & 7.3-149	Paragraph 279-4.18. The first sentence is poorly worded.
7.3-151 & 7.3-153	Paragraph f. and paragraph b. Reference is incorrect; should be 7.3.2.5.5.
7.3-157	Paragraph a, b, and f are poorly worded and/or poorly related to the respective criterion.
7.3-162 & 7.3-164	Paragraph e. and paragraph b. Reference is incorrect; should be 7.3.2.7.4.
7.3-168	Paragraph e. Reference is incorrect; should be 7.3.2.8.4.
T7.3-1	Question indication that CRVICS does not require any auxiliary supporting systems.
T7.3-2	Low water level trip setting should be - 41.8. Range of Suppression Pool and CST transmitters appear to be in error.
T7.3-3, T7.3-4 & T7.3-5	Trip settings for LPCI and LPCS permissives are not consistent between tables.
T7.3-10	Trip settings for R.V. levels 3, 2, and 1 should be included in the table, not referenced, for consistency. Also, it appears the the MSL high flow and drywell high pressure trips should be $\geq$140% and $\geq$2 psig. (Either these are wrong or errors exist In Tables 7.3-2, 3, 4, and 5)
T7.3-11	Table appears to be redundant to information in T7.3-10.
T7.3-15	Typing error under remarks for "Loss of one ESF DC bus".
T7.3-16	Nominal margin value should be shown in "psi" not "psig".
T7.3-18	Title doesn't indicate what system is involved. "Loss of Inst. Air" is listed as a failure mode while "Remarks" says air is not used in this system.

Page	Comment
T7.3-25	Table indicates a 6" change in level changes the margin by 20".
T7.3-27	First two failure modes indicated are not failure modes of the system being analyzed.
F7.3-2	Both ADS figures are incorrect. Electrical drawings show LPCI/LPCS pumps must be operating before the ADS receives a start signal.
F7.3-7	Title of figure does not say what system is involved.
T7.3-6	The "minimum operable channels" for reactor vessel low water level should be the same as for high dry-well pressure.
T7.3-7	The first two sentences explaining the asterisk seem to be unnecessary and in conflict with the remainder of the description.

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REGULATORY DOCKET FILE COPY

MEMORANDUM FOR: Olan Parr, Chief, Light Water Reactors Branch No. 1, DPM

FROM: Rodney M. Satterfield, Chief, Instrumentation and Control Systems Branch, DSS

SUBJECT: ADDITIONAL ICSB QUESTIONS - SUSQUEHANNA UNITS 1 AND 2

Plant Name: Susquehanna Units 1 and 2
Docket Numbers: 50-387, 388
Licensing State: OL
Milestone Number: 8
Responsible Branch: LWR #3
Project Leader: S. Miner
ICSB Reviewer: R. Gregory (Savannah River Plant)
Review Status: Incomplete

Enclosed are additional questions that were generated by the Savannah River Plant ICSB reviewer following his assessment of Section 7.3 of the Susquehanna 1 and 2 Final Safety Analysis Report. We anticipate that additional questions will be forthcoming as the review continues. We suggest that these questions be forwarded to the applicant as soon as possible. The applicant should be informed that there is a need to respond as promptly as possible to ensure that the reviewer completes his review on schedule.

The numbers that appear with the enclosed questions were generated by Savannah River. Please revise the numbers to be consistent with the previous round one questions.

ORIGINAL SIGNED BY
RODNEY M. SATTERFIELD
Rodney M. Satterfield, Chief
Instrumentation and Control Systems Branch
Division of Systems Safety

Enclosure:
As stated

cc; V. Moore
S. Miner
T. Dunning
A. Hadden (SRL)

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OFFICE	ICSB	ICSB				
SURNAME	TDunning:cc	RSatterfield				
DATE	1/31/80	2/3/80				

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032.
7.3.1.1a.1
SUSQ
5

Discussion of the Emergency Core Cooling Systems and the associated tables are incomplete and inconsistent. Correct and clarify the following:

1) The same instruments are used for Reactor Vessel low water level and Primary Containment high pressure for many ESF systems. The specification shown for these instruments in Tables 7.3-1 through 7.3-5 are not consistant. Correct trip settings, ranges, and accuracies shown for these instruments.

2) These tables have allotted columns for instrument response times and margins (of trip setting) to meet requirements of IEEE 279-1971 Section 3, but most data has been omitted. Response times should indicate minimum and/or maximum where applicable.

3) Table 7.3-1 has omitted all specifications for the Turbine overspeed instrument.

4) Figure 7.3-5 has several errors:
   - o It does not show two ADS logics as indicated in 7.3.1.1a.1.4.4.
   - o Referenced Figure 7.3-16 does not exist.
   - o It does not show low pressure interlocks to LPCI and CS required to initiate ADS as indicated in 7.3.1.1a.1.4.4.

5) Table 7.3-2 indicates only one reactor water level setpoint (-149 inches) for the ADS. Section 7.3.1.1a.1.4.4 indicates two level setpoints, a low and a lower water level.

6) Use of level switches with a range of -150"/0/+60" to initiate ADS and CS action with trip settings at -149 does not seem like conservative design. Justify the use of this range for this application. Discuss accuracy of the trip setting and how it is affected by normal and accident environmental conditions and long term drift.

7) Why are two ranges shown for LPCI pump discharge pressure (10-240 psig and 10-260 psig). Range shown for this instrument in Table 7.3-4 is 10-240 psig only.

8) Section 7.3.1.1a.1.4.5 on ADS Bypasses and Interlocks indicates that it is possible for the operator to manually delay the depressurizing action and states "This would reset the timers to zero seconds and prevent depressurization for <u>105</u> seconds." Table 7.3-2, Figure 7.3-8 Sht. 3 and Table 6.3-2 all indicate a time delay of 120 seconds. How is a time delay of 105 seconds achieved?

9) Explain why two ranges (50-1000 psig and 50-1200 psig) are listed for the Reactor Vessel Low Pressure instrument in Table 7.3-3.

10) Instrument ranges for pump discharge flow, Table 7.3-3, and pump minimum flow bypass, Table 7.3-4, are specified in inches of water but trip settings are in gpm. Supply ranges for these flow instruments in gpm.

11) Table 7.3-8 HPCI System Minimum Numbers of Trip Channels Required for Functional Performance does not agree with Table 7.3-1 HPCI Instrument Specifications. Table 7.3-8 does not list HPCI pump high suction pressure or Turbine

Overspeed as shown in Table 7.3-1. Table 7.3-8 lists two items, HPCI pump flow and HPCI pump discharge flow, not shown in Table 7.3-1.

12) Table 7.3-4 Low Pressure Coolant Injection - Instrument Specifications does not agree with Table 7.3-10 Low Pressure Coolant Injection System Minimum Number of Trip Channels Required for Functional Performance. Table 7.3-10 does not list Reactor low pressure or Pump discharge pressure as shown in Table 7.3-4. Table 7.3-10 lists several trip channels which are not shown in Table 7.3-4. These include Reactor vessel low water level inside shroud, Reactor vessel low flow, Primary containment high pressure, and Reactor vessel low water level (Recirculation Pumps).

13) Table 7.3-11 Core Spray System Minimum Numbers of Trip Channels Required for Functional Performance is incomplete. It does not list Pump Discharge Flow as shown in Table 7.3-1.

032. Discussion of the Primary Containment and Reactor Vessel
7.3.1.1a.2 Isolation Control System in Section 7.3.1.1a.2 and associated
SUSQ Tables 7.3-5, 7.3-7 and 7.3-12 are confused, incomplete and
6 inconsistent. Correct or clarify the following:

1) Several instruments listed in 7.3.1.1a.2.1 are not discussed in the text and/or do not appear in the tables. These include RWCS High Flow, RHRS High Flow, RICI High Flow, HPCI High Flow.

2) Several items only appear in Table 7.3-5 with no discussion. These include RCIC Turbine Steamline High Temperature and Low Pressure, HPCI Turbine Steamline High Temperature and Low Pressure, Reactor Building and Drywell Ventilation Exhaust High Radiation.

3) In Table 7.3-5, instrument ranges, setpoints, accuracies, and time responses have been omitted for many sensors. Several sensors discussed in the text are not listed at all. These include Condenser Vacuum, RHR High Temperature and Differential Temperature, RWCS Differential Temperature, Main Steamline Differential Temperature. It is understood that some setpoints will be selected based on operating conditions, but these sensors must be identified.

4) Table 7.3-12 is redundant. It has only one entry, serves no purpose and could be eliminated.

5) Section 7.3.1.1a.4.12, Main Steamline-Leak Detection, appears to serve no purpose since all items are discussed in other parts of this section on the PCRVICS.

6) Table 7.3-7, Trip Channel Required for PCRVICS, is incomplete. Many functions discussed in the text and/or listed in Table 7.3-5 are missing.

7) Section 7.3.1.1a.2.4.2 references Table 7.3-7 for instrument characteristics. These are actually shown in Tables 7.3-5.

032. It is the current staff position that Mark II suppression

7.3.1.1a.4 SUSQ 7	chamber sprays be actuated automatically instead of manually. Similar plants such as Zimmer and Shoreham are making this change. Identify any significant differences between these plants and Susquehanna in this reguard and justify the proposed manual system.
032. Table 6.2-12 7.3.1.1a.2.9 7.3.1.1a.2.11 SUSQ 8	Describe test method to be used to verify closing times for main steamline isolation valves are within limits of technical specifications. Identify any special design features to facilitate this test. Table 6.2-12 is referenced for closure times of main steamline isolation valves, but time has been omitted from that table. What is the range of acceptable closure times?
032. 7.3.1.1a.3 SUSQ 9	Review of the Main Steamline Valve Isolation Control System logic at Hatch 2 and similar plants determined that failure of a single relay could cause two redundant isolation valves to open. Has this problem been corrected in the Susquehanna design?
032. SUSQ 10	General Electric and other NSSS suppliers have reported that post-accident temperature conditions can affect reactor vessel water level instrumentation. 1) Describe the liquid level measuring systems within containment that are used to initiate safety actions or are used to provide post-accident monitoring information. Provide a description of the type of reference leg used

i.e., open column or sealed reference leg.

2) Provide an evaluation of the effect of post-accident ambient temperatures on the indicated water level to determine the change in indicated level relative to actual water level. This evaluation must include other sources of error including the effects of varying fluid pressure and flashing of reference leg to steam on the water level measurements.

3) Provide an analysis of the impact that the level measurement errors in control and protection systems (2 above) have on the assumptions used in the plant transient and accident analysis. This should incluue a review of all safety and control setpoints derived from level signals to verify that the setpoints will initiate the action required by the plant safety analyses throughout the range of ambient temperatures encountered by the instrumentation, including accident temperatures. If this analysis demonstrates that level measurement errors are greater than assumed in the safety analysis, address the corrective action to be taken. The corrective actions considered should include design changes that could be made to ensure that containment temperature effects are atuomatically accounted for. These measures may include setpoint changes as an acceptable corrective action for the short term. However, some form of temperature compensation or modification to eliminate or reduce temperature errors should be investigated as a long term

solution.

4) Review and indicate the required revisions, as necessary, of emergency procedures to include specific information obtained from the review and evaluation of Items 1, 2, and 3 to ensure that the operators are instructed on the potential for and magnitude of erroneous level signals. Provide a copy of tables, curves, or correction factors that would be applied to post-accident monitoring systems that will be used by plant operators.

032. Pressure switches 1N022 A through S are used to actuate the 16
SUSQ safety relief valves in the overpressure mode of operation as
11 described in section 5.2.2.4.

1) Describe the logic associated with these instruments including those associated with ADS relief valves (Figure 7.3-8 Sht. 3) and non-ADS relief valves.
2) Identify design criteria and requirements met by this system.
3) Justify the use of a single instrument to operate each relief valve and analyze the effects of single failures.

032. The purpose of the Recirculation Pump Trip (RPT) is to aid the
7.6.1a.8 Reactor Protection System (RPS) in protecting the integrity of
SUSQ the fuel barrier.
12
1) Is the RPT designed in accordance with all requirements for the RPS? If not, identify and justify any exceptions.
2) Plants such as Hatch 2 and Zimmer have provided

recirculation pump trips for reactor vessel low water level or high reactor pressure. Why have these not been provided for Susquehanna?