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Mr. Themis Speis, Chief
Reactor Systems Branch
Mail Stop 268
Phillips Building
U.S. Nuclear Regulatory Commission
Bethesda, MD 21609

Dear Mr. Speis:

Attached are initial questions relating to the Susquehanna Nuclear Station FSAR. The questions cover Sections 15.0, 15.1, 15.2, 15.3, 15.4, 15.5, 15.6, 4.6, 3.13, and 3.2.

Your assistance is requested in obtaining responses from the applicant.

Yours very truly,

W M Taylor

W. M. Taylor
Superintendent
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WMT:ssr
ATT

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Q211. (15.0) SSES 1	Provide a realistic range and permitted operating band for the exposure dependent parameters in Tables 4.4-1 and 15.0-2. In Table 15.0-2, provide assurance that values of parameters selected yield the most conservative results.
Q211. (15.0) SSES 2	Uncertainty exists on the correct value of APRM neutron flux scram setpoint to be used in transient analyses. The value indicated as input for transient analysis in Table 15.0-2 is 125% NBR. However, a value of 120% NBR is indicated in Tables 7.2-4 and 7.6-5. Explain this discrepancy. For the correct value of setpoint used in transient analyses, provide a breakdown of any uncertainty allowances that are added to the nominal value.
Q211. (15.0) SSES 3	Provide a listing of the transients and accidents in Chapter 15 for which operator action is required in order to mitigate the consequences.
Q211. (15.0) SSES 4	The response to question 211.113 does not provide sufficient detail on non-safety grade equipment and components which mitigate transients and accidents. Provide a table of the non-safety grade equipment and components assumed to mitigate consequences for each transient and accident in Chapter 15.
Q211. (15.0) SSES 5	The analysis of transients and accidents in Chapter 15.0 does not state which of the RPS time response delays in Table 7.2-5 is used in the REDY computer model (NEDO-10802). For each transient and accident in Chapter 15.0, specify whether the sensor

or overall delay time is used in the analysis and why the specified delay time is conservative.

Q211. (15.0) SSES 6

Confirm the following items for all transients in Chapter 15.0 which require control rod insertion to prevent or lessen plant damage.

a) All calculations were performed with the conservative scram reactivity curve No. 2 in Figure 15.0-2.

b) The slowest allowable scram insertion speed was used.

Q211. (15.0) SSES 7

a) In Table 1 of Figure 5.1-3a (Nuclear Boiler), the relief valve spring set pressure of 1130 psig for safety/relief valves B and E does not agree with a corresponding value of 1146 psig in Table 5.2-2 of the FSAR and in Table 1 of Drawing M-141, Rev. 9. Correct this setpoint discrepancy for safety mode (mechanical) actuation.

b) For transient analysis, credit has been taken for safety/ relief valve actuation in the relief mode. A more conservative approach would be to take credit for safety/ relief valve actuation in the safety mode, resulting in higher peak vessel pressures.

1) What effect on MCPR and peak vessel pressure does credit for safety/relief valve actuation in the safety mode have on transients analyzed in Chapter 15?

2) Are all equipment and components required for safety/ relief valve actuation in the relief mode safety grade?

Q211. (15.0) SSES 8

Modify Table 15.0-1 as follows:

a) Give calculated values of MCPR instead of the entry >1.06.

b) For the "feedwater controller failure at maximum demand" transient, correct the discrepancy in values for maximum vessel pressure, maximum steam line pressure, and MCPR that exists between Table 15.0-1 and Section 15.1.2.3.3.

1747 036

Q211. For transients and accidents in Chapter 15 in which it is stated
(15.0) that the operator initiates some corrective action, provide
SSES justification for any corrective actions by the operator prior to
9 20 minutes.

Q211. Discuss how the pre-operational and startup tests will be used to
(15.0) confirm flow parameters used in Chapter 15 analyses. Provide
SSES details of any previous test of components in test facilities
10 conducted to show satisfactory performance of the recirculation
and feedwater flow control systems and respective pumps. Describe
how this information was used in Chapter 15 analyses.

Q211. Analyze the turbine trip and generator load rejection transient
(15.0) from a safe shutdown earthquake event. Credit should not be
SSES taken for non-seismically qualified equipment which include:
11 a) Any equipment contained in a non-seismic structure; and
b) Any equipment which is not seismically qualified.

Q211. On page 4-7 of NEDO-10802, it is stated that the difference in
(15.0) trend of flow coastdown versus initial power between the
SSES analytical and experimental coastdown curves for Dresden Unit
12 No. 2 (a BWR/3) in Figure 4-11 was due in part to differences
between actual and computed jet pump efficiencies.

a) How has this effect been treated in analysis of SSES transients involving flow coastdown with two recirculation pump trip (RPT)?
b) Is this treatment applicable to Susquehanna which is a BWR/4?

Q211.
(15.1.1.2.1)
SSES
13

For the "loss of feedwater heating" transient, the sequence of events in Tables 15.1-1 and 15.1-2 for the automatic and manual flow control modes, respectively, are not described in sufficient detail to permit evaluation of transient results in Figures 15.1-1 and 15.1-2 and comparison with NSOA events in Figure 15A.6-21. For the more limiting manual flow control mode, no detail is presented in Table 15.1-2 between 2 and 40 plus seconds. Revise Table 15.1-2 to include NSOA events in Figure 15A.6-21 and additional detail between 2 and 40 plus seconds.

Q211.
(15.1.1.2.3)
SSES
14

The thermal power monitor (TPM) is not included in the Susquehanna design per response to question 211.118. However, it is indicated as the primary protection system for mitigating the consequences of the "loss of feedwater heating" transient in Section 15.1.1.2.2. What was used to scram the reactor in the manual mode? Modify Figure 15A.6-21 and Sections 15.1.1.2.2 and 15.1.1.2.3 accordingly.

Q211.
(15.1.1.3.2)
SSES
15

This section states that input parameters and initial plant conditions for the "loss of feedwater heating" transient are in Table 15.0-1. This should be changed to Table 15.0-2 in this section and in the corresponding sections of the remaining transients in Chapter 15 where this discrepancy occurs.

1747 038

Q211. (15.1.2.2.1) SSES 16	Correct discrepancies between events in Table 15.1-3 and NSOA Figure 15A.6-22 for the "feedwater controller failure at maximum demand" transient. Table 15.1-3 does not include the initial core cooling and reactor vessel isolation events indicated in Figure 15A.6-22.
Q211. (15.1.2.3.1) SSES 17	Explain the basis for the assumed feedwater flow controller failure at 135% flow. Is the indicated failure initiated at 0 seconds or does the failure begin at 0 seconds and increase to 135% flow at a later time. If the former is true, correct Figure 15.1-3 accordingly.
Q211. (15.1.2.3.1) SSES 18	Please correct the inadvertent combination of Section 15.1.2.3.2, beginning on page 15.1-7, with Section 15.1.2.3.1.
Q211. (15.1.2.3.3) SSES 19	Provide justification that analysis of this transient at 105% NBR steam flow is more restrictive than at low power. If so, delete reference to "low power" for NSOA event No. 22 in Table 15A.2-2. If not, re-analyze and make appropriate corrections.

1747 039

Q211. (15.1.3.3.3) SSES 20

a) It is not apparent from the text whether the "pressure regulator failure-open" transient is terminated by a low turbine-inlet pressure trip or a L8 trip. Trips indicated in various sections of the text are summarized below:

Section	Trip
15.1.3.2.1.1	Low pressure at the turbine inlet
15.1.3.3.2	Low pressure at the turbine inlet
15.1.3.3.3	L8 trip
Table 15.1-4	Low pressure at the turbine inlet

Specify which trip is most restrictive on thermal margins and revise applicable tables, sections, and figures of the FSAR.

b) It appears that less than the assumed 115% NBR steam flow in Section 15.1.3.3.2 was simulated at the beginning of the transient in Figure 15.1-4. Explain this discrepancy and make corrections, if necessary.

c) Safety/relief valve (SRV) actuation for this transient in the relief mode is not included in Tables 15.0-1 and 15.1-4 and Figure 15.1-4 for decay heat removal. Please explain.

Q211. (15.1.3.2.1) SSES 21

In Table 15.1-4,

a) Include safety/relief valve actuation times for the "pressure regulator failure-open" transient.

b) Indicate the value of steam flow simulated at time = 0, presumably 115% NBR per Section 15.1.3.3.2.

1747 040

Q211. (15.1.3.3.2) SSES 22

Specify the assumed operating mode (manual or automatic) of the recirculation flow control system for the "pressure regulator failure-open" transient and provide justification that the most conservative results on core thermal margins are obtained with the assumed operating mode.

1747 041

Q211. A qualitative presentation of results for the "inadvertent

(15.1.4.3.1) safety/relief valve opening" transient is given because analyses

SSES from earlier FSAR's indicated this event is not limiting from a

23 thermal margin standpoint.

a) Provide supporting data that justifies this condition (i.e., referenced plant and MCPR).

b) The discussion in Section 15.1.4.3.2 implies a quantitative analysis was made. A statement similar to that in Section 15.2.1.3.2 would be more appropriate.

1747 042

Q211. (15.2.1.2.1) SSES 24	For the "pressure regulator failure-closed" transient, correct the discrepancy that exists between the 5 psi setpoint difference for the backup pressure regulator in Sections 15.2.1.1.1 and 15.2.1.2.1 and a corresponding 10 psi setpoint difference in Section 10.3.2.
Q211. (15.2.1.3.3) SSES 25	It is stated that the pressure disturbance in the reactor vessel from failure of the primary pressure regulator in the closed mode is not expected to exceed flux or pressure scram trip setpoints. Is this conclusion based on quantitative results in earlier FSARs? If so, reference appropriate sections of these FSARs or provide a summary of the results.

1747 043

Q211. (15.2.2.3.2) SSES 26	In the evaluation of the "generator load rejection" transient, a full-stroke closure time of 0.15 seconds is assumed for the turbine control valves (TCV). Section 15.2.2.3.4 states that the assumed closure time is conservative compared to an actual closure time of more like 0.20 seconds. However, in Figure 10.2-2, <u>Turbine Control Valve Fast Closure Characteristic</u>, an acceptable TCV closure time of 0.08 seconds is implied. Explain this apparent non-conservative discrepancy and the effect it has on analyses in Chapter 15 requiring TCV closure.
Q211. (15.2.2.3.3.1) SSES 27	Explain why vessel steam and bypass flows in Figure 15.2-1 drop to zero at approximately 37 seconds instead of zero at 45-plus seconds from a L2 vessel level isolation in Table 15.2-1.
Q211. (15.2.2.4.1) SSES 28	During the "generator load rejection with bypass" transient, it is stated that peak pressure remains within normal operating range. Explain how this is accomplished since safety/relief valve actuation in the relief mode occurs from the pressure increase.

1747 044

Q211. (15.2.3.2.1.2) SSES 29	Correct NSOA Figure 15A.6-31, <u>Protection Sequence Main Turbine Trip - With Bypass Failure</u>, by reversing the indicated power levels. This error occurred during revision of this figure per Question 211.110.
Q211. (15.2.3.2.1.3) SSES 30	Would a turbine trip coupled with failure of the operator to put the mode switch in the startup position before reactor pressure decays to <850 psig [action (5)] be more restrictive on thermal margins than the "turbine trip with bypass failure" transient analyzed in Section 15.2.3.3.3.2?
Q211. (15.2.3.2.3.1) SSES 31	This section addresses the effect of single failures and operator errors for turbine trips at power levels >67%. a) What is the basis for power levels >67%? b) Explain the discrepancy with NSOA Figures 15A.6-25 and 15A.6-31 which refer to power levels >30%.
Q211. (15.2.3.3.2) SSES 32	In the evaluation of the turbine trip transients, 0.10 second is assumed for full-stroke closure time of the turbine stop valve. Demonstrate that turbine stop valve closure times smaller than 0.10 second do not result in unacceptable increases in MCPR and reactor peak pressure or provide either (1) justification that smaller closure time cannot occur or (2) a minimum closure time to be incorporated in the Technical Specifications.
Q211. (15.2.3.3.3.1) SSES 33	During the "turbine trip with bypass" transient, explain why vessel steam and bypass flows in Figure 15.2-3 drop to zero at approximately 37 seconds instead of zero at 45-plus seconds from a L2 vessel level isolation in Table 15.2-3.

1747 045

Q211. This section includes a detailed discussion of activity above the
(15.2.4.5) suppression pool, activity releases to the environs, and offsite
SSES radiological doses. Explain why this information was not included
34 in corresponding sections of other events in Chapter 15 requiring SRV actuation. For instance, the "generator load rejection with bypass failure" transient clearly has a higher peak vessel pressure and longer blowdown.

Q211. Table 15.2-5 does not list all significant events up to 40
(15.2.4.2.1) seconds for the "closure of all MSIV" transient. Include the
SSES following items:
35 a) Significant actions associated with attainment of applicable vessel level setpoints.

b) Recirculation pump runback if it was simulated in the analysis.

1747 046

Q211. (15.2.5.2.1) SSES 36

Include the time at which the turbine stop valves are closed in Table 15.2-10.

Q211. (15.2.5.3.3) SSES 37

This section states that the turbine bypass valve and main steam isolation valve closure would follow the main turbine and feedwater turbine trip about 5 seconds after they initiate during the transient. Based on this, the bypass valves should close at approximately 5.01 seconds instead of 12.1 seconds in Table 15.2-10 and Figure 15.2-6. Explain this apparent discrepancy.

1747 047

Q211. (15.2.6.2.1.1) SSES 38

Add the following items to Table 15.2-12 to be consistent with Figure 15A.6-28 for the "loss of auxiliary power transformer" transient:

a) Safety/relief valve actuation

b) Reactor vessel and containment isolation.

Q211. (15.2.6.2.1.2) SSES 39

Add the following items to Table 15.2-13 to be consistent with Figure 15A.6-29 for the "loss of all grid connections" transient:

a) Reactor vessel and containment isolation

b) Initiation of the standby AC power system.

1747 048

Q211. (15.2.7.2.2) SSES 40

It is indicated that credit is taken for safety/relief valve operation with "low setpoints" to remove decay heat since bypass valves become ineffective with MSIV isolation. Does this mean use of relief mode setpoints that are lower than the safety mode setpoints or does this imply use of setpoints lower than the relief mode values in Table 15.0-2.

1747 049

Q211. (15.2.9.2.1.1) SSES 41

For the "failure of RHR shutdown cooling" transient, the FSAR considers alternate shutdown cooling methods in the event the residual heat removal (RHR) system in the suction line may not be used because of valve failure. In the analysis, valves in the automatic depressurization system (ADS) were used to transfer fluid (steam, water or a combination of these) from the reactor vessel to the suppression pool. The RHR system removes the added heat by removing cooling water from the suppression pool and injecting it into the reactor vessel. We require that you perform a test or cite previous test results to demonstrate that the ADS valves can discharge the fluid flow under the most limiting conditions when the fluid is all water. Show that this alternate method is a viable means of shutdown cooling by comparing the system hydraulic losses with the available pump head. Hydraulic losses should be provided for each system component and, wherever possible, should be derived from experimental results.

Q211. (15.3.1.3.3.2) SSES 42 — Table 15.3-2 indicates that zero vessel steam flow does not occur until after 46 seconds. However, Figure 15.3-2 indicates zero steam flow occurs at approximately 36 seconds. Explain this discrepancy.

Q211. (15.3.1.3.2) SSES 43 — In the analysis of one and two recirculation pump trip events in Sections 15.3.1, a minimum design rotating inertia was used to obtain a predicted rate of decrease in core flow greater than expected. Specify the type inertia value (minimum, average, or maximum) used for each transient in Chapter 15 and the basis for selection of each. In the selection basis, include the effect (increase, decrease, or no change) on MCPR and reactor vessel pressure.

Q211. (15.3.1.3.3.2) SSES 44 — Include relief valve flow in Figure 15.3-2.

1747 051

Q211. a) Table 15.3-3 indicates that zero steam flow should not occur

(15.3.3.3.3) until after 41.7 seconds. However, Figure 15.3-3 indicates

SSES zero steam flow at approximately 35 seconds. Explain this

45 discrepancy.

b) Include relief valve flow in Figure 15.3-3.

1747 052

Q211. The narrative on page 15.4-13 states, "The water level does not
(15.4.4) reach either the high or low level set points." Table 15.4.3
SSES indicates a low level trip occurs 22.0 seconds after pump start.
46 Sector 3 of Figure 15.4-6 indicates a low level trip occurs
approximately 23.5 seconds after pump start. Further:

a) Table 15.4-6 indicates a low level alarm 10.5 seconds after pump start while Sector 3 of Figure 15.4.6 indicates this alarm occurs about 11.5 seconds after the pump starts.

b) Table 15.4-6 indicates vessel level beginning to stabilize 50.0 seconds after the pump starts. Sector 3 of Figure 15.4-6 shows no such indication.

Resolve these discrepancies.

Q211. Please identify the diffuser flow units in Sector 2 of Figure
(15.4.4) 15.4-6 (and also in Sector 2 of Figure 15.4-7). If this is %
SSES flow, explain why diffuser flow 1. drops to zero about 30 seconds
47 after the pump starts.

1747 053

Q211. (15.5.1) SSES 48	The narrative on page 15.5-3 and Table 15.5-1 both indicate full HPCI flow is established at approximately 19% of rated feedwater flow in one second. Explain why the curve of feedwater flow in Sector 1 does not show this change.

1747 054

Q211. (15.6.1) SSES 49

The FSAR indicates this transient is analyzed in Subsection 15.1.4. However, no analytical data (curves) are provided in Subsection 15.1.4. Please supply necessary information so that this transient can be evaluated concerning a decrease in reactor coolant inventory.

1747 055

Q211. A number of inconsistencies exist among narrative descriptions, tables, and figures in Appendix 15A relative to Control Rod Drive System. Please resolve.

(4.6)

SSES

50

a) Table 15A.6-2 indicates event 7 can occur in states C & D. Figure 15A.6-7 indicates applicability to states A, B, C, D. Narrative on page 15A-35 indicates any state.

b) Table 15A.6-2 indicates event 16 can occur in states A, B, & C. Narrative and Figure 15A.6-16 indicate applicability in states A & B only.

c) Figure 15A.6-17 and narrative on page 15A-39 indicate event 17 is applicable in states C & D. Definition indicates not applicable in state C.

d) Figure 15A.6-25 does not indicate event 25 applicable to state D only.

e) Figure 15A.6-28, Table 15A.6-2 and narrative page 15A-44 for event 28 are inconsistent for applicable states.

f) Narrative page 15A-50, Table 15A.6-4, Figure 15A.6-40 for event 40 are inconsistent for applicable state.

1747 056

Q211. (3.13.1) SSES 51

Regulatory Guide 1.29, Section C.1.e, specifies that portions of the steam systems of boiling water reactors extending from the outermost containment isolation valve up to but not including the turbine stop valve, and connected piping of 2 1/2 inches or larger nominal pipe size up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation, be classified Seismic Category I. You state on page 3.13-10 that your equivalent portion of the steam system is non-Seismic Category I. Justify your design deviation from the above requirements.

Q211. (3.13.1) SSES 52

Your response to Question 211.1 indicates that GE is currently preparing a Licensing Topical Report to provide an analytical basis for recirculation pump seal leakage. Provide this report.

Q211. (3.13.1) SSES 53

a) Item (5) on page 3.13-11, discusses those portions of structures, systems, or components (SSC) whose continued function is not required but whose failure could reduce the functioning of items important to safety. Provide a list of these SSC.

b) Regulatory Guide 1.29, Section C.4, requires that Appendix B of 10 CFR 50 should be applied to the above SSC. Provide justification for not including such items in the 10 CFR 50 Appendix B Quality Assurance Program.

Q211.

a) Provide a list of those structures, systems and components

(3.13.1) which form interfaces between Seismic Category I and

SSES non-Seismic Category I features.

54 b) Provide justification for not adhering to 10 CFR 50, Appendix B for such items (item (6), page 3.13-11).

1747 058

Q211. Regulatory Guide 1.29, Section C.1.b, requires that all reactor
(3.2.1) vessel internals to be Seismic Category I. Table 3.2-1 indicates
SSES that reactor vessel internals other than engineered safety
55 features are not Seismic Category I. Please justify.

Q211. In Table 3.2-1, fill in the following information, where missing:
(3.2.1) (1) Principal construction codes and standards (most pages).
(3.2.2) (2) Page 18, Main Steam System: Pressure vessels, heat exchangers
SSES (all information).
56 (3) Page 1, Nuclear Boiler System: Air supply check valves
(safety class).

Q211. The RHR pump return line as shown on P & I Diagram M-151 (Figure
(3.2.2) 5.4-13) penetrates into the Suppression Chamber as a Safety Class
SSES 2, Quality Group B line (pipe 18"-GBB-109). After penetration,
57 the quality group classification is changed to D. Standard Review Plan Section 3.2.2 states that changes in quality group classification are usually permitted only at valve locations, with the valve assigned the higher classification. Demonstrate that the safety function of the system is not impaired due to the fact that quality group classification changes at a point where no valve was located.

Q211. The RHR containment spray line piping (within isolation valve) is
(3.2.2) listed as Quality Group A, Safety Class 1, Seismic Category I
SSES (Table 3.2-1, page 4). In Figure 5.4-13 (P & ID M-151) this
58 line is indicated as 12" GBB-118, i.e. Quality Group B. Resolve

this inconsistency.

Q211. (3.2.2) SSES 59	Table 3.2-1, page 10, lists piping and valves forming a part of containment boundary of the Reactor Building Closed Cooling Water System as Quality Group B, Safety Class 2, Seismic Category I. Penetration of primary containment for this piping is not shown on any of the relevant P & I Diagrams. Show the above piping and valves on appropriate P & I Diagrams and indicate the classification of this piping.
Q211. (3.2.1) SSES 60	Confirm that the piping from the condensate storage tank to the HPCI pump suction is safety grade and seismically classified.
Q211. (3.2.1) SSES 61	Regulatory Guide 1.29 states that systems required for post accident containment heat removal should be Seismic Category I. Justify why the Reactor Building Closed Cooling Water System pumps and heat exchangers are not Seismic Category I.

1747 060