

SSS-FSAR

QUESTION 121.1

Provide a sketch of the Susquehanna reactor vessels (including dimensions) showing all longitudinal and circumferential welds, and all forgings and/or plates. Welds should be identified by a shop control number (such as a procedure qualification number), the heat of filler metal, type and batch of flux, and the welding process. Each forging and/or plate should be identified by a heat number and material specification.

RESPONSE:

Unit 1:

Vessel Beltline Material Identification - Susquehanna SES Unit 1.

A. Lower Shell Course (CBIN Dwg R-1, Rev. 4, Contract No. 683331)

1. Plates

<u>PC #</u>	<u>ID #</u>	<u>MELT #</u>	<u>SLAB #</u>
21	1	CB5083	1
21	2	C0770	2
21	3	C0814	2

2. Welds

The vertical welds and girth weld for this shell course were completed in the "field." Records were not kept on which of the electrodes, identified by heat and lot numbers, were used in the weld-up of the specific field welds of this shell assembly. It is assumed that any of the SMAW electrodes - type 8018 released for field welding could have been used on any or all of the associated seams in the beltline region.

Electrodes released for the field welding of these plates are as follows:

<u>Type</u>	<u>Lot No.</u>	<u>Heat No.</u>
SMAW Electrode Type 8018	B504B27AE	401S0371
SMAW Electrode Type 8018	629616	L320A27AG
SMAW Electrode Type 8018	402K9171	K315A27AE
SMAW Electrode Type 8018	411L3071	L311A27AE
SMAW Electrode Type 8018	494K2351	L307A27AD
SMAW Electrode Type 8018	C115A27A	402C4371
SMAW Electrode Type 8018	J417B27A	412P3611

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B. Lower Intermediate Shell Course

1. Plates

<u>PC #</u>	<u>ID #</u>	<u>MELT #</u>	<u>SLAB #</u>
22	1	C0803	1
22	2	C0776	1
22	3	C2433	1

2. Welds

The vertical welds and the girth weld for this shell course were completed in the "field." Records were not kept on which of the electrodes, identified by heat and lot numbers, were used in the weld-up of the specific field welds of the shell assembly. It is assumed that any of the SMAW electrodes (type 8018) released for field welding could have been used on any or all of the associated seams in the beltline region.

Electrodes released for the field welding of these plates are as follows:

<u>Type</u>	<u>Lot No.</u>	<u>Heat No.</u>
SMAW Electrode Type 8018	B504B27AE	401S0371
SMAW Electrode Type 8018	629616	L320A27AG
SMAW Electrode Type 8018	402K9171	K315A27AE
SMAW Electrode Type 8018	411L3071	L311A27AF
SMAW Electrode Type 8018	494K2351	L307A27AD
SMAW Electrode Type 8018	C115A27A	402C4371
SMAW Electrode Type 8018	J417B27AF	412P3611

Unit 2:

Vessel Beltline Material Identification = Susquehanna SES Unit 2

A. Lower Shell Course

1. Plates

<u>PC ID#</u>	<u>MELT #</u>	<u>SLAB #</u>
21-1	6C956	1-1
21-2	6C980	1-1
21-3	6C1053	1-1

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2. Welds

The vertical welds and girth weld for this shell course were completed in the "field." Records were not kept on which of the electrodes, identified by heat and lot numbers, were used in the weld-up of the specific field welds of this shell assembly. It is assumed that any of the SMAW electrodes - type 8018 released for field welding could have been used on any or all of the associated seams in the beltline region.

Electrodes released for the field welding of these plates are as follows:

<u>Type</u>	<u>Lot No.</u>	<u>Heat No.</u>
SMAW Electrode Type 8018	B504B27AE	401S0371
SMAW Electrode Type 8018	629616	L320A27AG
SMAW Electrode Type 8018	402K9171	K315A27AE
SMAW Electrode Type 8018	411L3071	L311A27AF
SMAW Electrode Type 8018	494K2351	L307A27AD
SMAW Electrode Type 8018	C115A27A	402C4371
SMAW Electrode Type 8018	J417B27A	412P3611
SMAW Electrode Type 8018	C109A27A	09M057
SMAW Electrode Type 8018	E204A27A	624263
SMAW Electrode Type 8018	F414B27AF	659N315

B. Lower Intermediate Shell Course

1. Plates

<u>PC ID#</u>	<u>MELT #</u>	<u>SLAB #</u>
22-1	C2421	3
22-2	C2929	1
22-3	C2433	2

2. Welds

The vertical welds and girth weld for this shell course were completed in the "field." Records were not kept on which of the electrodes, identified by heat and lot numbers, were used in the weld-up of the specific field welds of this shell assembly. It is assumed that any of the SMAW electrodes (type 8018) released for field welding could have been used on any or all of the associated seams in the beltline region.

Electrodes released for the field welding of these plates are as follows:

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<u>Type</u>	<u>Lot No.</u>	<u>Heat No.</u>
SMAW Electrode Type 8018	B504B27AE	401S0371
SMAW Electrode Type 8018	629616	L320A27AG
SMAW Electrode Type 8018	402K9171	K315A27AE
SMAW Electrode Type 8018	411L3071	L311A27AF
SMAW Electrode Type 8018	494K2351	L307A27AD
SMAW Electrode Type 8018	C115A27A	402C4371
SMAW Electrode Type 8018	J417B27AF	412P3611
SMAW Electrode Type 8018	C109A27A	09M057
SMAW Electrode Type 8018	E204A27A	624263
SMAW Electrode Type 8018	F414B27AF	659N315

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Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 REACTOR VESSEL
FIGURE 121.1-1

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Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 REACTOR VESSEL
FIGURE 121.1-2

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Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 REACTOR VESSEL
FIGURE 121.1-1

QUESTION 121.2

Supply the following information for each of the ferritic materials of the pressure retaining components in the reactor coolant pressure boundary of the Susquehanna plant:

- (1) The unirradiated mechanical properties as required by the testing programs in Section III of the ASME Code and Appendix G of 10 CFR Part 50 (test results to be presented should include Charpy V-notch, dropweight, lateral expansion, tensile, upper shelf energy, T_{NDT} and RT_{NDT}). If any of these properties have not been determined by a test method required by Appendix G of 10 CFR Part 50, state the actual test procedure used and/or the method used to estimate the test result together with a complete technical justification of the procedure used.
- (2) Identify the material(s) in the reactor coolant pressure boundary that will limit the pressure-temperature operating curves at the beginning-of-life.

For each reactor vessel beltline weld, plate or forging provide the following information:

- (3) The chemical composition (particularly the Cu, P and S content) and the maximum end-of-life fluence.
- (4) The relationship used to predict the shift in RT_{NDT} and percent decrease in upper shelf energy as a function of neutron fluence.
- (5) Identify the material(s) in the reactor coolant pressure boundary that will limit the pressure-temperature operating curves and the end-of-life.

RESPONSE:

- (1) The Susquehanna SES Unit No. 1 reactor pressure vessel was ordered prior to the issuance of Appendix G 10 CFR Part 50. The ferritic material for the pressure boundary was qualified by dropweight testing for the shell plate material and both dropweight and Charpy V-notch testing for the weld material. The test results, along with the specific requirements prevailing at the time of vessel ordering are summarized in the tables which follow.

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Impact Properties of SA533, Grade B, Class 1 Plate Material in Beltline Region		
Location	Plates	Drop Wt. NDTT (°F)
Lower Shell Course	21-1	-10°F
	21-2	-30°F
	21-3	-30°F
Lower Intermediate Shell Course	22-1	-10°F
	22-2	-10°F
	22-3	-50°F

Impact Properties of Weld Materials Employed In the Beltline Region				
Weld Material Identification	Charpy "V" (Ft/lb)	Test Temp. °F	Required	Drop Wt. NDTT °F
Lot #B504B27AE Ht #401SO371	57, 58, 62	-20	30 ft/lbs at 10°F	-80
Lot #629616 Ht #L320A27AG	51, 52	+10	"	-70
Lot #402K9171 Ht #K315A27AE	58, 58	+10	"	-70
Lot #411L3071 Ht #L311A27AF	51, 67	+10	"	-70
Lot #494K2351 Ht #L307A27AD	87, 96	+10	"	-80
Lot #C115A27A Ht #402C4371	82, 84, 92	+10	"	N/R
Lot #J417B27AF Ht #412P3611	52, 65, 69	-20	"	-80
Note: N/R - Not Reported				

Unirradiated fracture toughness properties (T_{NDT} , RT_{NDT} and upper shelf fracture energy) as required by Appendix G, 10 CFR Part 50, identifying the limiting material in the reactor vessel beltline region.

The Susquehanna SES Unit No. 2 reactor pressure vessel was ordered prior to the issuance of Appendix G, 10 CFR Part 50. The ferritic material for the pressure boundary was qualified by dropweight testing for the shell plate material and both

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dropweight and Charpy V-notch testing for the weld material. The test results, along with the specific requirements prevailing at the time of vessel ordering are summarized in the tables which follow.

Impact Properties of SA533, Grade B, Class 1 Plate Material in Beltline Region				
Location	Plates	Drop Wt. NDTT (°F)		
Lower Shell Ring	21-1	-20°F		
	21-2	-20°F		
	21-3	+ 10°F		
Lower Intermediate Shell Ring	22-1	-10°F		
	22-2	-20°F		
	22-3	-30°F		
Impact Properties of Weld Materials Employed In the Beltline Region				
Weld Material Identification	Charpy "V" (Ft/lb)	Test Temp. °F	Required	Drop Wt. NDTT °F
Lot #B504B27AE Ht #401SO371	57, 58, 62	-20	30 ft/lbs at 10°F	-80
Lot #629616 Ht #L320A27AG	51, 52	+ 10	"	-70
Lot #402K9171 Ht #K315A27AE	58, 58	+ 10	"	-70
Lot #411L3071 Ht #L311A27AF	51, 67	+ 10	"	-70
Lot #494K2351 Ht #L307A27AD	87, 96	+ 10	"	-80
Lot #C115A27A Ht #402C4371	82, 84, 92	+ 10	"	N/R
Lot #J417B27AF Ht #412P3611	52, 65, 69	-20	"	-80
Lot #C109A27A Ht #09MO57	43, 43, 44	+ 10	"	N/R
Lot #E204A27A Ht #624263	26, 38, 42, 50, 76	-20	"	-70
Lot #F414B27AF Ht #659N315	74, 76, 77	-10	"	-80
Note : N/R - Not Reported				

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(2) Intermediate shell plate 23-3, Heat Slab No. C1232-2 is initially limiting for Susquehanna Unit 1. Core beltline plates 22-1, 22-2, & 22-3, Heat & Slab Nos. 6C956-1-1, 6C980-1-1, & 6C1053-1-1 respectively are initially limiting for Susquehanna Unit 2 with respect to pressure-temperature operating curves at the beginning-of-life. The feedwater nozzles will also be limiting at lower pressures as indicated on Figures 5.3-4 and 5.3-5 for Units 1 & 2.

(3) Vessel Plate Material Susquehanna SES Unit 1

	C	Mn	P	S	Cu	Si	Ni	Mo
(Values are shown in percent)								
Lower Shell								
PC 21-1, Melt #B5083, Slab #1	.21	1.27	.010	.019	.14	.25	.48	.51
PC 21-2, Melt #C0770, Slab #2	.22	1.23	.008	.016	.14	.19	.50	.49
PC 21-3, Melt #C0814, Slab #2	.20	1.36	.011	.016	.13	.26	.51	.51
Lower Intermediate Shell								
PC 22-1, Melt #C0803, Slab #1	.21	1.30	.009	.019	.09	.24	.53	.52
PC 22-2, Melt #C0776, Slab #1	.22	1.34	.010	.010	.12	.27	.48	.48
PC 22-3, Melt #C2433, Slab #1	.18	1.30	.009	.015	.10	.23	.63	.57

	C	Mn	P	S	Cu	Si	Ni	Mo	Cr	Vn
(Values are shown in percent)										
Weld Material - Unit 1										
Type SMAW										
Electrode 8018										
Lot # and Heat #										
Lot #B504B27AE Ht #401S0371	.05	1.18	.013	.012	.03	.37	1.0 4	.56	.03	.02
Lot #629616 Ht #L320A27AG	.05	1.17	.015	.018	.04	.44	.99	.55	.05	.02

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	C	Mn	P	S	Cu	Si	Ni	Mo	Cr	Vn
Lot #402K9171 Ht #K315A27AE	.06	1.15	.015	.016	.03	.36	.98	.53	.05	.02
Lot #411L3071 Ht #L311A27AF	.05	1.20	.016	.019	.03	.46	.93	.50	.04	.02
Lot #494K2351 Ht #L307A27AD	.05	1.18	.015	.017	.04	.37	1.1 0	.57	.04	.02
Lot #C115A27A Ht #402C4371	.033	1.22	.009	.014	.02	.49	.92	.57	N/R	N/R
Lot #J417B27AF Ht #412P3611	.07	1.10	.016	.019	.03	.36	.93	.47	.03	.02
Note: N/R - Not Reported										

Vessel Plate Material - Unit 2

	C	Mn	P	S	Cu	Si	Ni	Mo
(Values are shown in percent)								
Lower Shell								
PC 21-1, Melt #6C956, Slab #1-1	.18	1.43	.012	.006	.11	.22	.55	.52
PC 21-2, Melt #6C980, Slab #1-1	.19	1.35	.011	.006	.10	.22	.56	.51
PC 21-3, Melt #6C1053, Slab #1-1	.18	1.37	.012	.010	.10	.30	.58	.50
Lower Intermediate Shell								
PC 22-1, Melt #C2421, Slab #3	.19	1.22	.007	.011	.13	.25	.68	.55
PC 22-2, Melt #C2929, Slab #1	.20	1.27	.006	.015	.13	.22	.64	.56
PC 22-3, Melt #C2433, Slab #2	.18	1.30	.009	.015	.10	.23	.63	.57

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	C	Mn	P	S	Cu	Si	Ni	Mo	Cr	Vn
(Values are shown in percent)										
Weld Material - Unit 2										
Type SMAW Electrode 8018 Lot # and Heat #										
Lot #B504B27AE Ht #401S0371	.05	1.18	.013	.012	.03	.37	1.04	.56	.03	.02
Lot #629616 Ht #L320A27AG	.05	1.17	.015	.018	.04	.44	.99	.55	.05	.02
Lot #402K9171 Ht #K315A27AE	.06	1.15	.015	.016	.03	.36	.98	.53	.05	.02
Lot #411L3071 Ht #L311A27AF	.05	1.20	.016	.019	.03	.46	.93	.50	.04	.02
Lot #494K2351 Ht #L307A27AD	.05	1.18	.015	.017	.04	.37	1.10	.57	.04	.02
Lot #C115A27A Ht #402C4371	.033	1.22	.009	.014	.02	.49	.92	.57	N/R	N/R
Lot #J417B27AF Ht #412P3611	.07	1.10	.016	.019	.03	.36	.93	.47	.03	.02
Lot #C109A27A Ht #09M057	.063	1.18	.009	.021	.03	.47	.89	.53	N/R	N/R
Lot #E204A427A Ht #624263	.051	1.08	.010	.023	.06	.38	.89	.50	N/R	N/R
Lot #F414B27AF Ht #659N315	.05	1.14	.015	.013	.04	.35	1.00	.49	.05	N/R
Note: N/R - Not Reported										

The maximum end-of-life fluence is 1.4×10^{18} n/cm² at 1/4T depth of the vessel beltline material for Units 1 and 2.

- (4) This information is contained in Subsection 5.3.1.4.1.7.
- (5) Core beltline plates 22-2 & 22-3, Heat. & Slab Nos. C0776-1 and C2433-1 are limiting at end-of-life for Susquehanna Unit 1. For Susquehanna SES Unit 2, the core beltline plates which were limiting at the beginning-of-life (see response to 121.2 (2)) are also limiting at the end-of-life. The feedwater nozzles are also limiting at lower pressures for Units 1 & 2 as shown on Figures 5.3-4 and 5.3-5.

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QUESTION 121.3

The FSAR states that compliance with Appendix G of 10 CFR Part 50 and Appendix G of Section III of the ASME Code was not possible for components purchased prior to the issuance of the Summer 1972 Addenda of the ASME Code without replacement of large amounts of materials, reworking of fabricated components and the revision of most all of the design analyses for the components.

The details of the method of compliance as stated in the FSAR are insufficient to identify the areas of noncompliance with Appendix G of 10 CFR Part 50. The applicant should state specifically those sections in which strict compliance with the regulations was not achieved.

The technical bases for the proposed alternate methods used to satisfy the requirements of those sections of Appendix G of 10 CFR Part 50 where strict compliance was not achieved should be presented. These bases should include technical justification to demonstrate that the proposed alternatives provide acceptable safety margins relative to the Appendix G requirements.

RESPONSE:

See revised Subsection 5.3.1.5.

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QUESTION 121.4

Paragraph II.C.2 of Appendix H. 10 CFR Part 50 states: "Surveillance capsules containing the surveillance specimens shall be located near but not attached to the inside vessel wall in the beltline region..." FSAR Section 5.3 indicates that the capsule holder brackets were welded to the reactor pressure vessel inner wall. Present sufficient design and fabrication detail to demonstrate that the capsule attachments were designed and constructed in accordance with accepted standards, such as the ASME Code Section III rules for attachments to vessels.

RESPONSE:

The surveillance brackets are welded to the clad material which surfaces the pressure vessel and are, therefore, not attached to the pressure boundary directly. As attached, the brackets do not have to comply with specifications of the ASME Pressure Vessel Code. See Figure 121.4-1.

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Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
SURVEILLANCE SPECIMEN BRACKETS
FIGURE 121.4-1

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Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
SURVEILLANCE SPECIMEN BRACKETS
FIGURE 121.4-1

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QUESTION 121.5

In FSAR Sections 1.6 and 5.3 General Electric Report NEDO-20631, "Mechanical Property Surveillance of Reactor Pressure Vessels for General Electric BWR-6 Plants," dated March 1975, is referenced.

This report has not been submitted for review. Therefore the information referenced by the GE report must be provided in the FSAR so that an evaluation of compliance with Appendix H of 10 CFR Part 50 can be made for this plant.

RESPONSE:

The report NEDO-20631 has been withdrawn as a reference for Susquehanna SES, and was replaced by report NEDO-21708, "Radiation Effects in Boiling Water Reactor Pressure Vessel Steels," dated October 1977. The NRC staff has been provided NEDO-21708 for review. NEDO-21708 addresses the requirements of Appendix H to 10CFR Part 50 and supports the current application of Regulatory Guide 1.99.

QUESTION 121.6

To provide assurance that high energy turbine missiles will not be produced at operating speed or design overspeed, provide documentation (including the results of material property testing) to show the degree of conformance of the turbine-generator with the guidelines in SRP 10.2.3, "Turbine Disk Integrity," Paragraph II, "Acceptance Criteria."

RESPONSE:

Turbine disk integrity is discussed in Section 10.2.3. Results from tests on the disks are given below.

High Pressure Rotor

The high pressure turbine rotors were forged from vacuum degassed NiCrMoV steel. Final rotor properties were verified by tests after a suitable quench and temper. The measured rotor properties, together with a 100% volumetric (ultrasonic) evaluation, form the basis for rotor material acceptance.

Material properties of high pressure rotors on turbines 170 X 592 and 170 X 593 have been examined and the rotors were found acceptable for their intended application. In particular, the high pressure rotors on above units have bore measured room temperature Charpy energies in excess of 50 foot pounds and bore measured 50% fracture appearance transition temperatures (FATT) below 50°F.

The ratio of fracture toughness K_{Ic} to maximum tangential stress for above rotors meets or exceeds $2\sqrt{\text{in.}}$. K_{Ic} for the above ratio was calculated from acceptance data by methods which are more conservative than methods described by J. A. Begley and W. A. Longsdon in Westinghouse Scientific Paper 71-1E7-MSLRF-P1. Rotor bore tangential stresses for the above ratio consist of the sum of centrifugal stress at 115% of rated speed, thermal stress and shrink stress.

The tangential stresses, as calculated above, were compared with measured rotor yield strength, as adjusted to rotor operating temperature, and it was found that stresses are lower than .75 of yield strength, as required by design criteria.

LOW PRESSURE ROTOR

The low pressure turbine wheels were forged from vacuum degassed NiCrMoV steel. Final wheel properties were verified by tests after a suitable quench and temper. The measured wheel properties, together with a 100% volumetric (ultrasonic) evaluation, form the basis for wheel material acceptance.

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Material properties of low pressure wheels on turbines 170 X 592 and 170 X 593 have been examined and the wheels were found acceptable for their intended application. In particular, all wheels on above units have surface measured room temperature Charpy energies in excess of 60 foot pounds and surface measured 50% fracture appearance transition temperatures (FATT) below 0°F.

The ratio of fracture toughness K_{IC} to maximum tangential stress for all above wheels meets or exceeds $2\sqrt{\text{in.}}$. K_{IC} for the above ratio was calculated from acceptance data by methods which are more conservative than methods described by J. A. Begley and W. A. Longsdon in Westinghouse Scientific Paper 71-1E7-MSLRF-P1. Wheel bore tangential stresses for the above ratio consist of the sum of centrifugal stress at 115%* of rated speed, thermal stress and shrink stress.

The tangential stresses, as calculated above, were compared with measured wheel yield strength, as adjusted to wheel operating temperature, and it was found that stresses are lower than .75 of yield strength, as required by design criteria.

*Note: The highest speed anticipated from loss of load with normal operation of the control system is 110% of rated speed.

QUESTION 121.7

(Reference Hatch Nuclear Plant Unit No. 2 response to items 121.15 and 121.18). The following information is necessary to demonstrate that the Susquehanna Unit Nos. 1 and 2 feedwater inlet nozzle thermal sleeve/sparger design has been evaluated with due consideration to nozzle cracking due to thermal cycling and that a program of schedule augmented inservice inspections, with a sensitive method that will assure detection, has been developed:

- (1) The technical basis to assure the structural integrity of both the feedwater inlet nozzle and the sparger.
- (2) An evaluation of the feasibility of automated ultrasonic testing (UT) fixtures installed on all feedwater inlet nozzles with particular attention on examination of the nozzle bore region.
- (3) An evaluation of the feasibility of performing the internal surface examination by magnetic particle methods.

Your response should contain:

- a) a description of the nozzle and sparger design including dimensions, materials of construction and weld locations.
- b) description of analyses and test data, referencing if necessary data previously submitted to the staff where directly appropriate for this plant.
- c) projected crack growth rates, stress levels and usage factors for both the nozzle and the sparger should be described in detail.
- d) any plant modifications that are planned to reduce the feedwater to reactor water temperature differential during low power operation.
- e) any instrumentation that will be installed in the reactor to verify the conclusions of the design analysis should be identified.

Several ultrasonic testing concepts and procedures have been used to examine the feedwater inlet nozzle regions in operating plants. Define the specific ultrasonic testing procedure that will be used for Susquehanna Unit Nos. 1 and 2. Discuss the influence of local grindouts on crack detection on your ultrasonic testing method.

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In addition, provide a description of the augmented inservice inspection (ISI) program to be implemented including scheduled surface examination, ultrasonic testing and verification of the leak tight integrity of the thermal sleeve to safe end joint on all nozzles. The essential elements of an acceptable program are given below:

Augmented Inservice Inspection Plan

- (1) Preservice Examination - Preservice UT examination should include all nozzle inner radius, bore, and safe end regions. In addition, a preservice surface examination should be performed on the accessible regions of all nozzle inner radii.
- (2) Inservice Examination - To confirm the continuing structural integrity, the following examinations should be performed:
 - (a) At each scheduled refueling outage, an external UT examination of all feedwater nozzle inner radii, bore and safe end regions.
 - (b) After 50 startup/shutdown cycles but prior to 70 cycles, a surface examination of the accessible regions of all nozzle inner radii. The definition of startup/shutdown cycles and the procedure for liquid penetrant examination is contained in report NUREG-0312, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking."
 - (c) Subsequent surface examinations of the accessible region of all nozzle inner radii should be performed at the earlier of (i) every other scheduled refueling outage, or (ii) at the scheduled refueling outage after 20 but prior to 40 startup/shutdown cycles after the last surface examination.
- (3) Thermal Sleeve to Safe End Joint - An examination method, such as a leak test should be developed to confirm the continuing structural and leak-tight integrity of the thermal sleeve to safe end joint.

Acceptance Standards

- (1) All UT indications elevated to be cracks should be verified by appropriate surface examination and removed by local grinding.

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- (2) All surface indications evaluated to be service induced cracks should be removed by local grinding.
- (3) The UT inspection personnel should be required to demonstrate supplemental qualifications by either (i) past successful experience in locating and identifying cracks in BWR feedwater inlet nozzles or (ii) performing a qualification test on a full size unclad nozzle mockup.

Recording and Reporting Standards

Requirements for recording of indications and reporting of inspection results are contained in report NUREG-0312.

RESPONSE:

Note: All responses are presented in the order they are found in the Question.

- (1) Discuss the technical basis to assure the structural integrity of both the feedwater inlet nozzle and the sparger, including:

- (a) A description of the nozzle and sparger design including dimensions, materials of construction and weld locations.

Description of feed water inlet nozzle.

<u>Part</u>	<u>Material</u>
Nozzle Forging	SA508CL II
Safe End	SA508CL I

Dimensions, location of weld and other details are provided in the following drawings:

79E902, Sheet 1	Susquehanna SES Units 1 and 2 - Figure(s) 121.1- 1a, b
137C5543 PT No. 4	Susquehanna SES Units 1 and 2 - Figure(s) 121.7-2

Description of sparger material, basic dimensions and weld locations are presented in the drawings:

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Susquehanna Unit 1 - Figure(s) 121.7-3a,b,c
Susquehanna Unit 2 - Figure(s) 121.7-4a,b,c

- (b) A description of analysis and test data referencing, if necessary, data previously submitted to the staff where directly appropriate for this.

The information for part b has been provided in the response and reference document cited for Question 112.7.

- (c) Projected crack growth rates, stress levels and usage factors for both the nozzle and the sparger should be described in detail.

The information for part c has been provided in the response and reference document cited for Question 112.7

- (d) Any plant modifications that are planned to reduce the feedwater to reactor water temperature differential during low power operation.

Susquehanna SES is currently evaluating specific modifications to fluid systems and operating procedures as discussed in General Electric Document NEDE-21821A.

- (e) Any instrumentation that will be installed in the reactor to verify the conclusions of the design analysis should be identified.

Due to demonstrated benefits from nozzle cracking fixes, no instrumentation has been installed for design verification.

- (2) Evaluate the feasibility of automated ultrasonic testing (UT) fixtures installed on all feedwater inlet nozzles with particular attention on examination of the nozzle bore region.

Currently, automated ultrasonic examination of the feedwater nozzle inner radii, safe-end, and bore region is not feasible. Preservice examinations on the nozzles will be performed utilizing a General Electric developed ultrasonic testing (UT) procedure. This procedure divides the nozzle inner surface into three regions, each

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of which is examined separately by a single angle beam shear wave technique. Current state of the art technology does not allow for automation of this technique due to the complexity of the technique and various scanning patterns involved; also, computer assisted signal discrimination equipment is not yet available for field usage.

Scanning solely of the nozzle bore region may be accomplished from the cylindrical section of the nozzle forging utilizing a temporary, removable track scanner. In terms of radiation exposure, (examination/set up time) and examination coverage, automation of only this portion of the examination is not beneficial and is not being considered for preservice activities.

- (3) Evaluate the feasibility of performing the internal surface examination by magnetic particle methods.

Magnetic prod inspection methods are not acceptable in this area. Due to limited access in which to perform the examination, maintenance of proper prod contact with the nozzle surface is difficult, possibly resulting in arc-strikes below the electrodes. These surface defects are localized heat affected zones of higher hardness than the surrounding metal. Should the arc strike be accompanied by localized cracking, then surface grinding would be necessary to restore the nozzle to its original surface condition.

Handheld magnetic yokes will not readily fit in the area between the sparger body and the nozzle radius while maintaining proper contact with the nozzle surface and still allow adequate access to perform the examination. Based on the above, magnetic particle examination methods are not considered feasible inside the reactor vessel with the present sparger configuration.

- (4) Define the specific ultrasonic testing procedure that will be used for Susquehanna Unit Nos. 1 and 2. Discuss the influence of local grindouts on crack detection on your ultrasonic testing method.

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Susquehanna SES will perform preservice examinations of all feedwater nozzle inner radii, safe end, and bore regions to provide a baseline for routine augmented inservice inspections outlined later in this response. Feedwater nozzle safe end examinations will be performed in accordance with ASME Section XI requirements to General Electric Company Procedure #ISE-QAI-322 "Ultrasonic Examinations of Similar and Dissimilar Metal Welds." The inner radii and bore regions will be performed in accordance with General Electric Company Generic Procedures listed below:

TP-508-0173	Rev. D	"Procedure for Nozzle Inner Radii Zone I Ultrasonic Examination"
TP-508-0174	Rev. D	"Procedure for Nozzle Inner Radius Zone 2 Ultrasonic Examination"
E50YP14	Rev. 0	"Procedure for Nozzle Inner Radius Zone 3 Ultrasonic Examination"

Susquehanna SES site specific procedures technically in accordance with these generic procedures are being generated.

For examination purposes, the inner surface has been divided into three regions' each of which is examined separately by a single angle beam shear wave technique. Examination of the nozzle inner radius will be performed by pulse-echo ultrasonic techniques from the exterior of the reactor pressure vessel by contacting the vessel plate surface. The nozzle inner bore region shall be examined from the outer blend radius and the cylindrical portion of the nozzle - the former requiring a special transducer wedge that complements the curvature of the contact area radius.

Should local grindouts be made in the examination surface creating a depression with definable sides, depth, and length, the ultrasonic techniques being used would obtain reflections from these cavities. Such reflections will be minimized by blending the grind cavity into the surrounding base metal in accordance with ASME

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requirements. This should result in improved detection sensitivity to actual cracking in the grindout area by eliminating spurious geometric indications from the grindout.

- (5) Provide a description of the augmented inservice inspection (ISI) program to be implemented including scheduled surface examination, ultrasonic testing, and verification of the leak tight integrity of the thermal sleeve to safe end joint on all nozzles.

AUGMENTED INSERVICE INSPECTION PROGRAM

Susquehanna SES will implement the reactor feedwater (RFW) nozzle inspection program described below. Justification for any deviations from recommended inspections in NUREG 0619 are presented following the response.

PRESERVICE EXAMINATION

Susquehanna SES will perform a preservice ultrasonic examination of reactor feedwater nozzle inner radii, bore, and safe end regions. All U.T. personnel and procedures will be fully qualified as required. In addition, a preservice liquid penetrant examination will be performed on accessible areas of all Unit #1 feedwater nozzle inner radius surfaces. Also, all nozzle forgings have previously been fully shop magnetic particle inspected and have met ASME Section III requirements. Full liquid penetrant examination will be performed on all Unit #2 feedwater nozzle forgings prior to installation of the spargers.

Inservice Examination

Susquehanna SES-1 will perform the following routine inservice inspections as follows:

1. Ultrasonic examination of the reactor feedwater nozzle inner radii and bore region will be performed every two (2) refueling cycles on one (1) RFW nozzle. The inspection interval begins with the first refueling cycle since the unclad nozzle and triple sleeve sparger was installed prior to plant start up. Safe end examinations will continue to be performed in accordance with ASME Section XI requirements.
2. Penetrant testing of the nozzle inner radii and bore region will be performed only as required to verify and characterize U.T. indications.

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3. Visual inspection of the sparger will continue to be performed in accordance with ASME Section XI requirements.
4. Verification of the thermal sleeve to safe end joint shall be made by performance of in-vessel physical leak testing or some alternate method such as on-line leaking monitoring. Susquehanna SES is presently pursuing the feasibility of the later.

In the event an indication is discovered by UT and found to result from service induced cracks propagating from the nozzle inner surfaces, the following action will be taken:

All accessible areas of remaining feedwater nozzles will be examined using penetrant techniques during the refueling outage in which the cracking is verified.

All surface indications determined to be service induced cracks will be removed by local grinding.

A RFW nozzle examination program for subsequent refueling outages will include the external ultrasonic examination of all feedwater nozzle inner radii, bore and safe end regions for each scheduled refueling outage for 3 consecutive outages. If no new indications are discovered, or if new indications are determined to not result from service induced cracks at the nozzle inner surfaces, the aforementioned program will be resumed. If after 3 additional outages no new indications resulting from surface induced cracks are detected, subsequent examinations will be scheduled in accordance with normal ASME Section XI requirements.

The conduct of surface examinations of accessible nozzle inner radius surfaces will continue to be used throughout plant life only to confirm or characterize new ultrasonic indications which are suspected to result from service induced cracks at the nozzle inner surfaces.

Thermal Sleeve to Safe End Joint

Susquehanna SES shall verify the integrity of the thermal sleeve-to-safe end joint by performance of in-vessel physical leakage testing or alternate methods such as on-line periodic leakage monitoring.

Recording and Reporting Standards

Susquehanna SES-1 will record crack indications and report inspection results in compliance with the requirements stated in NUREG 0619.

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JUSTIFICATION OF DEVIATION FROM RECOMMENDATIONS

Ultrasonic Examinations Frequency

Susquehanna SES will ultrasonically examine one RFW nozzle every second refueling outage. This is justified for the following reasons, which reflect a significant advance in the Susquehanna SES design and operating procedures towards the long term solution of the BWR nozzle cracking problems per NUREG 0619.

- a. Improved Design: The Susquehanna SES RFW thermal-sleeve-to-safe-end joint provides a near zero leakage design. This design essentially eliminates the primary historical initiating source of nozzle cracking in BWRs.
- b. No Nozzle Cladding: The Susquehanna SES-1 RFW nozzle surfaces are not clad. The likelihood of crack initiation in unclad nozzles is considerably reduced such that elimination of the nozzle cladding and installation of the triple sleeve sparger design may be all that is necessary to suppress cracking within the design lifetime.
- c. Proven Examination Technique: The ultrasonic examination equipment and personnel to be used in performing both baseline and inservice ultrasonic examinations will be qualified on a full scale mock-up of the nozzle, simulating the nozzle geometry and anticipated fatigue crack defects. Since the Susquehanna SES-1 reactor feedwater nozzles are unclad as stated in b) above, a more sensitive examination is possible due to lack of clad/basemetal interface.
- d. Augmented Examination Frequency: The above stated program provides RFW nozzle examination coverage at approximately one and one half times the frequency of the ASME Section XI requirements, i.e., all RFW nozzles will be examined within approximately seven years rather than within ten years.

The above factors, when combined, provide adequate assurance that the factors which have led historically to BWR RFW nozzle cracking have been virtually eliminated. Furthermore, any cracking which might occur from unanticipated sources will be discovered before propagating to a significant depth utilizing an augmented examination schedule with state-of-the-art qualified ultrasonic examination techniques.

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Surface Examinations

Susquehanna SES-1 will perform liquid penetrant examinations of the accessible internal surfaces of all RFW nozzles during the preservice inspection activities. In-service surface examinations necessitating removal of the spargers, will be performed only when indications of service induced cracking are discovered ultrasonically. This is justified as follows:

- a. Reduced probability of crack initiation and growth as stated in the justification above (Ultrasonic Examinations Frequency a thru f).
- b. Access: In order to obtain access to perform a penetrant surface examination of the RFW nozzle surfaces during a refueling outage, the vessel water level would have to be lowered below the level of the spargers and hydrolaser decontamination performed. Special shielded platforms would be required to minimize exposures.
- c. Removal of the current design sparger for routine penetrant examination may result in damage to the thermal sleeve sealing surface, resulting in increased likelihood of leakage.

Acceptance Standards

- (1) All UT indications evaluated to be cracks should be verified by appropriate surface examination and removed by local grinding.
- (2) All surface indications evaluated to be service induced cracks should be removed by local grinding.
- (3) The UT inspection personnel should be required to demonstrate supplemental qualifications by either (i) past successful experience in locating and identifying cracks in BWR feedwater inlet nozzles or (ii) performing a qualification test on a full size unclad nozzle mock-up.

Response:

- (1) Susquehanna SES will comply with this criteria as stated in 2(a) above.
- (2) Susquehanna SES will comply with this criteria as stated in 2(a) above.

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- (3) Susquehanna SES will utilize General Electric Company qualified procedures previously referenced. All personnel performing examinations at Susquehanna SES will be qualified in accordance with these procedures on a full nozzle mock-up.

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
FEEDWATER INLET NOZZLE SAFE END
FIGURE 121.7-1a

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
FEEDWATER INLET NOZZLE SAFE END
FIGURE 121.7-1b

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
FEEDWATER INLET NOZZLE SAFE END
FIGURE 121.7-2

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY SOLUTION ANNEALED AFTER ALL MACHINING & WELDING (EXCLUDING THERMAL SLEEVE & BRACKETS)
FIGURE 121.7-3a

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY NOZZLE AND THERMAL SLEEVE ARRANGEMENT FIGURE 121.7-3b

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER THERMAL SLEEVE INTERFERENCE FIT DETAILS
FIGURE 121.7-3c

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY
FIGURE 121.7-4a

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY NOZZLE AND THERMAL SLEEVE ARRANGEMENT FIGURE 121.7-4b

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER THERMAL SLEEVE INTERFERENCE FIT DETAILS
FIGURE 121.7-4c

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
FEEDWATER INLET NOZZLE SAFE END
FIGURE 121.7-1A

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
FEEDWATER INLET NOZZLE SAFE END
FIGURE 121.7-1B

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
FEEDWATER INLET NOZZLE SAFE END
FIGURE 121.7-2

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY SOLUTION ANNEALED AFTER ALL MACHINING & WELDING (EXCLUDING THERMAL SLEEVE & BRACKETS)
FIGURE 121.7-3A

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY NOZZLE AND THERMAL SLEEVE ARRANGEMENT
FIGURE 121.7-3B

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 1 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER THERMAL SLEEVE INTERFERENCE FIT DETAILS
FIGURE 121.7-3C

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY
FIGURE 121.7-4A

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER ASSEMBLY NOZZLE AND THERMAL SLEEVE ARRANGEMENT
FIGURE 121.7-4B

Security-Related Information

Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
UNIT 2 IMPROVED INTERFERENCE FIT FEEDWATER SPARGER THERMAL SLEEVE INTERFERENCE FIT DETAILS
FIGURE 121.7-4C

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QUESTION 121.8

We will require that your inspection program for Class 1, 2 and 3 components be in accordance with the revised rules in 10 CFR Part 50, Section 50.55a, paragraph (g) published in the February 12, 1976 issue of the FEDERAL REGISTER.

To evaluate your inspection program, the following minimum information is necessary for our review:

- (1) A preservice inspection plan to consist of the applicable ASME Code Edition and the exceptions to the Code requirements.
- (2) An inservice inspection plan submitted within six months of anticipated commercial operation.

The preservice inspection plan will be reviewed to support the safety evaluation report finding on compliance with preservice and inservice inspection requirements. The basis for the determination will be compliance with:

- (1) The Edition of Section XI of the ASME Code stated in your PSAR or later Editions of Section XI referenced in the FEDERAL REGISTER that you may elect to apply.
- (2) All augmented examinations established by the Commission when added assurance of structural reliability was deemed necessary. Examples of augmented examination requirements can be found in NRC positions on (a) high energy fluid systems in SRP Section 3.2, (b) turbine disk integrity in SRP Section 10.2.3, and (c) feedwater inlet nozzle inner radii.

Your response should define the applicable Section XI Edition(s) and subsections. If any examination requirements of the Edition of Section XI in your PSAR can not be met, a relief request including complete technical justification to support your conclusion must be provided.

The inservice inspection plan should be submitted for review within six months of anticipated commercial operation to demonstrate compliance with 10 CFR Part 50, Section 50.55a, paragraph (g). This plan will be evaluated in a safety evaluation report supplement. The objective is to incorporate into the inservice inspection program Section XI requirements in effect six months prior to commercial operation and any augmented examination requirements established by the

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Commission. Your response should define all examination requirements that you determine are not practical within the limitations of design, geometry, and materials of construction of the components.

Attached are detailed guidelines for the preparation and content of the inspection programs and relief requests to be submitted for staff review.

RESPONSE:

The inspection program for Class 1, 2 and 3 components has been provided (PLA-619, N. W. Curtis to B. J. Youngblood dated 1/27/81).

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QUESTION 121.9

Paragraph IV.A.2.a, Appendix G, 10 CFR Part 50, requires that a reference temperature, RT_{NDT} , be determined for each ferritic material of the reactor vessel and that this reference temperature be used as a basis for providing adequate margins of safety for reactor operation. Previously-submitted data are inadequate to define an RT_{NDT} for the reactor vessel ferritic materials; therefore, supply the following additional information:

- (a) If both CVN and dropweight tests were conducted for vessel beltline shell plates as stated in FSAR Subsection 5.3.1.5.1.2, supply the CVN test results in addition to the previously submitted dropweight test results (per response to Question 121.2). Calculate an RT_{NDT} for every shell plate, and explain in detail the method used to establish each RT_{NDT} value.
- (b) If only dropweight tests were conducted for vessel beltline shell plates as stated in the response to Question 121.2, explain in detail the method(s) used to establish an RT_{NDT} value for each vessel plate.
- (c) Supply both CVN and dropweight test results for every other ferritic vessel plate not addressed by items (a) and (b). This should include the upper shell and both lower and upper vessel heads. Calculate an RT_{NDT} value for each plate and explain in detail the method used to establish the RT_{NDT} values.
- (d) Identify every ferritic weld seam in the reactor vessel by weld wire, heat number, flux type, lot of flux and welding process. This should include any ferritic weld in the beltline region, upper shell, and lower and upper vessel heads. Submit CVN and dropweight test results in addition to the previously submitted beltline weld data. Calculate an RT_{NDT} for every ferritic weld seam, and explain in detail the method(s) used to establish each RT_{NDT} value.
- (e) Submit the correlation data used to establish an RT_{NDT} value of no less than -50°F when dropweight results are not available for weld material. This data should include weld wire and flux types, welding process, and heat treatment for each correlation weldment specimen. Explain in detail the analysis used to establish the -50°F value.

RESPONSE:

This information was provided by letter dated 5/20/81 (PLA-796, Curtis to Youngblood).

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QUESTION 121.10

Paragraph IV.A.3, Appendix G, 10 CFR Part 50, requires that materials for piping, pumps and valves meet the impact energy requirements of Paragraph NB-2332 of the ASME Code. Materials for bolting must meet the requirements of Paragraph NB-2333 of the ASME Code. To demonstrate compliance with Paragraph IV.A.3, supply all impact test data for the ferritic materials of these components. Identify each material by its ASME specification, heat or lot number, and dimensions when applicable. If any of the above data are not available, submit data from the literature and/or further tests, and analyses to demonstrate compliance with Appendix G.

RESPONSE:

This information was provided by letter dated 5/20/81 (PLA-796, Curtis to Youngblood).

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QUESTION 121.11

Paragraph IV.B, Appendix G, 10 CFR Part 50, requires the reactor vessel beltline materials have a minimum upper shelf energy of 75 ft-lbs in the transverse direction. Insufficient data have been supplied to demonstrate that all the beltline plates and welds meet this upper shelf requirement. Submit the following information to demonstrate compliance with Paragraph IV.B:

- (a) Impact energy data for all beltline plates (21-1, -2, and -3 of Unit No. 1 and 21-1, -2, and -3 of Unit No. 2) that will demonstrate that the plates in the vessel beltline will have 75 ft-lbs (in the transverse direction) for unirradiated material or that the upper shelf energy will not fall below 50 ft-lbs at the design fluency level. If these data are not available, submit data from the literature and/or further tests on similar base metal, and analyses used to define the upper shelf energy level.
- (b) Impact energy data for the following beltline weld materials that will demonstrate that the weld seams in the vessel beltline will have 75 ft-lbs for unirradiated material or that the upper shelf energy will not fall below 50 ft-lbs at the design fluency level. These welds, identified by lot number/heat number are: 629616/L320A27AG, 411L3071/L311A27AF, J417B27AF/412P3611, C109A27A/09M057 and E204A27A/624263. If these data are not available, submit data from the literature and/or further tests of weld material of the same weld wire and flux type, and analyses used to define the upper shelf energy level.

RESPONSE:

This information was provided by letter dated 5/20/81 (PLA-796, Curtis to Youngblood).

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QUESTION 121.12

The materials surveillance program uses three specimen capsules, that should contain reactor vessel steel specimens of the limiting base material, weld metal and heat-affected zone material. To help demonstrate compliance with Appendix H, 10 CFR Part 50, provide a table that includes the following information for each specimen:

- (1) Actual surveillance material;
- (2) Origin of each surveillance specimen (base metal: heat number, plate identification number; weld metal: weld wire, heat of filler material, production welding conditions, and plate material used to make weld specimens);
- (3) Test specimen and type;
- (4) Fabrication history of each test specimen;
- (5) Chemical composition of each test specimen.

Provide the location, lead factor and withdrawal time for each specimen capsule calculated with respect to the vessel inner wall. The above information should be submitted in tabular form as illustrated in Enclosure 1.

RESPONSE:

This information was provided by letter dated 5/20/81 (PLA-796, Curtis to Youngblood).

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QUESTION 121.13

Paragraph III.A of Appendix G, requires that ferritic materials of the reactor coolant pressure boundary be impact tested by means of Charpy V-notch and dropweight (when required by the ASME Code) tests. Supply the impact test data for the vessel nozzles, flanges and shell regions near geometric discontinuities to demonstrate compliance with Paragraph III.A. Each component material must be identified by heat number and location within the reactor coolant pressure boundary. Impact test data should include test temperatures, CVN energy, and/or mils lateral expansion.

RESPONSE:

This information was provided by letter dated 5/20/81 (PLA-796, Curtis to Youngblood).

QUESTION 121.14

The applicant has not submitted a Pre-service Inspection (PSI) Program for review. To evaluate compliance with 10 CFR Part 50.55a(g)(2), we will require a complete response to Question 121.8 concerning the PSI. All pre-service examination requirements defined in Section XI of the ASME Code that have been determined to be impractical must be identified and a supporting technical justification must be provided. The PSI program should include at least the following information:

- (a) For ASME Code Class 1 and 2 components, provide a table similar to IWB-2600 and IWC-2600 confirming that either the entire Section XI pre-service examination was performed on the component or relief is requested with a technical justification supporting your conclusion.
- (b) Where relief is requested for pressure retaining welds in the reactor vessel, identify the specific welds that did not receive a 100% pre-service ultrasonic examination and estimate the extent of the examination that was performed.
- (c) Certain ASME Code Class 1 and 2 vessel and piping system welds, that are 1/2 inch or less in nominal wall thickness, are subject to a Section XI pre-service volumetric examination. Confirm that a 100% volumetric examination was performed on these thin-wall weldments.
- (d) Where relief is requested for piping system welds (Examination Category B-J, C-F and C-G), provide a list of the specific welds that did not receive a complete Section XI pre-service examination including a drawing or isometric identification number, system, weld number, and physical configuration, e.g., pipe to nozzle weld, etc. Estimate the extent of the pre-service examination that was performed, the primary reason a complete examination is impractical, alternative and/or supplemental examination performed and that method of fabrication examination.
- (e) Describe the extent and method of pre-service volumetric examination of Class 1 integrally-welded supports in Examination Categories B-H and B-K-1.
- (f) Describe the extent and method of pre-service examination of Class 2 pressure-retaining bolting 2 inches in diameter or less.

RESPONSE:

The response to this question was submitted by letters dated 5/19/81 (PLA-813, Curtis to Youngblood), 6/16/81 (PLA-846, Curtis to Schwencer), and 4/23/82 (PLA-1053, Curtis to Schwencer).

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QUESTION 121.15

Paragraph 50.55a(b)(2)(iv) requires that ASME Code Class 2 piping welds in the Residual Heat Removal Systems, Emergency Core Cooling Systems and Containment Heat Removal Systems shall be examined. List the lines in these systems that were exempted from preservice volumetric and/or surface examination based on Paragraphs IWB-1220 and IWC-1220 of Section XI and provide a technical justification. The control of water chemistry to minimize stress corrosion described in Paragraph IWC-1220(c) of Section XI is not an acceptable basis for exempting ECCS components from examination because practical evaluation, review and acceptance standards cannot be defined. To satisfy the inspection requirements of General Design Criteria 36, 39, 42, and 45, the in-service inspection program must include periodic volumetric and/or surface examination of a representative sample of welds in the RHR, ECCS and Containment Heat Removal Systems.

RESPONSE:

The response to this question was submitted by letters dated 5/19/81 (PLA-813, Curtis to Youngblood), 6/16/81 (PLA-846, Curtis to Schwencer), and 4/23/82 (PLA-1053, Curtis to Schwencer).

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QUESTION 121.16

List the systems and line sizes that were exempted from pre-service examination based on Paragraph IWB-1220(b)(1) of the 1974 Edition of Section XI based on "normal makeup systems using on-site power."

RESPONSE:

The response to this question was submitted by letters dated 5/19/81 (PLA-813 Curtis to Youngblood), 6/16/81 (PLA-846, Curtis to Schwencer), and 4/23/82 (PLA-1053, Curtis to Schwencer).

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QUESTION 121.17

Your inservice inspection program must be revised to include periodic volumetric and/or surface examination of a representative sample of welds in the RHR, ECCS, and Containment Heat Removal Systems. In addition, we will require the following information concerning the examination of piping system welds:

- A) Your inservice inspection program is based upon the 1977 Edition through Summer 1978 Addenda of Section XI. However, in lieu of performing stress analysis on piping runs relative to weld sample selection for Category B-J and C-F welds as required by the Summer 1978 Addenda, you have elected to substitute the sample requirements of the 1974 Edition of Section XI. Describe the selection criteria used to determine your weld sample.
- B) Your program exempts under IWB-1220(a) components ≤ 4 inches nominal pipe size that are above the normal reactor pressure vessel water line. Provide the technical basis for applying this exclusion for examination.
- C) Your program exempts lines 18-GBB-109 and 6-GBB-109 on the basis of fluid chemistry control (IWC-1220(c) and IWC-1220(b)). Our position is that IWC-1220(c) is not a technically valid basis for exemption of inservice inspection requirements. Describe the exclusion from examination of these lines based only on IWC-1220(b). Discuss the operating conditions under which these lines are stagnant.

RESPONSE:

- A. Section 3.0 of the Susquehanna SES Unit #1 Inservice Inspection Plan addresses the applicable Code Edition and Addenda to be utilized in preparation of the specific weld sampling program for Class 1 Category BJ and Class 2 Category CF, piping welds. The extent of examination is further illustrated by composite Code tables found on Pages 3-3 through 3-7 of the ISI Plan. The weld selection criteria used will be that found in the specific paragraphs referenced in this section.

For all Code Class 1 pipe welds, examination selection criteria will be determined by Table IWB-2500 and Table IWB-2600 of the 1974 Edition, Summer 1975 Addenda.

For all Code Class 2 pipe welds, including the RHR, ECCS, and CHR systems, the weld sampling program to be

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utilized is specified by paragraph IWC-2411 of the 1974 Edition, Summer 1975 Addenda (Pages 3-6, 3-7 of Section 3.0).

These requirements are in compliance with 10 CFR 50.55 a (b) (2)(iv).

- B. Components 4-inch nominal pipe size and under that are above normal reactor pressure vessel water line are exempt per paragraph IWB-1220(a) based on reactor coolant make-up. The calculation for normal make-up capacity was performed by General Electric as described in G.E. Specification 22A2750, thereby, allowing 2" and 4" exemption below and above normal reactor water level.
- C. IWC-1220(c) was erroneously referenced here; the correct reference should be IWC-1220(b). The proper exemption, systems or portions of systems that do not operate during normal plant conditions, is being taken for RHR system piping containing a test return line and the connection to the containment spray header. These lines are isolated from the RHR system during normal plant operation and do not perform a safety function during normal operating modes.

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QUESTION 121.18

Identify the specific model(s) of General Electric low pressure turbine(s) installed in Susquehanna SES Units 1 & 2. Provide tables showing, for each of the seven wheels in a Susquehanna low pressure turbine, the weight and location of the wheels relative to the turbine center, and the (a) dimensions, (b) shapes, (c) weights and d) initial energy (or velocity) ranges of missiles postulated to be representative of missile-producing turbine wheel rupture at

- i. design overspeed, and
- ii. destructive overspeed.

RESPONSE:

The turbine numbers for SSES are 170 x 592 and 170 x 593 for units 1 & 2 respectively. Specifics of the turbine wheels are insignificant compared to the characteristics of the missile fragments, and were considered in their generation.

As stated in Section 3.5.1.3.2, the characteristics of the missiles postulated by the General Electric Company's Large Steam Turbine Division (GE) for destructive overspeed failures of the wheels in their 38" last stage buckets machines are given in Table 3.5-1 and Figure 3.5-1.

As stated in Section 3.5.1.3.4, GE has determined that the probability of a design overspeed failure in their 38" last stage bucket machines is insignificant compared with the probability of a destructive overspeed failure. Consequently, GE does not supply characteristics of missiles which could result from a wheel failure at design overspeed.

The details of the methodology used by GE in determining failure probabilities for the wheels in their 38" last stage buckets machines and in determining the characteristics of the missiles resulting from destructive overspeed failures are given in References 3.5-1, 3.5-2, 3.5-3 and 3.5-4.

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QUESTION 121.19

Provide tables listing all barriers and safety-related targets considered in the calculation of P2 (or P2 x P3). Provide schematic drawings (to scale) which show the location and orientation of barriers and targets relative to the turbine train.

RESPONSE:

No barriers were originally considered in the calculation of P2 x P3 except the targets structural walls and floor slabs. No credit was taken for the moisture separator or the radiation shield wall.

The targets considered in the original calculation were the control structure, the reactor buildings, the ESSW pumphouse, the steam tunnels, and the spent fuel pools. See response to Question 121.21 for revised evaluation of essential targets.

Scaled layout drawings are provided in Section 1.2 of the FSAR which show location and orientation of barriers, turbines, and targets.

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QUESTION 121.20

The derivations presented in 3.5.1.3.6 of the FSAR need clarification; numerous variables are poorly defined or not defined at all. The probability P3 must either be re-derived or an available document referenced which contains an adequate derivation.

RESPONSE:

Refer to revised Section 3.5.1.3 for the derivation.

QUESTION 121.21

Recalculate the P2 x P3 probabilities associated with each safety-related target for design overspeed and destructive overspeed, and show that the total P2 x P3 for each of these failure conditions is less than 10^{-3} per failure, as stipulated in Regulatory Guide 1.115.

RESPONSE:

I. Introduction

As noted in Section 3.13 of the FSAR, SSES was issued its construction permit over three years before Regulatory Guide 1.115 was formalized.

Because of this, we have proposed an alternate approach in the FSAR. However, in order to be responsive to this question we have evaluated our alternate method against Regulatory Guide 1.115.

A re-evaluation of our previous calculation changes P2 x P3 from 52×10^{-3} to $.60 \times 10^{-3}$. This reduction was made based on redundancy, credit for missile shielding, and more precise knowledge of essential shutdown equipment locations.

The data presented in this response is based on destructive overspeed missile spectrum data only. To date no missiles have been assumed by the turbines vendor to arise from a design overspeed event. However, for low trajectory missiles, the range of missile velocities which are higher than the minimum velocity necessary to cause damage is greater for destructive overspeed failures than for design overspeed failures. Therefore, destructive overspeed events result in higher total probabilities for damage. Furthermore, the destructive overspeed event encompasses the design overspeed event.

II. Reactor Building

An inspection of Table 3.5-3 shows that the major risk is the reactor building at 45×10^{-3} for P2x P3. This has two components, the risk for the roof slab at 818' and the risk for the west face of reactor building along column line "P".

A. Roof Slab

The original calculation assumed that unacceptable damage would occur if a turbine missile caused any spallation of the roof slab at elevation 818'; the roof slab risk was about 3×10^{-3} from our original calculation. However, there is no essential safe shutdown equipment at elevation 818' or on the two floors below at elevation 799' and 779'.

The computer code employed in the original calculation conservatively assumed that the missile velocity normal to the roof slab would be the same as if the roof slab were at the same elevation as the turbine. Actually, the difference in elevations between the 818' roof slab and the turbine axis is nearly 85 ft.; thus, the actual missile velocity normal to the roof slab should be reduced by approximately 75 ft./sec.

Because the floor slabs have difference thicknesses at different locations, the sum of the floor slab thickness at 818', 799' and 779' ranges from a minimum of 5'3" to a maximum of 8'3" of reinforced concrete. This mass of concrete, plus the reduction in strike velocity normal to the roof of 75 ft./sec. means that the roof slab risk to essential safe shutdown equipment below elevation 779' is negligible.

The reactor vessel is protected by massive shield plugs and the steel primary containment head assembly in the vertical direction, and is considered safe.

The only target at elevation 818' is the spent fuel pool. The P2 x P3 datum from our original calculation is unchanged at $.118 \times 10^{-3}$.

B. Face Wall

The reactor building face wall is primarily subject to low trajectory missiles with a risk of 42×10^{-3} from our original calculation. A review of FSAR Dwgs. M-223, Sh. 1 M-226, Sh. 1 shows that all low trajectory missiles would be subject to interference and slowing from the moisture separator and the radiation shield walls between the turbine low pressure hoods and the reactor building wall.

1. Credit for Shielding

The radiation shield wall is made of concrete block 3'3" thick which is equivalent to 2 ft. of poured concrete per modified NDRC formulae. The moisture separator has a carbon steel shell thickness of 1.25 inch. As each inch of carbon steel is equivalent to one foot of poured concrete, the total equivalent thickness for the moisture separator and the shield wall is greater than four feet of poured concrete. No credit was taken for the extensive internal steel baffling of the moisture separator.

The net effects of the equivalent concrete can be estimated using the methods in the SRP and our previously calculated data.

For low trajectory missiles P2 x P3 will in effect be proportional to:

$$\frac{V_2 - V_s}{V_2 - V_1}$$

Where V_2 = Maximum missile ejection strike velocity

Where V_1 = Minimum missile ejection strike velocity

Where V_s = Minimum missile ejection strike velocity required to spall the 3'0" reactor building wall.

The velocities are dependent on missile type, so it is necessary to consider the three stage groups (defined in FSAR Subsection 3.5.1.3.2) separately.

a) State Group I Missiles:

The analysis is based on the 'worst-case' missile which has $V_2 = 470$ ft./sec., $V_1 = 0$ ft./sec., a weight of 2000 lbs. and minimum projected rim area equivalent to a diameter of 16.47 in.

Assuming no barriers between the turbine and the reactor building, these missiles contribute 1.66×10^{-2} to P2 x P3. The minimum velocity required to spall a 3'0" wall is 150 ft./sec. This is V_s for the case with no barriers.

The minimum missile velocity required to perforate a 4'0" thick wall is 434 ft./sec. The residual velocity for a missile striking the barrier at 470 ft./sec. is 50 ft./sec. Since this is below the spalling threshold of 150 ft./sec. for the 3 ft. thick reactor building wall, the Stage Group I contribution to P2 x P3 will be zero.

b) Stage Group II Missiles:

The 'worst-case' missile has $V_2 = 550$ ft./sec., $V_1 = 0$ ft./sec., a weight of 3000 lb and a minimum projected rim area equivalent to a diameter of 19.48 in.

Assuming no barriers these missiles contribute 2.01×10^{-2} to P2 x P3.

The minimum velocity required to spall a 3'0" wall is 117.0 ft./sec. This is V_s for the no barrier case. The minimum velocity required to perforate a 4'0" wall with a residual velocity of 117.0 ft./sec. is 434.3 ft./sec. This is V_s for the barrier case.

The value of P2 x P3 would thus become

$$\begin{aligned} P2 \times P3 &= \frac{550 - 434.3}{550 - 117} \times 2.01 \times 10^{-2} \\ &= 5.37 \times 10^{-3} \end{aligned}$$

c) Stage Group III Missiles

The 'worst-case' missile has $V_2 = 610$ ft./sec. $V_1 = 400$ ft./sec., a weight of 6500 lb and a minimum projected rim area equivalent to a diameter of 26.63 in. The maximum projected rim area is

equivalent to a diameter of 39.99 in. Assuming no barriers these missiles contribute 5.01×10^{-3} to P2 x P3.

The minimum velocity required to spall a 3'0" wall is 71.5 ft./sec. For the maximum projected area the threshold for spalling is 84 ft./sec. The residual velocity for a missile striking a 4'0" wall at 400 ft./sec. (i.e., at V_1) is 154 ft./sec. for the minimum area. For the maximum projected rim area the residual velocity for a strike at 400 ft./sec. is 112 ft./sec. Hence the minimum residual velocity exceeds the maximum spalling threshold velocity for the reactor building wall. A similar analysis for penetration shows that stage III missiles will penetrate a 3'0" wall in 80% of the strikes after first penetrating a 4'0" wall.

2. Target Area Reduction

In the original calculation the target area for the reactor building face was a rectangle 90 feet high and 59 feet wide for a total area of 530 ft.

A review of essential shutdown cable and equipment has been performed for updating our Fire Protection Review Report for the new 10 CFR 50 Appendix R. This review is summarized in Q40.95. From this data base the spaces behind the target wall were examined for effects from spallation and penetration.

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- a. No target area above el 749' contained essential shutdown equipment.
- b. Elevation 719':
 - (1) On elevation 719' of the reactor building, damage could occur for the hydraulic control units (HCU) on only one side of the core. Moreover, the Standby Liquid Control can be considered a back-up. Also, three essential raceways, originally considered in the target area, E1K883, E1KJ23, and E1KJ24, are located about one hundred feet east of the target area. Only one essential raceway, E1K833, is in the target area. The actual target area of the wall is six feet wide between the stairwell and the elevator shaft on the northwest corner of the reactor building as shown on Dwg. M-243, Sh. 1, of the FSAR. Since there are no internal barriers which could restrict or contain the spallation products, for this target area the damage criterion is spallation. That is, any spallation is assumed to cause unacceptable damage, and, therefore, wall perforation would not add to the risk. The height of the target area is from 729' at the turbine deck to 749' at the floor slab or twenty feet.

Therefore, the actual essential spallation target area is 6' x 20' or 120 ft.².

The area correction factor for spallation is:

$$\frac{120 \text{ ft}}{5310 \text{ ft}} = .022$$

Applying this to each state group,

	<u>P2 X P3</u>	<u>Area Correction Factor</u>	<u>Subtotal</u>
Stage I0.0		.022	= 0.0
Stage II	$5.37 \times 10^3 \times$.022	= $.118 \times 10^{-3}$
Stage III	5.01×10^{-3}	.022	= $.110 \times 10^{-3}$
Risk Total for the 120 ft. target area			= $.228 \times 10^{-3}$

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- (2) No credit is taken for the 1'0" walls of the elevator shaft and the stairwell stopping the Stage III missile. The 2'0" concrete wall of the containment instrument gas room will resist missiles that enter the reactor building. Spallation products are assumed to be contained by either 1'0" or 2'0" walls. The damage criterion for this area is wall perforation only.

The area of risk for wall penetration is 23' wide. However, 6 ft. of this width has been accounted for above so that the net area is (23'-6') or 17' wide. Height is 20' (from elevation 729' to elevation 749'), for an area of 17' x 20' or 340 ft.

The area correction factor is then

$$\frac{340 \text{ ft.}^2}{5310 \text{ ft.}^2} = .064$$

Stage Group III missiles can penetrate the 3'0" reactor building wall 80% of the time after penetrating the 4'0" of shielding, depending on exit velocity and missile orientation. Thus the effective correction factor is 80% x .064 = .051.

<u>Missiles</u>	<u>P2 x P3</u>	<u>Area Correction Factor</u>	<u>Penetration Probability</u>
Stage I	0.0		
Stage II	0.0		
Stage III	$5.01 \times 10^{-3} \times$.051	$= .257 \times 10^{-3}$
Total Reactor Building Risk			
120 ft. area risk	$.228 \times 10^{-3}$		
	$.257 \times 10^{-3}$		
<u>+340 ft. area risk</u>			

Reactor Building total $.485 \times 10^{-3}$

III. Other Buildings

The other buildings as listed in table 3.5-3 contributed 7×10^{-3} for P2 x P3 in the original calculation. However, no credit was taken for redundancy of essential equipment. A "lob" missile is the only credible type for the diesel generator buildings. With this trajectory (low trajectories being prevented by the turbine pedestal) only one diesel generator could be damaged per turbine destruction event due to the substantial walls that separate the units and the near vertical trajectory. Losing one diesel would not prevent cold shutdown.

A similar argument can be applied to the steam tunnel. The outboard MSIVs could be lost but the primary containment's six feet of steel-lined concrete, the steam tunnel walls, the moisture separator and the radiation shield walls would protect the inboard MSIVs. The total risk of the diesel generator building and the steam tunnel was 5.5×10^{-3} . Applying the concept of redundant equipment reduces this to zero.

The control structure risk was originally calculated as $.73 \times 10^{-3}$ per turbine destructive event. This was based on the whole building volume above el 729' as a target. Again from our Appendix R review no essential equipment exists above elevation 783'. The roof and floor slab thicknesses equals a minimum of 5' 10.5" to a maximum of 8' 10.5". Also note that gravitational deceleration slows missiles by 75 feet per second at the roof slab elevation.

Including a factor for reduction of low trajectory missiles because of the moisture separator and radiation shield walls reduces the total control structure P2 x P3 to essentially zero for both high and low trajectory missiles.

IV. Quantified Revised Estimates

Applying the correction factors described above revises the P2 x P3 values given in FSAR Table 3.5-2 to the estimates below.

Target	<u>P2 x P3</u>	
	<u>Was</u>	<u>Revised Estimate</u>
Reactor Bldg	$45. \times 10^{-3}$	$.485 \times 10^{-3}$
Spent Fuel Pool	$.118 \times 10^{-3}$	$.118 \times 10^{-3}$
Steam Tunnel	$3. \times 10^{-3}$	0
Control Structure	$.73 \times 10^{-3}$	0
Diesel Generator Bldgs	2.5×10^{-3}	0
TOTAL	52×10^{-3}	$.603 \times 10^{-3}$

V. Non-Quantified Conservatism

The damage probability, P3, has numerous conservatisms as originally calculated. The method used was based on a modification of the NDRC method. We assumed that the rotor fragment always hits on edge and never on the flat face. A flat face hit would considerably reduce the delivered energy per unit area of the target which, intuitively, lowers the probability of spallation and/or penetration.

Energy transfer at impact was conservatively assumed to be 100% of the energy normal to the target face at ejection. No corrections were made for air resistance of the large, tumbling fragments, or less than complete mechanical coupling from glancing blows, ricochets, or the like.

We had further assumed that any spallation or penetration prevented safe shutdown. It is obvious that not all wall damage results in complete equipment failure and prevention of cold shutdown.

VI. Summary

In summary, examining both quantified and non-quantified conservatism in our calculations convinces us that the total P2 x P3 figure for SSES is well within the 1×10^{-3} number stipulated in Regulatory Guide 1.115.