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Energy Systems

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ET-NRC-93-3795
NSRA-APSL-93-0006
Docket No.: STN-52-003
Proj 674
January 14, 1993

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: DR. THOMAS MURLEY

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600

Dear Dr. Murley:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of September 23, 1992, October 1, 1992 and October 9, 1992. This transmittal is a partial response to those letters. A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A. The Westinghouse responses to the remainder of the requests for additional information contained in your letters of September 23, 1992 and October 1, 1992 will be provided prior to January 23, 1993.

If you have any questions on this material, please contact Mr. Brian A. McIntyre at 412-374-4334.

Nicholas J. Liparulo

Nicholas J. Liparulo, Manager
Nuclear Safety & Regulatory Activities

/nja

Enclosure

cc: B. A. McIntyre - Westinghouse
F. Hasselberg - NRR

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ET-NRC-93-3795
ATTACHMENT A
AP600 RAI RESPONSES
JANUARY 14, 1993

RAI Number	Subject
210.3	Seismic Piping Analyses
210.6	Pipe Break Analyses
210.7	Earthquake Fatigue Analyses
210.16	Reactor Vessel Internals Integration
210.18	Reactor Vessel Internals Testing
210.23	Reactor Vessel Internals Dynamic Analyses
220.10	Containment Level C Factor of Safety
220.11	Containment Head Buckling
220.12	Containment Head Buckling
230.1	Non-Category I Facilities (UBC)
230.3	Seismic Design Basis
230.4	Vibratory Ground Motion
230.5	Cumulative Absolute Velocity
230.23	Triaxial Acceleration Sensors
250.10	Steam Generator Tube ISI
250.16	Steam Generator Tube Fabrication Inspections
250.21	Steam Generator Tube ISI Provisions
251.2	RCP Flywheel Materials
251.3	RCP Flywheel Fracture Toughness
251.4	RCP Flywheel Fabrication
251.5	Regulatory Guide 1.14 Requirements for RCP Flywheel
251.6	RCP Flywheel ISI
251.7	Regulatory Guide 1.14 Requirements for RCP Flywheel
251.8	RCP Flywheel Overspeed
251.9	Regulatory Guide 1.14 Exception
251.10	RCP Flywheel Alternate ISIs

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ATTACHMENT A
AP600 RAI RESPONSES
JANUARY 14, 1993

RAI Number	Subject
251.11	Containment of RCP Flywheel Rupture
251.12	RCP Flywheel Rupture/Leak
251.13	Ultrasonic Uranium Inspection
251.14	RCP Flywheel Enclosure Inspection
251.15	RCP Flywheel Overspeed
251.16	RCP Flywheel Stresses
251.17	RCP Flywheel Enclosure and Weld Limits
251.18	RCP Flywheel Stresses
251.19	RCP Flywheel Enclosure and Weld Limits
251.20	RCP Shaft and Bearings
251.21	RCP Flywheel Stress Corrosion Resistance
251.22	RCP Flywheel Stress Corrosion Resistance
251.23	RCP Flywheel Material Specifications
252.2	LBB Application
252.12	Part Through-Wall Flaws in LBB Analysis
252.13	LBB for Feedwater and Steam Piping
252.14	Pressurizer Surge Line Stratification
252.19	Material Fatigue Resistance
252.21	Containment Vessel Coating
252.22	Containment Cathodic Protection
252.23	Containment Corrosion
252.24	Containment Vessel Access
252.25	Containment Vessel Exterior Surface
252.26	Containment Middle Annu'us Corrosion
252.27	Containment Corrosion from PCS Leakage
252.39	CRDM Conformance with Regulatory Guide 1.85
252.41	Use of ASME NQA-2

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ATTACHMENT A
AP600 RAI RESPONSES
JANUARY 14, 1993

RAI Number	Subject
252.44	Reactor Internals Materials
252.50	Use of ASME NQA-2 alternatives
252.51	RCPB Material Specifications
252.52	Table 5.2-1 Material Specifications
252.56	Environmental Impact on Fatigue Resistance
252.58	Fracture Toughness of Low-Alloy Materials
252.67	Grinding of ASS Materials
252.73	RCPB Conformance with Regulatory Guide 1.37
252.74	RCPB Conformance with Regulatory Guide 1.43
252.75	RCPB Exceptions to Regulatory Guide 1.44
252.82	RV Beltline Forging and Welds
252.83	RV Beltline Material Copper Content
252.85	RV Material Heat-Treating
252.86	PTS Fracture Toughness Requirements
252.87	Embrittlement Prediction Uncertainties
252.89	RV Conformance with Regulatory Guide 1.37
252.92	RV Material Surveillance Program Reference Temp
252.93	Capsule Withdrawal Schedule
252.94	Surveillance Capsule Standard Reference Materials
252.96	Surveillance Capsule Lead Factors
252.101	Environmental Effects on RV Material Fatigue
252.102	In-Place RV Thermal Annealing
252.104	Embrittlement Prediction Uncertainties
252.105	Pressure-Temperature Limit Calculations
252.106	Pressure-Temperature Limit Verification
252.110	Steam Generator Flow-Induced Vibration
252.120	Control of Preheat Temperature for Welding of Low-Allow Steel

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ATTACHMENT A
AP600 RAI RESPONSES
JANUARY 14, 1993

RAI Number	Subject
252.122	Conformance with Regulatory Guides 1.7 and 1.54
252.125	ESF Materials Corrosion Allowance
252.127	QA for Cleaning of Fluid Systems
252.132	ESF Materials
252.133	ESF Materials Fatigue Test Data
252.142	Fatigue Analysis of Materials used in the Steam and Feedwater System
281.2	Primary Water Chemistry
281.4	Coatings
281.7	Protective Coatings
281.8	ANSI Standard N101.2
410.42	Same as Q410.10 IRWST lines
420.2	PMS Failure Modes and Effects Analysis
435.8	Diesel Generator Control
435.26	Class 1E dc System Control Room Alarms and Indications
435.27	dc Interrupting Rating of Molded Case Breakers
435.28	Regulatory Guide 1.47 Indication for dc System
435.29	Battery Charger Capacity
435.38	Battery Condition Monitor
435.51	Electrical and Physical Separation when Using Spare Battery and Charger
435.61	Containment Electrical Penetration Protection (RG 1.63)
435.70	Failure of the Normal Lighting System During an Earthquake
435.71	Emergency Lighting Basis
440.2	WCAP 13345 (CMT Testing)
440.16	WCAP 13342 (ADS Testing)
440.28	Technical Specifications for Nonsafety Systems
460.5	Solid Radwaste Systems

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ATTACHMENT A
AP600 RAI RESPONSES
JANUARY 14, 1993

RAI Number	Subject
460.6	Seismic Design of Radwaste Management Systems
620.2	Locker Room Layout
720.1	PRA WCAPS

Question 210.3

Section 3.2.1.1.2 of the SSAR references Section 3.7 for the criteria used for the design of seismic Category II structures, systems, and components. In Section 3.7.3.13.3, "Interaction of Other Piping with Seismic Category I Piping," one of the analysis options for seismic Category II piping systems is "envelop ıg methods that limit stresses to the level D limits of Equation (9), ND-3653 of the ASME Code, Section III."

a. Provide a brief description of "enveloping methods."

b. It is the staff's understanding that, to be consistent with the definition of seismic Cate[illegible] in Subsection 3.2.1.1.2, the loads resulting from an SSE will be used in the Equation (9) calculation [illegible]ch will then be compared to the ASME Service Level D limits in ND-3655 of ASME Section III. If this interpretation is not correct, provide a description of how this criteria will be implemented.

Response:

a. Enveloping methods use the concept of enveloping a series of bounding analyses. This method will not be used for the AP600 piping systems.

b. The staff's understanding of SSE pipe stress limits in seismic Category II systems is correct. See the response to Q210.7 for additional information.

SSAR Revision:

SSAR Subsection 3.7.3.13.3 will be revised as follows:

~~Nonseismic Category I piping systems whose failure is not acceptable to the adjacent seismic Category I piping are analyzed by the nomograph method, other simplified dynamic analysis methods, or enveloping methods that limit stresses to the level D limits of Equation (9), ND-3653 of the ASME Code, Section III, to provide structural integrity and help prevent any unacceptable physical interaction with adjacent seismic Category I piping and components.~~

~~The nomograph method provides seismic restraint spacing based on the natural frequency of the supported piping. This support spacing provides confidence that the first natural frequency of the nonseismic Category I piping is beyond that value which is twice the resonant frequency of the input response spectra.~~
Nonseismic Category I piping systems whose failure is not acceptable to the adjacent seismic Category I piping are analyzed and designed in accordance with the requirements of the ASME/ANSI B31.1 Code. The structural integrity of the piping and supports during an SSE event is demonstrated as follows. Seismic analysis is performed as discussed in Subsection 3.7.3.8 for ASME Class 3 piping. Pipe supports are evaluated to the ASME Class 3 loading combinations with the SSE shown in Table 3.9-8. Piping components are evaluated to the ASME Class 3 loading combinations with the SSE shown in Tables 3.9-7 and 3.9-11.

Question 210.6

Section 3.6.2.1.1 of the SSAR states that "Breaks are not postulated in these sections of pipe, including the reactor coolant loop and pressurizer surge line, that meet the requirements for mechanistic break as described in Subsection 3.6.3." If the pipe cannot meet the limitations and acceptance criteria for the leak-before-break methodology as discussed in Paragraph II.D of Enclosure 1 to the draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," February 27, 1992, excluding high- and moderate-energy piping from the guidance of Section 3.6.2 and Branch Technical Position MEB 3-1 of the SRP is not acceptable. Identify the high- and moderate-energy pipe that will not meet this acceptance criteria and evaluate it in accordance with the guidance of Section 3.6.2 and Branch Technical Position MEB 3-1 of the SRP (see Q252.2-Q252.14).

Response:

The leak-before-break methodology is not applied to moderate-energy piping systems. This methodology is applied to the candidate high-energy lines in the nuclear island identified in Appendix 3E (see below). This appendix also identifies other high-energy lines in the nuclear island with diameters larger than 1 inch, for which pipe breaks are postulated as described in Subsection 3.6.2. The evaluation criteria for lines that do not satisfy the leak-before-break criteria are described in Subsection 3.6.2.

SSAR Subsection 3.6.2.1.1 will be revised and Appendix 3E added as follows:

SSAR Revision:

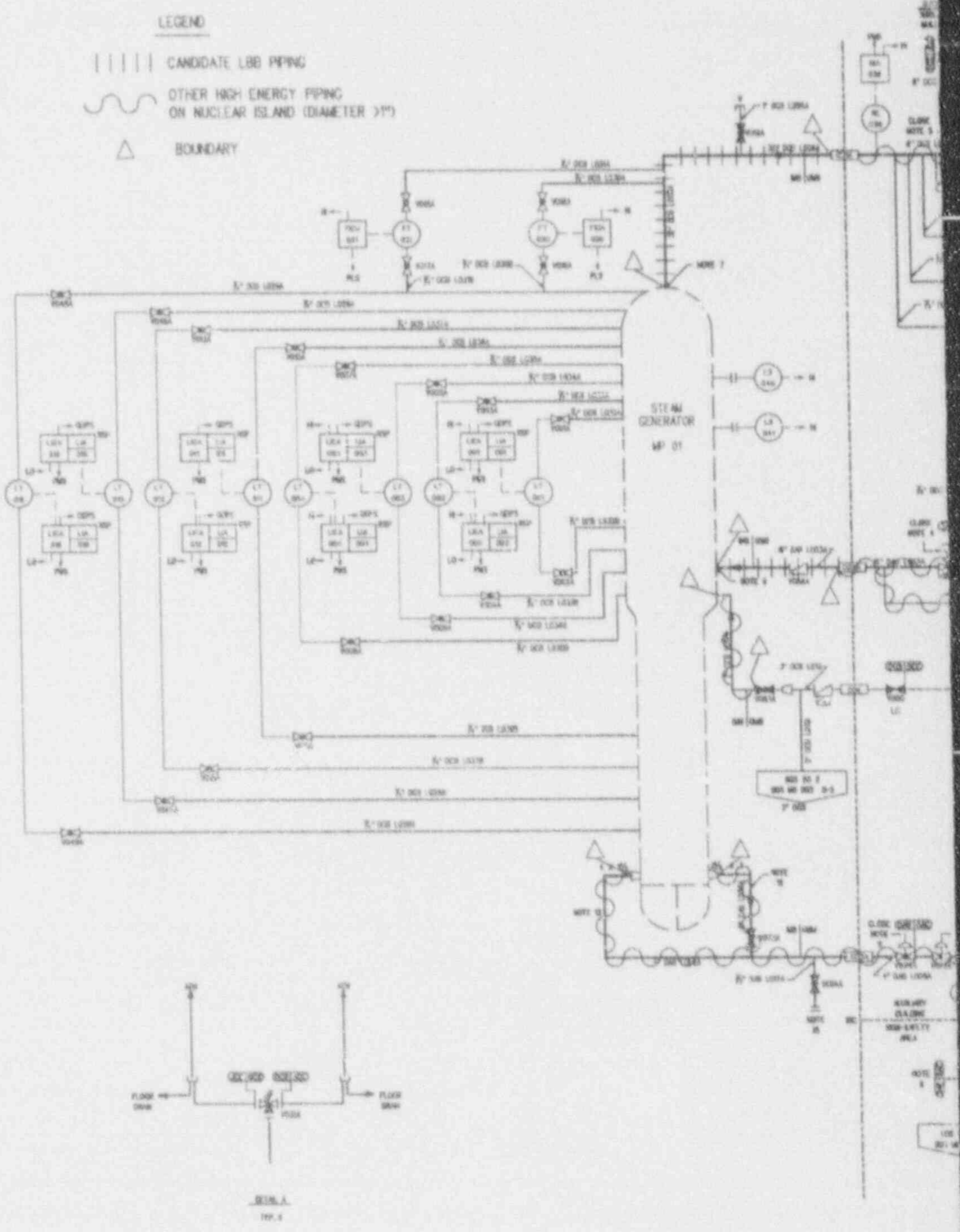
(Added to end of Subsection 3.6.2.1.1)

The leak-before-break methodology is applied to the candidate high-energy lines in the nuclear island identified in Appendix 3E. This appendix also identifies other high-energy lines in the nuclear island with diameters larger than 1 inch. The evaluation criteria for lines that do not satisfy the leak-before-break criteria are described in Subsection 3.6.2.

(Added as Appendix 3E)

Appendix 3E
High-Energy Piping in the Nuclear Island

This appendix identifies high-energy piping in the nuclear island with a diameter larger than 1 inch. Candidate leak-before-break piping is identified in Figure 3E-1 along with other piping for which high-energy pipe failures are postulated. The selection of the failure type is based on whether the system is high or moderate energy during normal operating conditions of the system. High-energy piping includes those systems or portions of systems in which the maximum normal operating temperature exceeds 200°F or the maximum normal operating pressure exceeds 275 psig. Piping systems or portions of systems pressurized above atmospheric pressure during normal plant conditions and not identified as high energy are considered moderate energy. Piping systems that exceed 200°F or 275 psig for 2 percent or less of the time during which the system is in operation or that experience high-energy pressures or temperatures for less than 1 percent of the plant operation time are considered moderate energy. In piping whose nominal diameter is greater than 1 inch but less than 4 inches, only circumferential breaks are postulated at each selected location. No breaks are postulated for piping whose nominal diameter is 1 inch or less.


LEGEND
CANDIDATE LBB PIPING
OTHER HIGH ENERGY PIPING ON NUCLEAR ISLAND (DIAMETER >1")
BOUNDARY
STEAM GENERATOR
FLOOR DRAIN
FLOOR DRAIN

AP600

SI APERTURE CARD

Also Available On Aperture Card

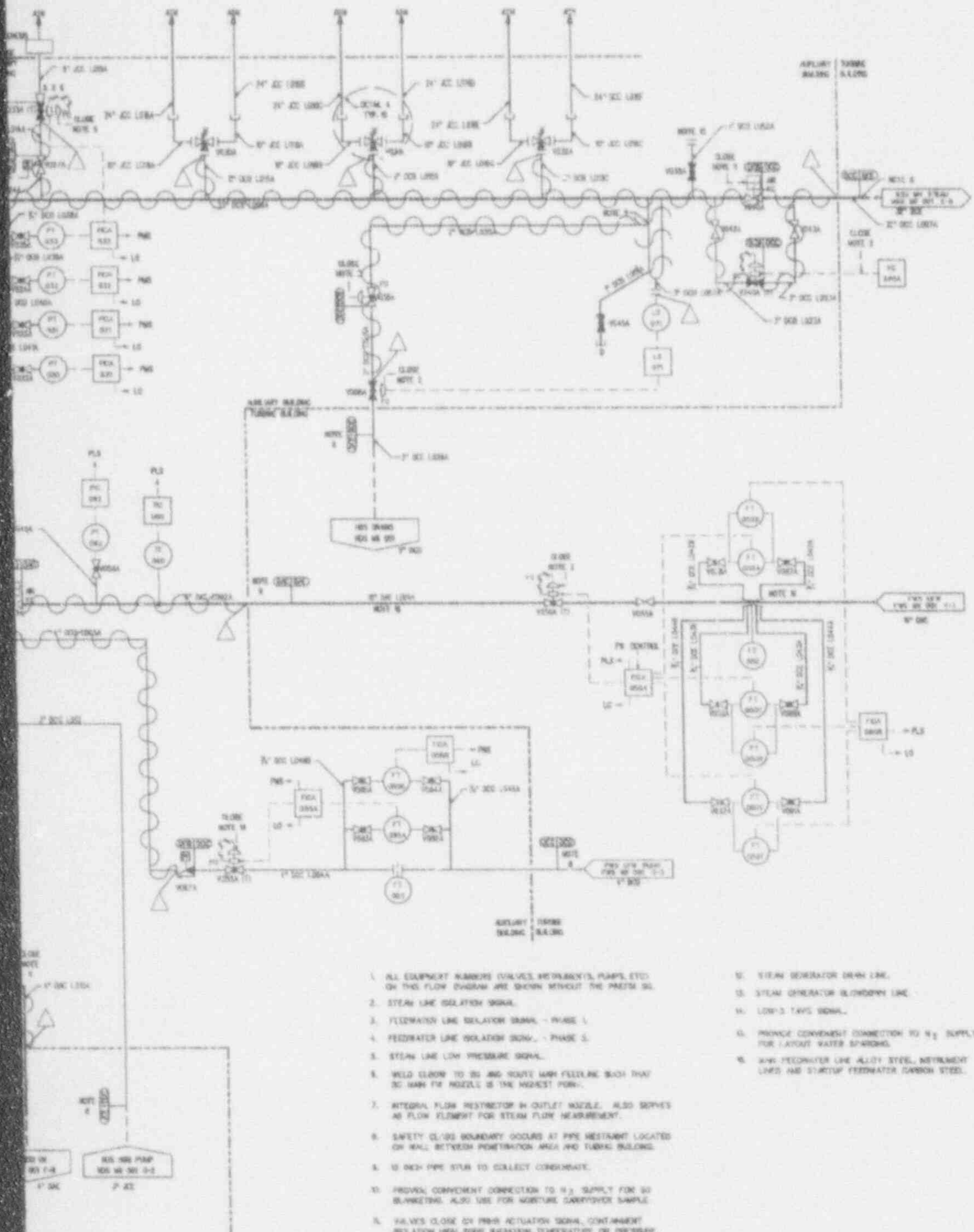
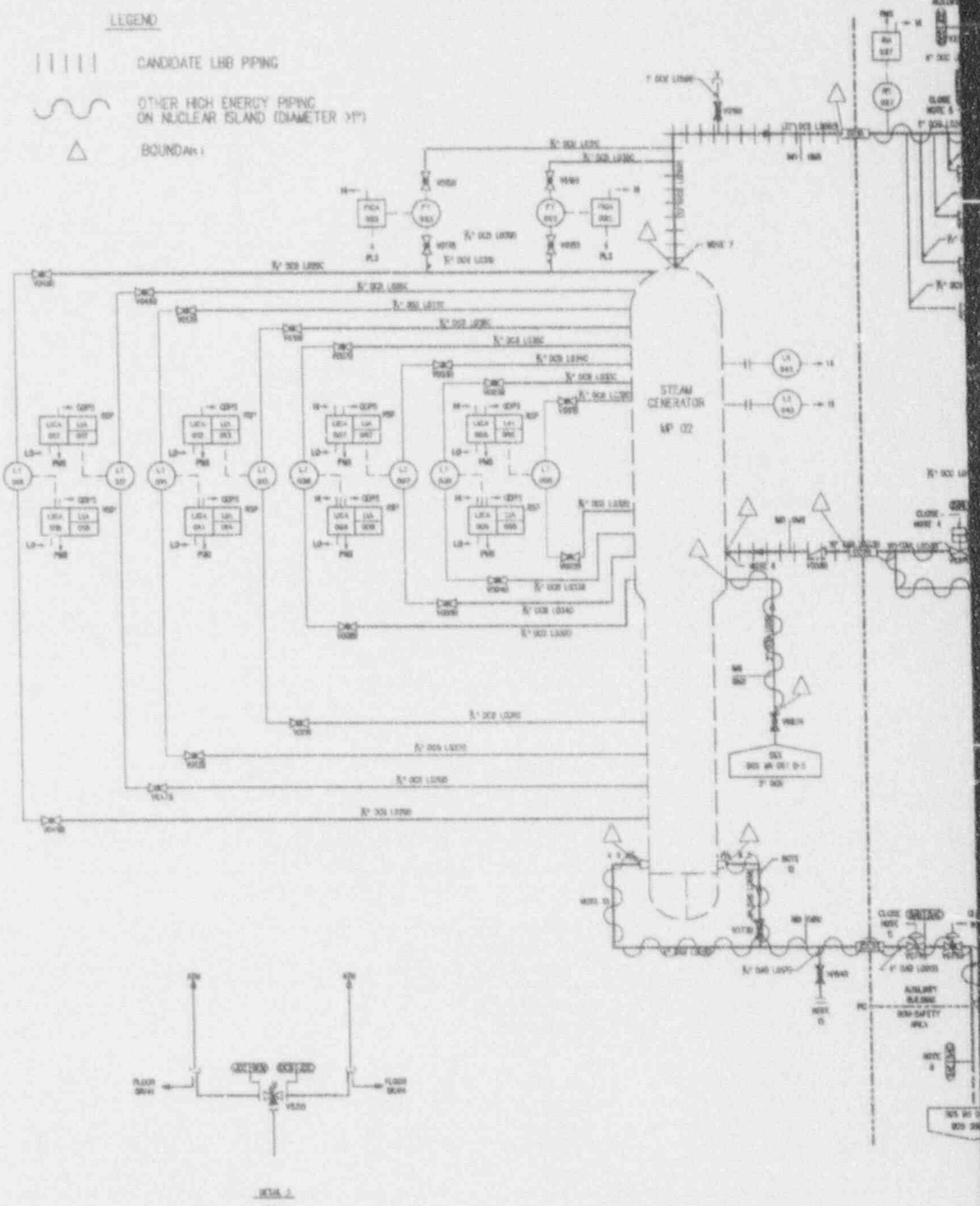


Figure 3E-1 (Sheet 1 of 10)

High Energy Piping - Steam Generator System

9301250260-01


LEGEND
CANDIDATE LBB PIPING
OTHER HIGH ENERGY PIPING ON NUCLEAR ISLAND (DIAMETER >1")
BOUNDARY
STEAM GENERATOR
FLOOR DRAIN

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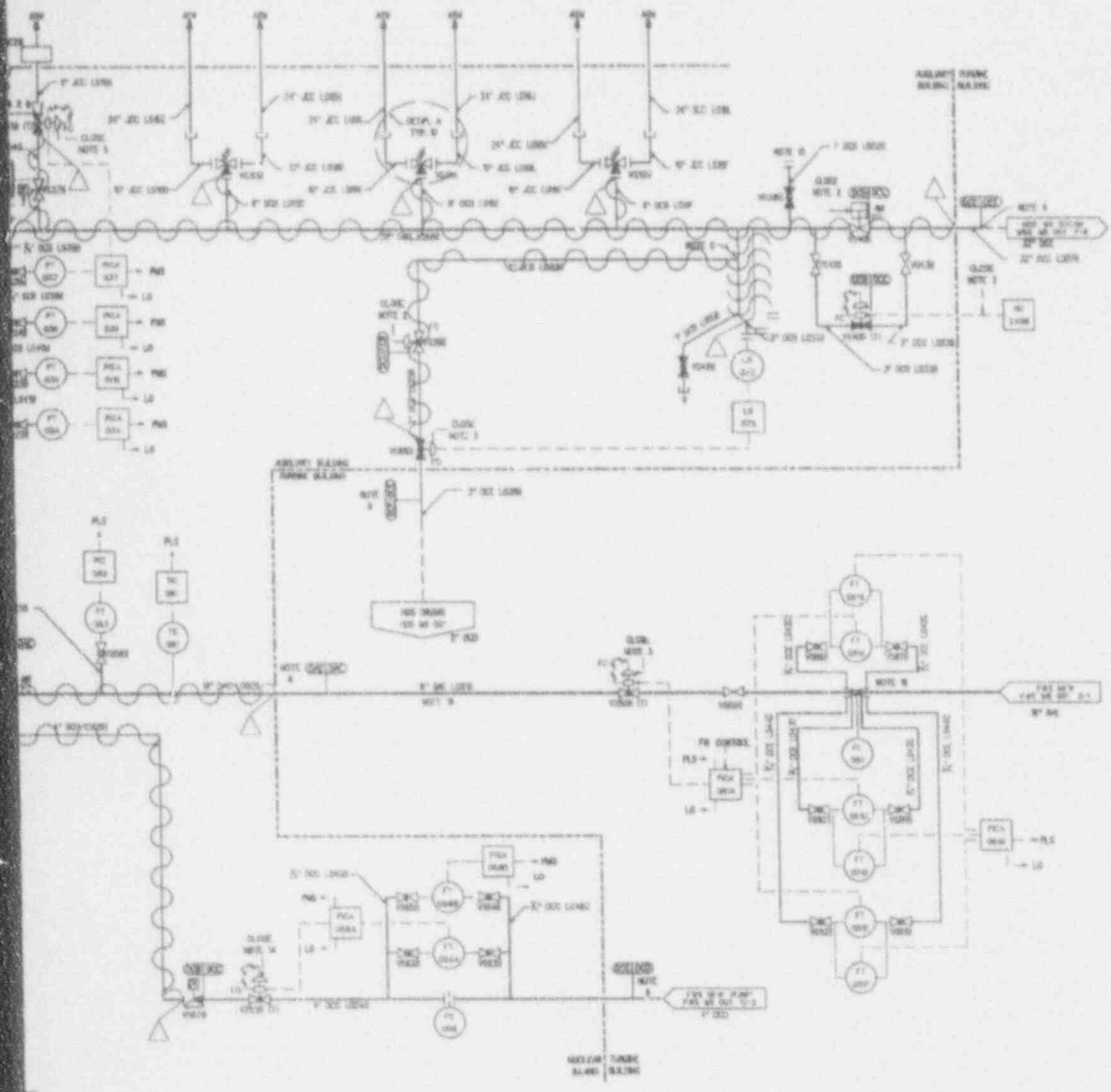
APERTURE
CARD

Also Available On
Aperture Card

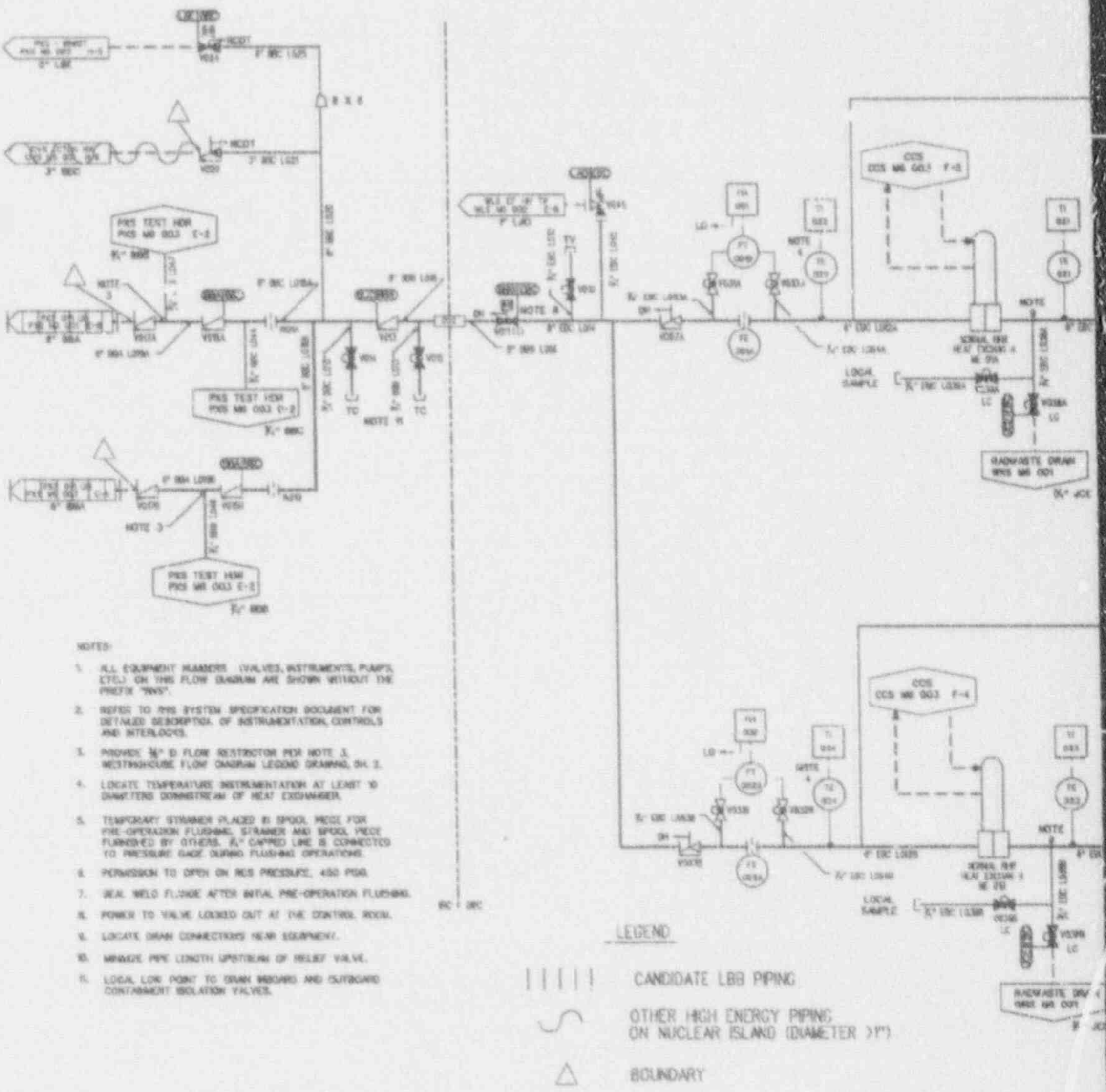


1. ALL EQUIPMENT NUMBERS (VALVES, INSTRUMENTS, PUMPS, ETC) ON THIS FLOW DIAGRAM ARE SHOWN WITHOUT THE PREFIX SG.
2. STEAM LINE ISOLATION SIGNAL.
3. FEEDWATER LINE ISOLATION SIGNAL - PHASE 1.
4. FEEDWATER LINE ISOLATION SIGNAL - PHASE 2.
5. STEAM LINE LOW PRESSURE SIGNAL.
6. WELD ELBOW TO SG AND ROUTE MAIN FEEDLINE SUCH THAT SG MAIN FW NOZZLE IS THE HIGHEST POINT.
7. INTEGRAL FLOW RESTRICTOR IN OUTLET NOZZLE. ALSO SERVES AS FLOW ELEMENT FOR STEAM FLOW MEASUREMENT.
8. SAFETY CLASS BOUNDARY OCCURS AT PIPE RESTRAINT LOCATED ON WALL BETWEEN PENETRATION AREA AND TURBINE BUILDING.
9. 12 INCH PIPE STUB TO COLLECT CONDENSATE.
10. PROVIDE CONVENIENT CONNECTION TO N_2 SUPPLY FOR SG BLANKETING. ALSO USE FOR MOISTURE CARRYOVER SAMPLE.
11. VALVES CLOSE ON PRHR ACTUATION SIGNAL, CONTAINMENT ISOLATION, HIGH SGBS RADIATION, TEMPERATURE, OR PRESSURE.
12. STEAM GENERATOR DRAIN LINE.
13. STEAM GENERATOR BLOWDOWN LINE.
14. LOW-3 TAVG SIGNAL.
15. PROVIDE CONVENIENT CONNECTION TO N_2 SUPPLY FOR LAYUP WATER SPARGING.
16. MAIN FEEDWATER LINE ALLOY STEEL, INSTRUMENT LINES AND STARTUP FEEDWATER CARBON STEEL.

Figure 3E-1 (Sheet 2 of 10)

High Energy Piping - Steam Generator System

9301250260-02


PXS TEST HDR
PXS M6 003 E-2
CCS
CCS M6 003 F-4
RADWASTE DRAIN
WRS M6 001
NOTES:
1. ALL EQUIPMENT NUMBERS (VALVES, INSTRUMENTS, PUMPS, ETC.) ON THIS FLOW DIAGRAM ARE SHOWN WITHOUT THE PREFIX "RNS".
2. REFER TO RNS SYSTEM SPECIFICATION DOCUMENT FOR DETAILED DESCRIPTION OF INSTRUMENTATION, CONTROLS AND INTERLOCKS.
3. PROVIDE 3/8" D FLOW RESTRICTOR PER NOTE 3, WESTINGHOUSE FLOW DIAGRAM LEGEND DRAWING, SH. 3.
4. LOCATE TEMPERATURE INSTRUMENTATION AT LEAST 10 DIAMETERS DOWNSTREAM OF HEAT EXCHANGER.
5. TEMPORARY STRAINER PLACED IN SPOOL PIECE FOR PRE-OPERATION FLUSHING. STRAINER AND SPOOL PIECE FURNISHED BY OTHERS. 3/4" CAPPED LINE IS CONNECTED TO PRESSURE GAGE DURING FLUSHING OPERATIONS.
6. PERMISSION TO OPEN ON RCS PRESSURE, 450 PSIG.
7. SEAL WELD FLANGE AFTER INITIAL PRE-OPERATION FLUSHING.
8. POWER TO VALVE LOCKED OUT AT THE CONTROL ROOM.
9. LOCATE DRAIN CONNECTIONS NEAR EQUIPMENT.
10. MINIMIZE PIPE LENGTH UPSTREAM OF RELIEF VALVE.
11. LOCAL LOW POINT TO DRAIN INBOARD AND OUTBOARD CONTAINMENT ISOLATION VALVES.
LEGEND
CANDIDATE LBB PIPING
OTHER HIGH ENERGY PIPING ON NUCLEAR ISLAND (DIAMETER >1")
BOUNDARY

SI
APERTURE
CARD

Also Available On
Aperture Card

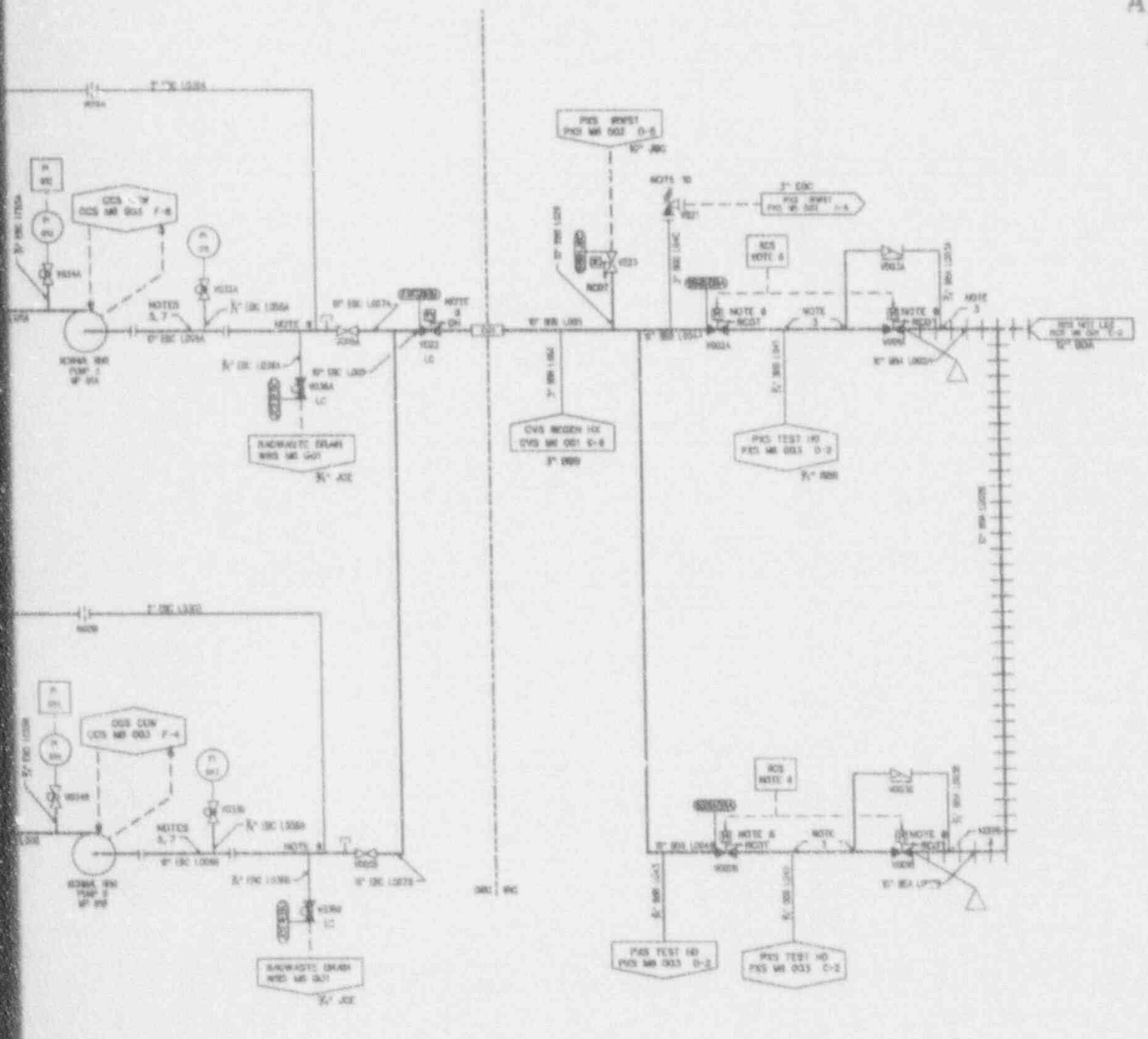
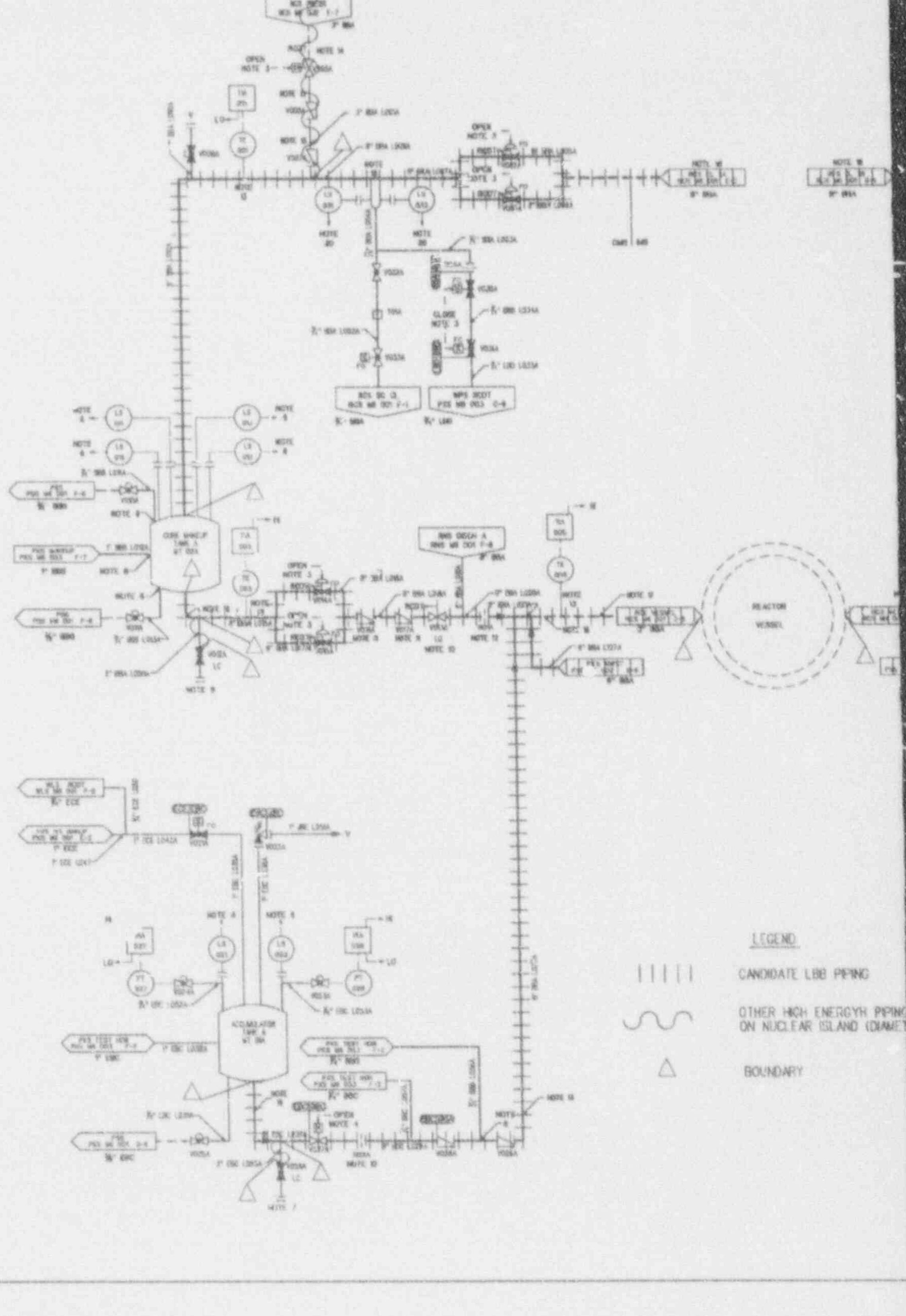


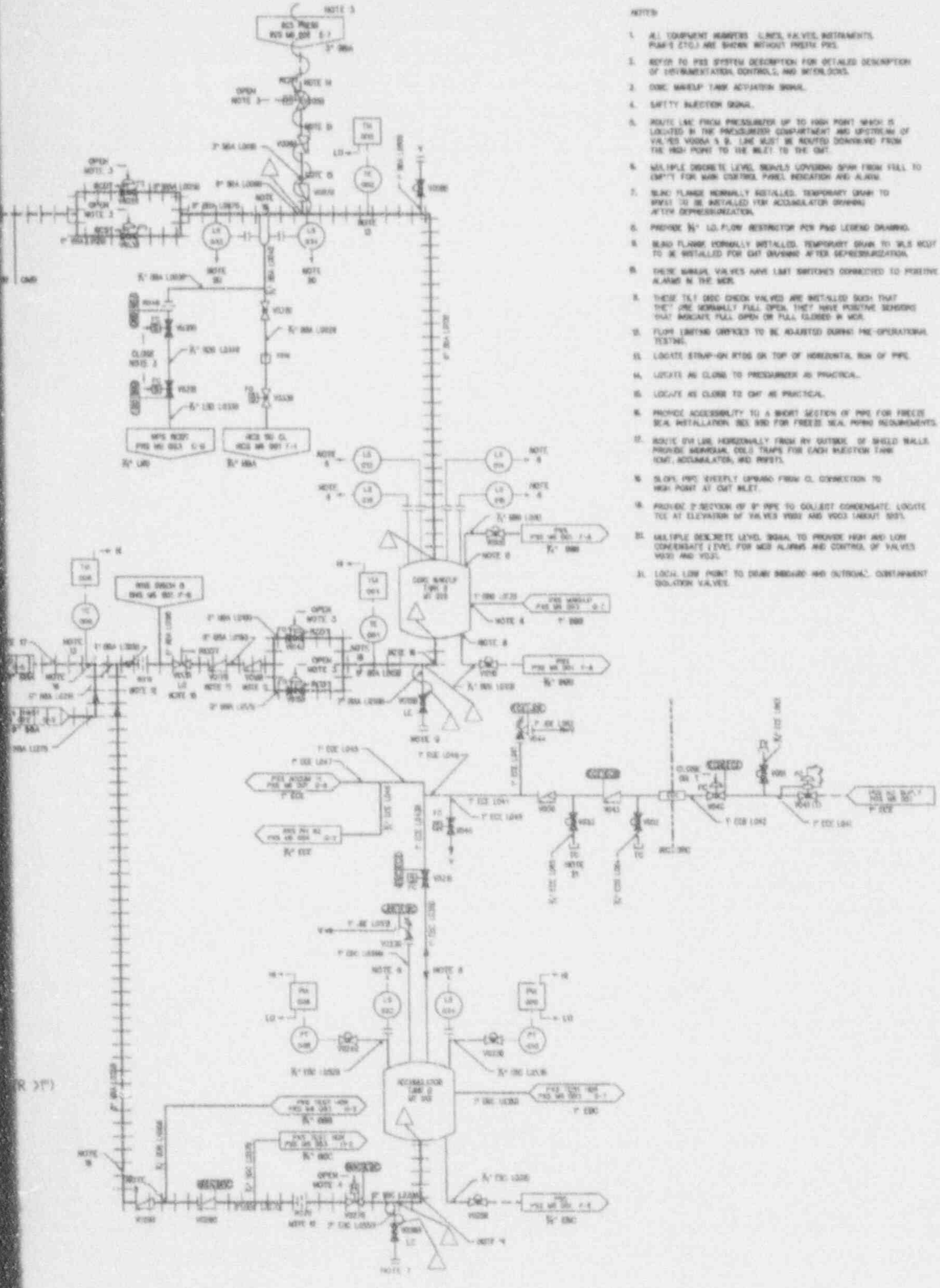
Figure 3E-1 (Sheet 3 of 10)

High Energy Piping - Normal Residual Heat Removal System

9301250260-03


REACTOR VESSEL
LEGEND
CANDIDATE LBB PIPING
OTHER HIGH ENERGYH PIPING ON NUCLEAR ISLAND (DIAMET
BOUNDARY

AP600



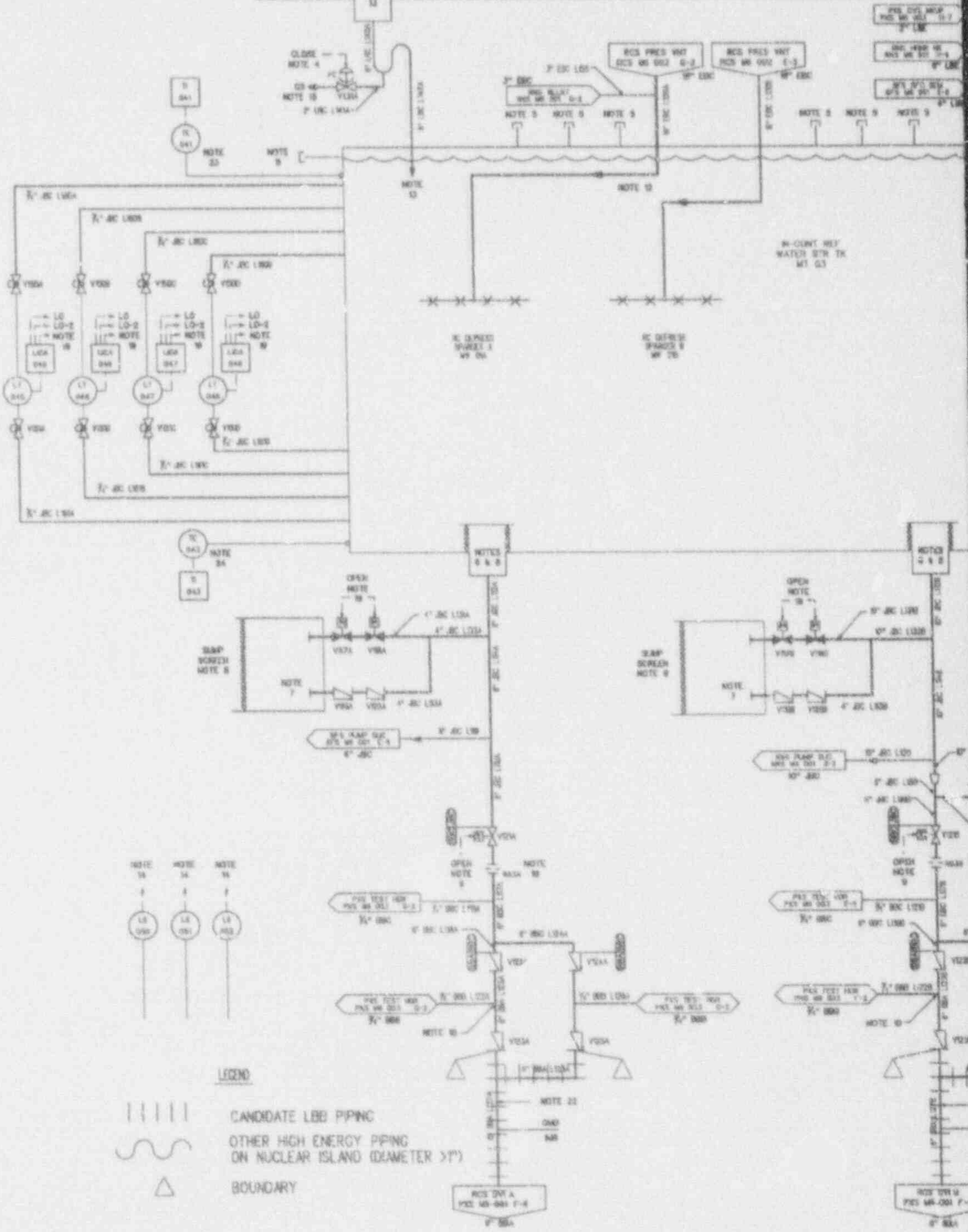
SI APERTURE CARD

Also Available On Aperture Card

Figure 3E-1 (Sheet 4 of 10)

High Energy Piping - Passive Core Cooling System

9301250260-04


LEGEND
CANDIDATE LBB PIPING
OTHER HIGH ENERGY PIPING
ON NUCLEAR ISLAND (DIAMETER >1")
BOUNDARY

SI APERTURE CARD

Also Available On Aperture Card

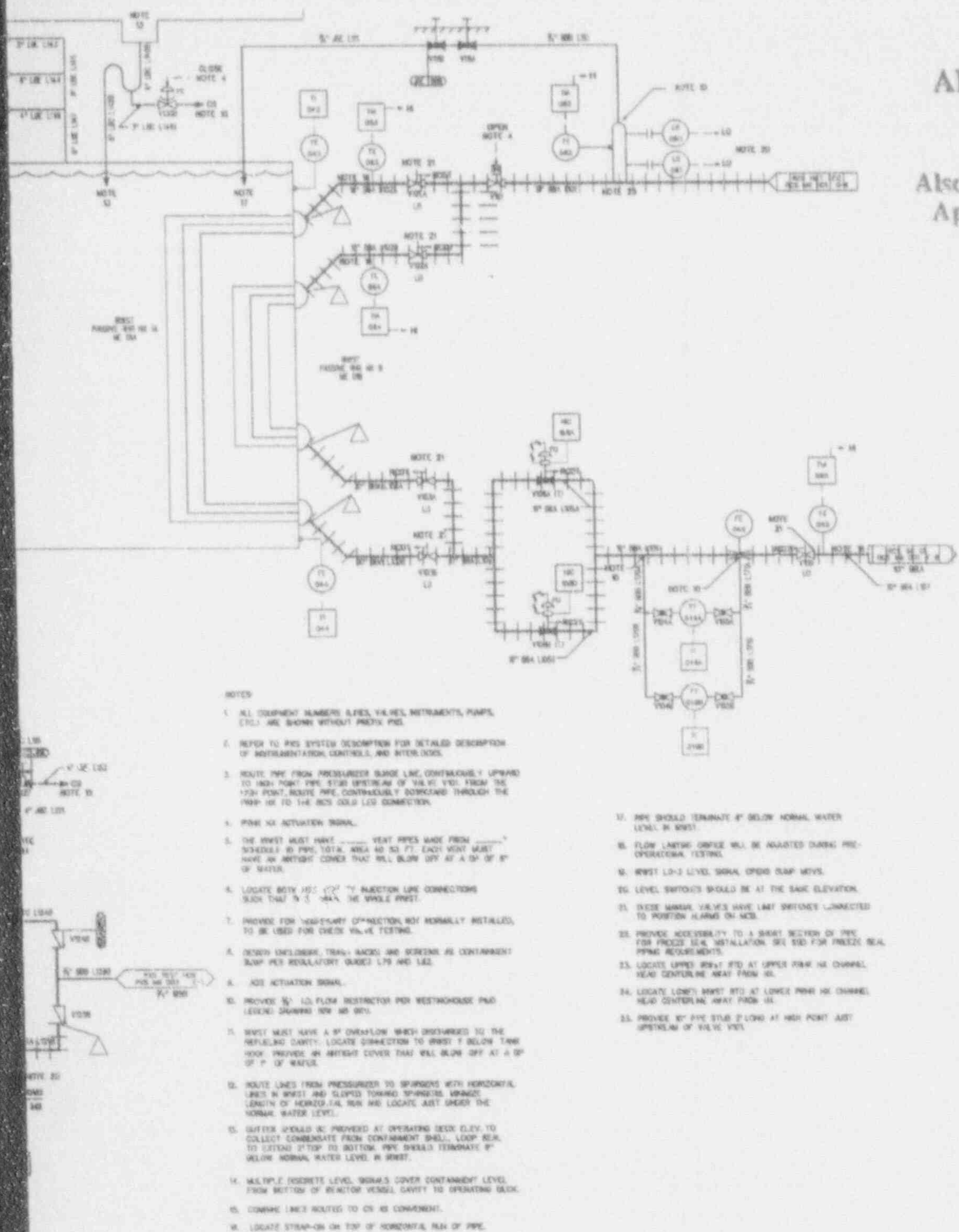
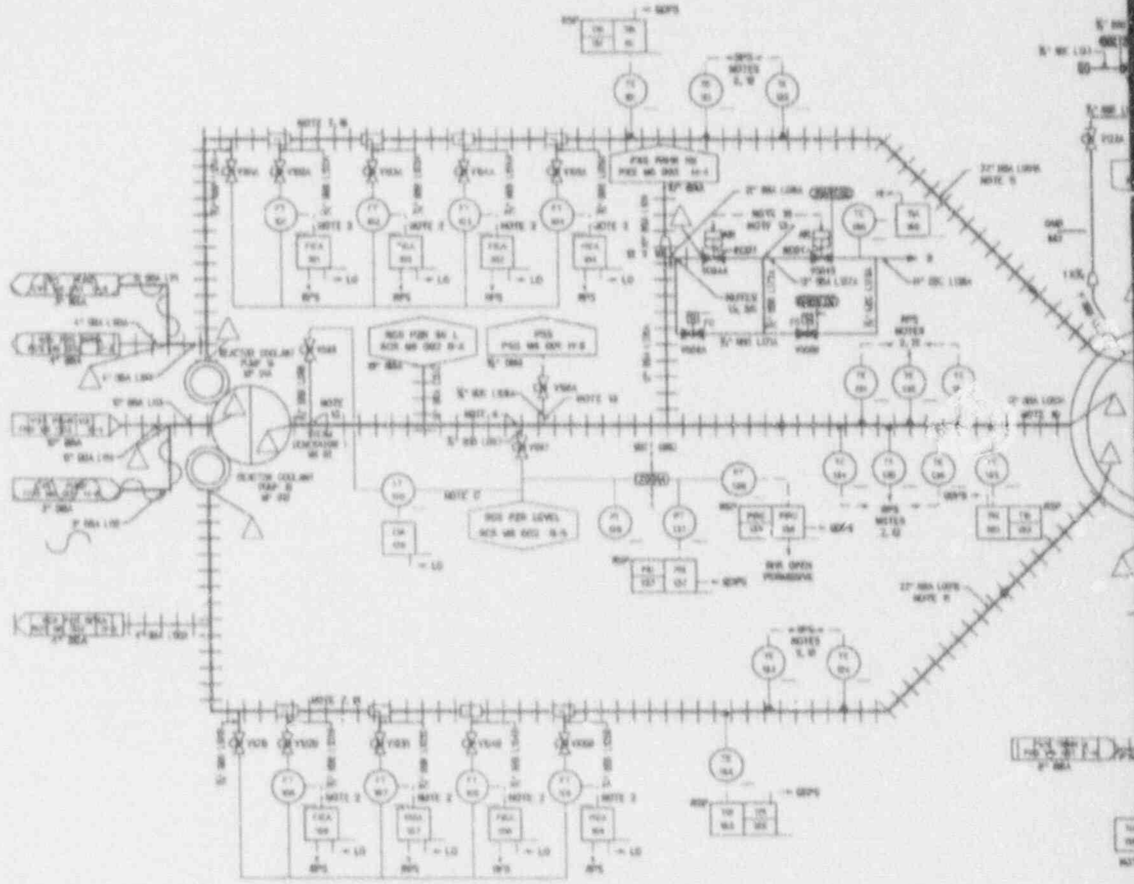


Figure 3E-1 (Sheet 5 of 10)

High Energy Piping - Passive Core Cooling System

9301250260-05



NOTES:

1. ALL EQUIPMENT NUMBERS, LINES, VALVES, INSTRUMENTS, PUMPS ETC. ARE SHOWN WITHOUT PREFIX RC.
2. REFER TO SYSTEM DESCRIPTION FOR DESCRIPTION OF ALL INSTRUMENTATION, CONTROLS, AND INTERLOCKS.
3. PRESSURIZER SPRAY SCOOP.
4. HEAD GASKET MONITORING CONNECTIONS FURNISHED WITH REACTOR VESSEL.
5. LOCATE CONNECTION IN BOTTOM OF REACTOR COOLANT PIPE, STRAIGHT DOWN. PROVIDE A 3 FOOT SECTION OF 1½ IN. PIPE.
6. LOCATE CONNECTION IN TOP HALF OF THE PIPING.
7. VELOCITY HEAD PROBE INSTALLATION PER REFERENCE (LATER). PROBE INCLUDES INTEGRAL ⅜" FLOW RESTRICTION FOR CLASS 1 TO 2 TRANSITION.
8. STRAP ON (SURFACE MOUNTED) TIC LOCATED AT BOTTOM OF PIPE.
9. HEAD VENT CONNECTION FURNISHED WITH VESSEL HEAD.
10. 31" INSIDE DIAMETER.
11. 22" INSIDE DIAMETER.
12. THERMOWELL MOUNTED FAST RESPONSE TEMPERATURE DETECTORS LOCATED IN SAME VERTICAL PLANE.
13. ⅜" I.D. FLOW RESTRICTOR FOR CLASS 1 TO CLASS 2 TRANSITION SIMILAR TO ARRANGEMENT SHOWN BY NOTE 3, WESTINGHOUSE FLOW DIAGRAM LEGEND DRAWING SHEET 2.
14. RPS - REACTOR PROTECTION SYSTEM.
15. RSP - REMOTE SHUTDOWN PANEL.
16. LOCATE VELOCITY HEAD PROBE AT LEAST 44 INCHES DOWNSTREAM OF THE PZR SPRAY SCOOP, ORIENTED ON THE OPPOSITE SIDE OF THE SCOOP.
17. LEVEL INSTRUMENTATION PROVIDED TO MONITOR RCS WATER LEVEL DURING MID-LOOP OPERATION FOLLOWING SHUTDOWN. LOCATE ONE TAP AT BOTTOM OF HOT LEG. FOR CHANNEL LT 160 TAP SHOULD BE UPSTREAM OF RNS CONNECTION. LOCATE OTHER TAP ON THE TOP OF THE HOT LEG AS CLOSE TO THE SG AS POSSIBLE. TAP AT SG TO BE HIGH-POINT OF TUBING RUN.
18. 4TH STAGE AUTOMATIC DEPRESSURIZATION SIGNAL.
19. QDPS - QUALIFIED DISPLAY PROCESSING SYSTEM.
20. LOCATE CONNECTION BELOW MID PLANE OF PIPING AND SLOPE DOWNWARD TOWARD VALVE TO FACILITATE WATER SEAL AT VALVE INLET.

LEGEND

CANDIDATE LBB PIPING

OTHER HIGH ENERGY PIPING ON NUCLEAR ISLAND (DIAMETER)1

BOUNDARY

SI APERTURE CARD

Also Available On Aperture Card

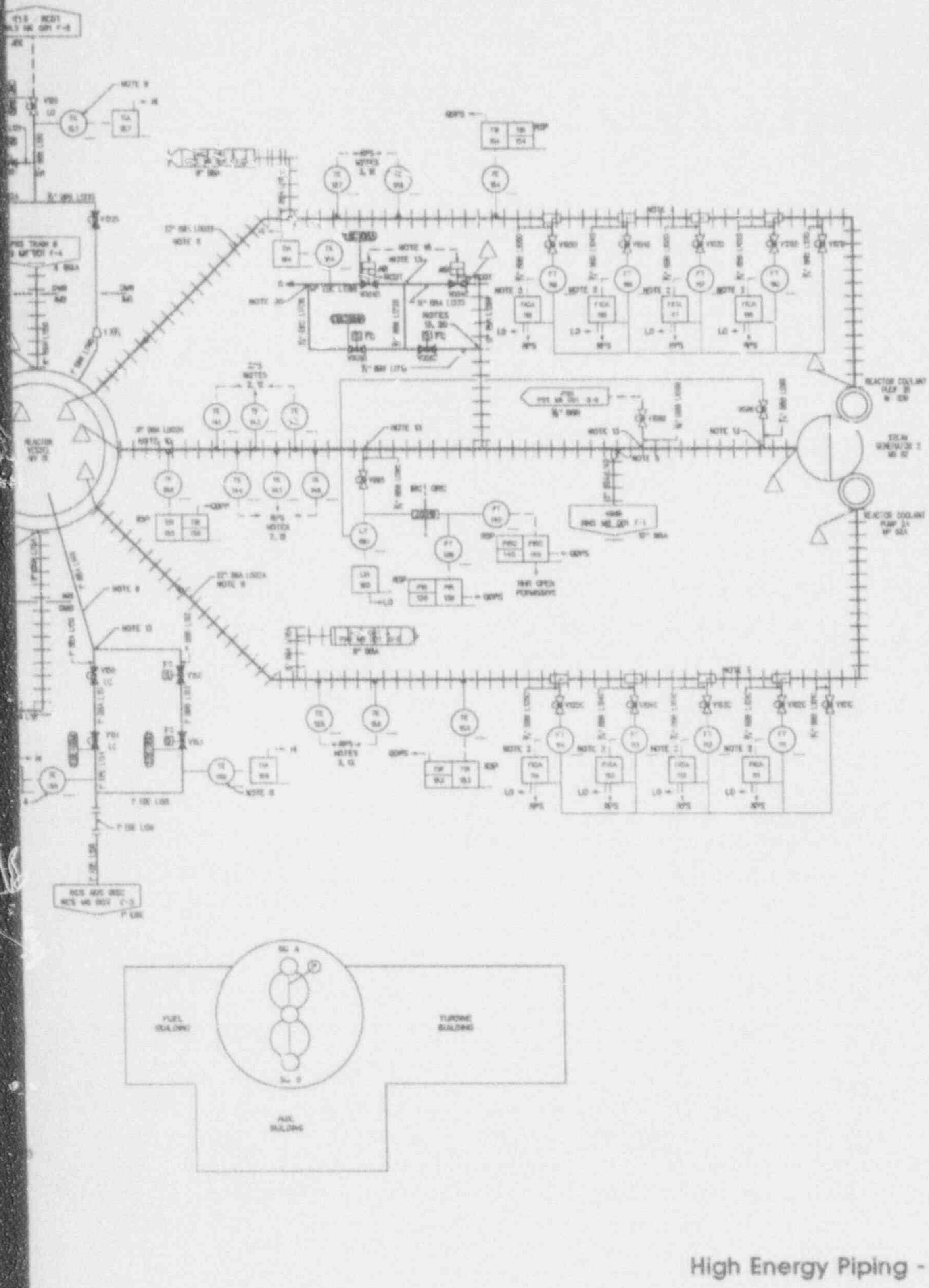


Figure 3E-1 (Sheet 6 of 10)

High Energy Piping - Reactor Coolant System

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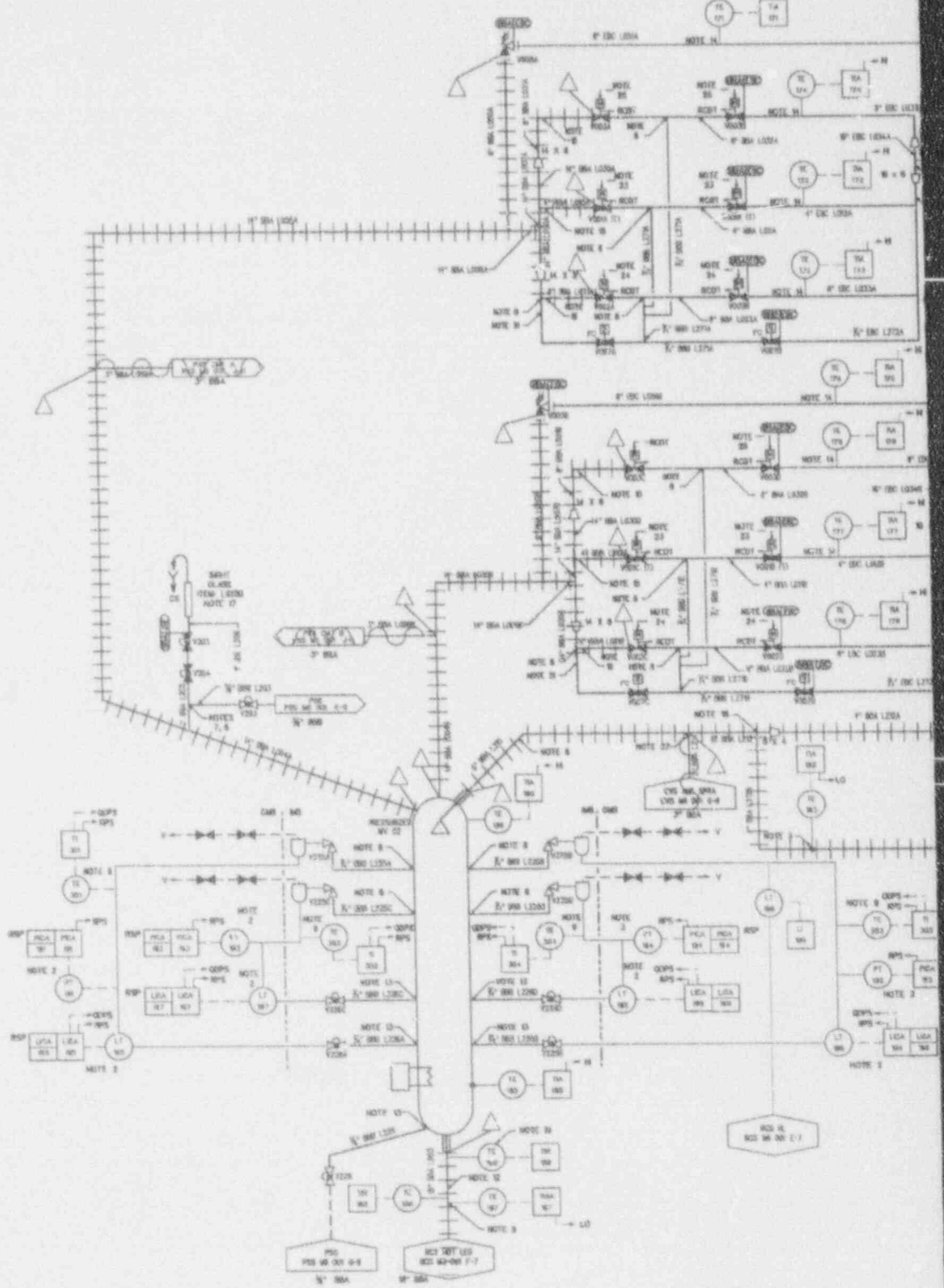

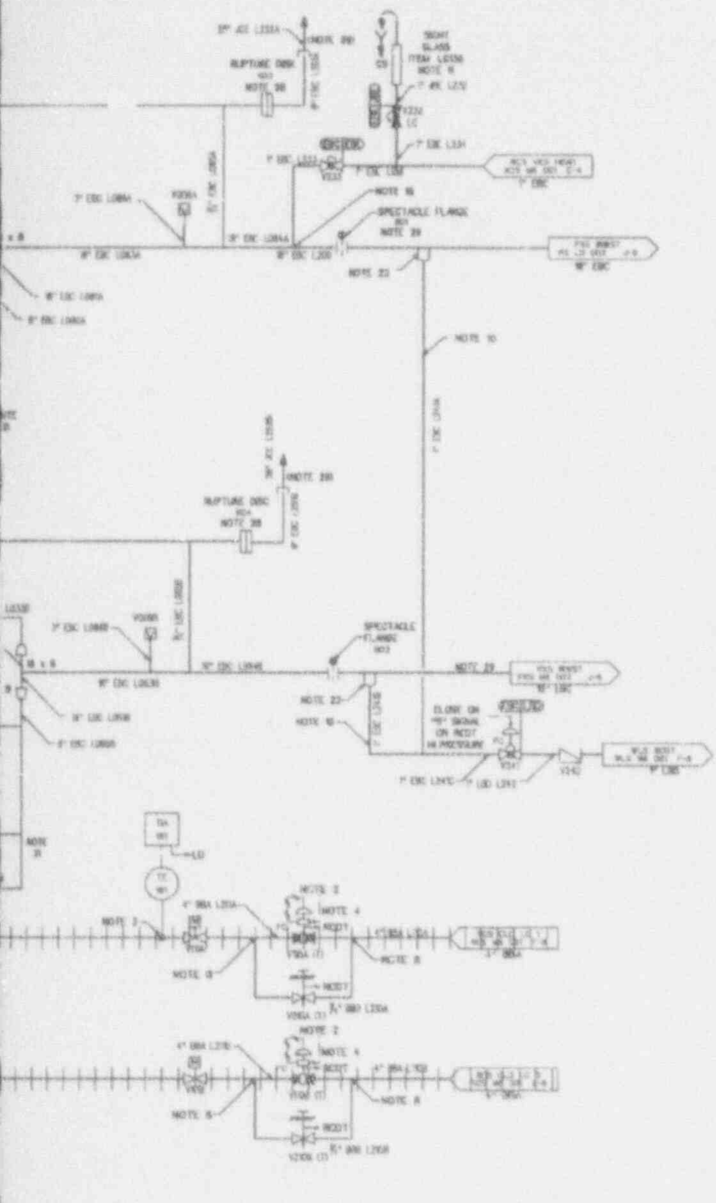

NOTES:

1. ALL EQUIPMENT NUMBERS, LINES, VALVES, INSTRUMENTS, PUMPS, ETC ARE SHOWN WITHOUT PREFIX RC.
2. REFER TO SYSTEM DESCRIPTION FOR DESCRIPTION OF ALL INSTRUMENTATION, CONTROLS, AND INTERLOCKS.
3. LOCATE TEMPERATURE DETECTOR APPROXIMATELY 2 FEET BELOW TEE.
4. LOCATE SPRAY VALVES IN HORIZONTAL PIPING AT OR BELOW THE BOTTOM PRESSURIZER LEVEL TAP. ¾" BYPASS INLET AND OUTLET CONNECTIONS SHOULD BE AS CLOSE AS POSSIBLE TO THE SPRAY VALVES.
5. LOCATE BOTH RTDS ON A HORIZONTAL RUN, APPROXIMATELY MIDWAY BETWEEN LOOP AND PRESSURIZER. TE-187 SHALL BE ON THE BOTTOM OF THE PIPE, AND TE-186 ON TOP.
6. ROUTE SPRAY PIPING TO MINIMIZE VOLUME EXPOSED TO STEAM AND TO PROVIDE A NORMALLY FILLED WATER SEAL EXTENDING AS CLOSE AS POSSIBLE TO THE PRESSURIZER.
7. LOCATE SAMPLE CONNECTION IN NORMALLY STEAM FILLED PORTION OF LINE.
8. PROVIDE .190 IN. (NO. 6 DRILL SIZE) FLOW RESTRICTOR FOR CLASS 1 - CLASS 3 TRANSITION SIMILAR TO ARRANGEMENT SHOWN BY NOTE 3, WESTINGHOUSE FLOW DIAGRAM LEGEND DRAWING SHEET 3.
9. STRAP-ON RTD FOR REFERENCE LEG DENSITY COMPENSATION, ONE CHANNEL LOCATED IN EACH VERTICAL RUN LONGER THAN 2 FEET. FOUR CHANNELS MINIMUM.
10. SLOPE CONTINUOUSLY DOWNWARD TO THE ROOT.
11. SIGHT GLASS PROVIDED FOR VENTING AND FILLING THE REACTOR VESSEL. LOCATE ABOVE REACTOR VESSEL HEAD ELEVATION.
12. SLOPED TO ALLOW COMPLETE DRAINAGE TO THE HOT LEG.
13. ¾" I.D. FLOW RESTRICTOR WITH PIPING FOR CLASS 1 - CLASS 2 TRANSITION AT PRESSURIZER LIQUID SPACE NOZZLES, SIMILAR TO ARRANGEMENT SHOWN BY NOTE 3, WESTINGHOUSE FLOW DIAGRAM LEGEND DRAWING SHEET 3.
14. STRAP-ON RTD LOCATED AT BOTTOM OF PIPE.
15. SLOPE PIPING TO PROVIDE WATER SEAL AT INLET OF DEPRESSURIZATION VALVES.
16. "TEE" JUNCTION SHOULD BE LOCATED CLOSE TO TOP OF THE PRESSURIZER IN THE WATER FILLED PORTION.
17. SIGHT GLASS PROVIDED FOR VENTING AND FILLING THE PRESSURIZER.
18. LOCATE CONNECTION CONSISTENT WITH MOST DIRECT ROUTING OF VESSEL HEAD VENT PIPING TO DEPRESSURIZATION VALVES DISCHARGE HEADERS.
19. LOCATE TE-189 AS CLOSE AS POSSIBLE TO THE PRESSURIZER NOZZLE.
20. RPS - REACTOR PROTECTION SYSTEM.
21. RSP - REMOTE SHUTDOWN PANEL.
22. FABRICATE BY CAPPING 12 INCH BRANCH OF 18 X 18 X 12 TEE, 12 INCHES LONG. LOCATE AS CLOSE AS POSSIBLE TO WALL OF IRWST.
23. 1ST STAGE AUTOMATIC DEPRESSURIZATION SIGNAL.
24. 2ND STAGE AUTOMATIC DEPRESSURIZATION SIGNAL.
25. 3RD STAGE AUTOMATIC DEPRESSURIZATION SIGNAL.
26. SAFETY VALVE DISCHARGE CHAMBER. ORIENT TO DIRECT DISCHARGE FLOW TO OUTSIDE OF PRESSURIZER CUBICLE.
27. CONNECT AUXILIARY SPRAY LINE TO THE BOTTOM OF THE MAIN SPRAY LINE IN THE WATER SEALED PORTION. THIS LINE SHOULD BE ROUTED TO FORM A VERTICAL COLD TRAP APPROXIMATELY 10 FEET LONG BETWEEN THE MAIN SPRAY LINE AND THE AUXILIARY SPRAY CHECK VALVE CS-V085.
28. PROVIDE MEANS FOR RETENTION OF RUPTURE DISC FRAGMENTS.
29. 900 LB SPECTACLE FLANGE INSTALLATION. OPEN SECTION IS NORMALLY IN LINE, BLANK SECTION USED FOR HYDRO-TESTING.
30. QDPS - QUALIFIED DISPLAY PROCESSING SYSTEM.
31. LOCATE CONNECTION BELOW MID PLANE OF PIPING AND SLOPE DOWNWARD TOWARD VALVE TO FACILITATE WATER SEAL AT VALVE INLET.

LEGEND

CANDIDATE LBB PIPING

OTHER HIGH ENERGY PIPING ON NUCLEAR ISLAND (DIAMETER >1")

BOUNDARY

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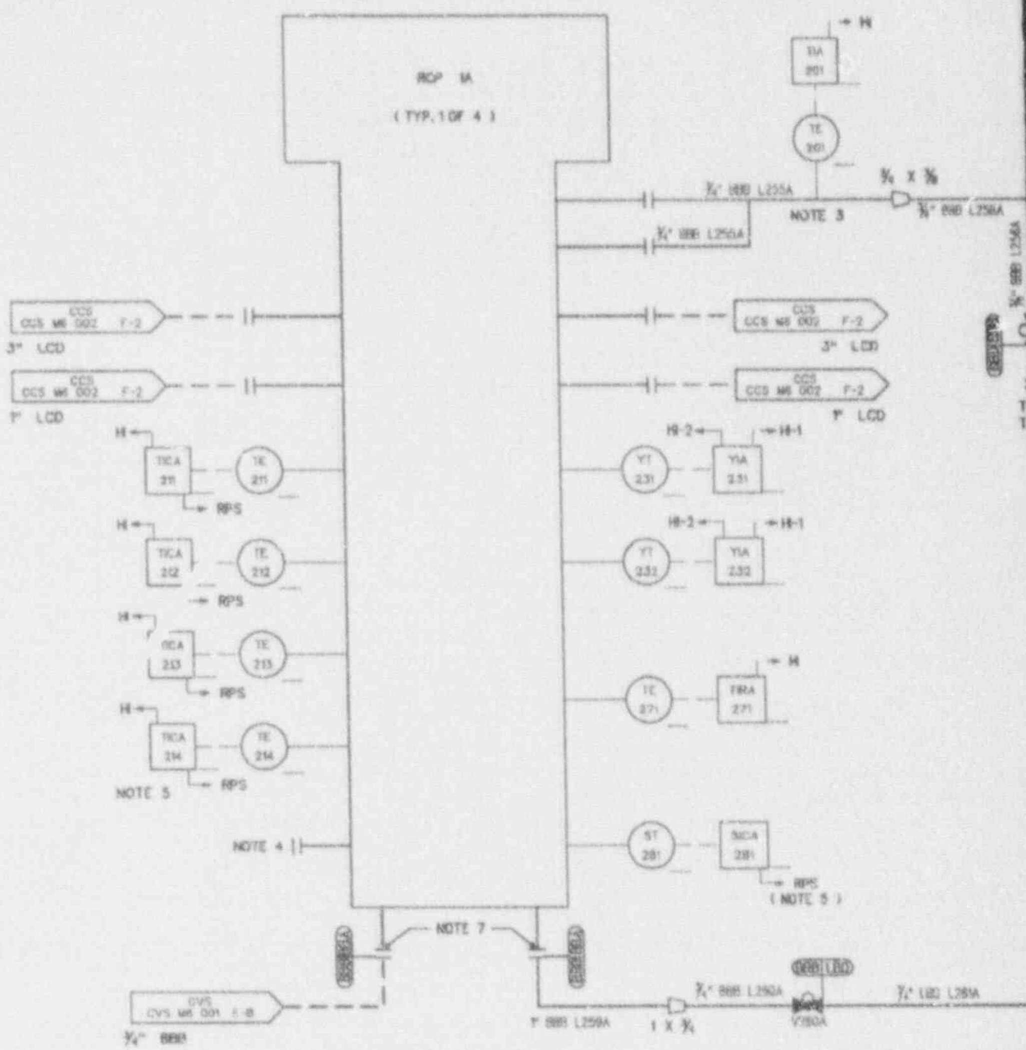
Figure 3E-1 (Sheet 7 of 10)

High Energy Piping - Reactor Coolant System

9301250260-07



REACTOR COOLANT PUMP INSTRUMENTATION				
	RCP A	RCS B	RCP C	RC[illegible]
BEARING WATER TEMPERATURE (TE -)	211 212 213 214	215 216 217 218	221 222 223 224	[illegible]
VIBRATION (YT -)	231 232	241 242	251 252	[illegible]
STATOR TEMPERATURE (TE -)	271	272	273	[illegible]
PUMP SPEED (ST -)	281	285	291	[illegible]
FLANGE LEAK-OFF TEMPERATURE (TE -)	201	202	203	[illegible]

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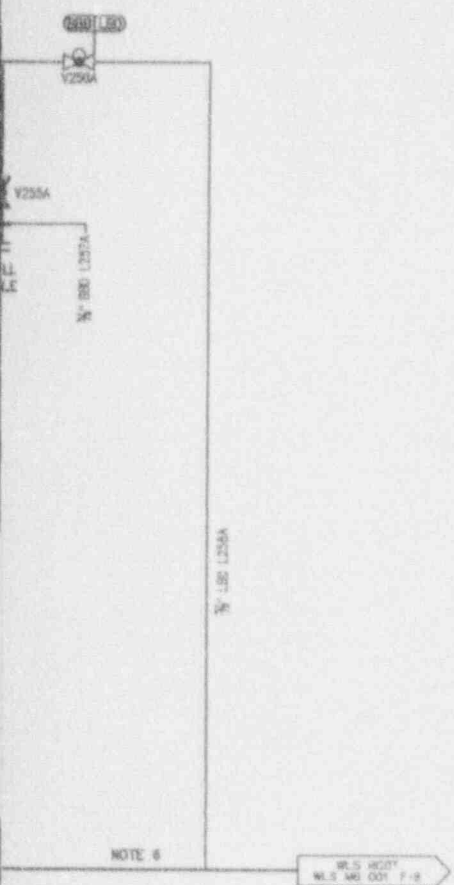


NOTES:

1. ALL EQUIPMENT NUMBERS, LINES, VALVES, INSTRUMENTS, PUMPS ETC) ARE SHOWN WITHOUT PREFIX RC.
2. REFER TO SYSTEM DESCRIPTION FOR DESCRIPTION OF ALL INSTRUMENTATION, CONTROLS, AND INTERLOCKS.
3. STRAP-ON RTD LOCATED AT BOTTOM OF PIPE.
4. INSTALL BLIND FLANGE ON LEAK-OFF CONNECTION FROM STATOR CAP
5. REFER TO SYSTEM SPECIFICATION DOCUMENT FOR DESCRIPTION OF ALL INSTRUMENTATION, INTERLOCKS AND CONTROLS.
6. SLOPE CONTINUOUSLY DOWNWARD TO RCDT.
7. ⅜" FLOW RESTRICTOR FOR CLASS 1 TO CLASS 2 TRANSITION MACHINED INTO PIPING FLANGE SUPPLIED WITH PUMP.
8. SUFFIX LETTERS FOR THE FOLLOWING ITEMS ARE A, B, C AND D, CORRESPONDING TO PUMPS A, B, C AND D RESPECTIVELY

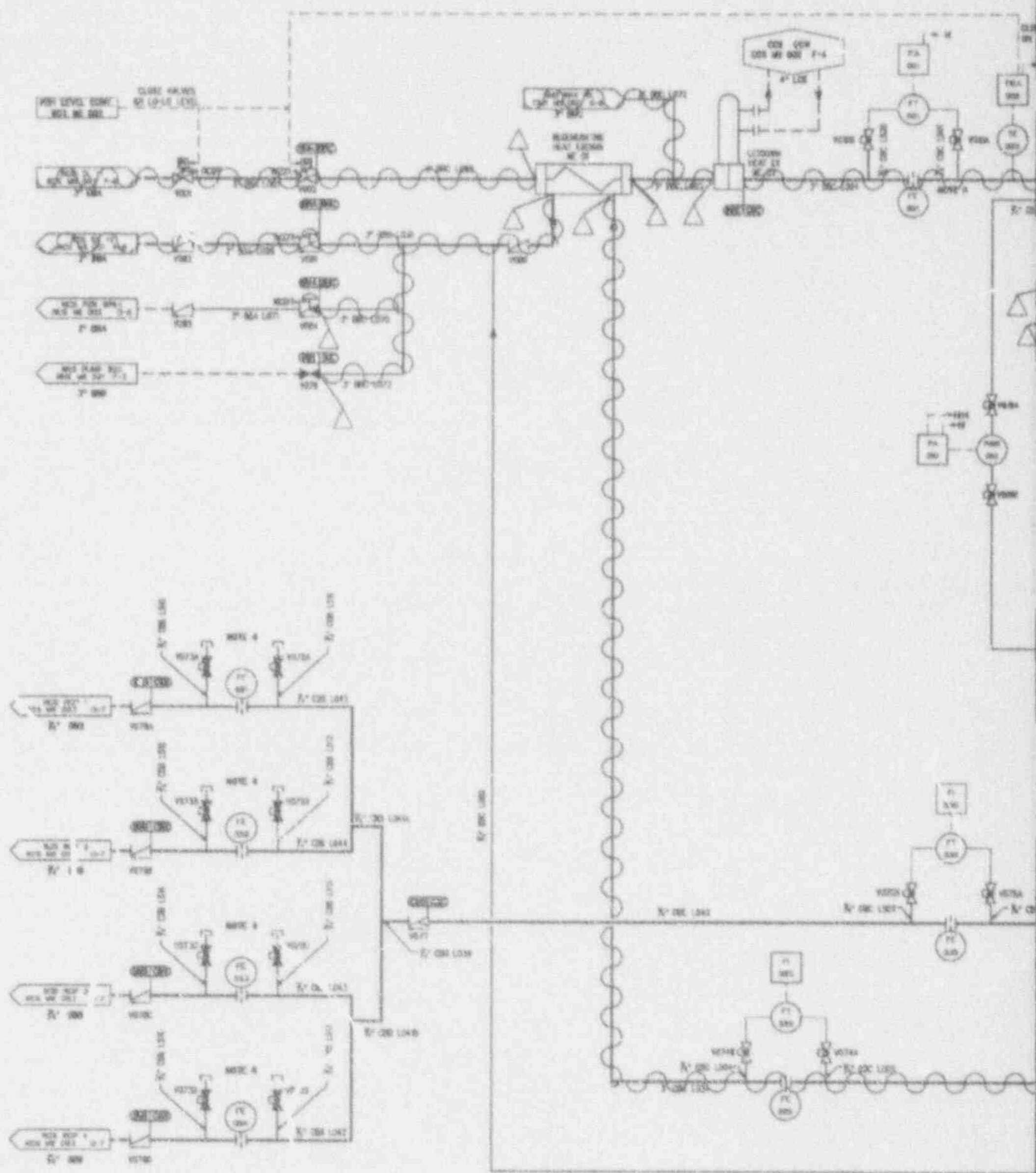
VALVES	LINES	
255	254	258
256	255	259
260	256	260
	257	261

D	SENSOR SUPPLIED WITH PUMP
5	YES
6	YES
7	YES
8	YES
1	YES
2	YES
4	YES
5	YES
4	NO

Figure 3E-1 (Sheet 8 of 10)

High Energy Piping - Reactor Coolant System

9301250260-08



AP600

SI APERTURE CARD

Also Available On Aperture Card

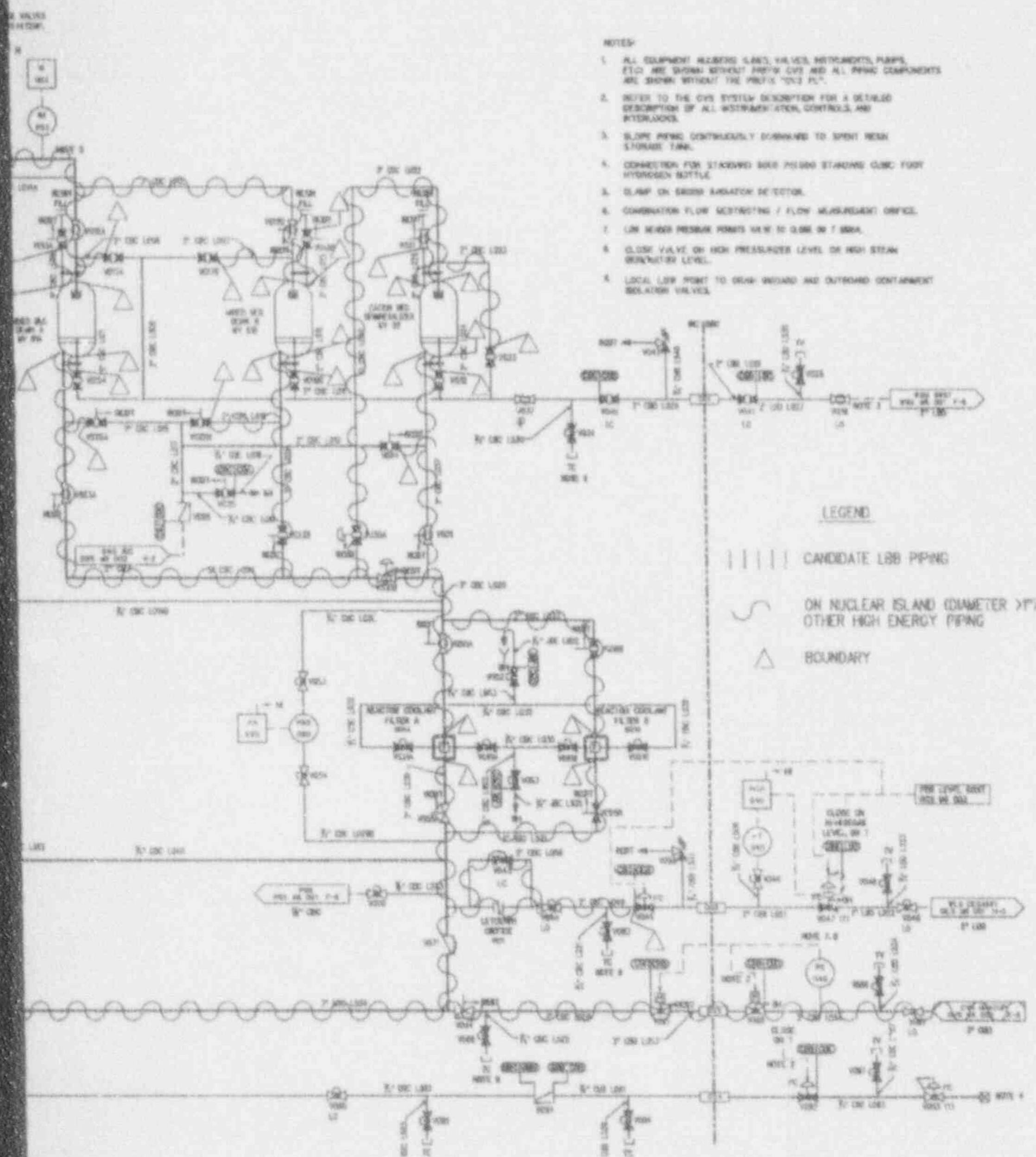
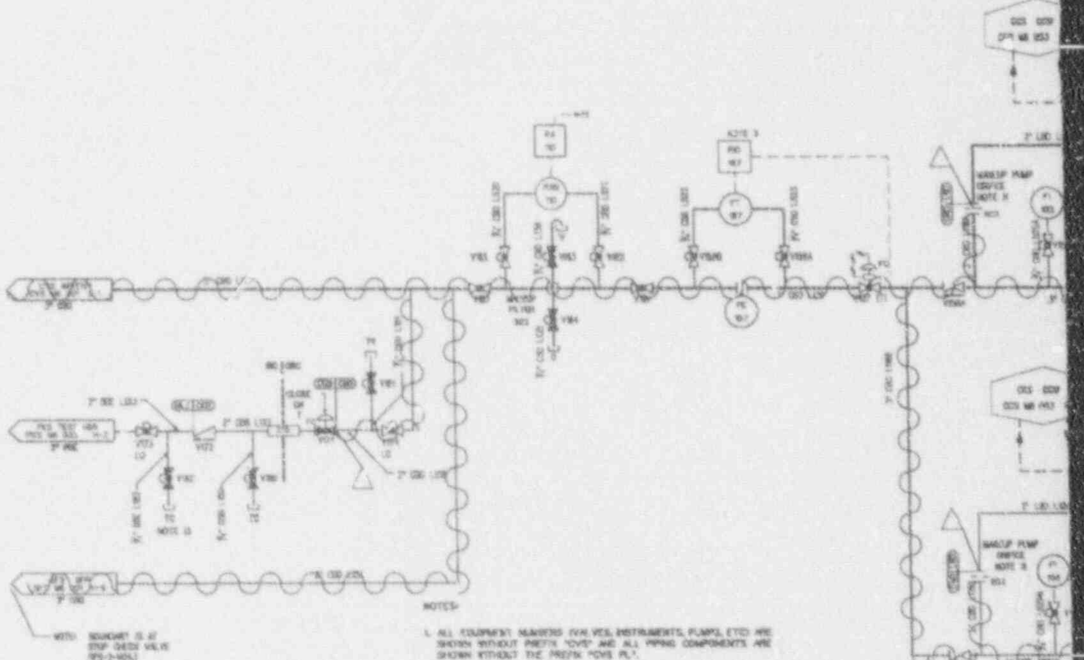


Figure 3E-1 (Sheet 9 of 10)

High Energy Piping - Chemical and Volume Control System

9301250260-09



NOTES:

1. ALL EQUIPMENT NUMBERS (VALVES, INSTRUMENTS, PUMPS, ETC) ARE SHOWN WITHOUT PREFIX "CVS" AND ALL PIPING COMPONENTS ARE SHOWN WITHOUT THE PREFIX "CVS PL".
2. REFER TO THE CVS SYSTEM DESCRIPTION FOR A DETAILED DESCRIPTION OF ALL INSTRUMENTATION, CONTROLS, AND INTERLOCKS.
3. PUMPS STOP & START AND FLOWRATE CONTROLLED BY MAKEUP CONTROL SYSTEM. MAKEUP CONCENTRATION IS ALSO CONTROLLED BY THE MAKEUP CONTROL SYSTEM BY POSITIONING VALVE V115.
4. LOOP SEAL TO EXTEND 12 INCHES BELOW AND ABOVE OVERFLOW NOZZLE.
5. LIMIT MAKEUP STEAM SUPPLY PRESSURE TO 75 PSIG.
6. BORIC ACID BATCHING TANK MUST BE LOCATED TO ALLOW GRAVITY DRAIN FROM BATCHING TANK INTO THE BORIC ACID TANK.
7. DURING PREOPERATIONAL FLUSHING A MESH STRAINER IS INSTALLED TEMPORARILY IN THE SPOOL PIECE. A LOCAL PRESSURE GAGE IS INSTALLED AT THE ¾ INCH CAPPED LINE TO MONITOR STRAINER PERFORMANCE. THESE COMPONENTS MUST BE REMOVED UPON COMPLETION OF FLUSHING.
8. VALVE FAILS WITH PUMP SUCTION ALIGNED TO BAT. ALIGNS TO BAT ON EITHER A REACTOR TRIP, A SOURCE RANGE FLUX DOUBLING SIGNAL, LOOP, OR AN "S" SIGNAL.
9. LOCATE CONNECTION AT SAME ELEVATION AS SFS CONNECTION.
10. TEMPORARY CONNECTION FOR HYDROSTATIC TEST PUMP.
11. BYPASS ORIFICE PROVIDED BY PUMP MANUFACTURER. PROVIDE MINIMUM OF 5 FEET STRAIGHT PIPE DOWNSTREAM OF ORIFICE.
12. CLOSE ON EITHER A REACTOR TRIP SIGNAL, A SOURCE RANGE FLUX DOUBLING SIGNAL, A LOOP, OR AN "S" SIGNAL.
13. LOCAL LOW POINT TO DRAIN INBOARD AND OUTBOARD CONTAINMENT ISOLATION VALVES.

SI
APERTURE
CARD

Also Available On
Aperture Card

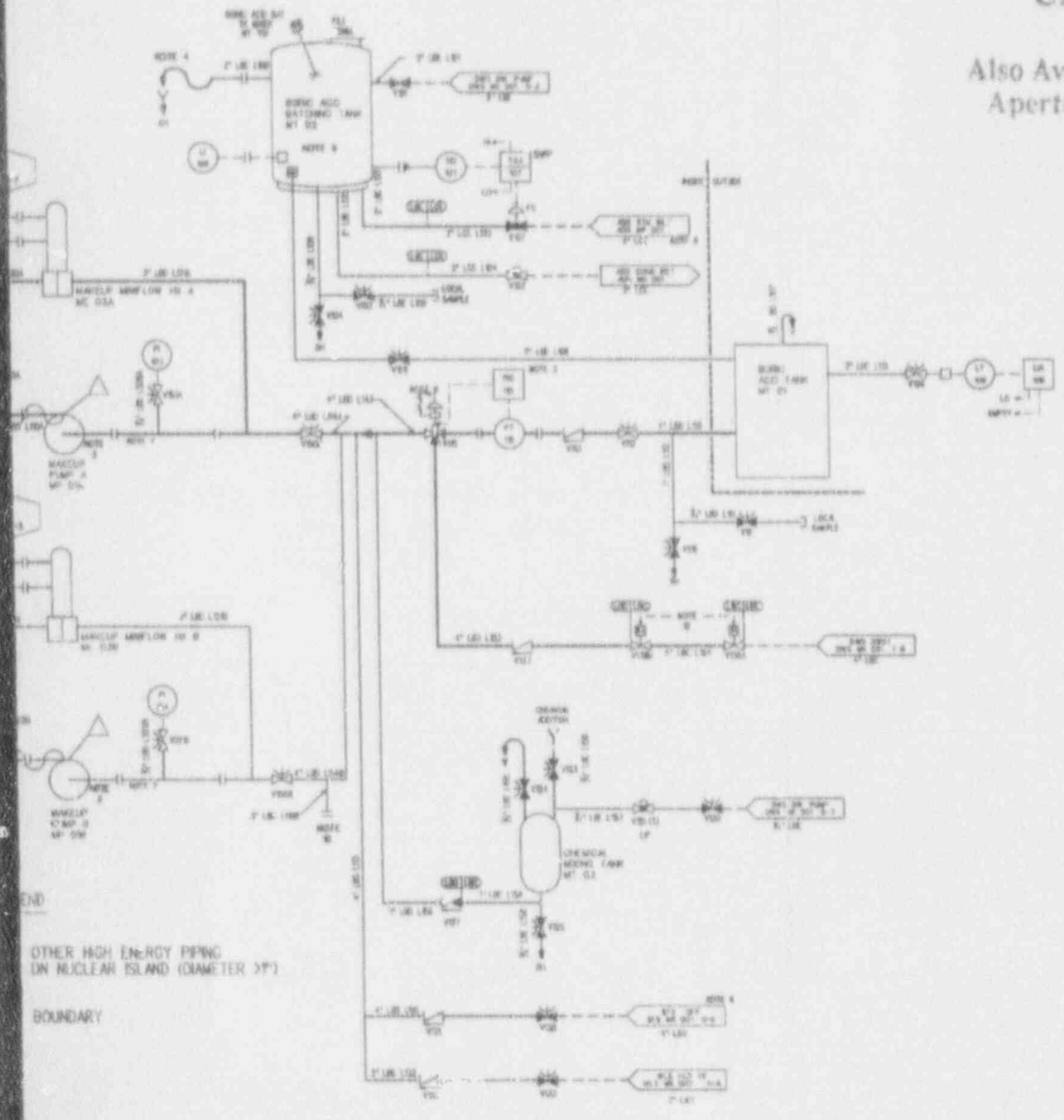


Figure 3E-1 (Sheet 10 of 10)

High Energy Piping - Chemical and Volume Control System

9301250260-10

Question 210.7

Section 3.7.3.2 of the SSAR, "Determination of Number of Earthquake Cycles," states that for cyclic motion due to earthquakes smaller than the SSE, subsystems sensitive to fatigue are evaluated by assuming two seismic events, each resulting in 10 full-stress cycles with magnitude equal to 50% of the calculated SSE response for structures and components. Discuss the technical justification for the selection of these values for the AP600 design.

Response:

The number of earthquake cycles will be increased in response to this question. The fatigue criterion for ASME components on recent plants with OBE events is five events with 10 high-stress cycles per event. For the AP600 plant there is no OBE event, and systems sensitive to fatigue are evaluated for seismic events with an amplitude not less than one-third of the SSE amplitude. The number of cycles is calculated based on IEEE-344-1987 to provide the equivalent fatigue damage of two SSE events with 10 high-stress cycles per event. This approach provides the equivalent fatigue damage of five OBE events with 10 cycles per event when the OBE amplitude is equal to 0.693 times the SSE. Therefore, the AP600 fatigue criterion is similar to the design criterion for recently licensed plants.

The relation between amplitude and cycles is shown below.

AMPLITUDE/SSE	EQUIVALENT CYCLES TO 1.0 SSE	TOTAL CYCLES
1.0	1.0	10
0.693	2.5	25
0.5	5.66	57
0.333	15.59	156

For ASME components, the fatigue damage caused by the equivalent of two SSEs is used:

for example, 5 events, 1/3 SSE, 63 cycles per event

For non-ASME components, two SSEs are used for consistency:

for example, 5 events, 1/2 SSE, plus 1 event, 1.0 SSE

The earthquake design requirements for ASME components are summarized below.

The effect on the ASME design for Class 1, 2, and 3 components and core support structures is described below:

Option A: Not more than 25 total seismic cycles.

If there are not more than 25 seismic stress cycles, then there is no seismic contribution to the cumulative usage factor (CUF) provided that:

1. The Level C stress limits are satisfied for the SSE stresses combined with normal 100 percent power loads.
2. The total number of Level C cycles from all events is not more than 25.

This could be applicable to portions of the component where the seismic load is defined as two SSE events with 10 high-stress cycles for each event.

Option B: More than 25 total seismic cycles.

If there are more than 25 seismic stress cycles, then there is a seismic contribution to the CUF. For this case the following apply:

1. The Level D stress limits are satisfied for the SSE stresses combined with normal 100 per cent power loads.
2. The Level B stress limits are satisfied for one-third of the SSE stresses combined with normal 100 percent power loads.
3. The CUF calculation is based on five earthquake events with a magnitude of one-third of the SSE and with 63 cycles per earthquake.

The effect on the ASME design for NF-component supports is described below:

Component supports shall satisfy the following:

1. Level D stress limits for the SSE stresses combined with normal 100 percent power loads..

The effect on the ASME design for piping is described below.

Class 1 Piping:

1. There are no earthquake loads for Level B primary stress service limits.

2. The Level B Equations 10, 11, and 14 calculations are based on five earthquake events with a magnitude of one-third of the SSE (including seismic anchor motions) and with 63 cycles per earthquake.

3. When Equation 10 cannot be satisfied, the following is met (in addition to Equations 11 and 12):

   S(sam) = C2*DO/2/I*(M1 + M2) less than or equal to 6.0 Sm

   M1 = range of thermal moments per Equation 12

M1 + M2 = larger of M1 plus one-half the range of SSE seismic anchor motion moments, or full range of SSE seismic anchor motion moments

Class 2 and 3 piping:

1. There are no earthquake loads for Level A, Level B, or Design Service Limits.

2. The following shall be met:

   P*DO/t*0.25 + 0.75*i*MA/Z + i/Z* (MC + M2) less than or equal to SA + 4.0 Sh

   MA, MC, SA, Sh, P, DO, t, i, and Z per Equation 11

   MC +M2 = larger of MC plus one-half the range of SSE seismic anchor motion moments, or full range of SSE seismic anchor motion moments

The SSAR will be revised as follows:

SSAR Revision:

3.7.3.2 Determination of Number of Earthquake Cycles

~~Seismic Category I structures, systems, and components are designed for one occurrence of the safe shutdown earthquake (SSE). In addition, subsystems sensitive to fatigue are evaluated for cyclic motion due to earthquakes smaller than the safe shutdown earthquake. Using linear elastic methods, these effects are considered by inclusion of two seismic events each resulting in 10 full stress cycles with magnitude equal to 50 percent of the calculated safe shutdown earthquake response for structures and components. The seismic input to the subsystem is taken as 50 percent of the safe shutdown earthquake input for subsystems qualified by non-linear analysis methods or by testing.~~

~~When seismic qualification is based on dynamic or equivalent static load testing for structures, systems, or components containing mechanisms that must change position in order to function, operability testing is performed for the safe shutdown earthquake preceded by two smaller earthquakes of magnitude equal to 50 percent of the safe shutdown earthquake.~~

Seismic Category I structures, systems, and components are evaluated for one occurrence of the safe shutdown earthquake (SSE). In addition, subsystems sensitive to fatigue are evaluated for cyclic motion due to earthquakes smaller than the safe shutdown earthquake. Using analysis methods, these effects are considered by inclusion of seismic events with an amplitude not less than one-third of the SSE amplitude. The number of cycles is calculated based on IEEE-344-1987 to provide the equivalent fatigue damage of two SSE events with 10 high-stress cycles per event. Typically, there are five seismic events with an amplitude equal to one-third of the SSE response. Each event has 63 high-stress cycles.

When seismic qualification is based on dynamic testing for structures, systems, or components containing mechanisms that must change position in order to function, operability testing is performed for the safe shutdown earthquake preceded by one or more earthquakes. The number of preceding earthquakes is calculated based on

IEEE-344-1987 to provide the equivalent fatigue damage of one SSE event. Typically, the preceding earthquake is one SSE event or five one-half SSE events.

(Paragraphs 18 and 19 of Subsection 3.9.3.1.2)

~~The effects of seismic events smaller than a safe shutdown earthquake are considered for components that must be evaluated for fatigue. For the purposes of fatigue evaluations, the smaller earthquake event is treated as a Level C event and the number of stress cycles is counted against the total of 25 cycles, which may be excluded from consideration in the fatigue analysis. The number of cycles of strong motion response for the smaller seismic event is postulated as 10 for each of two events.~~

~~The response magnitude of the smaller earthquake is equal to half the calculated response due to the safe shutdown earthquake.~~

The effects of seismic events on the design of components other than piping are considered in one of the following ways. When the SSE event is evaluated as a Level C event, there is no seismic contribution to the fatigue evaluation. In this case the SSE accounts for 20 cycles, corresponding to two SSE events with 10 stress cycles per event, of the 25 Level C stress cycles. Alternatively, a seismic event with an amplitude equal to one-third of the SSE is evaluated as a Level B event. In this case, the seismic contribution to the fatigue evaluation is based on five seismic events with an amplitude of one-third the SSE and with 63 cycles per event. The seismic evaluation of piping components is discussed in Subsection 3.9.3.1.5.

(Paragraph 9 of Subsection 3.9.3.2.2)

~~When seismic qualification is based on dynamic or equivalent static load testing for structures, systems or subsystems that contain mechanisms that must change position in order to function, operability testing is performed for the safe shutdown earthquake preceded by two smaller earthquakes of a magnitude equal to 50 percent of the safe shutdown earthquake.~~

When seismic qualification is based on dynamic or equivalent static testing for structures, systems, or subsystems that contain mechanisms that must change position in order to function, operability testing is performed for the safe shutdown earthquake preceded by one or more earthquakes. The number of preceding earthquakes is calculated based on IEEE-344-1987 to provide the equivalent fatigue damage of one SSE event. Typically, the preceding earthquake is one SSE event or five one-half SSE events.

(Paragraph 2 of Subsection 3.9.4.3)

~~For each loading combination the appropriate stresses due to pressure, component weight, hydraulic forces, thermal gradients, and seismic dynamic forces are evaluated and demonstrated to be less than the applicable stress limits. The cyclic stresses are combined with constant stresses to evaluate the fatigue usage due to cyclic loads. The transients used in the evaluation of cyclic loads are described in Subsection 3.9.1. The results of the stress evaluation are documented in a component stress report, as required by the ASME Code.~~

For each loading combination the appropriate stresses due to pressure, component weight, external loads, hydraulic forces, thermal gradients, and seismic dynamic forces are evaluated and demonstrated to be less than the applicable stress limits. The cyclic stresses are combined with constant stresses to evaluate the fatigue usage due to cyclic loads. The transients used in the evaluation of cyclic loads are described in Subsection 3.9.1. The effect of seismic events is considered in one of the following ways. When the SSE event is evaluated as a Level C event, there is no seismic contribution to the fatigue evaluation. In this case the SSE accounts for 20 cycles, corresponding to two SSE events with 10 stress cycles per event. of the 25 Level C stress cycles. Alternatively, a seismic event with an amplitude equal to one-third of the SSE is evaluated as a Level B event. In this case, the seismic contribution to the fatigue evaluation is based on five seismic events with an amplitude of one-third the SSE and with 63 cycles per event. The results of the stress evaluation are documented in a component stress report, as required by the ASME Code.

(Tables 3.9-3, 3.9-4, 3.9-5, 3.9-6, 3.9-7, 3.9-8, and 3.9-11)

Table 3.9-3

Loadings for ASME Class 1, 2, 3, CS and Supports
Primary Stress Producing Loads

Load	Description
P	Internal design pressure
PMAX	Peak pressure
DW	Dead weight
XL	Other specified external loads associated with the various service conditions(e.g., piping loads on nozzles)
SSE	Safe shutdown earthquake (inertia portion)
E	Earthquake smaller than SSE (inertia portion)
FV	Fast valve closure
RVC	Relief valve - closed system (transient)
RVOS	Relief valve - open system (sustained)
RVOT	Relief valve - open system (transient)
SRV	Dynamic effects due to safety-relief valve discharge
DU	Other transient dynamic event associated with Level B (Upset) service conditions
DE	Transient dynamic event associated with Level C (Emergency) service conditions
DF	Transient dynamic event associated with Level D (Faulted) service conditions during which, or following which, the piping system being evaluated must remain intact. This includes postulated pipe rupture events.

Table 3.9-4

Additional Loadings for ASME Class 1, 2, 3, CS and Supports

These loads are treated as loads producing secondary stress in the analysis of piping and as loads producing primary stress in the analysis of components and supports.

Load	Description
SSES	Seismic anchor motion portion of SSE
ES	Seismic anchor motion of earthquake smaller than SSE
TH	Thermal loads for the various service conditions
TNU	Service Level A and B (Normal and Upset) plant condition thermal loads, including thermal stratification and thermal cycling
TN	Service Level A (Normal) plant condition thermal loads
TU	Service Level B (Upset) plant condition thermal loads
TE	Service Level C (Emergency) plant condition thermal loads
TF	Service Level D (Faulted) plant condition thermal loads

Additional Loadings for ASME Class 1, 2, 3 Supports

Load	Description
HTDW	Hydrostatic test dead weight
SWE	Self weight excitation (Effect of the acceleration of the support mass on the support)

Table 3.9-5

Minimum Design Loading Combinations for ASME Class 1, 2, 3 and CS Systems and Components

Condition	Design Loading Combinations
Design	P + DW + XL
Level A Service	PMAX(1) + DW + XL
Level B Service	PMAX + DW + RVC + XL
	PMAX + DW + RVOS + XL
	PMAX + DW + RVOT + XL
	PMAX + DW + SRV(7) + XL
	PMAX + DW + DU + XL
	PMAX + DW + FV + XL
	PMAX + DW + SRSS(E + ES)(2)(8) + XL
Level C Service	PMAX + DW + DE + XL
	~~PMAX + DW + SRSS(2)(E + ES)~~
	PMAX + DW + SRSS (SSE + SSES)(8) + XL
Level D Service(6)	PMAX + DW + SRSS (SSE + SSES) + XL
	PMAX + DW + SRSS (SSE + SSES + RVC) + XL
	PMAX + DW + SRSS (SSE + SSES + SRV) + XL
	PMAX + DW + DF + XL
	PMAX + DW + SRSS (SSE + SSES + DF)(5) + XL
	PMAX + DW + RVOS + SRSS (SSE + SSES) + XL
	PMAX + DW + SRSS (SSE + SSES + RVOT) + XL

See following page for notes.

Notes for Table 3.9-5

1. The values of PMAX in the load combinations may be different for different levels of service conditions as provided in the design transients. For earthquake loadings PMAX is equal to normal operating pressure at 100 percent power.

2. SRSS equals the square root of the sum of the squares.

3. Design mechanical loads, such as the nozzle reactions associated with thermal expansion of piping systems, shall be combined with other loads in the loading combination expressions.

4. Appropriate loads due to static displacements of the steel containment vessel and building settlement should be added to the loading combinations expressions for ASME Code, Section III, Class 2 and 3 systems.

5. ASME Code, Section III, Class 1 and Class CS components are designed to the Level D service limits for this loading combination where DF represents a loss of coolant accident pipe rupture.

6. In combining earthquake loads and consequent plant transients, the timing and causal relationships that exist between DU, DE, DF, SSE, RVOS, RVOT, RVC, and SRV are considered for determination of the appropriate load combinations.

7. The pressurizer safety valve discharge is a Level C service condition.

8. Either of these two loading combinations shall be used.

AP600

Table 3.9-6

Minimum Design Loading Combinations for ASME Class 1 Piping

Condition	Design Loading Combinations for Primary Stress	Other Loadings(6)
Design	P + DW	---
Level A Service	--	P, RVC, RVOS, RVOT, SRV, FV, TN
Level B Service	PMAX(1) + DW + RVC PMAX + DW + RVOS PMAX + DW + SRV(2) PMAX + DW + DU PMAX + DW + FV PMAX + DW + RVOT	P, RVC, RVOS, RVOT, SRV, FV DU, TU, E, ES(3)
Level C Service	PMAX + DW + DE ~~PMAX + DW + E~~	TE(3) ~~ES(3)~~
Level D Service(4)	PMAX + DW + SRSS(5) (SSE + RVC) PMAX + DW + DF PMAX + DW + SRSS (SSE + SRV) PMAX + DW + SSE + RVOS PMAX + DW + SSE PMAX + DW + SRSS (SSE + RVOT) PMAX + DW + SRSS (SSE + DF)(7)	TF(3) SSES(3)

Notes:

1. The values of PMAX in the load combinations may be different for different levels of service conditions as provided in the design transients. For earthquake loadings PMAX is equal to normal operating pressure at 100 percent power.
2. Pressurizer safety valve discharge is classified as a Level C event.
3. See Table 3.9-911 for stress criteria.
4. The timing and causal relationships between safe shutdown earthquake and RVOS, RVOT, RVC, and SRV are considered.
5. SRSS equals the square root of the sum of the squares.
6. Other loadings are used in analyses that include secondary stresses, which include analysis of cyclic fatigue.
7. This load combination is for primary loop piping only.

Table 3.9-7

Minimum Design Loading Combinations for ASME Class 2 and 3 Piping

Condition	Design Loading Combinations for Primary Stress	Other Loadings[6]
Design	P + DW	---
Level A Service or Level B Service	PMAX[1] + DW + RVC PMAX + DW + RVOS PMAX + DW + SRV PMAX + DW + DU PMAX + DW + FV PMAX + DW + RVOT	PMAX + DW + TNU[2]
Level C Service	PMAX + DW + DE ~~PMAX + DW + E~~	TE[3] ~~ES[3]~~
Level D Service[4]	PMAX + DW + SRSS[5](SSE + RVC) PMAX + DW + DF PMAX + DW + SRSS(SSE + SRV) PMAX + DW + SSE PMAX + DW + SSE + RVOS PMAX + DW + SRSS(SSE + RVOT)	TF[3] SSES[3]

Notes:

1. The values of PMAX in the load combinations may be different for different levels of service conditions as provided in the design transients. For earthquake loadings PMAX is equal to normal operating pressure at 100 percent power.
2. Appropriate loads due to static displacements of the steel containment vessel and building settlement should be added to the loading combinations expressions for Class 2 and 3 systems.
3. See Table 3.9-~~9~~11 for stress criteria.
4. In combining earthquake loads and consequential plant transients, the timing of the loads is appropriately considered.
5. SRSS equals the square root of the sum of the squares.
6. Other loadings are used in analyses that include secondary stresses.

Table 3.9-8

Minimum Design Loading Combinations for Supports for ASME Class 1, 2, 3 Piping and Components

Condition	Design Loading Combinations(2)
Design	DW
Level A Service	DW + TH
Level B Service	DW + TH + SRSS(RVC + SWE)
	DW + TH + SRSS(RVOT + SWE)
	DW + TH + SRSS(DU + SWE)
	DW + TH + SRSS(SRV + SWE)(6)
	DW + TH + RVOS
	DW + FV
Level C Service	DW + TH + SRSS(1) (DE + SWE)
	~~DW + TH + SRSS(E + ES + SWE)~~
Level D Service(5)	DW + TH + SRSS (SSE + SSES + SWE)
	DW + TH + SRSS (SSE + SSES + RVC + SWE)
	DW + TH + SRSS (SSE + SSES + RVOT + SWE)
	DW + TH + RVOS + SRSS (SSE + SSES + SWE)
	DW + TH + SRSS (DF + SWE)(4)
	DW + TH + SRSS (SSE + SSES + SRV + SWE)
	DW + TH + SRSS (SSE + SSES + DF)(3)
Hydrostatic Test	HTDW

Notes:
1. SRSS equals the square root of the sum of the squares.
2. Appropriate loads due to static displacement of the steel containment vessel and building settlement should be added to the loading combinations expressions for Class 2 and 3 systems.
3. Class 1 component supports are designed for this condition, where DF represents a limiting pipe rupture.
4. For piping supports, an acceptable alternative is to permit support failure and evaluate the consequences on piping system integrity and operability.
5. In combining earthquake loads and consequential plant transients, the timing of the loads is appropriately considered.
6. The pressurizer safety valve discharge is a Level C service condition.

Table 3.9-11

Interim Stress Criteria for ASME Class 1, 2, and 3 Piping Loads (Secondary Stress Producing Loads)

Loading	Stress Criteria
TE, TF	Equation 10a, ASME III, NC 3653.2(1) (Allowable is 3.0 S_c)
~~ES~~	~~$C_2 M_{ES} D_o/2I \leq 4.5 S_m$ and $P_{ES}/A \leq 1.125 S_m$~~
~~SSE~~	~~$C_2 M_{SSES} D_o/2I \leq 6.0 S_m$ and $P_{SSES}/A \leq 1.5 S_m$~~
	~~Where:~~
	~~C_2 = ASME III, NB 3600, stress index~~ ~~M_{SSES} = Resultant moment for SSES~~ ~~P_{SSES} = Axial force from SSES~~
ES,SSES	When Equation 10 cannot be satisfied, including ES loading, the following is met for Class 1 piping:
	S(sam) = C2*DO/2/I*(M1 + M2) less than or equal to 6.0 Sm
	Where: M1 = range of thermal moments per Equation 12 (TN,TU) M1 + M2 = larger of M1 plus one-half the range of SSE seismic anchor motion moments (SSES), or full range of SSE seismic anchor motion moments
	For Class 2 and Class 3 piping:
	P*DO/t*0.25 + 0.75*i*MA/Z + i/Z* (MC + M2) less than or equal to SA ≤ 4.0 Sh
	where: MA, MC, SA, Sh, P, DO, t, i, and Z per Equation 11 MC +M2 = larger of MC plus one-half the range of SSE seismic anchor motion moments (SSES), or full range of SSE seismic anchor motion moments

Notes:

1. Applicable to Level C and Level D plant events for which the piping system must maintain an adequate fluid flow path.

(Fourth paragraph of Appendix 3D, Subsection 3D.4.1.2)

~~Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Plants" - The guide prescribes acceptable values of damping used in elastic modal dynamic seismic analysis of seismic Category I structures, systems, and components. The AP600 equipment qualification program is based on Regulatory Guide 1.61 and on values considered to be acceptable based on past NRC acceptances. The safe shutdown earthquake (SSE) damping values used for the qualification of mechanical and electrical equipment are listed in Table E.1 of Attachment E.~~

Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Plants" - The guide prescribes acceptable values of damping used in elastic modal dynamic seismic analysis of seismic Category I structures, systems, and components. The AP600 equipment qualification program is based on Regulatory Guide 1.61 and on values considered to be acceptable based on past NRC acceptances. The safe shutdown earthquake (SSE) damping values used for the qualification of mechanical and electrical equipment are listed in Table 3.7.1-1 of Chapter 3.

(First paragraph of Subsection E.4.4 of Attachment E to Appendix 3D)

~~The AP600 makes use of a small earthquake having the intensity of one-half of the safe shutdown earthquake at the safety-related equipment mounting location to simulate the fatigue effects of smaller earthquakes that may occur before the postulated safe shutdown earthquake. These small earthquakes correspond to the operating basis earthquakes (OBEs) referenced in IEEE 344-1987. Two of these small earthquakes are used to vibrationally age the equipment before the safe shutdown earthquake. When qualification by analysis is used, each earthquake (1/2 SSE, or SSE) is assumed to contain 10 peak stress cycles. These stress cycles are used to verify that the equipment is not subject to failure due to low-cycle fatigue.~~

The AP600 makes use of a small earthquake having the intensity of one-half of the safe shutdown earthquake at the safety-related equipment mounting location to simulate the fatigue effects of smaller earthquakes that may occur before the postulated safe shutdown earthquake. These small earthquakes correspond to the operating basis earthquakes (OBEs) referenced in IEEE 344-1987. When qualification by testing is used, five of these small earthquakes are used to vibrationally age the equipment before the safe shutdown earthquake. When qualification by analysis is used, two safe shutdown earthquake events are used to simulate the fatigue aging effects. Each event contains 10 peak cycles. These stress cycles are used to verify that the equipment is not subject to failure due to low-cycle fatigue.

(Second, third, and fourth paragraphs of Subsection E.5.1 of Attachment E to Appendix 3D)

~~Furthermore, the test input simulates the multidirectional nature of earthquake. The preferred method for meeting this requirement is to the use a triaxial test table capable of pr g three statistically independent, orthogonal input motions. In this case the seismic testing consists of two 1/2 safe shutdown earthquake tests and one safe shutdown earthquake test in one orientation.~~

~~Using a biaxial test table is acceptable if it is justified that the horizontal and vertical test inputs conservatively simulate the three dimensional nature of the seismic event. One acceptable approach is to mount the equipment on the test table with its front-to-back axis oriented at 45 degrees to the horizontal drive axis and scale the horizontal component of the input by a factor of the square root of two. Statistically independent inputs are preferred and, if used, the test can be performed in two stages, with the equipment rotated 90 degrees about the~~

~~vertical axis. In this case, the two 1/2 safe shutdown earthquake inputs need to be applied only in the first orientation.~~

~~If a dependent biaxial test table is used, the test is performed in four stages. The first stage involves two 1/2 safe shutdown earthquake tests and one safe shutdown earthquake test in the first orientation. The second, third, and fourth orientations are obtained by successively rotating the equipment 90 degrees clockwise from its previous position. One safe shutdown earthquake test is performed in each of the last three orientations.~~

Furthermore, the test input simulates the multidirectional nature of the earthquake. The preferred method for meeting this requirement is to use a triaxial test table capable of producing three statistically independent, orthogonal input motions. In this case the seismic testing consists of five 1/2 safe shutdown earthquake tests and one safe shutdown earthquake test in one orientation.

Using a biaxial test table is acceptable if it is justified that the horizontal and vertical test inputs conservatively simulate the three-dimensional nature of the seismic event. One acceptable approach is to mount the equipment on the test table with its front-to-back axis oriented at 45 degrees to the horizontal drive axis and to scale the horizontal component of the input by a factor of the square root of two. Statistically independent inputs are preferred and, if used, the test can be performed in two stages, with the equipment rotated 90 degrees about the vertical axis. In this case, the five 1/2 safe shutdown earthquake inputs need to be applied only in the first orientation.

If a dependent biaxial test table is used, the test is performed in four stages. The first stage involves five 1/2 safe shutdown earthquake tests and one safe shutdown earthquake test in the first orientation. The second, third, and fourth orientations are obtained by successively rotating the equipment 90 degrees clockwise from its previous position. One safe shutdown earthquake test is performed in each of the last three orientations.

(First sentence of the first paragraph of Subsection E.5.2.4 of Attachment E to Appendix 3D):

~~The aging effect of the two 1/2 safe shutdown earthquake earthquakes can be simulated by exposing the equipment to two sinusoidal sweeps at one half of the safe shutdown earthquake required input motion level in each orthogonal axis.~~

The aging effect of the five 1/2 safe shutdown earthquake earthquakes can be simulated by exposing the equipment to two sinusoidal sweeps at one-half of the safe shutdown earthquake required input motion level in each orthogonal axis.

Question 210.16

As indicated in Section 3.9.2.3 of the SSAR, the reactor vessel internals in the AP600 are similar in size and configuration to the 3-loop reactor at the H.B. Robinson plant with additional design changes from several reference reactors. However, the AP600 is not a 3-loop reactor, and effects of those design changes, although their acceptability were individually verified by separate tests in different reactors or lab conditions, may interact and result in unacceptable dynamic response. Since flow-induced excitations are complex and sensitive to a simultaneous effect of several parameters, such as configuration of flow path, pressure, temperature, flow velocity, etc., provide details of the evaluation to show how a combination of analysis, testing, and comparison to the results in several reference plants was used to verify the acceptability of flow-induced vibrations of the internals under operational transients and steady-state conditions. In addition, describe acceptance criteria and verify that the above stated evaluation, including detail drawings and calculations, was properly documented.

Response:

H. B. Robinson is the original three-loop plant prototype. As indicated in the RAI, the three-loop plant internals are similar in size and configuration to the AP600 internals. Therefore, the long, successful operation of these internals provides one part of the basis for verification of the adequacy of the AP600 internals. However, vibration assessments have been performed on numerous internals configurations, including those shown in Subsection 3.9.2.3 of the SSAR, providing a broad data base for prediction of the AP600 internals flow-induced vibration behavior. These data and analytical models of the AP600 internals configurations will be used to demonstrate the adequacy of the AP600 internals.

The approaches to be used for this program are summarized in the following. Turbulence and reactor coolant pump excitations are discussed.

- **Turbulence Excitation**

Lower Internals Response To Inlet Nozzle And Downcomer Turbulence

The dominant excitation of the lower internals is flow turbulence generated at the reactor vessel inlet nozzles and in the downcomer annulus.

Data from plant preoperational vibration measurement data and scale model flow test data for lower internals designs having neither a circular thermal shield nor neutron pads in the core barrel-reactor vessel downcomer annulus characterize the forcing functions due to inlet nozzle and downcomer annulus turbulence. These data, which are for four inlet nozzles as in the AP600 design, will be used to establish a forcing function for four-loop-size internals and will be scaled to the lower velocity, small size of the AP600 internals. Known behavior of response variation with flow from scale model tests will assist in the scaling evaluation. The resulting forcing function will be applied to a model of the AP600 reactor vessel core barrel, reflector, and core. The resulting responses will be compared to allowable high-cycle fatigue limits at key areas on the lower internals and to design interface loads at the reactor vessel/core barrel lower restraints. The small variation of the excitation with temperature during heatup and cooldown will also be evaluated in the analyses.

The vibration of the core barrel due to inlet nozzle and downcomer turbulence is proportional to flow velocity raised to a power greater than 2. Since the velocities are significantly lower than those of previous plants, the vibration levels of the AP600 are expected to be lower, providing a basis for the expectation that the analyses will show that the high-cycle fatigue stresses will be acceptable for the AP600.

The reflector will be modeled in the analysis so that vibration of the lower internals in modes that include motion of the reflector and reflector core barrel interface loads can be calculated. These results will be used to demonstrate that the lower internals design with the reflector will have adequate margins against flow-induced vibration. The preoperational vibration measurement program for the first AP600 will include transducers to confirm the vibrating response and adequacy of the reflector for flow-induced vibration.

The vortex suppression ring has been designed and tested so that fluctuations in the flow patterns in the lower reactor vessel head plenum that have been observed in some previous plant designs will not be significant in the AP600. Analytical estimates of the response of the vortex suppression ring and its supporting structure will consider turbulence excitation, base excitation due to vibration of the core barrel, and the potential for vortex shedding.

Upper Internals Components

Coolant flow exiting the core outlet converges on the reactor vessel outlet nozzles so that the highest velocities and flow turbulence excitation levels occur on the guide tubes and support columns located near the outlet nozzles. The UPPLEN code is used to calculate the velocities and flow forces due to the coolant flowing across these components. The integrated effects of these crossflows produce beam deflections and end reactions that will be compared to similar results for upper internals components for which vibration measurements have been made. Since the guide tube and support column designs for the AP600 plant are identical to previous designs, the adequacy of the AP600 components can be verified.

As noted in the initial submittal, the outlet nozzle velocities in the AP600 design are lower than the corresponding velocities in previously tested plants. Additionally, the outermost components of the AP600 design are more distant from the outlet nozzles than in previously tested designs. This provides a high confidence that the AP600 upper internals components will have adequate margins.

- **Reactor Coolant Pump-Related Excitation**

Plant data shows that internals vibration responses include contributions at reactor coolant pump rotating speed and impeller blade-passing-related frequencies. Laboratory and plant test data have been used to develop an analytical computer model (ACSTIC) to estimate the pump-related excitation forces on the reactor vessel internals. The calculated vibratory loads are added to turbulence-induced loads to determine the net vibration levels and high-cycle fatigue margins.

Summary

In summary, an extensive assessment of the adequacy of the AP600 reactor vessel internals against flow-induced vibration is planned to be finalized in the first quarter of 1994. This analysis will utilize plant and scale model vibrations measurement results to verify the adequacy of the internals for high cycle fatigue. As discussed in the response to Q210.18, the lower internals of the first plant will be instrumented so that preoperational vibration measurements can be obtained to confirm the adequacy of the core barrel with the reflector. Acceptance criteria for the analysis are the ASME Code allowable high-cycle fatigue stresses and design loads at interfaces calculated directly or inferred by comparison of the AP600 results to previously analyzed/tested designs.

SSAR Revision: NONE

Question 210.18

Since the AP600 design has different coolant loop configuration from the design of H.B. Robinson plant (see Q210.16), and it also has incorporated additional design changes from several reference plants, it is difficult to visualize the assertions that the reactor internals of the H.B. Robinson design is the valid representative for the AP600 internals. A vibration measurement program should be implemented per RG 1.20 during the preoperational test for either the first AP600 internals or the internals similar to the AP600 but with some design modifications (the Non-prototype Category II). Provide detailed information regarding the vibration measurement program, including numbers, types and locations of sensors, the basis of sensor selection and analyses for predicting levels of response of individual sensors. In addition, acceptance criteria of vibration measurements should also be described (Section 3.9.2.4).

Response:

As indicated in the response to Q210.16, data for internals responses in addition to the H. B. Robinson experience are available for geometries similar to the AP600 geometry for verification of the adequacy of the AP600 internals.

Consistent with the guidelines of Regulatory Guide 1.20, a preoperational vibration measurement program will be conducted on the first AP600 plant to confirm the adequacy of the core barrel-reflector configuration used in this design.

Westinghouse has successfully completed preoperational vibration measurement programs on seven plants. These programs included strain gages and accelerometers mounted on the internals to measure structural responses during hot functional test heatup, steady operation of several combinations of reactor coolant pumps, and startup and shutdown of reactor coolant pumps.

A preoperational measurement program is intended for the initial AP600 plant. The test plans will be developed in conjunction with the analysis described under Response 210.16. These plans will include:

- The locations and types of transducers to be installed
- The bases used to establish expected and acceptable vibration levels and expected natural frequencies
- The conditions at which data are to be acquired

SSAR Revision: NONE

Question 210.23

Section 3.9.2.6 of the SSAR indicates that the results of dynamic analysis of reactor internals have been compared to the results of preoperational testing in reference plants. Describe the analytical model used and provide details of the comparison.

Response:

The documentation of the comparison of predicted vibration levels to measured responses will be in two parts:

1. Comparison of results expected based on scale model data, previous plant vibration measurement data, and analysis to subsequent plant measurement data. The plan for this documentation is discussed in the response to Q210.16.

2. Comparison of predicted and actual AP600 plant measurement. The first AP600 plant will report on these comparisons.

SSAR Revision: NONE

Question 220.10

The argument for a reduced Level C factor of safety of 1.5 in the first two bullets of Paragraph 6 of Section 3.8.2.4.2.2 of the SSAR is not clear, based on the comment in Q220.9 and the following definitions:

a. The theoretical buckling load is calculated from an analytical model which does not include imperfections.

b. The predicted buckling load represents the load at which buckling is actually expected. It includes imperfection effects. The predicted buckling load may be found as the theoretical buckling load times the capacity reduction factor.

c. The allowable buckling load is the predicted load divided by the factor of safety. The capacity reduction factor (not the factor of safety) is intended to include imperfection effects.

The last bullet indicates that a reduced factor of safety is permissible because of the low probability of Level C loading. However, this has already been recognized by ASME when it permits a 20 percent increase in allowable stresses for Level C over level A (Paragraph 6). Justify the use of a factor of safety of 1.5 for Level C loading.

Response:

Subsection 3.8.2.4.2.2 of the SSAR will be revised as shown in the response to Q220.9 to clarify the factor of safety used in establishing an appropriate capacity for evaluation of certain severe accidents. The factor of safety of 1.5 is not intended to be applied for loadings for which ASME Service Level C is appropriate.

SSAR Revision: NONE

Question 220.11

In the case of the torispherical head, the theoretical buckling load is 176 psig. With a capacity reduction factor of 0.79, the predicted buckling load is 137 psi. With a Level C buckling factor of safety of 2.5, the Level C allowable buckling load would be 55 psig and not the 70 psig stated in Paragraph 6 of Section 3.8.2.4.2.2 of the SSAR. Clarify why no capacity reduction factor was used in the 70 psig calculation.

Response:

Subsection 3.8.2.4.2.2 of the SSAR will be revised as shown in the response to Q220.9 to clarify the discussion of the capacity reduction factors and the safety factors. Capacity reduction factors are used in evaluating the predicted capacity of the head. The proposed revision to the subsection explains why the capacity reduction factor of 0.79 was not considered.

SSAR Revision: NONE

Question 220.12

Paragraph 4 of Section 3.8.2.4.2.2 of the SSAR states that buckling of the head is not a consideration in the ultimate capacity of the containment because of post-buckling considerations. This argument is used again in bullet three of Paragraph 6 of the same section to justify a reduced factor of safety of 1.5 for Level C buckling. The argument is based upon the post-buckling behavior of only two tests. Provide evidence that the post-buckling strains for this head do not exceed acceptable limits. For example, what strain levels exist in the head at 144 psig, which is above the initial buckling load of 137 psig predicted in the SSAR?

Response:

Subsection 3.8.2.4.2.2 of the SSAR will be revised as shown in the response to Q220.9 to clarify the discussion of the capacity reduction factors and the safety factors. Table 220.12-1 shows the maximum strains and corresponding stresses in the knuckle region of the vessel head predicted by the BOSOR-5 analysis prior to buckling. Table 220.12-2 shows the stresses at the location of the maximum circumferential compressive stress. The BOSOR-5 analysis does not extend beyond the pressure of 174 psi predicted as the theoretical initiation of buckling. Post-buckling behavior of the heads is based on a large number of tests of ellipsoidal and torispherical heads (See for example reference 220.12-1.)

Reference

220.12-1: Bushnell, D., "Elastic-Plastic Buckling of Internally Pressurized Ellipsoidal Pressure Vessel Heads," Welding Research Council Bulletin 267, May 1981.

SSAR Revision: NONE

Table 220.12-1: Stresses in Knuckle Region at the Location of Maximum Effective (Von Mises) Membrane Strain, ee

Pressure psig	ee %	Sphi[a] ksi	Stht[b] ksi
140	0.175	34.40	-24.80
150	0.186	36.86	-26.10
160	0.197	39.37	-27.50
165	0.205	40.90	-27.98
170	0.219	41.81	-26.95
173c	0.235	43.45	-25.03

a. Meridional stress.
b. Circumferential stress.
c. 'BOSOR5' pressure=174 psig.

Table 220.12-2: Stresses in Knuckle Region at the Location of Maximum Circumferential Compression

Pressure psig	Sphi[a] ksi	Stht[b] ksi
140	34.29	-24.85
150	36.79	-26.18
160	39.37	-27.50
165	40.88	-28.00
170	41.60	-27.19
173[c]	42.21	-26.48

a. Meridional stress.
b. Circumferential stress.
c. 'BOSOR5' pressure=174 psig.

Question 230.1

Section 3.7 of the SSAR states that Non-Category I facilities are designed in accordance with the Uniform Building Code (UBC), Zone 2A, requirements. Clarify the intent of this statement which implies that any sites in Zones 2B, 3, and 4 with more severe seismic requirements are excluded from the standard design. Note that this requirement will exclude a large part of the western United States from site selection.

Response:

Acceptability of the AP600 at a specific site with respect to seismic requirements is based on the magnitude of the safe shutdown earthquake and the soil conditions as outlined in SSAR Section 2.5. Certain locations, including much of California are outside the seismic interface parameters.

The UBC, Zone 2A requirements are imposed only on items that are not safety-related. These UBC requirements are imposed to protect the utility investment. They have been established by the utilities in developing the Utility Requirements Document, and cover most of the locations where the safe shutdown earthquake would be equal to or less than 0.3g.

In practice, many of the major structures of the AP600 have seismic capability exceeding the minimum UBC Zone 2A level since they are designed to seismic Category I or II criteria. Applicability of the UBC criteria at a given site is normally determined by the utility since the Uniform Building Code is not mandated at all locations in the United States. Seismic adequacy of the non-seismic items can be addressed on a case-by-case basis at those sites where the UBC would require a larger seismic input.

SSAR Revision: NONE

Question 230.3

Section 3.7 of the SSAR states that the operating basis earthquake (OBE) has been eliminated as a design requirement for the AP600. Regulatory Guide 1.143 allows radwaste buildings to be designed to the OBE. Regulatory Guide 1.27 allows certain parts of the ultimate heat sink to be designed for the OBE. What is the AP600's seismic design basis for these facilities?

Response:

The design of radwaste buildings and equipment is addressed in the response to question 460.6. The ultimate heat sink for the AP600 is the atmosphere. Heat is transferred to the atmosphere by the passive containment cooling system, including the containment vessel and shield building. These are seismic Category I and are designed for the safe shutdown earthquake.

SSAR Revision: NONE

Question 230.4

Section 3.7 of the SSAR states that the operating basis earthquake (OBE) has been eliminated as a design requirement for the AP600. Appendix A of 10 CFR Part 100 requires that if vibratory ground motion exceeding the OBE occurs, shutdown of the plant will be required. State what the AP600 excitation level is specified for plant shutdown purposes.

Response:

The criterion for initiating a plant shutdown following a seismic event will be exceedance of a specified response spectrum limit or a cumulative absolute velocity limit, as described in SSAR Subsection 3.7.4.4. Plant-specific procedures will be developed as described in the response to Q230.5.

SSAR Revision: NONE

Question 230.5

Section 3.7 of the SSAR states that the cumulative absolute velocity (CAV) approach according to EPRI Report NP-5930 will be used for plant shutdown criteria following an earthquake. The CAV calculation discussed in EPRI NP-5930 has been amended. The standardized CAV calculation is discussed in EPRI Report TR-100082. The guidelines for nuclear plant response to an earthquake are discussed in EPRI Report NP-6695. State in greater detail what the AP600 plant procedures are following an earthquake occurrence.

Response:

Plant procedures following an earthquake are the responsibility of the Combined License applicant. These procedures will follow the guidance of the EPRI documents referenced in the question. The SSAR will be revised to clarify this responsibility.

SSAR Subsection 3.7.4.4, "Comparison of Measured and Predicted Responses," will be revised to read as follows and indicated references added:

SSAR Revision:

The recorded seismic data is used by the combined license applicant operations and engineering departments to evaluate the effects of the earthquake on the plant structures and equipment.

The combined license applicant will prepare plant-specific procedures covering response to an earthquake. These procedures will follow the guidance of EPRI Reports NP-5930 (Reference 1), TR-100082 (Reference 17), and NP-6695 (Reference 18).

The criterion for initiating a plant shutdown following a seismic event will be exceedance of a specified response spectrum limit or a cumulative absolute velocity limit. The seismic instrumentation system is capable of computing the cumulative absolute velocity as described in EPRI Report NP-5930 (Reference 1) and EPRI Report TR-100082 (Reference 17).

(References 17 and 18)

17. EPRI Report TR-100082, "Standardization of the Cumulative Absolute Velocity," December 1991.

18. EPRI Report NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," December 1989.

Question 230.23

Section 3.7.4.2 of the SSAR indicates that four triaxial acceleration sensors will be installed at an AP600 plant. Regulatory Guide 1.12 "Instrumentation for Earthquakes" is presently being revised by the NRC staff. The draft guide calls for 7 or 8 triaxial acceleration sensors at various locations within the plant site. Discuss in detail the AP600 position with respect to amending the SSAR to comply with the RG 1.12 revision.

Response:

The AP600 conformance with the current version of Regulatory Guide 1.12 is discussed in Appendix 1A of the SSAR. The proposed locations of the triaxial acceleration sensors have been based on a review of the seismic analyses of the nuclear island described in Subsection 3.7.2 of the SSAR. Table 230.23-1 shows the locations proposed in the proposed revision to Regulatory Guide 1.12 (DG-1016) and identifies the location of the equivalent AP600 sensor or justifies why such a sensor is not proposed.

SSAR Revision: NONE

Table 230.23-1 Comparison of AP600 Sensor Locations With DG-1016	
DG-1016 Proposed Location	AP600
Free field	Free field location to be defined by combined license applicant.
Containment foundation	Nuclear Island basemat (elevation 66'6") near column lines 9 and L.
Two elevations (excluding the foundation) on a structure internal to the containment	One sensor on the east wall of east steam generator compartment at elevation 138'. A second sensor is not proposed because access to a sensor at a lower elevation is inconsistent with maintaining occupational radiation exposures ALARA, and the seismic analyses show such a location to be unnecessary. The response of the CIS is well represented by the response at elevation 138'. A possible location would be at elevation 107', but the analyses show that the response at this location is similar to that of the basemat because the stiffness of the containment structures at low elevations.
Two independent Category I structure foundations where the response is different from that of the containment structure	All Category I structures are part of the nuclear island and have a common basemat. No additional sensors are required.
An elevation (excluding the foundation) on each of the independent Category I structures selected above	One sensor is proposed at elevation 229' on the shield building. This will measure response of the nuclear island structures (shield building and auxiliary building, which are integrally connected).

Question 250.10

Describe the steam generator tube inservice inspection program, such as the inspection technique, provisions for the selection and sampling of tubes, the inspection intervals, the actions to be taken in the event defects are identified, and reporting requirements (Section 5.4.2).

Response:

Technical Specification SR 3.4.4.2 requires verification of steam generator tube integrity in accordance with the Steam Generator Tube Surveillance Program. The Steam Generator Tube Surveillance Program is the responsibility of the combined license applicant and shall be managed and implemented using controls similar to those used for the inservice inspection program. The initial Steam Generator Tube Surveillance Program shall be subject to a review and approval process equivalent to that required for the inservice inspection program, and changes to the surveillance program will be processed in the same manner as relief requests for the inservice inspection program. The surveillance program specifies the details of the inspection including tube selection and sampling (as well as sample expansion), inspection interval, inspection technique, the actions to be taken when defects are identified, and reporting requirements. This program will be consistent with the recommendations of Regulatory Guides 1.83 and 1.121 and supplemented by the recommendations of the industry-prepared "PWR Steam Generator Examination Guidelines, Revision 2" (EPRI Report NP6201, December 1988) and, where appropriate, technical support documents for defect-specific repair limits (e.g., EPRI Report NP-6884-L, "PWR Steam Generator Tube Repair Limits: Technical Support Document for Expansion Zone PWSCC in Roll Transitions," and EPRI Report TR-100407, "PWR Steam Generator Tube Repair Limits: Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates").

SSAR Subsection 5.4.2 will be revised as follows:

SSAR Revision:

(Added to Table 1.8-1:

Item No.	Interface	Interface Type	Matching Interface Item	Section or Sub-section
5.1	Steam Generator Tube Surveillance Requirements	Requirement of AP600	Combined License Applicant Program	5.4.2

(Subsection 5.4.2.5, fifth paragraph)

The minimum requirements for in-service inspection of steam generators, including tube plugging criteria, are established as part of the Technical Specifications. The steam generator tube integrity is verified in accordance with a Steam Generator Tube Surveillance Program. The Steam Generator Tube Surveillance Program is the responsibility of the Combined License applicant. These requirements are consistent with the ASME Code, Section XI, and Regulatory Guide 1.121. Section XI of the ASME Code provides general acceptance criteria for indications of tube degradation in the steam generator. Specific conformance with Regulatory Guide 1.121 is discussed in Section 1.9.

Question 250.16

When in the fabrication procedure will the shop examination of the tubing be performed? Describe the procedures and precautions taken to ensure the integrity of the tubes during final assembly, shipment, and installation of the steam generators (Section 5.4.2).

Response:

The baseline inspection is typically performed at the site. The Combined License holder has the option of specifying a shop examination of the tubing. Should the Combined License holder specify a baseline examination at the manufacturing facility, it is done after all fabrication operations on the tubes and tube to tubesheet welds are completed.

The integrity of the tubes during final assembly, shipping and installation are ensured by the procedures and processes which have proved themselves to be reliable over the years.

The following change will be made to SSAR Appendix 1A, page 1A-41:

SSAR Revision:

Reg. Guide 1.83, Rev. 1, 7/75 - Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes

General	Conforms	A program for inservice inspection of AP600 steam generator tubing is established and performed in accordance with the guidelines of this regulatory guide. The baseline inspection will be performed at the site in accordance with Regulatory Position C.3.a. Should the Combined License applicant request a baseline examination at the manufacturing facility, it will be performed in accordance with Regulatory Position C.3.a. ~~Westinghouse intends to use the option of the last paragraph of Section B, which permits the shop examination of tubing to serve as an adequate baseline inspection, provided that the examination is done in accordance with the requirements of the ASME Code, Section III Subsection NB, Article 2550. Shop inspection is planned to support the AP600 construction schedule. The Combined License applicant may at its option perform the inspection prior to operation of the AP600 plant in accordance with Regulatory Position C.3.a.~~
C.2	Exception	The specification of equipment in Regulatory Position C.2.c does not represent state of the art equipment for gathering and storing eddy current information. When ~~the baseline~~ an eddy current inspection of a ~~for the~~ AP600 steam

		generator is done in the manufacturing facility, more capable equipment than that specified in the regulatory guide is used. The steam generator design is compatible with robotic eddy current inspection equipment.
C.3	Exception	As noted in the comment on Criteria Section C.2, ~~the baseline~~ any eddy current inspection ~~may be~~ done in the manufacturing facility ~~using~~ uses equipment of more current technology than that specified in Criteria Section C.2.

Question 250.21

Where will the provisions for inservice inspection of steam generator tubes be implemented, e.g., plant technical specifications (Section 5.4.2)?

Response:

The provisions for inservice inspection of steam generator tubes are implemented through the Steam Generator Tube Surveillance Program (see response to Q250.10).

SSAR Revision: NONE

Question 251.2

Westinghouse proposes to use a depleted uranium alloy casting in an Inconel alloy welded enclosure to construct the pump flywheel. These materials are not addressed in Section 5.4.1.1 of the SRP and Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity." Provide technical justifications for the use of these materials (Section 5.4.1).

Response:

As noted in Subsection 5.4.1 of the SSAR, the AP600 canned motor reactor coolant pump uses a fundamentally different approach to demonstrate safe operation of the flywheel than the design approach for which Section 5.4.1.1 of the Standard Review Plan and Regulatory Guide 1.14 were developed. Of prime importance in the consideration of flywheel integrity is minimizing the potential for generation of missiles from the flywheel in conformance with the requirements of General Design Criteria 4. The AP600 approach is to demonstrate that fragments from a postulated flywheel fracture do not penetrate the surrounding pressure boundary and thus do not become missiles. See the response to Q250.11 for additional information on the analysis of the retention of flywheel fragments. This basis of containing postulated fragments is the same as for the rotor and other rotating parts in previous shaft seal pump designs. The approach behind the recommendations of Section 5.4.1.1 of the Standard Review Plan and Regulatory Guide 1.14 is to minimize the potential for a flywheel fracture by extensive testing and inspection of the flywheel.

Although conformance with the recommendations in the regulatory guide is not required to demonstrate safe operation of the pump, many of the recommendations are followed in the design and fabrication of the flywheel assembly for operational reliability. Since the AP600 design does not rely on flywheel material integrity to minimize the potential for the generation of missiles, the quality assurance requirements inherent in the use of ASME Code pressure boundary quality material as suggested by the Standard Review Plan are not required. The design requirements for the flywheel assembly materials are selected to provide a high level of operational reliability. The basis for the design requirements for the flywheel assembly materials is outlined below.

The flywheel assembly is a uranium-alloy casting or forging surrounded by a nickel-chromium-iron alloy enclosure. The material strength used for the analyses that demonstrate flywheel integrity is based on the material specification outlined in Table 5.4-2. The material toughness is demonstrated by the yield strength and elongation. See the response to Q251.3 for additional information on the fracture toughness properties of the uranium alloy. Since the uranium alloys to be used in the flywheel were not developed for use as pressure boundary materials, ASME Code material specifications do not exist. See the response to Q251.23 for additional information on the material specification. Nevertheless, quality assurance practices can confirm that the minimum material requirements are met. The nickel-chromium-iron Alloy 600 material used in the enclosure is a commonly used material with established material specifications.

The uranium alloy does not come in contact with the reactor coolant. The Alloy 600 enclosure material has been shown to be compatible with reactor coolant in other applications. The operating temperature of the coolant surrounding the flywheel assembly is substantially less than the reactor coolant system operating temperature, so stress corrosion cracking of the Alloy 600 is not an issue. See the response to Q251.21 for additional information on the resistance to stress corrosion cracking of the flywheel enclosure.

SSAR Revision: NONE

Question 251.3

Westinghouse indicates that the fracture toughness guidelines in Section 5.4.1.1 of the SRP and Regulatory Guide 1.14 are not applicable to depleted uranium alloy castings. Provide information on the fracture toughness properties for this material and propose fracture toughness requirements with technical justifications (Section 5.4.1).

Response:

The fracture toughness of the uranium alloy casting is approximately 50 ksi $\sqrt{in.}$ between 100°F and 200°F based on available data. Over the same temperature range the minimum impact energy (Charpy V-notch) is 10 foot-pounds. The material specification for the flywheel material includes a requirement for this minimum impact energy. The material specification does not include a fracture toughness requirement, but the properties and processing specified define a material that meets the 50 ksi $\sqrt{in.}$ minimum.

Calculation of the critical flaw sizes is based on the 50 ksi $\sqrt{in.}$ fracture toughness. For the case of an axial-radial crack along the length of the bore at the flywheel design speed, the critical flaw size is larger than 2 inches deep. The critical flaw size for other flaw cases (such as semicircular and semieliptical along the bore with a axis less than the length of the bore) is even deeper.

SSAR Subsection 5.4.1.3.6.3 will be revised as follows:

SSAR Revision

(Fifth paragraph of Subsection 5.4.1.3.6.3)

The key parameters for the uranium alloy specification are defined in Table 5.4-2. These parameters include the minimum ultimate and yield tensile strength. Nil ductility transition and upper shelf energy are not specified in the requirements for the uranium alloy. These are characteristics of steel not duplicated in the uranium alloys. The material specification has appropriate testing to confirm that the fracture toughness used in the flywheel evaluation is satisfied. A Charpy V-notch test is required~~ for information~~. A portion of the uranium is machined off to obtain specimens for tensile and impact tests and to inspect the micro structure.

(Table 5.4-2)

Mechanical Requirements

Ultimate Tensile Stress	110 ksi Min.
Yield Stress	55 ksi Min.
Elongation	10% Min.
Reduction of Area	25% Min.
Charpy V-notch	10 ft-lb Min.

Question 251.4

Provide information on the fabrication process and resulting quality for the depleted uranium alloy casting (Section 5.4.1).

Response:

The melting of the depleted uranium alloy for the flywheel casting or forging billet is done under vacuum or inert atmosphere to provide a high-quality product. The vacuum or inert atmosphere prevents reaction of the uranium with air and minimizes the potential for the formation of voids. Because of the density of the uranium alloy, slag and other impurities tend to float to the top of the molten metal and porosity in the cast material is not a problem. The molds for the casting are treated to minimize the contamination of the uranium with carbon. The rest of the manufacturing process is controlled to minimize the contamination of the uranium alloy with carbon and hydrogen. Excessive carbon reduces the ductility of the uranium alloy. Hydrogen contamination may induce delayed cracking. Because of the thickness of the flywheel, the final heat treatment is a solution anneal in a vacuum furnace followed by a slow cooling. Other heat treatments such as annealing followed by water quenching and aging hardening are not appropriate for a thick uranium alloy flywheel. See the response to Q251.22 for additional discussion of the heat treatment.

SSAR Revision: NONE

Question 251.5

Section 1A of the SSAR indicates that the AP600 design meets the guidelines of Regulatory Position 1.d in Regulatory Guide 1.14. However, the flywheel, including the enclosure welds, will not be inspected. Discuss how the flywheel design meets Regulatory Position 1.d.

Response:

The uranium alloy flywheel is not subject to welding operations, including repair welding, or any other finishing operations that use thermal methods. The component parts of the enclosure are connected together with seal welds. These welds are not needed to provide the structural strength needed to resist the forces on a spinning flywheel assembly. Nevertheless the seal welds that connect the parts of the enclosure are inspected following fabrication by radiography and liquid penetrant. ASME Code, Section III criteria for structural welds are used as guidelines to establish welding and inspection requirements. See the response to RAI 251.14 for additional information on the analysis and inspection of the enclosure seal welds. The enclosure represents only a small fraction of the energy in a rotating flywheel assembly. The location of the seal welds are such that there is minimal effect on the fracture analysis.

SSAR Revision: NONE

Question 251.6

Regulatory Positions 2.c, 2.d, and 2.e in Regulatory Guide 1.14 recommends that an analysis be submitted for staff review. Provide the analysis with appropriate technical justifications. Further, because no inservice inspection for the flywheel is being proposed, describe the flaw size assumed in its analysis (Section 5.4.1).

Response:

Regulatory Positions 2.c., 2.d., and 2.e. in Regulatory Guide 1.14 recommend that analyses be conducted to predict the critical speed for ductile failure, nonductile failure, and excessive deformation of the reactor coolant pump flywheel. As noted in Subsection 5.4.1 of the SSAR and the response to Q251.2, the approach to demonstrate safe operation of the AP600 canned motor reactor coolant pump flywheel differs from the design approach for which Regulatory Guide 1.14 was developed. The AP600 design approach of demonstrating that postulated flywheel fragments are contained by the pump structure limits the significance of the analysis of critical flywheel failure speeds.

The analysis completed for the flywheel structural evaluates the stress intensity levels at the normal speed and at the design speed of 125 percent of normal. The calculated stress levels are evaluated against ASME Code, Section III, Subsection NG stress limits and the recommended stress limits in Positions 4.a. and 4.c. of the Standard Review Plan 5.4.1.1.of one-third and two-thirds of yield stress for normal speed and design speed respectively. See the responses to Questions 251.16, 251.17, 251.18, and 251.19 for additional information on the evaluation of stress in the flywheel assembly. The margin inherent in these limits provides an appropriate degree of margin to failure at the normal and design speeds.

The flaw size assumed in the evaluation of fracture toughness is described in the response to Q251.3.

See the response to Q251.7 for the SSAR revision related to critical flywheel failure speed analysis.

SSAR Revision: NONE

Question 251.7

Section 1A of the SSAR indicates conformance with Regulatory Position 2.f in Regulatory Guide 1.14. Provide information to support this statement.

Response:

As noted in Subsection 5.4.1.3.6.3 of the SSAR and in the response to Q251.2, the design speed (125 per cent of normal speed) envelopes all expected and postulated overspeed conditions including overspeeds due to postulated pipe ruptures. See the response to Q251.8 for a discussion of the size of postulated pipe ruptures. This limitation on the potential overspeed along with the design approach of demonstrating that postulated flywheel fragments are contained by the pump structure limits the significance of the analysis of critical flywheel failure speeds. The analysis completed for the flywheel stress report evaluates the stress intensity levels at the normal speed and the design speed of 125 percent of normal. The calculated stress levels satisfy the ASME Code, Section III, Subsection NG stress limits. The calculated primary stress levels are less than the recommended stress limits in Positions 4.a. and 4.c. of the Standard Review Plan 5.4.1.1 of one-third and two-thirds of yield stress for normal speed and design speed, respectively. See the responses to Questions 251.16, 251.17, 251.18, and 251.19 for additional information on the evaluation of stress in the flywheel assembly.

The flywheel structural analysis verifies that the failure modes outlined in Positions 2.c, 2.d, and 2.e of Regulatory Guide 1.14 do not occur at the design speed. The flywheel stress evaluation noted above demonstrates an appropriate margin against these failure modes. In addition, the design of the canned motor pump mitigates the effects of hypothetical failures by these modes, as outlined below.

The response to Q251.11 discusses the containment of fragments from a postulated flywheel fracture. The mode of failure, ductile or nonductile, would not alter the capacity of the surrounding pump structure to absorb the energy of the fragments and prevent the generation of missiles from the flywheel assembly.

Regulatory Guide 1.14 defines excessive deformation as any deformation that could cause separation of the flywheel from the shaft. Because of the restriction of the lateral movement of the flywheel assembly by the surrounding structure and axial movement by the thrust bearings, the loss of shrink fit would not be expected to result in substantial movement of the flywheel assembly or significant separation of the assembly from the shaft. This restriction in movement of the flywheel assembly and the adjacent location of the journal bearing to the flywheel assembly minimize the potential for a structural failure of the shaft during a hypothetical overspeed transient sufficient to result in excessive deformation.

Neither separation of the flywheel assembly from the shaft nor structural failure of the shaft would result in a loss of safety-related function of the canned motor pump during an overspeed transient. That safety-related function is the maintenance of the primary pressure boundary. Neither separation of the flywheel assembly nor structural failure of the shaft would degrade the pressure boundary of the pump. The safety-related function of providing flow during coastdown of the pump is not germane during an overspeed transient.

SSAR Subsection 5.4.1.3.6.3 and Appendix 1A will be revised as follows:

SSAR Revision:

(Eighth paragraph of Subsection 5.4.1.3.6.3)

An analysis of the ~~critical~~ flywheel failure modes of ~~speeds based on~~ ductile failure, nonductile failure and excessive deformation of the flywheel is performed to evaluate the flywheel design. The analysis is performed to determine that the critical flywheel failure speeds based on these failure modes are greater than the design speed. The critical flywheel failure speeds are not the same as the critical speed identified for the rotor. The ~~smallest of the~~ critical flywheel failure speeds are ~~calculated is~~ greater than the design speed. The overspeed condition for a postulated pipe rupture accident is less than the ~~smallest of~~ the critical flywheel failure speeds.

(Appendix 1A)

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.14, Rev. 1, 8/75 - Reactor Coolant Pump Flywheel Integrity			
2.c-e	ASME Code, Section III	Exception	The limits and methods of ASME Code, Section III, Paragraph F-1331.1(b), (replacement for Paragraph F-1323.1) are not directly applicable to a uranium alloy casting. The calculated stress levels in the flywheel are evaluated against the ASME Code, Section III, Subsection NG stress limits used as guidelines and the recommended stress limits in Positions 4.a and 4.c of the Standard Review Plan 5.4.1.1.
2.f.		Exception	The calculated stress levels in the flywheel satisfy the ASME Code, Section III, Subsection NG stress limits used as guidelines and the recommended stress limits in Position 4.a of the Standard Review Plan 5.4.1.1.
2.g		Conforms	

Question 251.8

Section 1A of the SSAR indicates conformance with Regulatory Position 2.g in Regulatory Guide 1.14, relating to the flywheel overspeed due to a postulated pipe rupture. Section 5.4.1.3.6.3 of the SSAR appears to assume the application of leak-before-break (LBB) for all high-energy piping 10 cm (4 in) in diameter or larger. Since the outcome of the staff's review of the application of LBB to the AP600 design is uncertain, the staff recommends that Westinghouse discuss how the flywheel conforms with RG 1.14 if the criteria of Section 3.6.2 and BTP MEB 3-1 is used to determine pipe break size.

Response:

The criteria of Section 3.6.2 and Branch Technical Position (BTP) MEB 3-1 are used to determine pipe break size and locations for those piping systems that do not satisfy the requirements for mechanistic pipe break criteria. See the response for RAI 210.6 for the high energy lines to which leak-before-break methodology is applied. Based on previous NRC decisions and the design practice for the AP600, the reactor coolant loop piping demonstrates leak-before-break and satisfies the requirements for mechanistic pipe break. Appendix 3b of the SSAR outlines a sample leak-before-break analysis of the reactor coolant loop. The success of application of mechanistic pipe break criteria to branch lines connected to the reactor coolant loop is based on the results of the leak-before-break analysis and NRC review.

The overspeed analysis of the AP600 reactor coolant pump flywheel is based on the design speed of 125 per cent of normal speed. The AP600 pipe rupture overspeed is expected to be enveloped by the design speed since the reactor coolant main loops and most or all of the branch line piping with a nominal diameter of 4 inches and greater are being qualified for LBB. The pipe rupture overspeed is expected to be substantially less than any of the calculated critical flywheel failure speeds.

As noted in the response to RAI 250.2, the approach used to demonstrate the safe operation of the flywheel is containment of the fragments from a postulated fracture by the surrounding pump structure. For a postulated flywheel fracture at the flywheel design speed there is a large amount of margin in the calculated capability of the pump structure to contain flywheel fragments. Thus even in the event of a postulated failure of a flywheel during a hypothetical break of a reactor coolant loop pipe, it is not expected that additional breaks in the reactor coolant pressure boundary would be created nor would missiles be generated by the flywheel.

The thirteenth paragraph of SSAR Subsection 5.4.1.3.6.3 will be revised as follows:

SSAR Revision:

Pipe rupture overspeed is based on a break of the largest branch line pipe connected to the reactor coolant system piping that is not qualified for leak-before-break criteria. The exclusion of the reactor coolant loop piping and branch line piping of 4 inches or larger size from the basis of the pump loss of coolant accident overspeed condition is based on the provision in GDC 4 to exclude dynamic effects of pipe rupture when a leak-before-break analysis demonstrates that appropriate criteria are satisfied. See Subsection 3.6.3 for a discussion of leak-before-break analyses. The criteria of Section 3.6.2 are used to determine pipe break size and location for those piping systems that do not satisfy the requirements for mechanistic pipe break criteria.

Question 251.9

Section 1A of the SSAR indicates that Westinghouse is taking exception to Regulatory Position 4.a in Regulatory Guide 1.14. Propose an alternative to this position with appropriate technical justifications.

Response:

A spin test is done on the flywheel assembly after the enclosure is welded closed. Inspection of the flywheel inside the assembly is not practical. Because of the density of the uranium, radiographic examination is also not a practical option.

The uranium alloy flywheel is ultrasonically inspected following final machining and prior to assembly of the enclosure around the flywheel. The ultrasonic inspection conforms with the requirements of the ASME Code, Section III, paragraph NB-2574, for ferritic steel castings, including the use of the procedures outlined in SA-609 (ASTM-A-609). See the response to Q251.13. Machined surfaces of the uranium flywheel undergo liquid penetrant inspection prior to final assembly. The liquid penetrant inspec' conforms with the requirements of the ASME Code, Section III, paragraph NB-2576, including the use of the procedures outlined in SA-165 (ASTM-A-165).

In-process controls during the assembly of the enclosure onto the flywheel are used to provide for the quality of the completed assembly. The spin test of the completed assembly confirms the quality of the flywheel assembly. Since the basis for safe operation of the flywheel assembly is the retention of the fragments from a postulated fracture by the structure of the pump, inspection subsequent to the spin test is not required.

SSAR Subsection 5.4.1.3.6.3 will be revised as follows:

SSAR Revision:

(Sixth paragraph of Subsection 5.4.1.3.6.3)

The uranium is ultrasonically inspected following final machining. The acceptance criteria for the ultrasonic inspection are based on criteria in the ASME Code, Section III, and are done in conformance with the procedures outlined in ASTM-A-609 ~~ASTM-A-388~~ (Reference 3) with modifications as required for use on a uranium alloy. Thermal methods are not used for finishing operations on the uranium. Following ~~any~~ finishing operations on the casting, the outside surface and the inside bore are subject to liquid penetrant inspections in conformance ~~consistent~~ with the requirements of ASTM-E-165 (Reference 4) ~~with modifications as required for use with a uranium alloy~~. In-process controls used during the construction of the flywheel assembly also provide for the quality of the completed assembly.

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(Reference 3. of Subsection 5.4.15)

3. ASTM-A-609-78, ~~388-86,~~ Standard Specification for Longitudinal Beam Ultrasonic Inspection of Carbon and Low-alloy Steel Castings. ~~Practice for Ultrasonic Examination of Heavy Steel Forgings.~~

(Appendix 1A)

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.14, Rev. 1, 8/75 - Reactor Coolant Pump Flywheel Integrity			
4.a	ASME Code, Section III NB-2545 or NB-2546, NB-2540, NB-2530	Exception	The inspections and guidelines referenced in the regulatory guide were developed for steel ~~and similar materials.~~ flywheels in shaft seal pumps. Inspection of the flywheel inside the flywheel assembly following a spin test is not practical. The ultrasonic inspection of the flywheel prior to final assembly is in conformance with the requirements of the ASME Code, Section III, paragraph NB-2574, for ferritic steel castings, including the use of the procedures outlined in SA-609 (ASTM-A-609). Machined surfaces of the uranium flywheel undergo liquid penetrant inspection prior to final assembly. The liquid penetrant inspection conforms with the requirements of the ASME Code, Section III, paragraph NB-2576, including the use of the procedures outlined in SA-165 (ASTM-A-165).

Question 251.10

Performance of inservice inspection of the flywheel should be considered. If the ISI procedures in Section 5.4.1.1 of the SRP is not applicable to uranium flywheels, propose alternative inservice inspection procedures with appropriate technical justifications (Section 5.4.1).

Response:

Inservice inspection of the uranium alloy flywheel would be very labor intensive and involve significant radiation exposure. Since the surrounding structure of the pump would contain flywheel fragments even in the worse case fracture, inservice inspection would add little if any to the safety of pump operation. The technical justification of no inservice inspection is the analysis that shows that the fragments of a fractured flywheel would not penetrate the pressure boundary of the pump to become missiles. On this basis a flywheel fracture is an operational reliability consideration rather than a safety-related consideration. The use of inspections and in-process controls during fabrication of the flywheel assembly and a spin test of the completed assembly also provide verification of the initial quality of the assembly. The use of vibration monitoring of the pump during operation provides an indication of rotating part stability and thus integrity. This allows any necessary maintenance to be performed as needed for operational reliability.

As noted in the response to RAI 251.2 the design approach to the flywheel in the AP600 canned motor reactor coolant pump is fundamentally different than that for previous shaft seal reactor coolant pump designs. The canned motor pump design was selected for several safety related and operational reasons. Inherent in the design of a canned motor reactor coolant pump is the location of the flywheel assembly within pressure housing and the flywheel enclosure in contact with reactor coolant. To make the flywheel readily accessible for an inservice inspection of marginal utility, many advantages of the canned motor pump would have to be foregone. Routine inservice inspection of the flywheel is neither recommended nor advantageous.

SSAR Revision: NONE

Question 251.11

Section 1A of the SSAR states that a flywheel rupture will be contained within the stator shell. Provide an analysis and technical justifications supporting this statement.

Response:

The canned motor reactor coolant pump has an outer shell that comprises the pressure boundary. The shell is analyzed to demonstrate that in the event of a postulated flywheel fracture, the surrounding pump structure is sufficient to prevent missiles from leaving the pump. The analysis considers that portion of the shell, including the flange, and motor end cap around the flywheel assembly between the top and bottom elevations of the assembly as the barrier to missile generation. The full analysis is included in a flywheel structural analysis and is summarized below. The pump flywheel structural analysis will be finalized by June 1, 1993.

The analysis of the capacity of the surrounding pump structure to contain the fragments of a postulated flywheel failure is done using the energy absorption equations of Reference 1. The containment of missile-like metal disk fragments is by a two-stage process. Stage 1 involves inelastic impact and transfer of momentum to include an effective target mass. To show that the fragments do not perforate the surrounding structure, the energy dissipated in plastic compression and shear strain and the local impact area must be sufficient to account for the loss in kinetic energy of the system. For the nonperforation case, the process enters Stage 2, which involves dissipation of energy in plastic tension strain over extended volumes of shell material. For containment, the energy dissipated in plastic strain in Stage 2 must account for the residual kinetic energy on the system. In predictive calculations it is more conservative to consider Stage 2.

For the AP600 reactor coolant pump analysis, the uranium inset in the flywheel assembly is assumed to fracture at the design speed of 125 percent of normal speed. The worst-case scenario of fragment size and number was derived analytically, using methods in the paper noted above, to determine the mass and velocity combination that would produce the most severe impact on the surrounding pressure boundary components. The following conservative assumptions are also made:

1. End plates and welds of the flywheel enclosure and the coolant surrounding the flywheel assembly have negligible energy-absorbing capability.

2. Only the mass in the stator shell and flange and the motor end cap between the elevation of the top and bottom of the flywheel assembly are considered to absorb energy.

3. Closure bolts and joint effects were not considered to be affected.

4. The minimum material properties were used.

The analysis results show that the fragments impact the surrounding pump structure with a kinetic energy of less than 10 percent of the tensile energy-absorbing capability of the surrounding pump structure. Thus the components around the flywheel contain the flywheel fragments with only a small portion of the energy-absorbing capability available. The energy absorbed by the flywheel enclosure is also only a small fraction of the energy-absorbing capability available in the pump structure and is not relied on to demonstrate containment of the fragments.

References:

1. Hagg, A. C., and Sankey, G. O., "The Containment of Disk Burst and Fragments by Cylindrical Shells," ASME Journal for Power, April 1974, pp. 114-123.

SSAR Revision: NONE

Question 251.12

Section 1A of the SSAR indicates that a "small" flywheel rupture or leak in the enclosure will not result in stresses in the pressure boundary to cause a break. Provide information to clarify what is the intent of the term "small" flywheel rupture. The staff is concerned with the rupture of the flywheel into large fragments of high energy.

Response:

The canned motor pump design is evaluated for a spectrum of postulated uranium flywheel fractures. A fracture that ruptures the flywheel enclosure is bounded by the analysis of the worst-case fracture that shows that the fragments are contained as noted in the second paragraph of Subsection 5.4.1.3.6.3. A fracture that deforms the enclosure enough to bring it in contact with the surrounding structure is bounded by the analysis described in Subsection 5.4.1.3.6.2. A small fracture in the context of the SSAR discussion is one that may unbalance the assembly but any resulting fragment is contained by the enclosure without sufficient deformation to result in interference with the surrounding structure. The discussion of these faults on the low end of the spectrum are included for completeness of the discussion of postulated flywheel fractures.

SSAR Revision: NONE

Question 251.13

Section 5.4.1.3.6.3 of the SSAR indicates that ultrasonic inspection of the uranium following final machining will be based on ASTM A388 as modified for uranium. Identify any modifications to the application of ASTM A388 to the AP600 design with appropriate technical justifications. In addition, demonstrate that this preservice inspection is equivalent to that in Section III of the ASME Code.

Response:

ASTM A388 is a standard for use of ultrasonic inspections on steel forgings. ASTM A609, which is a standard for use of ultrasonic inspections on ferritic steel castings, will be used as the standard for ultrasonic inspection of the uranium flywheel. Changes to the practices specified in the standard to account for use on uranium include the use of uranium reference blocks and potential additional restrictions on the couplants used. The size and frequency of transducers may also be different than the standard, although the inspection of a prototype flywheel casting was done with a transducer size and frequency in the range designated in the standard. Areas of the standard that are not dependent on the type of material inspected such as personnel qualification requirements, surface conditions, procedure, and data reporting should not have to be modified. See the response for RAI 251.9 for additional discussion of inspection of the uranium flywheel.

It is not the intent that the inspection of the uranium alloy flywheel be equivalent in every respect to inspections required of components built to the requirements of the ASME Code, Section III. The requirements for the flywheel are chosen to provide high operational reliability. There are no pressure boundary functions associated with the flywheel assembly that require the use of the ASME Code.

SSAR Revision: NONE

Question 251.14

Demonstrate that the construction of the flywheel enclosure meets Section III of the ASME Code, including inspection (Section 5.4.1).

Response:

Since the flywheel enclosure is not a pressure boundary and is not relied upon to contain fragments from a postulated flywheel fracture, there is no requirement to meet the requirements of the ASME Code, Section III for construction of the enclosure. Additionally, the enclosure contributes only a small portion of the energy in a rotating flywheel assembly. The function of the enclosure is to isolate the uranium alloy from the reactor coolant circulating in the reactor coolant pump. A leak in the enclosure could result in an out-of-balance condition for the flywheel assembly or, over the long term, the possible introduction of depleted uranium into the reactor coolant. Neither of these events represents a catastrophic failure and both would be addressed by other systems. Sensors in the pump detect vibration of the pump and the chemical and volume control system includes provisions to reduce contaminants in the reactor coolant. The uranium would be detected by periodic sampling of the reactor coolant by the primary sampling system.

The welds connecting the pieces of the enclosure are seal welds and are not relied on to contribute to the strength of the enclosure. The ASME Code, Section III criteria for structural welds are used to establish welding requirements and inspection requirements for the enclosure. As noted in the ninth paragraph of SSAR Subsection 5.4.1.3.6.3, the welds are subject to dye penetrant and radiographic tests. The ASME Code Subsection NG stress limit criteria are used as guidelines to evaluate the stress in the enclosure components and the seal welds for normal and design speeds. The use of the ASME Code, Section III to establish design, fabrication, and inspection requirements was selected to provide operational reliability and availability.

SSAR Revision: NONE

Question 251.15

Demonstrate that the design overspeed of the flywheel is at least 10% above the highest anticipated overspeed (Section 5.4.1).

Response:

The requirement for the AP600 is that the design speed (125 per cent of normal speed) be greater than or equal to anticipated overspeed conditions due to electrical faults and overspeed conditions due to postulated pipe breaks. Anticipated overspeed conditions are those due to electrical faults including turbine overspeed events. Because of design of the turbine control system (see SSAR Subsection 10.2.2.3), reactor coolant pump overspeed resulting from an electrical fault is expected to be less than the design speed. See the response to RAI 251.8 for a discussion of flywheel overspeed due to postulated pipe rupture.

Since the basis for safe operation of the pump with respect to flywheel integrity is the containment of flywheel fragments by the pump structure rather than the prevention of fracture (see the responses to RAIs 251.2 and 251.11) a 10% margin between calculated overspeed and the design speed is not required to assure safe operation.

SSAR Revision: NONE

Question 251.16

Show that the combined stresses for the uranium flywheel at the normal operating speed, due to centrifugal forces and the interference fit of the wheel on the shaft, is less than 1/3 of the minimum specified yield strength (Section 5.4.1).

Response:

The flywheel structural analysis verifies that the primary stresses in the uranium due to centrifugal forces at the normal operating speed are less than one-third of the minimum yield strength. The combination of primary and secondary stresses is evaluated using stress limits in the ASME Code, Section III, Subsection NG. The secondary stresses are due to the interference fit of the uranium on the shaft. The allowable stress values developed applying ASME Code, Section III, factors (Appendix III) to the mechanical properties of uranium are satisfied for analyzed stresses at normal operating speed.

SSAR Revision: NONE

Question 251.17

Discuss how the limit in Q251.16 is met for the flywheel enclosure and associated welds (Section 5.4.1).

Response:

The evaluation of the flywheel enclosure does not use the limit of one-third of minimum yield strength as a criterion for normal operating speed. The flywheel enclosure prevents contact of coolant with the uranium flywheel. No credit is taken in the analysis of the flywheel missile generation for the retention of the fragments by the enclosure, and the flywheel enclosure contributes only a small portion of the energy in a rotating flywheel assembly. The evaluation of the stress in the flywheel enclosure components and the seal welds connecting the components for normal and design speeds uses the criteria in Subsection NG of the ASME Code as a guideline. The ASME Code limits are satisfied for analyzed stresses at the normal operating speed and the design speed.

SSAR Revision: NONE

Question 251.18

Show that the combined stresses for the uranium flywheel at the design overspeed, due to centrifugal forces and the interference fit, is less than 2/3 of the minimum specified yield strength (Section 5.4.1).

Response:

The flywheel structural analysis verifies that the combined stresses in the uranium flywheel due to centrifugal forces and the interference fit at the design speed of 125 percent of normal speed are less than the limit of two-thirds of the minimum yield strength. The combination of primary and secondary stresses is also evaluated using stress limits in the ASME Code, Section III, Subsection NG. The secondary stresses are due to the interference fit of the uranium on the shaft. The allowable stress values developed applying ASME Code, Section III, factors (Appendix III) to the mechanical properties of uranium are satisfied for analyzed stresses at the design speed.

SSAR Revision: NONE

Question 251.19

Discuss how the limit in Q251.18 is met for the flywheel enclosure and associated welds (Section 5.4.1).

Response:

The evaluation of the flywheel enclosure does not use the limit of two-thirds of minimum yield strength as a criterion for design speed conditions. The criteria in the ASME Code, Section III, Subsection NG, are used as a guideline for stress limits. The ASME Code limits are satisfied for analyzed stresses at the design speed. See the response for Q251.17.

SSAR Revision: NONE

Question 251.20

Demonstrate that the shaft and the bearings supporting the flywheel will be able to withstand any combination of loads from normal operation, anticipated transients, the design basis of loss-of-coolant accident, and the safe shutdown earthquake (Section 5.4.1).

Response:

The containment of fragments from a postulated fracture of the flywheel is not dependent on the support of the shaft and flywheel by the bearings. Postulated failures of the bearings and shaft would result in the rotating assembly being slowed to a stop. Bearing or shaft failures would be indicated by vibration or temperature sensors. A postulated failure of a bearing or shaft that allowed excessive lateral movement would result in contact between one or more rotating parts and the surrounding structure thereby slowing the rotation. A postulated failure of a bearing or shaft that allowed excessive axial movement would not remove the restriction provided by the pump internals including the impeller and suction adapter. Thus a failure that would allow axial movement would not result in significant movement of the flywheel assembly out of the flange area.

Based on this information, the effect of these loads on the shaft and bearings is of interest with regard to operational reliability but not with regard to safe operation. The shaft and bearing supports are evaluated for loads due to seismic events.

SSAR Revision: NONE

Question 251.21

Identify the materials for the flywheel enclosure and associated welds. Provide technical justifications to show that the flywheel enclosure and associated welds are resistant to stress corrosion cracking, especially if Inconel 600 or 182 materials will be used (Section 5.4.1).

Response:

The material of construction of the flywheel assembly construction is nickel-chromium-iron Alloy 600. The material for the welding filler metal is Alloy 182 which has a similar composition. Since the coolant surrounding the flywheel assembly is normally at a relatively low temperature (approximately 165°F) compared to the bulk of the reactor coolant system, primary water stress corrosion cracking that has been found in some Alloy 600 parts exposed to reactor coolant at higher temperatures would not be expected.

SSAR Revision: NONE

Question 251.22

Demonstrate that the uranium flywheel is resistant to stress corrosion cracking or other potential degradation mechanism in a reactor coolant environment (Section 5.4.1).

Response:

The uranium alloy flywheel is sealed in the nickel-chromium-iron alloy encloure and is not in contact with the reactor coolant or other fluid. See the response to RAI 251.14 for additional discussion of the seal welds. The uranium alloy flywheel is heat treated by solution annealing in a vacuum furnace and slowly cooled. This heat treatment minimizes the potential for residual stresses. The heat treatment process also removes hydrogen from the material to reduce the potential for hydrogen embrittlement. Since the depleted uranium alloy is not in contact with reactor coolant or any other fluid and operates at a relatively low temperature, degradation of the material is not expected.

The fourth and ninth paragraphs of SSAR Subsection 5.4.1.3.6.3 will be revised as follows to reflect this response.

SSAR Revision:

The reactor coolant pump flywheel assembly is fabricated from a high-quality, depleted uranium alloy casting or forging. Castings are poured using a process to minimize the formation of voids, cracks, or other flaws. The forging process is also controlled to minimize the formation of flaws. ~~In addition, the heat treatment cycle~~ Subsequent to casting or forging, the flywheel is heat treated by solution annealing in a vacuum furnace and slowly cooled. This heat treatment minimizes the potential for residual stresses. The heat treatment process also removes hydrogen from the material to reduce the potential for hydrogen embrittlement. ~~is selected and controlled to prevent defect formation.~~

The uranium is sealed ~~contained~~ within a welded nickel-chromium-iron alloy enclosure to prevent contact with the reactor coolant or any other fluid. The enclosure minimizes the potential for corrorsion of the flywheel and contamination of the reactor coolant with depleted uranium. The enclosure material specifications are ASTM-B-168 and ASTM-B-564. Even though the welds of the flywheel enclosure are not external pressure boundary welds, these welds are made using procedures and specifications that follow the rules of the ASME Code. A dye penetrant and radiographic test of the enclosure welds is performed in conformance with these requirements.

Question 251.23

Table 5.4-2 in the SSAR lists the flywheel material specifications. Provide the technical basis for these specifications.

Response:

The material specification information including ultimate tensile strength and yield strength provided in Table 5.4-2 is based on material testing by the material supplier. The composition of the alloys, including the limits on the constituent elements, is also based on the experience of the material supplier. The production of uranium flywheel is controlled to minimize the formation of voids or other defects. The heat treatment process is controlled to provide the required material properties. See the response to RAI 251.22 for a discussion of the heat treatment. Quality assurance testing of the material verifies that the material supplied conforms to the material specification. Ultrasonic and liquid penetrant inspection are performed on the uranium flywheel to verify the absence of unacceptable defects. See the response to RAIs 251.9 and 251.13 for a discussion of the ultrasonic and liquid penetrant inspections.

SSAR Revision: NONE

Question 252.2

Section 3.6.3 of the SSAR indicates that the leak-before-break (LBB) methodology will be used to eliminate the dynamic effects of postulated pipe ruptures from the design basis. The SSAR indicates that the scope of LBB application is high-energy ASME Code Section III Class 1, 2, and 3 piping of 10 cm (4 in) in nominal diameter or larger. Identify specific piping being considered for LBB applications (see Q210.6).

Response:

The leak-before-break methodology is applied to the candidate high-energy lines in the nuclear island identified in Appendix 3E. See the response to Q210.6.

SSAR Revision: NONE

Question 252.12

Section 3.6.3.3 of the SSAR indicates that "part through-wall flaws" may be considered at the critical locations. This is not consistent with the requirements of a LBB analysis. Provide information to clarify this statement.

Response:

Stability is established based on postulated part through-wall flaws when the normal condition stresses are very low relative to the faulted condition stresses. The low normal condition stresses result in a large calculated value for the through-wall leakage crack. This situation could occur in small-diameter piping systems (4 to 6 inches) that are well designed for normal condition thermal expansion loading. In this case the potential for a through-wall crack is very low since the normal condition stresses are low and the use of part through-wall flaws for the stability evaluation provides a satisfactory level of safety.

SSAR Revision: NONE

Question 252.13

Section 3.6.3 of the SSAR discusses feedwater and steam piping. The staff has not approved the application of LBB for these piping for power reactors. Provide additional discussion relating to potential susceptibility of feedwater and steam piping to degradation mechanisms, such as water/steam hammer and erosion/corrosion.

Response:

The steam and feedwater piping in the AP600 plant is not susceptible to failure from the effects of erosion/corrosion and steam/water hammer. The steam piping from the steam generator out to the containment penetration contains a low water content, which minimizes the potential for erosion/corrosion. The steam piping in operating plants with similar characteristics has not experienced erosion/corrosion. The AP600 feedwater piping from the containment penetration to the steam generator nozzle is made of erosion/corrosion-resistant materials such as SA 335 P22 or P91. The effects of steam/water hammer are considered in the design of the steam and feedwater piping. Slow-closing check valves are used in the feedwater piping system to minimize the effects of water hammer. The layout of the feedwater piping prevents significant backflow of steam from the steam generator by incorporating an elbow with a downward slope at the steam generator inlet nozzle and by keeping the highest elevation of the piping at the steam generator inlet nozzle. This layout minimizes void formation at all fluid flow levels.

SSAR Revision: NONE

Question 252.14

The pressurizer surge line is potentially susceptible to thermal stratification. If the surge line is within the LBB scope, describe the ASME Section III fatigue "cumulati.. usage factor" for the surge line for the projected 60-year plant design life and the considerations given to the thermal stratification loads in the LBB analysis (Section 3.6.3).

Response:

The pressurizer surge line is within the LBB scope. The ASME Section III cumulative usage factor calculation for the pressurizer surge line includes transients caused by thermal stratification for the 60-year plant design life. This calculation includes the effect of thermal striping caused by the oscillations of the boundary between two fluids with different temperatures. The leak-before-break analysis includes the thermal expansion loads caused by thermal stratification. Specifically, the leak rate calculation will be performed using lower-bound loads under normal operating conditions, including the effects of thermal stratification. The flaw stability evaluations will be based on the maximum loads, including the effects of thermal stratification. The SSE loads are combined with the thermal stratification loads at normal 100 percent power.

SSAR Revision: NONE

Question 252.19

Recent fatigue test data indicates that the effects of the environment could significantly reduce the fatigue resistance of materials.[1] The specific concern relates to the reactor water and temperature environment and its synergistic interactions with strain rate. The recent data indicate that the design fatigue curves in ASME Section III Class 1 requirements may not be as conservative as originally intended. Describe the procedures that explicitly account for the effects of the environment in the fatigue analysis of components. (Cladding on base metal is not a structural material and should not be considered adequate to isolate the base metal from the effects of the environment. This is because the cladding may be breached exposing the base metal to the water environment. Further, the cladding does not insulate the base metal from the reactor temperature.) (Section 3.6.3).

Response:

The AP600 plant fatigue design for ASME Class 1 components is based on ASME Code, Section III, 1989 Edition and 1989 Addenda. The environmental effects described in the literature are being addressed as an industry issue. The basis of AP600 fatigue design will remain as described until a unified design approach representing the industry and regulatory positions is derived.

SSAR Revision: NONE

1. K. Iida, J. Fukakura, M. Higuchi, H. Kobayashi, S. Miyazono, and M. Nakao, "Survey of Fatigue Strength Data of Nuclear Structural Materials in Japan," Abstract of DBA Committee Report, 1988. (Presented to the American Society of Mechanical Engineers, Subgroup on Fatigue Strength, on December 5, 1988, in New York City, NY.) (Enclosure in letter, from J. Craig [NRC] to E. Griffing [Nuclear Management and Resources Council] dated July 2, 1991.)

2. M. Higuchi and K. Iida, "Fatigue Strength Correction Factors of Carbon and Low-Alloy Steels in Oxygen-Containing High-Temperature Water," Nuclear Engineering and Design, Volume 129, 1991, pp. 293-306.

3. J. B. Terrell, "Effect of Cyclic Frequency on the Fatigue Life of ASME SA-106-C Piping Steel in PWR Environments," Journal of Materials Engineering, Volume 10, Number 3, 1988, pp. 193-203.

Question 252.21

Section 3.8.2.6 of the SSAR indicates that the containment vessel is coated to a level just below the concrete. Provide technical justifications for not coating the portion of the containment vessel that is embedded in concrete.

Response:

The seals, provided at elevation 108 feet inside the containment vessel and at elevation 100 feet outside the vessel, prevent moisture and water from entering between the concrete and the steel vessel.

The waterproofing membrane and waterstops protect the containment vessel from exterior groundwater or flooding.

Cavities inside the containment have drains leading to sumps, thereby avoiding accumulation of water on the floors.

The above described provisions protect the portion of the containment vessel that is embedded in concrete without the need for surface coating.

SSAR Revision: NONE

Question 252.22

Discuss the need for cathodic protection of the containment vessel to protect from ground water corrosion and stray current corrosion. Also, describe considerations for the location and type of cathodic protection anodes, i.e., deep bed versus mat-type anodes (Section 3.8.2).

Response:

As discussed in Subsection 3.4.1.1.1 of the SSAR, the seismic Category I structures are protected against flooding by waterproofing membranes and waterstops. Cathodic protection is not required because water from flooding or ground water is not able to enter the steel containment vessel. These measures also prevent corrosion due to stray currents.

SSAR Revision: NONE

Question 252.23

Article NE-3121 of the ASME Code requires the consideration of corrosion in the design of the containment. Specifically, the containment thickness is to be increased over that determined by the design formulas in the ASME Code to account for corrosion. Provide a corrosion allowance for the projected 60-year plant design life with the associated technical basis (Section 3.8.2).

Response:

Corrosion of the containment vessel has been addressed in SSAR Subsection 3.8.2.6. The coatings are discussed in SSAR Subsection 6.1.2. Additional thickness has been provided in the embedment transition zone. No increase in thickness is required in the portions of the bottom head that are fully embedded in concrete, nor in the vessel above the transition zone, which is protected by the inorganic zinc coating. This coating will be inspected and maintained as part of the coating performance monitoring program described in Subsection 6.1.2. Containment corrosion is also discussed in responses to Questions 252.24 through 252.28.

SSAR Revision: NONE

Question 252.24

Demonstrate that the containment vessel is designed and provided with access to permit the performance of inspection, maintenance, and repair of all exterior and interior surfaces of the containment vessel, except for the portion embedded in concrete (Section 3.8.2).

Response:

Except for the portion embedded in concrete, the containment vessel has access to permit inspection, maintenance, and repair of all exterior and interior surfaces as discussed below.

The containment air baffle, adjacent to the exterior surface of the containment vessel, is configured to permit visual inspection and maintenance of the containment vessel as described in Subsection 3.8.4.1.3 of the SSAR. Other areas adjacent to the exterior surface of the containment vessel provide free access to it for inspection, maintenance, and repair.

There are no structures adjacent to the interior surface of the containment vessel that would preclude its inspection, maintenance, and repair. The in-containment refueling water storage tank, which is the closest structure to the containment vessel, is separated from it by 2'-0". The operating deck, at elevation 135'-3", is 2 feet thick and is separated from the containment vessel by 3 inches. Some HVAC ductwork and equipment are also within a few inches from the containment vessel wall. However, this will not preclude inspection, maintenance, and repair of the containment vessel wall.

SSAR Revision: NONE

Question 252.25

The exterior surface of the containment vessel may be exposed to weather conditions. Discuss the effects of weather on the corrosion of the exterior surface of the containment vessel (Section 3.8.2).

Response:

The exterior of the containment vessel will be coated with an inorganic zinc coating as discussed in SSAR Subsection 6.1.2. The use of the inorganic zinc coating in conjunction with a coating performance monitoring program will maintain the corrosion resistance of the containment vessel exterior. The maintained inorganic zinc will provide corrosion protection for the vessel exterior for the life of the plant.

SSAR Revision: None

Question 252.26

Discuss the potential for corrosion of the containment vessel within the middle annulus area of the shield building, i.e., the area bounded above by a seal and below by concrete. For example, trapped moisture or fluid may cause accelerated corrosion of the containment vessel (Section 3.8.2).

Response:

The exterior of the containment vessel, including the middle annulus area, is coated with inorganic zinc coating as discussed in SSAR Subsection 6.1.2. The middle annulus area is also provided with ventilation from the VAS system as discussed in Subsection 9.4.3.2.4 of the SSAR. These features, along with coating performance monitoring, provide protection against corrosion.

SSAR Revision: NONE

Question 252.27

Discuss the potential effects of corrosion on the reactor vessel containment due to a leak in the passive containment cooling system water storage tank atop the shield building (Section 3.8.2).

Response:

Water leaking from the passive containment cooling water tank could leak onto the containment vessel and then flow down to the floor of the upper annulus. The slope of this floor would direct the flow away from the vessel and into the floor drains. The inorganic zinc coating on the exterior of the containment provides corrosion protection in case of leakage from the passive containment cooling water tank. The inorganic zinc coating will be maintained through coating performance monitoring.

SSAR Revision: None

Question 252.39

Discuss conformance of the control rod drive system with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and Regulatory Guide 1.85. Provide technical justifications for any deviations, or provide acceptable alternatives (Section 4.5.1).

Response:

Regulatory Guide 1.37 references ANSI Standard N45.2.1, which has been incorporated into NQA-2 Part 2.1. The technical requirements specified in ANSI Standard N45.2.1 and NQA-2 Part 2.1 are compatible. Therefore, compliance with NQA-2 Part 2.1 satisfies Regulatory Guide 1.37.

Regulatory Guide 1.85 addresses the acceptability of code cases for materials under ASME Section III Division 1. Compliance with this regulatory guide is discussed in SSAR Appendix 1A.

SSAR Revision: NONE

Question 252.41

Section 4.5.1.4 of the SSAR indicates that the guidance in ASME NQA-2 will be used. However, ASME NQA-2 is not listed in Regulatory Guide 1.37 or 1.28, "Quality Assurance Program Requirements." Provide technical justifications for using ASME NQA-2.

Response:

USNRC Standard Review Plan Section 17.3, Revision 0, "permits the use of up-to-date industry consensus standards." The ANSI N45.2 series of standards referenced by the current revisions of many regulatory guides have been replaced by ASME NQA-1 and NQA-2. ANSI N45.2.1, which is referenced in Regulatory Guide 1.37, has been incorporated into NQA-2 Part 2.1. The technical requirements specified in ANSI N45.2.1 and NQA-2 Part 2.1 are compatible. Therefore, compliance with NQA-2 Part 2.1 satisfies Regulatory Guide 1.37.

SSAR Revision: NONE

Question 252.44

Section 4.5.2.1 of the SSAR indicates that only a few materials will be used for the reactor internals and core supports. If other materials will also be used, identify them and address related concerns that have been raised on the control rod drive structural materials (see Q252.29 - Q252.41), if they are applicable to the material used for the reactor internals or core supports.

Response:

Only stainless steels of the types 304LN, type 316 strain hardened, type 316, Nitronic 50, and 403 (Modified) material are used in the reactor internals and core support structures. Cobalt based alloys are not used as base material although weld deposited cobalt hardfacing is currently used on the wear surface of the radial key, clevis inserts and alignment pins. Refer to RAI 252.31 for a discussion of programs to replace the cobalt hardfacing materials. Refer to RAI 252.15 for a discussion of LN grade materials.

SSAR Revision:

4.5.2.1 Materials Specifications

The major core support material for the reactor internals is SA-182,SA-479 or SA-240 Type 304LN stainless steel. For threaded structural fasteners the material used is strain hardened Type 316 stainless steel. Remaining internals ~~P~~parts not fabricated from Type 304LN stainless steel typically include wear surfaces such a guide tube cards (Nitronic™ 50), and hardfacing on the radial key, clevis insert, alignment pin (Stellite™ 156 or low cobalt hardfaces); dowel pins (Type 316); hold down spring (Type 403 stainless steel (modified)); and irradiation specimen springs (Type 302 Stainless Steel). Core support structure and threaded structural fastener materials are specified in the ASME Code, Section III, Appendix I as supplemented by Code Cases N-60 and N-4. ~~bolts and dowel pins, which are fabricated from Type 316 stainless steel. The holddown spring is Type 403 stainless steel. There are no other materials used in the reactor internals or core support structures that are not included in ASME Code, Section III, Appendix I.~~

Question 252.50

Section 4.5.2.5 of the SSAR indicates that the guidance in ASME NQA-2 will be used with the reactor internals and core support structures. However, ASME NQA-2 is not listed in Regulatory Guide 1.37 or 1.28. Provide technical justifications for using ASME NQA-2.

Response:

USNRC Standard Review Plan Section 17.3, Revision 0, "permits the use of up-to-date industry consensus standards." The ANSI N45.2 series of standards referenced by the current revisions of many regulatory guides have been replaced by ASME NQA-1 and NQA-2. ANSI N45.2.1, which is referenced in Regulatory Guide 1.37, has been incorporated into NQA-2 Part 2.1. The technical requirements specified in ANSI N45.2.1 and NQA-2 Part 2.1 are compatible. Therefore, compliance with NQA-2 Part 2.1 satisfies Regulatory Guide 1.37.

SSAR Revision: NONE

Question 252.51

Table 5.2-1 of the SSAR lists "typical" material specifications for the reactor coolant pressure boundary (RCPB). Specify the actual materials for staff review.

Response:

RCPB materials will be those listed in Table 5.2-1.

SSAR Subsection 5.2.3.1 will be revised as follows:

SSAR Revision:

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Materials Specifications

Table 5.2-1 lists ~~typical~~ material specifications used for the principal pressure-retaining applications in Class 1 primary components and reactor coolant system piping. Material specifications with grades,classes or types are included for the reactor vessel components, steam generator components, reactor coolant pump, pressurizer, core makeup tank, and the passive residual heat removal heat exchanger. Subsection 5.4.3 defines reactor coolant piping. See Subsection 4.5.2 for ~~typical~~ material specifications used for the core support structures and reactor internals. See appropriate sections for internals of other components. Engineered safeguards features materials are included in Subsection 6.1.1.

Table 5.2-1 material specifications are the ~~representative of~~ materials used in the AP600 reactor coolant pressure boundary ~~and reactor internals~~.

The materials used in the RCPB conform to the applicable ASME Code rules.

Question 252.52

Provide more details relating to the material specifications in Table 5.2-1 of the SSAR. For example, the reactor coolant piping is listed as SA376. However, SA376 can be further characterized by "type" with different properties. Section 3B.2.2 of the SSAR indicates that Type 316LN will be used (see Q252.15). Provide detailed information on RCPB materials in Table 5.2-1 of the SSAR.

Response:

Table 5.2-1 of the SSAR will be revised as follows:

SSAR Revision:

Table 5.2-1 (Sheet 1 of 4)

Reactor Coolant Pressure Boundary Materials Specifications

Component	Material	Class, Grade, or Type
Reactor Vessel Components		
Head plates (other than core region)	SA-533 or SA-508	GR B, CL 1 or CL 3
Shell courses	SA-508	CL 3
Shell, flange, and nozzle forgings	SA-508	CL 3
Nozzle safe ends	SA-182	F316LN
Appurtenances to the control rod drive mechanism (CRDM)	~~SB-166 or~~ SB-167 ~~and~~or SA-182	TP690 or F304LN, F316LN
Instrumentation tube appurtenances, upper ~~lower~~ head	~~SB-166 or~~ SB-167 ~~and~~or SA-182, SA312, SA376	TP690 or F304LN, F316LN
Closure studs~~, nuts, washers, inserts and adapters~~	SA-540	GR B23 or GR B24, CL 3
~~Core support pads~~	~~SB-166 or SA-508~~	
Monitor tubes and vent pipe	SA-312 or SA-376 or SB-166, SB-167 ~~or SA-182~~	TP304LN, TP316LN or TP690
~~Seal ledge~~	~~SA-616 or SA-533~~	
~~Vessel supports and head lift lugs~~	~~SA-533~~	

Table 5.2-1 (Sheet 2 of 4)

Reactor Coolant Pressure Boundary Materials Specifications

Component	Material	Class, Grade, or Type
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-CR-Fe Weld Metal F-Number 43	
Steam Generator Components		
Pressure plates	SA-533	GR B, CL 1
Pressure forgings (including nozzles and tube sheet)	SA-508	CL 3
Nozzle safe ends	SA-182~~Stainless Steel Weld Metal Analysis A-8~~	F316LN
Channel heads	SA-508	CL 3
Tubes	SB-163	TP690TT
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Inconel~~Ni-Cr-Fe~~ Weld Metal UNSN06052 and W86152 ~~F-Number 43~~(Code Cases 2142 and 2143)	
Manway~~Closure~~ studs/nuts	SA-193, SA-194	GR B7
Pressurizer Components		
Pressure plates	SA-533	GR B, CL 1
Pressure forgings	SA-508	CL 3
Nozzle safe ends	SA-182	F316LN
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Inconel~~Ni-Cr-Fe~~ Weld Metal UNSN06052 and W86152~~F-Number 43~~(Code Cases 2142 and 2143)	

Table 5.2-1 (Sheet 3 of 4)

Reactor Coolant Pressure Boundary Materials Specifications

Component	Material	Class, Grade, or Type
Manway~~Closure~~ studs/nuts	SA-193, SA-194	GR B7
Reactor Coolant Pump		
Pressure forgings	SA-182 or SA-336	F304LN, F316LN
Pressure casting	SA-351 or SA-352	CF3A
Tube and pipe	SA-213; SA-376 or SA-312	TP304LN, TP316LN
Pressure plates	SA-240	304LN, 316LN
~~Bar materials~~	~~SA-479~~	
Closure bolting	SA-193~~, SA-320,~~or SA-540 ~~or SA-453~~	GR B7 or GR B24,CL 4
Reactor Coolant Piping		
Reactor coolant pipe	SA-376	TP304LN, TP316LN
Reactor coolant fittings, branch nozzles	SA-376, SA-182	TP304LN, TP316LN
Surge line	SA-376	TP304LN, TP316LN
RCP piping other than loop and surge line	SA-312 and SA-376	TP304LN, TP316LN
CRDM		
Latch housing	SA-336~~SA-182~~	F304LN, F316LN
Rod travel housing	SA-336~~SA-182, SA-479~~	F304LN, F316LN
Welding materials	Stainless Steel Weld Metal Analysis A-8	
Valves		
Bodies	SA-182 or SA-351	F304LN, F316LN or CF3A
Bonnets	SA-182,SA-240 or SA-351	F304LN, F316LN, 304LN, 316LN or CF3A

Table 5.2-1 (Sheet 4 of 4)

Reactor Coolant Pressure Boundary Materials Specifications

Component	Material	Class, Grade, or Type
Discs	SA-182, SA-564 or SA-351	F304LN, F316LN or GR 630 or CF3A
Stems	~~SA-182~~SA-479 or SA-564	F316, F316LN or GR 630
Pressure retaining bolting	SA-453 or SA-564	GR 660 or GR 630
Pressure retaining nuts	SA-453 or SA-194	GR 6 or TP410
Core Makeup Tank		
Pressure plates	SA-533 or SA-240	GR B, CL 1 or 304L, 304LN, 316L, 316LN
Pressure forgings	SA-508 or SA-182, SA-336	CL 3 or F304L, F316L
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43	
Passive Residual Heat Removal Heat Exchanger		
Pressure plates	SA-240~~SA-533~~	304L, 304LN
Pressure forgings	SA-336~~SA-508~~	F304L, F304LN
Cladding and buttering	Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-FE Weld Metal F-Number 43	
Tubing	SA-376, SA-213~~SB-163,nickel-iron-chromium alloy or stainless steel~~	TP304LN, TP316LN

Question 252.56

Recent fatigue test data indicate that the effects of the environment could significantly reduce the fatigue resistance of materials (see Q252.19). The specific concern relates to the reactor water and temperature environment and its synergistic interactions with strain rate. The recent data indicate that the design fatigue curves in ASME Section III Class 1 requirements may not be as conservative as originally intended. Describe the procedures that explicitly account for the effects of the environment in the fatigue analysis of components in the RCPB. (Cladding on base metal is not a structural material and should not be considered adequate to isolate the base metal from the effects of the environment. This is because the cladding may be breached, exposing the base metal to the water environment. Further, the cladding does not insulate the base metal from the reactor temperature.)

Response:

The AP600 fatigue design for ASME Class 1 components of the reactor coolant pressure boundary is based on ASME Code, Section III, 1989 Edition and 1989 Addenda. The environmental effects described in the literature are being addressed as an industry issue. The basis of AP600 fatigue design will remain as described until a unified design approach representing the industry and regulatory positions is derived.

SSAR Revision: NONE

Question 252.58

Section 5.2.3.3.1 of the SSAR indicates that Westinghouse has conducted a test program to show that the fracture toughness properties of low-alloy materials are "adequate." Demonstrate that the requirements of Subsection NB of Section III of the ASME Code are satisfied by the materials used in the RCPB.

Response:

Subsection 5.2.1.1 satifies to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, for the design and fabrication of the reactor coolant pressure boundary (RCPB) components. Therefore, the requirements of Subsection NB of Section III of the ASME Code are satisfied by the materials used for the RCPB.

SSAR Revision: NONE

Question 252.67

Grinding of austenitic stainless steel materials may introduce susceptibility to stress corrosion cracking. EPRI Report NP-6780-L provides certain controls on grinding recommended by the industry. Describe the controls Westinghouse recommends be imposed on grinding (Section 5.2.3).

Response:

The EPRI guidelines for controls on grinding are applicable to BWRs. Westinghouse does not recommend specific controls on grinding because of other factors in the AP600 design that eliminate susceptibility to stress corrosion cracking, i.e., LN-grades of austenitic stainless steels and environment. The response to Q252.15 provides further discussion of stress corrosion cracking susceptibility.

SSAR Revision: NONE

Question 252.73

Section 1A of the SSAR discusses conformance of the AP600 design with regulatory guides. The SSAR proposes exceptions to Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Wate. Cooled Nuclear Power Plants." The proposed alternative to Regulatory Guide 1.37 is based on staff guidance in Regulatory Guide 1.28, "Quality Assurance Program Requirements." Clarify the basis for the application of ASME NQA-2, which is not discussed in Regulatory Guide 1.28. Revise Section 1A of the SSAR accordingly, if appropriate.

Response:

USNRC Standard Review Plan Section 17.3 Revision 0, "permits the use of up-to-date industry consensus standards." The ANSI N45.2 series of standards referenced by the current revisions of many regulatory guides have been replaced by ASME NQA-1 and NQA-2. ANSI N45.2.1, which is referenced in Regulatory Guide 1.37, has been incorporated into NQA-2 Part 2.1. The technical requirements specified in ANSI N45.2.1 and NQA-2 Part 2.1 are compatible. Therefore, compliance with NQA-2 Part 2.1 satisfies Regulatory Guide 1.37.

The "Clarification/Summary Description of Exceptions," discussion for Regulatory Guide 1.37 in SSAR Appendix 1A will be revised as follows:

SSAR Revision:

The ANSI N45.2 series of standards that are referenced by the current revisions of the Quality Assurance Regulatory Guides have been replaced by ASME NQA-1 and NQA-2. ~~Refer to the Regulatory Guide 1.28 position.~~ ANSI N45.2.1, which is referenced in Regulatory Guide 1.37, has been incorporated into NQA-2 Part 2.1. The technical requirements specified in ANSI N45.2.1 and NQA-2 Part 2.1 are compatible. Therefore, compliance with NQA-2 Part 2.1 satisfies Regulatory Guide 1.37.

Question 252.74

Section 1A of the SSAR indicates conformance with the guidance in Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." Discuss how Regulatory Position C.3 will be met. Specifically, clarify that a procedure qualification will be established in accordance with Regulatory Position C.2 even though Regulatory Position C.1 is not applicable. Describe whether Regulatory Position C.3 will be met if the production welding procedure does not conform to the qualified procedure. Revise Section 1A of the SSAR accordingly.

Response:

To meet the requirements of Regulatory Position C.3, Westinghouse requires that vendor welding procedures and procedure qualification records be submitted for approval before they are used for manufacturing. Vendor plants and operations are regularly audited and monitored for compliance in accordance with the Westinghouse N-Stamp Quality Assurance Plan. All noncompliances are reported to Westinghouse for disposition. The disposition will necessarily comply with all ASME Code and Regulatory Guide requirements.

It is not the intent of Westinghouse to impose the conditions of Regulatory Position C.2 on either its vendors or itself since only fine grain materials are being used for the AP600.

SSAR Appendix 1A will be revised as follows:

SSAR Revision:

Reg. Guide 1.43, Rev. 0, 5/73 - Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

C.1-3	N/A	The AP600 material, specifically ASME SA-533 and SA-508 Class 3 made to a fine grain practice, is not subjected to the controls in this regulatory guide.

Question 252.75

Section 1A of the SSAR proposes exceptions to Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Provide technical justifications for not following the guidance in Regulatory Guide 1.44 or propose an acceptable alternative.

Response:

Appendix 1A of the SSAR will be changed to reflect that the criteria in Regulatory Guide 1.44 will be met.

SSAR Revision:

Reg. Guide 1.44, Rev. 0, 5/73 - Control of the Use of Sensitized Stainless Steel

C.1-6 Conforms

Question 252.82

Table 5.3-1 in the SSAR gives the same percentage of residual elements for the reactor vessel beltline forging and welds. Provide technical justifications for not lowering the residual element contents for the welds.

Response:

Maximum limits are specified for the copper, nickel, phosphorus, and vanadium contents in the reactor vessel welds (see the response to Q252.83). Using the existing trend curves, the maximum projected end of life, 54 EFPY, ΔRT_{NDT} is 24°F. Further reductions in alloying elements are not considered necessary for irradiation and may alter the properties of the weld.

SSAR Revision: NONE

Question 252.83

When the copper content of the reactor vessel beltline material is reduced, the susceptibility of the material to neutron irradiation may become dominated by other elements. Discuss the effects of not lowering the contents of nickel, phosphorous, and vanadium (Section 5.3).

Response:

Maximum limits are specified for the nickel, phosphorus and vanadium contents of the reactor pressure vessel beltline material, i.e., places where the fluence exceeds 1 x 10^{17}n/cm^2 at the end of life. Using the existing trend curves and the maximum projected end of life, 54 EFPY, ΔRT_{ndt} is 23°F. Further reductions in alloying elements are not considered necessary for irradiation and may alter the properties of the steel.

SSAR Table 5.3-1 will be revised as follows:

SSAR Revision:

Table 5.3-1

Maximum Limits for Elements of
the Reactor Vessel

Element	Beltline Forging (%)	As Deposited Weld Metal (%)
Copper	0.03	0.03
Phosphorus	0.012	0.012
Vanadium	0.05	0.05
Sulfur	0.015	0.015
Nickel	1.0	1.0

Question 252.85

Provide information to show that the reactor vessel materials will be heat-treated to achieve a fine grain microstructure (Section 5.3).

Response:

The reactor vessel material, SA508 Class 3, for the AP600 is supplied in a quenched and tempered condition to meet the minimum mechanical properties required by the ASME SA508 specification. The quenched and tempered condition of SA508 Class 3 material has a fine grain microstructure.

SSAR Revision: NONE

Question 252.86

Table 5.3-3 in the SSAR shows the value for "RT_{PTS}" required by 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." Provide details for this calculation, including assumptions and margins. The calculation should be based on the projected 60-year plant design life.

Response:

In 10 CFR 50.61, the NRC staff selected a conservative, uniform method for determining plant-specific values of RT_{PTS} at a given time in plant life.

For comparison with the screening criteria, the value of RT_{PTS} for the reactor vessel must be calculated for each weld and plate or forging in the beltline region using the following methodology:

RT_{PTS} = initial RT_{NDT} + M + ΔRT_{PTS}

where:

Initial RT_{NDT} = the initial reference temperature of the unirradiated material

M = Margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and the calculational procedures

$\Delta RT_{PTS} = (CF) * f^{0.28 - 0.1 * \log(f)}$

where:

CF = chemistry factor

f = EOL (54 EFPY) surface fluence

The following assumptions were used in the analysis:

- f for the forging is 1.632×10^{19} n/cm^2 (E > 1.0 MeV)
- f for the lower girth weld is 2.1×10^{18} n/cm^2 (E > 1.0 MeV)
- Wt. % Cu for the forging and weld metal is 0.03%
- Wt. % Ni for the forging and weld metal is 1.0%
- The initial RT_{NDT} for the forging and weld metal is -20°F

The following margins were used in the calculations:

For the forging:

M = 34°F for base metal with measured values of initial RT_{NDT}
M = 48°F for base metal with generic values of initial RT_{NDT}

Thus, since there is no actual measured data, the margin value chosen for the calculations was the generic value of 48°F.

For the lower girth weld:

M = 56°F for welds with measured values of initial RT_{NDT}
M = 66°F for welds with measured values of initial RT_{NDT}

Thus, since there is no actual measured data, the margin value chosen for the calculations was the generic value of 66°F.

Based on the 60-year design life (54 EFPY), the end-of-life (EOL) RT_{PTS} calculated values are:

Forging EOL RT_{PTS} = 51°F

Lower girth weld EOL RT_{PTS} = 70°F

SSAR Table 5.3-3 will be revised as follows:

SSAR Revision:

Table 5.3-3

End-of-Life RT_{NDT} and Upper Shelf Energy Projects

	Unirradiated		End-of-life		54 EFPY
	RT_{NDT} (°F)	USE (ft-lb)	RT_{NDT} (°F) 1/4T ID	USE (ft-lb) 1/4T Ib	RT_{PTS} (θ°F)
Beltline	-20	> 75	< 270	> 50	51
Head	10	N/A	< 270	> 50	N/A
Flange	10	N/A	< 270	> 50	N/A
Weld	10	N/A	< 270	> 50	N/A
Forging					51
Lower girth weld					70

Question 252.87

Westinghouse uses the guidance in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to estimate the extent of neutron embrittlement. However, there are uncertainties in neutron embrittlement prediction procedures. For example, Regulatory Guide 1.99, Revision 1, would predict a reference temperature shift of 30°C (54°F) based on the phosphorous content, which is not addressed in Regulatory Guide 1.99, Revision 2. Thus, in calculating the shift in the reference temperature, Method 1 or Method 2 (as discussed below) should be used, whichever is more limiting:

Method 1:

A shift should be calculated based on Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

Method 2:

A shift should be calculated accounting for the phosphorous content and technical justifications for the methodology should be provided. Or, as an alternative, a shift may be estimated using the following equation:

$$A = [\,40 + 1000\,(\,\%Cu - 0.08\,)$$

$$+ 5000\,(\,\%P - 0.008\,)\,]\;[\,f / 10^{19}\,]^{1/2}$$

where

A = predicted shift, °F

f = fluence, n/cm^2 (E > 1 MeV)

%Cu = weight percent of copper
(If %Cu $\leq$ 0.08, use 0.08.)

%P = weight percent of phosphorus
(If %P $\leq$ 0.008, use 0.008.)

Describe how this approach in estimating the reference temperature shift is met (Section 5 3).

Response:

The susceptibility of reactor pressure vessel steel to neutron irradiation has been monitored through surveillance programs. Trend curves are used to project the change in material properties as a function of irradiation fluence. Originally, these trend curves were based on data from test reactor simulations. As more in-plant surveillance data became available, the trend curves were modified to include the additional data. ASME SA 508 pressure vessel material is included in a number of operating plant surveillance capsule programs; thus, irradiation data is included in the current trend curves. Therefore, the current trend curves will be representative of the AP600 reactor pressure vessel. Thus, these trend curves should be used to project the change in vessel properties with accumulated fluence. It is expected that these projections, using the existing trend curves, will conservatively bound the actual change in properties.

Furthermore, Regulatory Guide 1.99, Revision 1, is a guide that describes the general procedures acceptable to the NRC staff on an interim basis for predicting the effects of residual elements of copper and phosphorus on neutron radiation damage to the low-alloy steels currently (1977) used for light-water-cooled reactor vessels. Regulatory Guide 1.99, Revision 1, also states, "Research and construction experience with low-residual-element compositions of these steels is accumulating rapidly and is expected to provide a firm basis for acceptable procedures in the near future."

Regulatory Guide 1.99, Revision 2, was based on the latest available data as of 1988 and includes considerably more data obtained from actual operating plants than was available at the time Regulatory Guide 1.99, Revision 1, was developed. Regulatory Guide 1.99, Revision 2, data indicates that the copper and nickel contents of the material are the dominant contributors when projecting the shift in RT_{NDT} of the material.

The Regulatory Guide 1.99, Revision 1, calculation requires an assumption that the lowest wt. % of copper in the material is 0.08 percent. However, the maximum allowable amount of copper is 0.03 percent in the AP600 beltline material. If this calculation is performed by using the value of 0.03 weight percent copper, the adjusted-reference-temperature predicted by Regulatory Guide 1.99, Revision 1, is 13°F.

The Regulatory Guide 1.99, Revision 1, calculation is also based on the assumption that the temperature shift in the Charpy V-notch curve for the irradiated material relative to that for the unirradiated material is measured at the 50 foot-pound energy level or measured at the 35-mil lateral expansion level. If these calculations are to be compared to the current calculational methods of Regulatory Guide 1.99, Revision 2, it would require an additional assumption that the adjustment of the reference temperature at the 30 foot-pound level is equal to the adjustment at the 50 foot-pound level and is also equal to the adjustment at the 35-mil lateral expansion level defined in Revision 1. In 1977 this was not believed to be the case as evidenced by Regulatory Guide 1.99, Revision 1, which states "The 'shift' of the adjusted reference temperature is defined in Appendix G as the temperature shift in the Charpy V-notch curve for the irradiated material relative to that for the unirradiated material, measured at the 50-foot-pound energy level or measured at the 35-mil lateral expansion level, whichever temperature shift is greater."

Thus, Westinghouse believes that the current Regulatory Guide 1.99, Revision 2, methodology, which is now part of 10 CFR 50, is the most reliable method to use when predicting the shifts in RT^{NDT} for the AP600 reactor vessel beltline material. Therefore, only results obtained by calculations using Regulatory Guide 1.99, Revision 2, will be reported.

SSAR Revision: NONE

Question 252.89

Section 5.3.2.4 of the SSAR discusses conformance with regulatory guides. The applicant should also discuss conformance with Regulatory Guide 1.37.

Response:

Section 5.3.2.4 will be revised. See also the response to Q252.73 for discussion on conformance with Regulatory Guide 1.37.

SSAR Subsection 5.3.2.4 will be revised as follows:

SSAR Revision:

Welding of ferritic steels and austenitic stainless steels is discussed in Subsection 5.2.3. Subsection 5.2.3 includes discussions on the degree of conformance with Regulatory Guide 1.44. Section 1.9 discusses the degree of conformance with Regulatory Guides, including 1.31 and 1.34 (if applicable), as well as 1.37, 1.43, 1.50, 1.71, and 1.99.

Question 252.92

The reactor vessel materials surveillance program depends on the estimated shift of the reference temperature according to ASTM E185-82. For establishing the surveillance program, estimate the shift in the reference temperature using the following methods:

(i) Method 1 discussed in Q252.80.

(ii) Method 2 discussed in Q252.80.

(iii) A shift should be assumed to be greater than 56°C (100°F) but less than 111°C (200°F).

The shift estimate should be based on Item (i), (ii), or (iii), whichever results in the largest temperature shift.

Because of uncertainties in current methods in estimating neutron embrittlement, the staff has established a minimum shift estimate in Item (iii) in developing a surveillance program for design certification. The staff concludes that the reactor vessel materials surveillance program plan should be based on a reasonably conservative estimate of the temperature shift. This is because it may be technically difficult to backfit an existing surveillance program should the actual temperature shift be higher than that estimated.

Describe how this approach in estimating the shift in the reference temperature for the surveillance program is met (Section 5.3).

Response:

Items (i) and (ii) reference Q252.80. This should be Q252.87. Based on the response to Q252.87, the methodology given in Regulatory Guide 1.99 Revision 2 will be used to calculate ARTs and ΔRTNDTs for the beltline materials.

Based on Regulatory Guide 1.99 Revision 2 and the current available data, the maximum predicted ΔRT_{NDT} of the beltline material is 24°F at EOL (54 EFPY). This calculated ΔRT_{NDT} would require the use of the withdrawal schedule given in ASTM E185-82 for a predicted transition temperature shift at the vessel inside surface of $\leq$ 100°F, which requires the withdrawal of only three capsules over the life of the vessel.

However, since ASTM E185-82 was intended to cover only a 40-year design life and the preliminary nature of the available data, the withdrawal schedule for the surveillance program will be based on the requirements in ASTM E185-82 for a vessel with a predicted transition temperature shift > 100°F but $\leq$ 200°F as suggested in item (iii) of the question.

SSAR Revision: NONE

Question 252.93

Appendix H to 10 CFR Part 50 requires the reactor vessel materials surveillance program to meet ASTM E185-82. ASTM E185-82 has been applicable to plants designed for 40 years, i.e., 32 effective full-power years (EFPYs) at end-of-life. Thus, the staff finds that the schedule in ASTM E185-82 should be maintained for 40 years (32 EFPYs).

Further, the schedule in ASTM E185-82 should be supplemented to address the period between 40 and the projected 60 years for the AP600. Propose a capsule withdrawal schedule beyond 32 EFPYs to demonstrate compliance with Appendix H of 10 CFR Part 50 to the end of the AP600's proposed design life of 60 years. One option may be to maintain the time interval between the last two capsule withdrawals within 32 EFPYs throughout the rest of plant design life, or at the end of the proposed 60-year plant design life, whichever is earlier.

For example, if a design certification applicant estimates that the reference temperature shift is greater than 56°C (100°F) but less than 111°C (200°F) using the procedures discussed in Q252.85, the capsule withdrawal schedule in ASTM E185-82 would require one capsule each to be withdrawn at 3, 6, 15, and 32 EFPYs, along with certain restrictions on fluence levels. This schedule should be followed for up to 32 EFPYs. In addition, propose a schedule beyond 32 EFPYs to the end of the proposed 60-year plant design life (Section 5.3).

Response:

Based on Regulatory Guide 1.99, Revision 2, and the current available data, the maximum predicted ΔRT_{NDT} of the beltline material is 24°F at EOL (54 EFPY). This would require only the use of the ASTM E185-82 withdrawal schedule for a predicted transition temperature shift of $\leq$ 100°F at the vessel inside surface. If this withdrawal schedule were to be followed, it would require that only three surveillance capsules be tested through EOL of the AP600 reactor vessel.

Since ASTM E185-82 was intended to cover only a 40-year design life and since available data is only preliminary, the withdrawal schedule for the AP600 surveillance program will be based on the requirements in ASTM E185-82 for a reactor vessel with a predicted transition temperature shift > 100°F but $\leq$ 200°F. In addition, to be even more conservative, the program will schedule five capsules to be withdrawn instead of four.

The AP600 surveillance capsule program will consist of eight surveillance capsules and be governed by the following withdrawal schedule:

CAPSULE	WITHDRAWAL TIME
1ST	When the accumulated neutron fluence of the capsule is 5×10^{18} n/cm^2.
2ND	When the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel 1/4T location.
3RD	When the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location.

4TH	When the accumulated neutron fluence of the capsule corresponds to a fluence not less than once or greater than twice the peak EOL vessel fluence.
5TH	EOL
6TH	Standby
7TH	Standby
8TH	Standby

The preceding schedule may be implemented regardless of a 40- or 60-year reactor vessel design life and will meet the requirements of ASTM E185-82 and Appendix H to 10 CFR 50.

SSAR Revision: NONE

Question 252.94

Provide information on the inclusion of standard reference materials in its surveillance capsules (Section 5.3).

Response:

According to ASTM E185-82, the inclusion of correlation monitor materials in a reactor vessel surveillance program is optional. Thus, correlation monitor materials will not be included in the AP600 reactor vessel surveillance program.

SSAR Revision: NONE

Question 252.96

Describe the "lead factors" for the surveillance capsules (Section 5.3).

Response:

The lead factors are based on an AP600 DOT3.5 R-Theta reactor internals heat generation rate analysis.

The surveillance capsule lead factor (LF) is defined as:

LF = [The Surveillance Capsule Fast Neutron Flux (E > 1.0 MeV)] ÷
[The Pressure Vessel Maximum Fast Neutron Flux (E > 1.0 MeV)]

Based on the following assumptions the surveillance capsule lead factors will range from 1.2 to 3.0;

- The pressure vessel maximum fast neutron flux = 9.58E+09 n/cm^2-sec.
- The radial location of the pressure vessel maximum fast neutron flux is 199.7 centimeters (pressure vessel inner surface).
- The azimuthal locations of the pressure vessel maximum fast flux is $\pm$ 23.2° about the center of the reactor core flats.
- The surveillance capsule center is at a radius of 73.31 inches.
- The R-Theta model does not include the surveillance capsule or the capsule holder geometry.

The range of surveillance capsule lead factors (1.2 to 3.0) defined herein is based on a non-modeled capsule/holder analysis. Standard plant analyses show a 25 percent increase in fast neutron flux with the surveillance capsule/holder modeled. An analysis will be performed for the combined operator license application with the capsule/holder modeled in order to more accurately define the surveillance capsule lead factors and azimuthal locations.

SSAR Revision: NONE

Question 252.101

The staff's concern relating to the environmental effects on fatigue has been discussed in Q252.19 and Q252.49. This is applicable to all materials. Address this concern for the reactor vessel materials (Section 5.3).

Response:

The AP600 fatigue design for ASME components of the reactor vessel is based on ASME Code, Section III, 1989 Edition and 1989 Addenda. The environmental effects described in the literature are being addressed as an industry issue. The basis of AP600 fatigue design will remain as described until a unified design approach representing the industry and regulatory positions is derived.

SSAR Revision: NONE

Question 252.102

Discuss design considerations for facilitating an in-place reactor vessel thermal annealing treatment should this become necessary (Section 5.3).

Response:

The feasibility of annealing a commercial nuclear reactor pressure vessel was first addressed by Mager et al.[1,2,3] Similar studies were reported by Server.[4] Guidelines for an in-situ thermal annealing treatment of nuclear reactor pressure vessels are given in ASTM E509.[5] These five documents form the basis of the design considerations for facilitating an in-place reactor vessel thermal annealing treatment of the AP600 should this become necessary. These are discussed in more detail below.

ANNEALING DESIGN CONSIDERATIONS	AP600 DESIGN
General insitu annealing feasibility	Feasibility established in references 1 through 4. Since the vessel configuration/shape of the AP600 is similar to the generic vessels investigated in these studies, the general conclusions are similar for the AP600. These studies evaluated vessel designs, effects on the vessel materials, insulation, concrete, RCS piping and associated equipment supports, internals and fuel storage capabilities, personnel radiation exposure, vessel integrity through thermal/stress analyses, and safety concerns.

Knowledge of : 1. reactor vessel material chemical composition 2. mechanical properties 3. fabrication techniques 4. fabrication history 5. nondestructive test results 6. anticipated stress levels in high fluence zones 7. neutron fluence 8. fluence rate energy spectrum 9. expected operatingtemperature 10. power history 11. initial RT_{NDT} 12. initial Charpy upper shelf energy for materials of concern in the beltline region	 1. Known and documented for AP600. 2. Known and documented for AP600. 3. Known and documented for AP600. 4. Known and documented for AP600. 5. Known and documented for AP600. 6. Known and documented for AP600. 7. Responsibility of license holder during service, estimated neutron fluence for design life documented for AP600. 8. Responsibility of license holder during service, estimated fluence rate energy spectrum documented for AP600. 9. Design expected operating temperature documented for AP600. 10. Responsibility of license holder. 11. Known and documented for AP600. 12. Known and documented for AP600.
Available sample material from heats of base metal, weld metal , and heat-affected zones located in reactor vessel beltline for additional surveillance capsules.	For the AP600, archive material from heats of material located in the reactor vessel beltline will be available in sufficient quantity to fabricate two additional capsules if it should become necessary.
Surveillance program in accordance with requirements of ASTM E-185.	For the AP600, the surveillance program conforms to ASTM E-185-82.
Transition temperature and upper shelf Charpy energy level data versus neutron irradiation.	For the AP600, Regulatory Guide 1.99, Rev. 2 and the material surveillance capsule program are used.
Accessibility to the reactor vessel to allow inspection and temperature monitoring.	For the AP600, inspections and temperature monitoring can be done on the inside and outside of the reactor vessel, if it should become necessary.

Degree of material properties recovery for the specific materials in the beltline of the vessel.	For the AP600, sufficient material (either surveillance program materials or archive materials) are available for determining material properties recovery, if it should become necessary.
Radiation exposure of personnel for annealing.	Reference 3 documents an annealing procedure which quantifies and minimizes the radiation exposure of personnel. For the AP600, a similar procedure would be applicable, if it should become necessary.
Detailed annealing procedure.	References 1, 2, and 4 provide annealing procedures. For the AP600, similar procedures would be applicable, if it should become necessary.
Detailed thermal and structural analysis to demonstrate acceptable: - dimensional stability - thermal stress gradients - residual stress - concrete temperatures.	References 1 and 4 document thermal and stress analyses which conclude that in-situ annealing is feasible and that vessel integrity and associated structures are not a problem if proper annealing equipment design and temperature control features are instituted. Based on the AP600 similarities with the vessels used in these studies a similar conclusion is reached.
Storage of core, internals, and coolant.	For the AP600 storage of the core, internals, and coolant will not impact an annealing operation, if it should become necessary.
Annealing verification.	Material available from the AP600 surveillance capsule program and archive materials can be used to determine annealing effectiveness, if it should become necessary.
Annealing equipment laydown area.	Sufficient space is available in the AP600 design for annealing equipment assembly, maintenance, and disassembly.

[1] T. R. Mager and R. D. Rishel, NP-2493: Development of a Generic Procedure for Thermal Annealing an Embrittled Reactor Vessel Using a Dry Annealing Method, EPRI Research Project 1021-1, Interim Report, July 1982.

[2] T. R. Mager et al., NP-2712: Feasibility of and Methodology for Thermal Annealing an Embrittled Reactor Vessel, EPRI Research Project 1021-1 Final Report, November 1982.

[3] T. R. Mager et al., NP-6113: Thermal Annealing of an Embrittled Reactor Vessel - Feasibility and Methodology, EPRI Research Project 1021-1 Final Report, January 1989.

[4] W. L Server, NUREG/CR-4212: In-Place Thermal Annealing of Nuclear Reactor Pressure Vessels, Prepared for Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, April 1985.

[5] ASTM E 509-86, "Standard Guide for In-Service Annealing of Light-Water Cooled Nuclear Reactor Vessels".

SSAR Revision: NONE

Question 252.104

There are uncertainties in neutron embrittlement prediction procedures. Thus, for establishing the reactor vessel pressure-temperature limits prior to the availability of valid plant specific surveillance data, estimate the shift in the reference temperature using either Method 1 or 2 as described in Q252.80, whichever is more limiting. The reference temperature shift should be based on the proposed design life of 60 years. The COL applicant will be requested to commit to reviewing the continued applicability of the pressure-temperature limits when plant specific surveillance data become available. Provide information to show that this approach in establishing pressure-temperature limits is met (Section 5.3.3).

Response:

The reference to Q252.80 should be Q252.87. Based on the response to Q252.87, the methodology given in Regulatory Guide 1.99, Revision 2, will be used to estimate the shift in the reference temperature.

The applicability of the pressure-temperature limits to a specific plant will be reviewed by the combined operator license holder as creditable surveillance capsule data becomes available.

SSAR Revision: NONE

Question 252.105

Provide details for the pressure-temperature limit calculations, including assumptions and margins. Estimate the shift in the reference temperature according to Q252.97. Further, identify any deviations from the recommended calculational procedures in Section 5.3.2 of the SRP (Section 5.3.3).

Response:

The reference to Q252.97 should be Q252.87. Based on the response to Q252.87, the methodology given in Regulatory Guide 1.99, Revision 2, will be used to estimate the shift in reference temperature.

The heatup and cooldown limit curves were calculated using the most limiting value of the adjusted reference temperature (ART). The methodology used to calculate the ART is given in the Regulatory Guide 1.99, Revision 2. A description of the methodology and assumptions used in the analysis follows:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

where

Initial RT_{NDT} = reference temperature of the unirradiated material

ΔRT_{NDT} = (CF) * $f^{(0.28-0.10 \log f)}$

CF = chemistry factor, a function of copper and nickel content

$f = f_{surf} * (e^{-0.24X})$

f_{surf} = neutron fluence at the inner wetted surface of the vessel (10^{19} n/cm^2, E > 1.0 MeV)

X = depth into the vessel wall measured from the vessel inner surface (inches)

Margin = 2 * $[(\sigma_I)^2 + (\sigma_\Delta)^2]^{1/2}$

σ_I = standard deviation for the initial RT_{NDT}

σ_Δ = standard deviation for ΔRT_{NDT}

The assumptions used in calculating the 1/4T and 3/4T ARTs at the end of the 60-year design life (54 EFPY) for the forging and lower girth weld are as follows:

1) f_{surf} for the forging is 1.632 x 10^{19} n/cm^2 (E > 1.0 MeV)

2) f_{surf} for the lower girth weld is 2.1×10^{18} n/cm^2 (E > 1.0 Mev).
3) reactor vessel thickness at the beltline region is 8 inches.
4) weight percent Cu for the forging and lower girth weld metal is 0.03%.
5) weight percent Ni for the forging and lower girth weld metal is 1.0%.
6) The initial RT_{NDT} for the forging and lower girth weld metal is -20°F.

The shift in the reference temperature at 1/4T and 3/4T for the forging weld metal is discussed in the response to Q252.104.

The "Margin" is the quantity, °F, that is to be added to obtain conservative, upper-bound values of adjusted reference temperature. If a measured value of initial RT_{NDT} for the material in question is used, σ_I may be taken as zero. If a generic value is used, σ_I should be obtained from the same set of data. The standard deviations from Regulatory Guide 1.99, Revision 2, for ΔRT_{NDT}, σ_Δ is 17°F for base metal and forgings and 28°F for weld metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}. The value of the initial RT_{NDT} is an estimated value. For this reason, a value of 17°F is used for σ_I for both the forging and the weld metal. Thus, the following margins were used in the calculations:

M = 39°F @ 1/4T location of the forging
M = 37°F @ 3/4T location of the forging
M = 39°F @ 1/4T location of the lower girth weld
M = 36°F @ 3/4T location of the lower girth weld

The calculated end-of-life (54 EFPY) 1/4T and 3/4T ARTs for the forging and weld metal are as follows:

ART @ 1/4T of forging = 40°F
ART @ 3/4T of forging = 32°F
ART @ 1/4T of lower girth weld = 38°F
ART @ 3/4T of lower girth weld = 28°F

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Section III of the ASME Boiler and Pressure Vessel Code. The calculational procedure employed is described in detail in Reference 1, which is in accordance with the USNRC Standard Review Plan. Mechanics of the OPERLIM code which is used to develop the allowable pressure-temperature relationships for normal heatup and cooldown rates, including criticality limits, are contained in Reference 2.

References:

1. Westinghouse Proprietary Class 3 Report "Basis for Heatup and Cooldown Limit Curves," WCAP-7924A.
2. Westinghouse Proprietary Class 2 Report "Documentation and Verification of the OPERLIM Computer Code," WCAP-9186.

SSAR Revision: NONE

Question 252.106

Demonstrate that its pressure-temperature limits are in accordance with Appendix G to 10 CFR Part 50. For example, verify that the limit for the closure flange is satisfied (Section 5.3.3).

Response:

The current pressure-temperature limit meets Appendix G flange requirements as follows:

- When the core is critical and when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F (67°C) for normal operation.

- Using a 60-psig pressure margin and a 10°F temperature margin for instrumentation errors, the limiting pressure and temperature for the flange region are Limiting Pressure = 561 psig for normal operation and Limiting Temperature = 140°F for normal operation.

Details of the overall approach used to determine heatup and cooldown curves and to meet all the fracture toughness requirements outlined in Appendix G are contained in Reference 1.

References:

1. WCAP-7924A, "Basis for Heatup and Cooldown Limit Curves," Westinghouse Proprietary Class 3.

SSAR Revision: NONE

Question 252.110

Section 5.4.2.3.3 of the SSAR discusses flow-induced vibrations with special emphasis on fluid elastic vibration. Provide the results of prototype tests and calculations to support the discussion.

Response:

The flow-induced vibration analysis of the AP600 steam generator using final design information, including support configuration and tube bundle fluid flow rates, is not complete. However, extensive testing and evaluation of the tube bundle designs for the Model F and Delta-75 steam generators have been performed. The Model F tube bundle has the same size tubes on a square pitch. The Delta-75 steam generator has the same tube bundle configuration, including tube size and triangular pitch, as the AP600 steam generator. The analytical models used to evaluate tube vibration have been validated with a number of flow tests using various tube sizes and pitch geometries.

The following discussion outlines testing and analysis performed for the Model F and Delta-75 steam generators that support the AP600 steam generator tube bundle configuration and are used to evaluate the potential for flow-induced vibrations. Two regions are of greatest interest in the evaluation of flow-induced vibration of steam generator t . The first of these is the inlet area at the bottom of the tube bundle where the water flowing down the annulus between the shell and wrapper turns and enters the tube bundle. The second area of interest is the U-bend region at the top of the tube bundle. In both of these regions the fluid is more or less in cross-flow over the tubes. A summary of the testing and analysis for each of these regions follows.

One of the items of interest in the inlet area is the effect of the flow distribution baffle on the flow fields and therefore on the vibration response. Basic vibration response to the flow distribution baffle was studied in cold flow tests with tube bundle sectors, including several tubes instrumented with both strain gages and accelerometers. Field tests of operating steam generators instrumented with accelerometers in the same region (References 1 and 2) verified that already acceptable straight-leg vibration levels in tube spans above the first tube support plate in conventional Model 51 steam generators had been reduced by an additional factor of 4 in Model F units with the flow distribution baffle.

Cold flow vibration tests were conducted on a 49-tube test model in water with flow oriented 45 degrees to the square pattern to quantify tube response and qualify analytical models used to predict response to cross-flow in the inlet region both with and without the flow distribution baffle. These tests provided measured tube vibration response to simulated inlet cross-flow from 10 to 200 percent of nominal full-power conditions and verified analytical predictions of frequencies and vibration amplitudes for this region of the steam generator. Reduction in vibration levels by more than a factor of 4 for tubes with flow distribution baffle contact versus tests without flow distribution baffle tube interaction was also demonstrated in these tests. The absence of tube response to either of the potential vortex shedding or fluid-elastic mechanisms supported the design bases.

Subsequent cold-flow tests were conducted on a 15-degree sector of the lower tube bundle region of the Model F steam generator. This model had seven times more tubes and concentrated on flows in the 100 to 140 percent of nominal full-power range. Test series were done with tube arrays oriented for flow at 0 to 45 degrees through the square pattern, and tubes were configured both with and without support at the flow distribution baffle. The model included the next two support plates. Tube vibration amplitudes, secondary fluid velocities, and dynamic forces at two support plates were measured for incremental flows ranging from 10 to 140 percent of nominal. Frequencies and vibration amplitudes were again consistent with analytical predictions. Tube dynamic characteristics at the support plates also indicated that the potential for tube degradation in this region is small at expected flow rates.

Reduced tube vibration response in the periphery of the inlet region, where flow excitation is generally higher, is a consequence of the shorter span between tube contacts with the flow distribution baffle in place. The Delta 75 has a similar spacing, and it has an increased probability of support due to slightly reduced clearances in the broached holes relative to both the tested and prior conventional design.

Correlations and empirical constants used in analyses of vortex shedding, fluid-elastic excitation, and turbulence had been derived based on years of laboratory testing at the Westinghouse Science and Technology Center. Vortex shedding is theoretically possible for the outermost tubes in the inlet cross-flow region. This had been demonstrated previously in carefully controlled laboratory tests when flow through the square array was staggered 45 degrees from the inline orientation. Two discrete Strouhal numbers that enveloped open literature predictions characterized tube response in the first and fifth tube rows in this orientation. No vortex shedding response was evident even in the carefully controlled laboratory environment for the inline flow configuration.

Periodic tube response characterized by a moving peak in the response spectrum as velocity increased could not be found in the 49-tube inlet cross-flow model for the full range of 10 to 200 percent of full-power nominal flow rates even when tested in the 45-degree, staggered-flow orientation. Tube response to vortex shedding, if present, was therefore too small to be observed over small, random turbulence effects. This was true for tests both with and without the flow distribution baffle in place. This is consistent with analyses based on expected fluctuating dynamic lift coefficients and correlation lengths characteristic of the steam generator flow distribution.

Both the potential for vortex shedding in carefully controlled tests, and its absence for full-size steam generator operating conditions are also applicable for triangular arrays (References 3 and 4]. Thus, in addition to extensive Westinghouse experience with square arrays, it is generally noted that vortex shedding is only a potential design problem in the peripheral tube rows of large arrays in liquid flows (Reference 4). Conservative tube response calculations are typically made assuming that vortex shedding occurs in the inlet region even though such response is not expected.

Root mean square tube displacements from the 49-tube water flow tests were consistent with measurements from an operating plant (Reference 1) and with analytical predictions, which were made using empirical constants that envelope the magnitude of tube response to the random turbulent force spectrum typical of operating steam generators. These constants had been derived earlier by Westinghouse based on single cylinder data from Y. C. Fung (Reference 5). Appropriate constants for both peripheral and interior tubes were demonstrated to be conservative.

Fluid-elastic tube vibration did not occur in any of the 49-tube model tests. This is consistent with analyses based on threshold instability constants determined from previous laboratory tests. Instability constants are a function of the tube array pattern and spacing, and appropriate values have also been determined from similar laboratory tests for triangular configurations similar to the Delta-75. Conservative reference values for use with the gap pitch velocity (area ratio times upstream velocity) were defined for both arrays with a 1.42 pitch-to-diameter ratio. Results are consistent with data available from literature. For the pitch-to-diameter ratio of interest, triangular arrays are more stable, so that operating experience is conservative relative to fluid-elastic response. Tube vibrations in the straight-leg region are therefore known to be small and predictable using conventional approaches along with empirical constants determined from tests appropriate to the steam generator design configuration.

Boundary conditions for vibration analyses are obtained from qualified three-dimensional thermal-hydraulic codes such as ATHOS (Reference 6).

Effects of the wrapper inlet, tube lane blocking devices, flow distribution baffle central cutout, flow distribution baffle broached flow areas, tube support plate trifoil flow areas, annular flow area between plates and wrapper, and tube array geometry are included in the straight-leg portion of the overall model. Resulting velocity and density distributions are used to scale forcing functions and set boundary conditions for evaluation of the limiting vibration mechanisms. Vibration analyses are conducted using qualified finite element analytical models using approaches derived from extensive testing at the Westinghouse Science and Technology Center (Refernces 7, 8, and 9).

Three potential secondary flow-excitation mechanisms are addressed: 1) vortex shedding, 2) turbulence, and 3) fluid-elastic excitation. The first is typically not a practical concern, except possibly for the outer few rows of tubes in the inlet region of steam generators for which nonuniform, two-phase turbulent flow exists throughout most of the tube bundle. Backup analyses are conducted for these outer-row tubes even though experimental results and field experience indicate that there is no vortex shedding response even here (probably due primarily to close tube spacing with high inlet turbulence and nonuniform velocities over the inlet spans). Strouhal numbers covering the range of values determined from carefully controlled laboratory experiments and external literature yield a range of potential synchronization frequencies. Correlation over the entire inlet flow span is then assumed, and tube response is calculated using an upper bound lift coefficient following methodology outlined by Connors (Reference 9). Calculated vibration amplitudes are typically less than the small calculated turbulence amplitudes, which are consistent with measured amplitudes in operating plants with years of operation without measurable tube wear. Resulting tube bending stresses are more than two orders of magnitude below ASME Code limits. Tube response to uncorrelated wake shedding in the bundle interior is covered by evaluation of random turbulence excitation.

Secondary flow turbulence throughout the bundle produces random tube displacements in a narrow frequency band that includes the natural frequency of the tube for the existing support configuration. Tube motions at locations that contact support plates are typically characterized by small amplitude displacements without liftoff, so that small fretting wear coefficients apply.

Fluid-elastic tube vibration is potentially more severe than either vortex shedding or turbulence because it is a self-excited mechanism: relatively large tube amplitudes can feed back proportionally large driving forces if an instability threshold is exceeded. Tube support spacing incorporated into the design of the tube support system provides tube response frequencies in such a way that the instability threshold is not exceeded for secondary fluid flow conditions. A factor of at least 2 is used as a guideline in establishing support spacing. This is typically imposed by requiring that the calculated stability ratio (effective velocity/threshold velocity) be 0.5 or less. This approach provides margin against initiation of fluid-elastic vibration for tubes effectively supported at nominal locations.

Fluid-elastic instability analyses are performed for straight-leg tube spans using fluid flow conditions from qualified thermal-hydraulic analyses. The methodology follows that of Connors (Reference 9) using the appropriate threshold instability constant for the Delta-75 array. Typical stability ratios are much less than unity, indicating ample margins against initiation of fluid-elastic vibration in straight-leg tube spans.

Instability is not predicted even when conservative support conditions covering unexpected loss of contact at any one support location are postulated. Analyses for steam generators with square arrays subjected to similar flow excitation yield results consistent with laboratory tests, which did not produce any fluid-elastic response (see above), and with field experience, which has shown no eddy current test indications attributed to wall thinning at any support location of Model F-type steam generators for operating times up to 10 years. Greater design margins apply to the Delta-75 as a result of the higher instability constant obtained from tests with the triangular array.

Analyses are also performed for other postulated support conditions to demonstrate margin against instability even if dense corrosion products are postulated to form in the tube/support clearance. (This is a conservative assumption based on tests and operating experience with broached 405 SS supports.) In the limit this is assumed to result in the tube being "clamped" or "fixed" rather than "pinned" so that the positive damping of the tube is reduced, thereby possibly reducing the margin against initiation of fluid-elastic instability (frequency is simultaneously increased). Appropriate reduced damping values are used following the same analytical approach defined by Connors (Reference 9).

Results of vibration analyses are used both to assess satisfaction of tube stress limits and to demonstrate adequate margins against unacceptable wear. Static stresses resulting from interaction with the tube sheet, flow distribution baffle, and tube support plates are also included when evaluating the tube. Maximum pressure plus thermal interaction effects are considered when calculating tube sheet rotation (bowing) and differential expansion contributions. Flow distribution baffle/tube interaction stresses are acceptably small because of design tolerances and special fabrication alignment procedures. Typical tube responses to flow-induced excitation in the straight-leg region of feedring steam generators are benign, and vibrations have small effects on margins against tube stress and fatigue limits.

Antivibration bars (AVBs) maintain tube-to-tube spacing, stiffen the tube bundle, and restrain vibration of the tubing U-bends above the top tube support plate. They are assembled in an advanced design configuration that was developed during a comprehensive program conducted over the past several years (Reference 10).

The advanced design configuration program was undertaken to eliminate the small percentage of tubes with moderate wall thinning in the U-bend region attributed to tube vibration and wear after 5 to 8 years of operation in some conventional operating steam generators with .875" OD x .050" T tubing and two sets of chromium-coated nickel-chromium-iron Alloy 600 AVBs. The mechanism leading to tube/AVB wear was established (fluid-elastic rattling within tube-to-AVB gaps), and design changes have been incorporated into the advanced configuration to provide enhanced margins against vibration-induced tube wear. These changes include a tighter tube-to-AVB fitup by both design and assembly, specifying the AVB material to be the same as the 405 stainless steel used in the Model F steam generator straight leg tube supports, and increasing the width of the AVB consistent with tube dryout and leak-before-break constraints. Nineteen steam generators incorporating some (four steam generators) or all (fifteen steam generators) of the advanced features have been fabricated (with the eight in service).

The basic objective of the advanced features is to avoid or minimize the consequences of fluid-elastic rattling (so called fluid-elastic vibration in the "support inactive mode," or "double-span behavior") between loosely fit AVBs in the U-bend region. Similar conclusions were reported by KWU relative to U-bend tube wear in a KWU steam generator configuration with relatively loose tube/support strip fitup. This mechanism is also the focus of ongoing studies described by typical literature from the United States (Reference 11), France (Reference 12), and England (Reference 13).

Extensive vibration testing has been conducted during the past 10 years to refine conventional design approaches and to support development of advanced U-bend/AVB design configurations. Basic design information for tube vibration in prototypic steam-water flow was generated during Model Boiler 2 (MB-2) tests in a 0.01 power scale model of the Model F steam generator. There was no evidence of periodic vortex shedding in these tests, which also provided a threshold instability constant when tested with AVBs removed (tubes were fluid elastically stable with AVBs installed). Turbulent tube response characteristics in the U-bend region with AVBs in place were enveloped by calculations using the same force spectra scaling factors qualified for the straight-leg region.

A quarter scale 12-row x 8-column square array of aluminum U-bends was tested in the wind tunnel in the same fluid-elastic vibration regime as steam generators with steam-water flows. These tests provided additional information, especially in establishing threshold fluid-elastic instability constants for various support conditions in the U-bend region. Results were consistent with MB-2 values for loose support conditions but were lower for various tight-fitup conditions. A lower bound value for U-bend analyses which is not the same as that for straight-leg response was derived from these tests. Tube/AVB dynamic interaction correlations were also established as a function of flow and fitup conditions.

Subsequently, a similar series of tests was conducted on a triangular array of the same size tubes in the same basic test rig, which has eight rows and nine columns of tubes configured to match the Delta-75 orientation in the U-bend. Results confirmed that this triangular array has an increased margin against instability in the U-bend region as it does in the straight-leg region.

Two series of wind tunnel experiments were conducted on cantilever tubes designed to simulate the response of curved U-bend tubes. A 7-row x 5-column array of full-size tubes mounted in such a way that orthogonal stiffnesses differed to match U-bend response provided two kinds of information. Basic fitup effects on tube response to both fluid-elastic and turbulent excitation were determined first. Contact with an AVB on one side of a gap with zero or very small preloads was effective in suppressing fluid-elastic tube response for low rates up to the maximum tested value that produced a stability ratio of four when tested without contact. Preloads on the order of 0.1 pound eliminated all impact/sliding motions with liftoff, which would otherwise result from both turbulence and fluid-elastic excitation for the tested diametral clearances. Both the threshold instability constant and turbulent tube response correlations were consistent with those derived from earlier tests. Then the test rig was modified and used to refine basic fluid-elastic driving force correlations for use in properly controlling mechanical shaker tests of full-size steam generator U-bends.

Mechanical excitation tests followed on full-size 0.687" OD x 0.040" T U-bends to characterize the wear-producing forces and motions at tube/AVB intersections. Parametric tests covered a range of fitup conditions subject to simulated out-of-plane fluid-elastic excitation, in-plane turbulence, and out-of-plane turbulence. Initial tests with four AVB intersections led to the fundamental conclusion that out-of-plane fluid-elastic vibration within tube/AVB gaps is the likely explanation for wear that had been observed in some operating steam generators. Subsequent tests with six AVB intersections simulated the excitation forces and fitup conditions characteristic of advanced design configurations. Wear-producing forces and motions were determined and recorded in the form of work rates for use in wear calculations (Reference 14). These work rates were verified by independent testing on the same full-size tube using a simulated negative damping feedback loop (Reference 15) in addition to the original effective sinusoidal force simulation. A semiemperical wear calculation was developed (Reference 16) in which measured work rates from these tests are scaled to pertinent operating conditions using appropriate parameters from the thermal-hydraulic report and vibration analyses.

Thermal-hydraulic and tube vibration analyses follow the same general approach in the U-bend region as for the straight-leg region. Effects of the AVBs on flow distributions are obtained from qualified thermal-hydraulic models with explicit treatment of their size and location. There is no potential for flow peaking near the bend region of AVBs (involving small-radius tubes not supported by AVBs) in advanced design configurations as has been observed in some operating steam generators with conventional design and fabrication bases. (This is a consequence of explicit control of insertion depth during assembly.)

Basic vibration analyses employ the same qualified analytical models following a modal decomposition approach similar to that used for the straight-leg region. Threshold instability constants and scaling factors for random turbulence response are derived from results of testing summarized above. Analyses cover the range of possible fitup conditions determined by inspection of tube bundles during and after fabrication.

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SSAR Revision: NONE

Question 252.120

Section 1A of the SSAR indicates that the guidance in Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," will not be applied to ASME Code Class 2 and 3 components in the AP600. Provide technical justifications for not following this guidance or provide acceptable alternatives to this guidance. In addition, provide confirmation that preheat requirements in the ASME Section III Code will be satisfied. Revise Section 1A of the SSAR accordingly.

Response:

Westinghouse requires that all welding procedures for class 2 and 3 components be qualified in accordance with Section IX and the appropriate Subsection of Section III. This includes preheat and post weld heat treatment requirements. Where low alloy steels are used for Class 2 and 3 components, all ASME Boiler and Pressure Vessel Code requirements for welding of these materials are imposed. The use of low hydrogen welding materials is also required. These controls, have in the past, been adequate to assure the integrity and reliability of these components. These controls are also adequate for the AP600.

SSAR Section 1A under Reg. Guide 1.50 will be revised as follows:

SSAR Revision:

Reg. Guide 1.50, Rev. 0, 5/73 - Control of Preheat Temperature for Welding of Low-Alloy Steel

General	ASME Code, Sections III and IX	N/A	The guidelines of this regulatory guide are followed during the initial fabrication of low-alloy steel components of the AP600. This regulatory guide is considered as applicable to ASME Code , Section III, Class 1 components. The AP600 practice for Class 1 components is in agreement with the guidance of this regulatory guide except for Regulatory Positions C.1(b) and 2. For AP600 Class 2 and 3 components, the guidelines provided by this regulatory guide are not applied, however all requirements of the ASME Boiler and Pressure Vessel Code are imposed.

Question 252.122

Indicate conformance with Regulatory Guides 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," as related to Section 6.1.1 of the SRP. Provide technical justifications if the guidance contained in these documents will not be followed, or provide alternatives with bases to demonstrate the equivalency (Section 6.1.1).

Response:

The first part of the question deals with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." The combustible gas that will be involved with respect to the corrosion of coatings inside containment is hydrogen. The hydrogen control system is described in SSAR Subsection 6.2.4. It complies with Regulatory Guide 1.7 or, in some cases, utilizes more conservative assumptions consistent with advances in ALWR design, such as the source term. Additional conformance details are provided in SSAR Appendix 1A.

The second part of the question deals with Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." Regulatory Guide 1.54 imposes ANSI N101.4 for the quality assurance (QA) of coatings since ANSI N101.4 is used to implement 10 CFR Part 50, Appendix B, for coatings. The following paragraphs are taken from ANSI N101.4:

> 1.2.2 The standard, N101.4, specifically applies to protective coatings and their application to surfaces for nuclear facilities as listed in ANSI N101.2 and/or ANSI N5.9 and comprises requirements applicable to Class I and Class II service levels.
>
> 1.2.2.1 Class I service level applies to those systems and components of nuclear facilities which are essential (1) to prevent postulated accidents which could affect the public health and safety or (2) to mitigate the consequences of these accidents. The quality assurance for Class I service level shall conform to the requirements of this standard.
>
> 1.2.2.2 Class II service level applies to those systems and components of nuclear facilities which are essential to the attainment of the intended normal operating performance. The quality assurance and/or documentation for Class II service level is not mandatory and shall be used only to the extent required by the project specification.

SSAR Subsection 6.1.2.1.5 justifies why containment coatings in the AP600 are classified as non-safety-related. The designation as non-safety-related means the coatings are not Class I as described in ANSI N101.4. This classification of coatings as non-safety-related does not require implementation of a QA program for coatings per Regulatory Guide 1.54. Even though the coatings function is not safety related and does not require a QA program to be implemented, many measures are being taken so that the coatings are qualified for use and are properly applied and inspected. Furthermore, the performance of the coatings will be monitored and they will be maintained.

Subsection 6.1.2.1.5 of the SSAR thoroughly discusses this issue. The AP600 position on Regulatory Guide 1.54 is given on Page 1A-27 of the SSAR.

SSAR Revision: None

Question 252.125

Provide a corrosion allowance for the materials used in the engineered safety features of the AP600 for the projected 60-year plant design life along with the technical basis to support the allowance (Section 6.1.1).

Response:

The engineered safety features described in Section 6.0 are fabricated from materials best suited for that system's purpose. Several systems are air handling systems, which do not have corrosion concerns. Other systems are fabricated from stainless steel. The systems with corrosion allowances are listed below.

Containment: Corrosion potential of the containment is discussed in Questions 252.22 through 252.28.

Passive Containment Cooling System: The pipes and valves of this system are fabricated from stainless steel, which does not require corrosion allowances.

Containment Hydrogen Control System: This system does not have components that contain fluids.

Passive Core Cooling System: The pipes and valves of this system are fabricated from stainless steel, which does not require corrosion allowances. The N_2 supply line to the accumulator is carbon but does not require a corrosion allowance.

Main Control Room Emergency Habitability System: The valves and tanks that are part of this system contain air, which does not require corrosion allowances.

Fission Product Control: The retention of fission products is accomplished by the containment described above.

Containment Isolation System: Each system that penetrates the containment has valves and a section of pipe of Class B quality level. The penetrations are of different materials and contain different fluids or gases. The required corrosion allowances follow:

Systems penetrations with stainless steel components, corrosion allowances not required:
PSS, RNS, SFS, PXS, CVS, FHS, WLS, DWS
Systems penetrations containing air or nitrogen not requiring corrosion allowances:
CAS, VFS, VWS, CNS, PXS
SGS penetrations: Corrosion control of this system is discussed in Q252.139
CNS penetrations: Controlled water chemistry and low velocities in this system do not require corrosion allowances.
FPS penetrations: Wall thickness of this pipe is much greater than required for the pressures. This provides adequate corrosion allowance.

SSAR Revision: NONE

Question 252.127

Section 6.1.1.6 of the SSAR indicates that the AP600 design conforms to Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." However, Section 1A of the SSAR indicates that Westinghouse is proposing exceptions to Regulatory Guide 1.37. Provide information to clarify your intent. Revise Section 1A of the SSAR accordingly.

Response:

Subsection 6.1.1.6 will be revised. See also the response to Q252.73 for clarification of intent.

SSAR Subsection 6.1.1.6 will be revised as follows:

SSAR Revision:

~~Components and systems included in the engineered safety features are cleaned in conformance with Regulatory Guide 1.37.~~ See Subsection 1.9.1 for a discussion on the provisions of Regulatory Guide 1.37 for the cleaning of components and systems."

Question 252.132

Identify where Inconel 600 and 182 are applied in the ESFs. Operating experience indicates that these materials are susceptible to cracking. If these materials will be used, discuss any special measures to be taken to reduce the susceptibility to cracking and provide test data to demonstrate that the materials are not susceptible to cracking for the projected 60-year plant design life (Section 6.1.1).

Response:

There are no Inconel 600 nor 182 components in the ESFs.

SSAR Revision: NONE

Question 252.133

Recent fatigue test data indicate that the effects of the environment could significantly reduce the fatigue resistance of materials (see Q252.19). The specific concerns relate to the reactor water and temperature environment and its synergistic interactions with the strain rate. Describe the procedures that explicitly account for the effects of the environment in the fatigue analysis of materials used in the ESFs (Section 6.1.1).

Response:

The AP600 fatigue design for engineered safety features components is based on ASME Code, Section III, 1989 Edition and 1989 Addenda. The environmental effects described in the literature are being addressed as an industry issue. The basis of AP600 fatigue design will remain as described until a unified design approach representing the industry and regulatory positions is derived.

SSAR Revision: NONE

Question 252.142

Recent fatigue test data indicate that the effects of the environment could significantly reduce the fatigue resistance of materials (see Q252.19). The specific concerns relate to the reactor water and temperature environment and its synergistic interactions with the strain rate. Describe the procedures that explicitly account for the effects of the environment in the fatigue analysis of materials used in the steam and feedwater system (Section 10.3.6).

Response:

The AP600 steam and feedwater systems are not ASME Class 1. The ASME portions of these systems are designed to section NC or ND of the ASME Code, Section III, 1989 Edition and 1989 Addenda. The environmental effects described in the literature are being addressed as an industry issue. The basis of AP600 fatigue design will remain as described until a unified design approach representing the industry and regulatory positions is derived.

SSAR Revision: NONE

Question 281.2

Section 5.2.3.2.1 of the SSAR discusses the primary water chemistry for AP600. Is the RCS water chemistry consistent with the guidelines of EPRI Reports NP-6780-L and NP-7077, "PWR Primary Water Chemistry Guidelines: Revision 2," November 1990 that are identified in Chapter 1 of the ALWR Utility Requirements Document for passive plants, Volume III? Identify differences between the primary water chemistry of the AP600 and these guidelines, and provide justification for the deviations.

Response:

The AP600 primary water chemistry guidelines are consistent with the guidelines of the EPRI reports. There are no differences between the AP600 guidelines and those recommended in the EPRI reports. In addition, Table 5.2-2 includes more parameters (i.e., requirements) than are included in the EPRI reports.

SSAR Revision: NONE

Question 281.4

Discuss coatings in Section 6.1.1 of the SSAR in accordance with Acceptance Criterion II.B.4 of Section 6.1.1 of the SRP.

Response:

The coatings inside the containment are not safety-related as explained in the response to Q252.122. Even though they are not safety related, the coating materials will still be qualified, applied, and inspected in accordance with stringent specification requirements. SSAR Subsection 6.1.2.1.5 addresses steps that will be taken so that coatings perform as intended even though they are not safety-related.

SSAR Revision: NONE

Question 281.7

Discuss conformance protective coatings (organic materials) with the guidance in Regulatory Guide 1.54 and provide technical justifications for any deviations (Section 6.1.2).

Response:

See the response to Q252.122.

SSAR Revision: NONE

Question 281.8

Provide technical justifications for not using ANSI Standard N101.2 or propose an acceptable alternative (Section 6.1.2).

Response:

Coatings will be DBA tested in accordance with ASTM D 3911. That ASTM document was prepared as a replacement for the DBA testing requirements contained in ANSI Standard N101.2. ANSI Standard N101.2 has been withdrawn and ASTM standards were prepared to replace the ANSI standard. SSAR Subsection 6.1.2.1.5 addresses "testing and application" and references the applicable ASTM test standards.

SSAR Revision: NONE

Question 410.42

Two lines are routed from the IRWST to each of the PXS compartments. The six inch line is routed to PXS-A and the 10-inch line is routed to PXS-B. What is the purpose of these lines? Why are these lines sized differently? What is the effect if the PXS compartment overflows (Section 3.4.1)?

Response:

See the response for Q410.10. Questions 410.10 and 410.42 are identical.

SSAR Revision: NONE

Question 420.2

Section 7.2.2.1 of the SSAR states that the probabilistic risk assessment, in lieu of a failure modes and effects analysis, provides a quantification in terms of system unavailability for the failure of the protection systems. R.G. 1.70 "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants" states that the applicant should submit a failure mode and effects analysis (FMEA) for the protection systems and components. General principles of reliability analysis of nuclear power generating station protection systems (IEEE STD 352) emphasize that the FMEA is a qualitative analysis. The probabilistic risk assessment does not provide the same level of detailed information as provided by the FMEA. Provide the FMEA for the protection systems and components of the AP600 I&C systems.

Response:

In accordance with Regulatory Guide 1.70 and IEEE Standard 352, a FMEA for the AP600 protection system will be provided. Completion is planned for April 1993. The FMEA will be submitted as WCAP-13594, "Advanced Passive Plant Protection System FMEA."

SSAR Subsections 7.2.2.1 and 7.3.2.1 will be revised, and a reference to WCAP-13594 will be added as follows:

SSAR Revision:

(7.2.3 References)

1. WCAP-13594 (P), "Advanced Passive Plant Protection System FMEA."

7.2.2.1 Failure Modes and Effects Analysis (FMEA)

~~The probabilistic risk assessment (in lieu of a failure modes and effects analysis) provides a quantification in terms of system unavailability for the failure of the protection system to perform an automatic or manual reactor trip when required.~~

~~The protection system is designed to address credible common mode failures. Design attributes include the following:~~

- ~~System design for verification and validation~~
- ~~System verification and validation process~~
- ~~Functional diversity~~
- ~~A system functional tester~~
- ~~Fail-safe design principles~~
- ~~System qualification~~
- ~~System maintenance features.~~

~~These design features prevent credible common mode failures in the protection system. Therefore, protection system common mode failure is not considered a credible failure during design basis accidents. In addition, the probabilistic risk assessment uses component reliabilities that include both random and common mode failure probabilities. The assessment shows that the contribution of the protection system along with the nonsafety-related diverse actuation system (discussed in Subsection 7.7.1.11 and shown on Figure 7.2-1 Sheet 29) provides a reliability that supports the overall acceptable core melt frequency.~~

A failure modes and effects analysis is performed on the protection system. Through the process of examining all feasible failure modes, it is concluded that the AP600 protection system maintains all safety functions during single point failures. The failure modes and effects analysis is documented in Reference 1 of Subsection 7.2.3.

7.3.2.1 Failure Modes and Effects Analyses

The ~~probabilistic risk assessment (in lieu of a~~ failure modes and effects analysis~~)~~ (Reference 1 of Subsection 7.2.3) provides ~~a quantification of the probability~~ an analysis of failures of ~~automatic and manual safeguards system actuation signals together with the respective mechanical parts of each safety~~ the protection system. The protection system is designed to address common mode failures. ~~as discussed in Subsection 7.2.2.1.~~

Question 435.8

Section 8.3.1.1.1 states that the diesel generator is capable of being manually paralleled with the preferred power supply for periodic testing. Discuss the design provisions for the diesel generator control to handle a simultaneous loss of offsite power.

Response:

Generally, periodic diesel generator testing is performed, one diesel generator at a time, during normal plant operation. During this testing, the main generator is synchronized with the offsite power source and supplies the plant auxiliaries via the unit auxiliary transformers. Should offsite power be lost during this operation mode, the main turbine generator has the capability to withstand 100 percent load rejection and continues to supply the plant auxiliaries. If the onsite standby diesel generator periodic testing is in progress during an offsite power loss under the stated conditions, additional design controls are not required. The periodic testing may be terminated using the standard test procedure without causing any adverse effects on the onsite standby power source.

For a postulated loss of offsite power concurrent with the main turbine generator trip while a diesel generator is undergoing periodic testing, the following loading scenario will take place.

The system loads required for the normal plant operation that are fed from the permanent non-safety-related bus would immediately overload the associated diesel generator. This excessive loading condition would cause a drop in generator frequency. The following design features protect the diesel generator from adverse effects.

Should the diesel generator frequency drop below the set frequency value (the frequency setpoint determined on the basis of vendor-specific data for the selected diesel generator), a signal would be generated to trip the diesel generator breaker. The underfrequency tripping would take place only when the following additional permissives are satisfied simultaneously: a) Diesel generator is in a "TEST" mode, b) Diesel generator is operating and is connected to the associated permanent non-safety-related bus as signified by "diesel generator breaker closed position."

When the diesel generator breaker trips, the diesel generator is isolated from the bus load, but is still maintained in a "no load running" condition via the operation of the governor controls. With no power source at the permanent non-safety-related bus, all designated load breakers and the normal power source incoming breaker would be tripped via the bus undervoltage relay operation. The diesel generator test mode would be reset. Since the generator is still producing power at the rated voltage and frequency, the diesel generator breaker would be reconnected to the associated dead bus and, operation of the automatic load sequencer would be initiated to supply the defense-in-depth systems loads.

SSAR Subsection 8.3.1.1.1, eighth paragraph, will be revised as follows:

AP600

SSAR Revision:

The onsite standby power system powered by the two onsite standby diesel generators supplies power to selected loads in the event of a loss of normal, and preferred ac power supplies. Those loads that are priority loads for investment protection based on their specific functions (permanent non-safety loads), are assigned to annex building buses ES1 and ES2. These plant permanent non-safety loads are divided into two functionally redundant load groups (degree of redundancy for each load is described in the sections for the respective systems). Each load group is connected to a separate bus ES1 and ES2. Each bus is backed by a non-Class 1E onsite standby diesel generator. In the event of a loss of voltage on these buses due to a turbine generator trip concurrent with a loss of preferred power source, the diesel generators are automatically started and connected to the respective buses. The source incoming breakers on switchgear ES1 and ES2 are interlocked to prevent inadvertent connection of the onsite standby diesel generator and preferred/maintenance ac power sources to the 4.16 kV buses at the same time. The diesel generator, however, is capable of being manually paralleled with the preferred power supply for periodic testing. Design provisions protect the diesel generators from excessive loading beyond the design maximum rating, should the preferred power be lost during periodic testing. The control scheme, while protecting the diesel generators from excessive loading, does not compromise the onsite power supply capabilities to support the defense-in-depth loads. See Subsection 8.3.1.1.2 for starting and load sequencing of standby diesel generators.

Question 435.26

The specific requirements for monitoring the dc power systems are derived from the generic requirements in Section 7.4 of IEEE Standard 946-1985. In summary, these general requirements state that the dc system (batteries, distribution systems and chargers) shall be monitored to the extent that it can be shown to be ready to perform its intended function. Accordingly, the guidelines used in the staff's review of the dc power system designs are that the following indications and alarms of the Class 1E dc power system should be provided in the control room:

- battery current (ammeter-charge/discharge)
- battery charger output current (ammeter)
- dc bus voltage (voltmeter)
- battery charger output voltage (voltmeter)
- battery discharge
- dc bus undervoltage and overvoltage alarm
- dc bus ground alarm (for ungrounded systems)
- battery breaker(s) or fuse(s) open alarm
- battery charger trouble alarm (one alarm for a number of abnormal conditions which are usually indicated locally)
- amp-hour discharge meter for safety-related batteries that are required to provide dc power for 72 hours
- an alarm whenever the charger goes into a current limiting condition
- a temperature indicator to measure the battery room ambient temperature

Indicate the provisions for the monitoring of the Class 1E power systems. Justify any proposed deviations from the above list.

Response:

The AP600 Class 1E dc power system design conforms to the indication and alarm requirements recommended in Section 7.4 of IEEE Standard 946-1985 for the control room. Section 7.5 of the SSAR describes the AP600 design philosophy for providing the safety-related display information in the control room. The control room functional-based task analyses (see SSAR Subsection 18.8.2.1.2) will determine the means by which the information is presented to the operator.

SSAR Revision: NONE

Question 435.27

The staff recognizes that standard molded-case ac breakers can be used in dc circuits. However, the dc interrupting rating will generally be one-half to one-third of the ac value. Many manufacturers publish no dc application data at all for these breakers. Verify that the molded case breakers will have adequate dc interrupting rating.

Response:

The limitation of the availability of molded-case circuit breakers with a high dc interrupting rating is known in the industry. However, there are manufacturers who can supply molded case circuit breakers with UL-listed interrupting ratings for dc circuits. A description of the application of molded-case circuit breakers in the AP600 dc distribution system follows.

AP600 design generally utilizes fusible disconnect switches in the Class 1E dc system. If a molded-case circuit breaker is used in a particular circuit, it will be sized to meet the dc interrupting rating requirement. Proper documentation will be obtained to ensure that the molded-case breakers have adequate dc interrupting rating.

The non-Class 1E dc power system has molded-case circuit breakers. These breakers will have UL-listed current interrupting ratings for dc applications.

SSAR Revision: NONE

Question 435.28

State if the AP600 design conforms to the following regarding the adequacy of safety-related dc power systems:

a. The position of circuit breakers or fused disconnect switches associated with the battery charger, battery and dc bus supply should be monitored to conform to the recommendations of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

b. The voltage variation for an associated battery bus during any expected accident mode of operation should be within design specifications.

Response:

a. The AP600 design conforms to the recommendation of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." The position of the fused disconnect switches associated with the battery charger, battery, and dc bus supply will be monitored to provide information to the alarm system. See also Appendix 1A of the SSAR.

b. The operating range for the safety-related dc power system is 105 to 140 vdc, as stated in the last paragraph of SSAR Subsection 8.3.2.1. This voltage range envelopes the design basis accident conditions and the batteries have been sized to provide adequate voltage at the end of the battery duty cycle.

SSAR Revision: NONE

Question 435.29

State if the battery charger has sufficient capacity to operate all non-accident shutdown loads assuming the battery is not available.

Response:

The AP600 battery chargers have sufficient capacity to operate the non-accident shutdown loads without the associated battery.

SSAR Revision: NONE

Question 435.38

Section 8.3.2.1.1.1 of the SSAR states that the AP600 battery monitoring system detects battery open circuit conditions and monitors battery voltage. It does not mention other conditions, such as the open-circuited intercell connectors, high terminal resistance, sulfated plates, or any condition involving abnormally low (or high) terminal voltage. As a safeguard against these no-fault conditions, it may be preferable to use a battery condition monitor with these capabilities. Has Westinghouse considered using such a monitor? Discuss the use of such a monitor in the AP600 design.

Response:

The AP600 battery monitoring system detects the battery open-circuit conditions including the open-circuited intercell connectors. It also monitors the battery high and low terminal voltage. Continuous monitoring of the battery parameters such as voltage and current will provide the pre-programmed alarm for high terminal resistance. Checking sulfated plates or other anomalous conditions requires periodic inspections, which will be covered in the plant procedures to be prepared by the combined license applicant.

SSAR Revision: NONE

Question 435.51

Section 8.3.2.1.1.1 of the SSAR states that a single battery bank with a spare battery charger is provided for the Class 1E dc and the UPS systems. In the case of a failure or unavailability of the normal battery bank and the battery charger, permanently installed cable connections allow the spare to be connected to the affected bus by using a plug-in type disconnect. Explain how the AP600 design will maintain the electrical and physical separation between the redundant safety systems when a system is powered from the backup dc source.

Response:

Separate cables a e permane 'ly installed between each of the four divisions and the spare battery bank w.th twist-lock, plug-in connectors (male) at each end.

Each division and the spare battery include a fused transfer switch between the battery and the switchboard. These transfer switches include a twist-lock, plug-in connector (female).

To connect one division to the spare battery, the permanently installed cable for that division is plugged into its associated connector at one end. The other end is plugged into the connector for the spare battery.

Since the spare battery transfer switch has only one plug, it is possible to connect only one division's cable at a time. When the spare battery is connected to a given division, it becomes that division, and only the cable of that division is energized. The cables associated with the other divisions will remain disconnected at both ends. Thus, electrical and physical separation between the redundant safety systems will be maintained when a system is powered from the spare battery.

SSAR Revision: NONE

Question 435.61

Provide a description of the design of electrical penetration protection to address the following guidance (see RG 1.63):

a. The system shall, with precision and reliability, automatically disconnect power to the penetration conductors when currents through the conductors exceed the established protection limits.

b. All primary and backup breaker overload and short circuit protection systems shall be qualified, including seismically, for the service environment. However, the seismic qualification for non-Class 1E circuit breaker protection systems should, as a minimum, assure that the protection systems remain operable during an operating basis earthquake.

c. The circuit breaker protection system trip set points and breaker coordination between primary and backup protection shall have the capability for test and calibration. Provisions for test under simulated fault conditions should be provided. For designs where protection is provided by a combination of breaker and a fuse or two fuses in series, administrative control should be exercised to check the sizes and types of fuses and their proper application. Operating experience has shown that wrong sizes of fuses were used for the protection of the circuits in many plants.

d. No single failure shall cause excessive currents in the penetration conductors which will degrade the penetration seals.

e. Where external control power is needed for tripping breakers, the circuitry for tripping primary and backup breakers shall be independent, physically separated, and powered from separate sources.

Response:

The AP600 design will conform to the requirements in Regulatory Guide 1.63, which endorses IEEE standards 317-1983 and 741-1986 with the exception that the AP600 will use the latest (IEEE 741-1990) version of the industry standard.

The following is an item-by-item response:

a. Each penetration conductor will be protected with primary and backup protective devices. The protective devices shall be set to trip and disconnect the power source when current through the penetration conductor exceeds the established limit before causing any damage.

b. The primary and backup protective devices used to protect penetration assembly for Class 1E circuits will be qualified for the service environment.

For non-Class 1E circuits, IEEE 741 states "Designs which use dual protection for penetration circuits provide adequate assurance that the protection will function acceptably under all plant conditions. Since the plant design basis is that a seismic event will not cause a loss-of-coolant accident (LOCA) or high energy line break (HELB) inside containment, seismic qualification of protective devices for non-Class 1E circuits that penetrate the containment is not required."

Therefore, the protective devices for non-Class 1E circuits that penetrate the containment are not required to be seismically qualified.

c. The AP600 medium voltage and 480V load center breakers are draw-out type and can be racked out for necessary test. The 480V molded case circuit breakers can be bench tested for their proper application including simulated fault conditions. The administrative control are the responsibility of the Combined License applicant.

d. The AP600 penetration protection device, with primary and backup protective devices, will prevent degradation of the penetration seals due to excessive current in the penetration conductors caused by a single failure.

e. The AP600 design will provide independent and physically separate dc control power sources for tripping of primary and backup breakers, where required.

SSAR Revision: NONE

Question 435.70

Section 9.5.3.1.1 of the SSAR specifies the lighting systems to be composed of normal, emergency, and security lighting systems. The normal lighting system is designed to meet the visual requirements for all areas of the plant to permit personnel to perform all required tasks. The system is part of the plant permanent non-safety systems. The power supply is provided from non-Class 1E ac power distribution system at 480/277 V and 208/120 V levels. The normal lighting system is powered from diesel-backed buses and the lighting load is distributed between the two onsite standby diesel generator buses. The lighting circuits from the normal lighting system to the individual lighting fixtures are staggered as much as possible, with the staggered circuits fed from separate buses, so as to ensure that some lighting is retained in a room in the event of a bus or circuit failure. Describe whether this lighting (including lighting fixtures and light bulbs) has been analyzed to assure that its failure during an earthquake will not adversely impact the Class 1E functions.

Response:

The intent for the AP600 lighting system design for the main control room and remote shutdown areas is to have a manufacturer design and fabricate a total ceiling modular system that meets the seismic Category II requirements. The emergency lighting will be designed to meet the seismic Category I requirements. In other areas, the lighting system will have seismically qualified lighting fixtures and mountings as required to prevent an adverse impact on the adjacent Class 1E equipment due to an earthquake.

SSAR Revision: NONE

Question 435.71

The emergency lighting system is designed to provide a minimum illumination level in areas of the plant where emergency operations require reading and control functions. These areas include the main control room and remote shutdown panel and the emergency access/egress between vital and non-vital areas. Section 9.5.3.2.2 of the SSAR states that the emergency lighting for the main control room and remote shutdown area is supplied from the Class 1E dc and the UPS system. The control room emergency lighting is integrated with normal lighting. Emergency lighting in areas outside the main control room and remote shutdown area is accomplished by 8 hour, self-contained battery pack lighting units. In some existing nuclear power plants, the access and egress routes have been supplied with Class 1E dc system and backed by dc 8 hour, self-contained battery pack units. Earlier in the design, lighting for the access and egress routes were powered from the Class 1E 120 V uninterruptible power supplies (WCAP-13202). Justify the deviation. Also, describe whether the emergency lighting fixtures are staggered and fed from the separate divisions.

Response:

The AP600 emergency lighting design is based on the criteria established in Utility Requirements Document (URD), Volume III, Chapter 11, Section 8.5. The emergency lighting for areas outside the main control room and remote shutdown area is provided by 8-hour, self-contained battery pack lighting units. The normal supply to these units is provided from the diesel-backed MCCs. The rationale in the URD for not powering all emergency lights from Class 1E ac and dc power sources follows:

Although some plants have all emergency lights powered from emergency ac and dc power sources, those methods introduce complexity into those vital systems, increase their size, and inherently reduce their reliability.

The emergency lighting circuits are staggered as much as practical. The staggered circuits receive power from separate divisions.

SSAR Revision: NONE

Question 440.2

WCAP-13345, Rev. 2, "AP600 Core Makeup Tank Test Specification," is dated November 1991. Since that time, substantial changes have been made in both the test program and the design of the test article; these have been communicated to the staff during meeting presentations. A revised and updated copy of the test specification, with current information on test article design, instrumentation, test matrix, should be submitted for staff review.

Response:

WCAP-13345, "AP600 Core Makeup Tank Test Specification," is currently being updated and will be forwarded to the NRC in February 1993. (See responses to Questions 440.4 and 440.5.)

SSAR Revision: NONE

Question 440.16

The test descriptions in Sections 7.A and 7.B, and in Tables 7.1 and 7.2 of WCAP-13342, Rev. 0 do not include detailed test procedures. The updated test specification requested in Q440.11 should include sufficient detail on test methods, facility conditions, and data acquisition, including step-by-step procedures, for the staff to determine if an adequate range of data on component performance will be provided.

Response:

When Westinghouse prepares a test specification, the detailed test procedures are not included since they are the responsibility of the testing organization. The testing organization develops the test procedures, which are then reviewed and approved by Westinghouse. The procedures are operator instructions on how to operate the facility to obtain the test conditions specified in the test specification. The test specification (WCAP-13342) does provide adequate information for the staff to determine the adequacy of the tests including information on the facility initial conditions, data acquisition, range of conditions, expected data, instrumentation and method of testing.

SSAR Revision: NONE

Question 440.28

Technical Specification 3.9 in Chapter 16 of the SSAR sets forth limiting conditions for operation and Surveillance Requirements for refueling operations regarding boron concentration, unborated water isolation valves, nuclear instrumentation, and refueling cavity water level. These LCOs do not address the proposed LCOs for TS improvements discussed in Chapter 7 of NUREG-1449. Address the following concerns:

a. What are the LCOs or the requirements on the configurations of the CVCS, NRHRS and Spent Fuel Pit Cooling System, offsite and onsite ac power, and containment integrity for shutdown and low power operation conditions.

b. What consideration has been made regarding automatic requirements to go to Mode 5, i.e., cold shutdown, to ensure optimal RHR capability?

c. While LCO 3.9.2 specifies that each valve used to isolate unborated water sources shall be secured in the closed position, it also notes that valves may be opened during planned boron dilution or makeup activities. Provide examples as to when a planned boron dilution or makeup activities will be needed during refueling operation.

Response:

a. Technical specifications identify the required operability of safety-related equipment during all plant conditions, including shutdown and refueling. The technical specifications incorporate the proposed improvements discussed in Chapter 7 of NUREG-1449, based on requirements for the AP600 passive safety-related systems.

   The passive residual heat removal heat exchangers are required to be operable until RCS draining is initiated. Once the RCS is depressurized, the ADS valves connected to the pressurizer are opened and the core makeup tanks are isolated. The IRWST is required to be operable until flood-up of the refueling cavity is initiated. The IRWST, in conjunction with the open ADS valves, protects the plant during mid-loop operations. The normal decay heat removal system is lost in this condition. The IRWST is automatically actuated if a sustained loss of shutdown decay heat removal occurs during mid-loop operations.

   Because of the design of the passive safety systems, there are no requirements for operability of on-site and off-site ac power, with the exception of 120-vac instrument power provided by the Class 1E dc and uninterruptable power supply system.

Containment integrity is required during Modes 1 through 4 and during mid-loop operations. Technical Specifications 3.6.1 and 3.6.3 address containment integrity. They are currently incomplete regarding applicability during mid-loop operations. They will be revised to reflect the requirements for containment integrity during mid-loop operations for Mode 5 and Mode 6.

Technical Specification 3.7.5 addresses the requirements to provide spent fuel pit cooling capabilities whenever spent fuel is being stored.

There are no technical specifications for the operability of the chemical and volume control system or the normal residual heat removal system since these systems do not provide safety-related injection or cooling functions. However, Technical Specification 3.1.10 contains requirements for the unborated water isolation valves in the chemical and volume control system and for containment isolation valves in both systems.

b. The AP600 Technical Specifications consider the appropriate shutdown equipment capabilities and requirements prior to requiring a change to Mode 5. This prevents automatically forcing the operator to transition to Mode 5, when remaining in a different higher mode, such as hot standby, may provide greater plant safety.

c. While the plant is in Mode 6 conditions with the reactor vessel head removed, injection flow tests of both the accumulators and the core makeup tanks are performed. Following the flow tests, the tanks are refilled by the CVS makeup pumps, using a blended flow from the demineralized water storage tank and the boric acid tank. This makeup operation requires opening certain administratively locked-closed valves.

The SSAR will be revised as follows:

SSAR Revision:

1. Technical Specification:	3.6.1 - Containment
Applicability:	Modes 1, 2, 3, 4, and 5 Modes 5 and 6, loops not full
2. Technical Specification:	3.6.2 - Containment Air Locks
Applicability:	Modes 1, 2, 3, 4, and 5 Modes 5 and 6, loops not full
3. Technical Specification:	3.6.3 - Containment Isolation Valves
Applicability:	Modes 1, 2, 3, 4, and 5 Modes 5 and 6, loops not full

Question 460.5

Provide the following information regarding solid radwastes (Section 11.4):

a. Estimates of solid waste volumes expected to be shipped annually for wet solid wastes and dry solid wastes separately.

b. A discussion of compliance with Position III.1 of BT? ETSB 11-3 regarding the storage capacity for accumulated filter sludges.

c. A discussion of compliance with Position III.2 of BTP ETSB 11-3 regarding storage volume for solidified wastes (both wet and dry solid wastes) available in the plant.

Response:

a. Estimates of solid waste volumes expected to be shipped annually are provided below for wet solid wastes and dry solid wastes. These volumes are consistent with SSAR Table 11.4-4.

Wet Solid Waste	Shipping or Disposal Volume, ft^3/yr
Spent Charcoal and Ion Exchange Resins	314.8
Mixed Liquid Wastes	17.0
Chemical Wastes	19.8
Total Wet Solid Wastes	351.6
Dry Solid Wastes	
Spent Filter Cartridges	4.7
WGS Charcoal	9.8
Compactible DAW	991.3
Non-Compactible DAW	363.4
Mixed Solid Wastes	7.5
Total Dry Solid Wastes	1376.7

Under normal conditions, there are no wastes generated from the secondary-plant cycle. If radioactivity is detected in the steam generator blowdown and reaches a predetermined level (due to primary and secondary leakage), the blowdown is diverted from the condensate polishers to the liquid radwaste system for processing as described in SSAR Subsections 10.4.8 and 11.2.2.1.5. Thus, condensate polishing resins are not a normal source of radwaste requiring shipping and disposal. Should plant operation continue with leakage from the primary to the secondary side utilizing the condensate polishers, a shipping (disposal) volume of up to 300 ft^3/month could be produced by the condensate polishing system, as indicated in SSAR Subsection 11.4.2.1.

b. The AP600 plant does not include filters that generate sludge-type wastes. Also, tanks and sumps are designed to minimize the formation of sludge deposits, and the particulate matter that can cause sludge deposits is transported to and removed by the cartridge filters in the liquid radwaste system. Therefore, there are no storage provisions for accumulating sludges.

c. Branch Technical Position ETSB 11-3 specifies in Position III.2. that storage areas for solidified wastes should be capable of accommodating at least 30 days of waste generation at normal generation rates and that these storage areas should be indoors. The storage durations of the storage areas in the Radwaste Building are evaluated for four general types of waste and container categories discussed below.

Spent Ion Exchange Resins and Filter Charcoal in High-Integrity Containers (HICs)

Although the shipping or disposal volume is nearly independent of container size (based on equal filling efficiency, e.g., 90%), the storage duration for the filled HICs is dependent on the number of containers which is indirectly proportional to container size. To be conservative, it is therefore assumed that the spent resins and filter bed charcoal will be dewatered in HICs that will fit into a Type B shipping cask (i.e., the SEG 3-82B, formally HN-200). Normally, 250 ft^3/yr of spent resin and charcoal is expected to be generated (SSAR Table 11.4-4), with an activity of 950 curies (SSAR Table 11.4-6). This resin can be mixed to produce a uniform specific activity of 3.8 Ci/ft3. A 70-ft^3 HIC filled to 90% would contain about 240 Curies, well within the cask's capability. About four of these HICs are required per year. The three onsite storage casks can each hold one of these HICs, and the resulting storage duration is about 9 months. The two spent resin container fill stations may also be used for storage until it is necessary to begin filling another HIC. The two fill stations and two of the onsite storage casks (reserving one onsite storage cask for high-activity filter drums) provide about 12 months of storage. These storage times are in addition to the pre-packaging storage times provided by the spent resin tanks as described in SSAR Subsection 11.4.2.2.1.

High Activity Filter Cartridges in Drums

As indicated in SSAR Table 11.4-4, packaging of the CVS reactor coolant filter cartridges is expected to normally generate 0.35 drums of waste per year. This is based on a generation rate of 2 ft^3 of filter cartridges every 18 months. Based on a drum volumetric loading of 50 percent, about one-half of a drum is filled every 18 months (2 ft^3/7.55 ft^3 x 0.5), and about one drum is produced every 36 months. The high-activity filter storage tube module may be used to store all filter cartridges normally generated every 18 months. Thus, after a drum is filled with high-activity filters, encapsulated, and sealed, it may remain in the processing cask for about 17 months before it is necessary to begin to package the filter cartridges stored in the storage tube module to clear space for the next batch of spent filters. Therefore, a storage duration of about 17 months is available for high-activity filters using only the processing cask. One of the onsite storage casks could also be used for high activity filter drum storage if necessary.

Other Wastes in Drums

Based on SSAR Table 11.4-4, about 11 drums are produced each year containing wastes other than high activity filter and mixed wastes. The Radwaste Building (Proprietary Figure 1.2-29) has a packaged waste storage room that may be used to store both drums and boxes. Using two storage locations for palletized drums stacked three high, 24 drums can be stored. This provides about 28 months of storage for the normal expected generation rate. Stacking only two pallets high provides about 18 months of storage. Without stacking about 9 months of storage is available for the normal generation rate.

Mixed wastes are accumulated in drums and are sent to a special prefabricated storage building at an expected rate of about three drums per year.

Wastes in Boxes

Based on SSAR Table 11.4-4, about 12.7 boxes are generated per year. Fourteen box storage locations are available in the packaged waste storage room. Without stacking and with stacking two and three high, about 1, 2, and 3 years of storage are provided, respectively.

Maximum truck loading is expected to be 28 boxes. This is equivalent to the 14 storage locations stacked two high. At the normally expected generation rate, it takes 26 months to produce a truck load.

In summary, indoor storage is provided for all categories of packaged wastes well in excess of 30 days, based on normally expected waste generation rates.

SSAR Revision: NONE

Question 460.6

The staff concludes that the proposed seismic design of the structures that house the liquid, gaseous and solid radwaste management systems as well as the proposed design of the applicable components of the gaseous waste management systems do not meet RG 1.143 seismic design guidelines in the sense that the operating basis earthquake (OBE) has been eliminated in the AP600 design (see Appendix 1A of the SSAR). Provide justification for the deviations or revise the design criteria for the above to meet the applicable guidelines of RG 1.143. Also, clarify how the AP600 design meets Position C.1.1.3 for liquid radwaste management system (Sections 11.2, 11.3, and 11.4).

Response:

Conformance with Regulatory Guide 1.143 is discussed in SSAR Appendix 1A. Regulatory Guide 1.143, Position C.1.1.3, specifies that "foundations and walls of structures that house the liquid radwaste system should be designed to the seismic criteria described in Regulatory Position 5 of this guide to a height sufficient to contain the maximum liquid inventory expected to be in the building." Equipment for the liquid radwaste sytsem is located in the Auxiliary Building and in the high bay area of the Radwaste Building. The Auxiliary Building is seismic Category I. The high bay area of the Radwaste Building is seismic Category II (see SSAR Table 3.2-2 and Subsection 3.7.2.8) and is designed and analyzed to prevent failure under the safe shutdown earthquake. Analysis methods and design criteria for seismic Category II structures are given in the response to Q230.11. The seismic Category II design results in greater seismic capability than required by Regulatory Guide 1.143 for structures housing radwaste systems.

The Radwaste Building is designed to retain the maximum contents of the detergent waste and monitor tanks, the chemical waste tank, and the spent resin tanks. Based on SSAR Tables 11.2-2 and 11.4-12, the maximum expected contents of these five tanks is 21,500 gallons. The immediate retention area is the high bay area of the Radwaste Building floor area bounded by column lines GR to HR and the north and south walls of the tank rooms. This area is 745 ft^2. The maximum flood height is about 4 feet. The threshold of the door to the tank rooms is 5'6" above the floor, and the pipeway to the Auxiliary Building is sealed by a seismically designed bulkhead. Water would also flow through floor and module drain lines and through penetrations to the module area between column line 4R and the tank rooms. The floor area increases to 1050 ft^2, and the maximum flood level is about 2'9". The curb wall east of the spent resin container fill stations along column line 4R is at least 3'0" high and any penetrations are at least 3'0" above the floor or are sealed. Water will continue to flow through floor drains and any low penetrations to the spent resin container fill stations. The flood area expands to 1250 ft^2, and the final maximum flood level is about 2'4". This evaluation does not take any credit for any initially empty sump volume or any sump discharge to the liquid radwaste system in the Auxiliary Building. Penetrations to areas adjacent to these retention areas are sealed by seismically designed bulkheads or penetrations if below the maximum flood levels.

Thresholds of all Radwaste Building external openings are raised at least 2 inches above the general floor area. This is adequate to retain at least 30,000 gallons in the event of fire water system activation. This is equivalent to about 30 minutes of operation at 1000 gpm. This provision also allows for retention of the small volumes of decontamination and detergent waste in the Radwaste Building.

AP600

The guard bed and the delay beds in the gaseous radwaste system are designed for seismic loads following the intent of Regulatory Guide 1.143. These are the only AP600 components used to store or delay the release of gaseous radioactive waste. The beds are located in the seismic Category I Auxiliary Building at elevation 66'6". Seismic loads for this equipment will be established using one-half of the safe shutdown earthquake floor response spectra. The equipment and supports will be designed in accordance with commercial codes (ANSI B31.1 and AISC).

SSAR Tables 3.2-2 and 3.2-3 and the last paragraph of Subsection 3.7.2.8 will be revised as follows:

SSAR Revision:

(Table 3.2-2)

~~Solid Radwaste Building~~	~~C-II~~
Radwaste Building	
- Columns AR - GR	NS
- Columns GR - IR	C-II

(Table 3.2-3 [Sheet 85 of 107]):

WGS MV 01 GUARD BED	12	D	II(a)	ANSI B31.1
WGS MV 02A DELAY BED A	12	D	II(a)	ANSI B321.1
WGS MV 02B DELAY BED B	12	D	II(a)	ANSI B321.1

(a) Guard and delay beds are designed using one-half of the safe shutdown earthquake floor response spectra.

(Subsection 3.7.2.8)

The non-Category I structures that are close enough to the nuclear island that their collapse could affect safety functions are the annex I and II buildings, the high bay area of the radwaste building (column lines GR to IR), and the turbine building. These structures are analyzed and designed to prevent their failure under the safe shutdown earthquake.

Westinghouse

Question 620.2

There are no showers in the ladies' locker area and no access to the men's showers. How does the design address this need (Figure 1.2 24, p. 1.2-46)?

Response:

On Figure 1.2-24, p. 1.2-46, there is a curbed opening between rooms 40319 and 40320. The women's showers are in room 40324. The layout of these rooms is shown in more detail in Proprietary Volume 1, p. 1.2-21, Figure 1.2-24.

SSAR Revision: NONE

Question 720.1

Provide the following referenced documents:

a. "EPRI MAAP 4.0 Users Manual for Probabilistic Risk Assessment," WCAP-13408, June 1992.

b. "Letter Report to EPRI on the AP600 Refueling Outage Plan," NPDP-FS-0067, June 1988.

c. Westinghouse Scientific Paper 71-1E7-MSLRF-P1 (Section 10.2.6).

Response:

a. "EPRI MAAP 4.0 User's Manual for Probabilistic Risk Assessment," WCAP-13408, June 1992
The EPRI MAAP 4.0 Users Manual is being prepared as an EPRI document. As such, the PRA citation for the Users Manual is incorrect and will be revised to reflect the correct EPRI report number upon completion. The MAAP 4.0 Code is undergoing the final phase of the independent design review process which will be completed in February 1993. At that point EPRI plans to finalize the draft users manual such that it should be available by early summer 1993.

b. NPDP-FS-0067, "Letter Report to EPRI on the AP600 Refueling Outage Plan," June 1988
This Westinghouse internal memorandum contains the "AP600 Preliminary Refueling Outage Plan Overview and Summary," dated May 25, 1988. The AP600 refueling plan is currently being revised. Upon completion the PRA citations to the refueling outage plan will be updated, and a copy of the revised refueling plan will be submitted to the NRC. In the interim, to facilitate NRC staff review, copies of the preliminary refueling outage plan were provided to the NRC on December 15, 1992 as an attachment to Westinghouse letter ET-NRC-92-3779.

c. Westinghouse Scientific Paper 71-1E7-MSLRF-P1, "Correlation of Fracture Toughness and Charpy Properties for Rotor Steels," July 1971
This Westinghouse scientific paper is referenced in SSAR Subsection 10.2.3.2. A review of the SSAR citation has determined that the referenced document is not utilized as a supporting technical document. The citation to this document can be removed with no technical impact on the SSAR content. Therefore, the AP600 SSAR will be revised by removing the citation to this document.

The SSAR and PRA will be revised as follows:

SSAR Revision:

(last sentence of the first paragraph of SSAR Subsection 10.2.3.2)

Material fracture toughness needed to maintain this ratio is verified by mechanical property tests on material taken from the rotor~~, using correlation methods which are more conservative than that presented in Reference 1~~.

(Subsection 10.2.6)

~~10.2.6 References~~

~~1. Begley, J. A.,M and Logsdon, W. A., Westinghouse Scientific Paper 71-1E7-MSLRF-P1.~~

PRA Revisions:

1. The AP600 PRA reference citations to the MAAP 4.0 User's Manual will be revised as follows. Following publication by EPRI, the citations will be updated to reflect the correct report number and publication date.

   (Reference 1.7-2, Page 1-5)

   2. "EPRI MAAP 4.0 User's Manual~~ for Probabilistic Risk Assessment~~," ~~WCAP-13408 (Proprietary) and WCAP-13409 (Nonproprietary), June 1992~~ EPRI NP-xxxx, [to be published].

   (Reference 7.7-2, Page 7-12)

   2. "EPRI MAAP 4.0 User's Manual~~ for AP600 PRA~~," ~~WCAP-13408 (Proprietary) and WCAP-13409 (Nonproprietary), June 1992~~ EPRI NP-xxxx, [to be published].

   (Reference 10.4-1, Page 10-7)

   1. "EPRI MAAP 4.0 User's Manual~~ for AP600 PRA~~," ~~WCAP-13408 (Proprietary) and WCAP-13409 (Nonproprietary), June 1992~~ EPRI NP-xxxx, [to be published].

   (Reference 11.5-1, Page 11-5)

   1. "EPRI MAAP 4.0 User's Manual~~ for the AP600 PRA~~," ~~WCAP-13408 (Proprietary) and WCAP-13409 (Nonproprietary), June 1992~~ EPRI NP-xxxx, [to be published].

(Reference 12.7-1, Page 12-3)

1. "EPRI MAAP 4.0 User's Manual ~~for AP600 PRA,~~" ~~WCAP-13408 (Proprietary) and WCAP-13409 (Nonproprietary), June 1992~~ EPRI NP-xxxx, [to be published].

(Reference 13.4-5, Page 13-3)

5. "EPRI MAAP 4.0 User's Manual ~~for AP600 Probabilistic Risk Assessment,~~" ~~WCAP-13408 (Proprietary) and WCAP-14409 (Nonproprietary), June 1992~~ EPRI NP-xxxx, [to be published].

(Reference 14.5-1, Page 14-6)

1. "EPRI MAAP 4.0 User's Manual ~~for Probabilistic Risk Assessment,~~" ~~WCAP-13408 (Proprietary) and WCAP-13409 (Nonproprietary), June 1992~~ EPRI NP-xxxx, [to be published].

(Reference 15.5-2, Page 15-3)

2. "EPRI MAAP 4.0 User's Manual ~~for Probabilistic Risk Assessment,~~" ~~WCAP-13408 (Proprietary) and WCAP-13409 (Nonproprietary), June 1992~~ EPRI NP-xxxx, [to be published].

2. The following AP600 PRA reference citations will be revised to reflect the revision to the AP600 refueling plan. Following publication, the citations will be updated to reflect the correct report number and publication date.

(Reference B.8-3, Page B-7)

3. ~~Letter Report to EPRI on AP600 Refueling Outage Plan, NPDP-FS-0067, June 1988~~ "AP600 Typical Refueling and Maintenance Outage Schedule," WCAP-xxxx (proprietary), WCAP-xxxx (nonproprietary) [to be published].

(Reference F.6-3, Page F-57)

3. ~~Letter Report to EPRI on AP600 Refueling Outage Plan, NPDP-FS-0067, June 1988~~ "AP600 Typical Refueling and Maintenance Outage Schedule," WCAP-xxxx (proprietary), WCAP-xxxx (nonproprietary) [to be published].