Question 231.1

The fuel performance code (PAD 3.3) used for the South Texas safety analyses has recently been approved by us with some restrictions which are identified in a letter dated February 9, 1979, from J. Stolz, NRC to T. Anderson, Westinghouse. Provide an assessment as to whether those restrictions effect the results of the safety analyses presented in the FSAR. If so, revise the analyses accordingly and provide the results.

Response

The effects of the restrictions on the PAD Code given in the letter dated February 9, 1979, from J. Stolz, NRC to T. Anderson, Westinghouse was assessed. It was determined that the restrictions on the use of the code during normal operation had no effect on the safety analyses presented in the UFSAR. There was also a restriction on the use of the code to analyze transients. An analysis was conducted which conservatively bounded possible transient phenomena. Based on this analysis, it was also concluded that there is no effect on the results of the safety analyses discussed in Chapter 4.2.1.3 of the UFSAR.

Question 231.2

Predicted cladding collapse times for South Texas Units 1 and 2 have been calculated with the model as given in WCAP-8377. We have approved the use of this model, by letter to Westinghouse dated February 14, 1975, subject to provisions that no alternations were made to the specified curves used as input to the model. Provide assurance that these provisions have been satisfied.

Response

No alternations have been made to the specified curves used as input to the cladding collapse model described in WCAP-8377.

Question 231.3

Provide the basis for the internal fuel rod gas pressure criteria presented in the FSAR. We note that these criteria are the same as those approved in our review of WCAP-8963 (see letter from J. Stolz, NRC to T. Anderson, Westinghouse) in which an acceptable basis is provided. Therefore, a reference to WCAP-8963 will provide an acceptable response to this request. Due to the restrictions to the fuel performance code (PAD 3.3) imposed at high burnups, as discussed in Request No. 231.1, above, determine the effects of these restrictions on satisfying the rod pressure criteria.

Response

The basis for the internal fuel rod gas pressure criteria presented in the UFSAR is described in WCAP-8963. The restrictions on the fuel performance code (PAD 3.3) as discussed in Request No. 231.1 were reviewed. It was determined that these restrictions had no effects on satisfying the rod pressure criteria.

Question 231.6 Deleted

CN-2839

Question 231.7

Recent PWR experience has shown that fretting wear has occurred between control rods and thimble tubes at a location associated with the fully withdrawn "parked" rod position. Provide assurance, either through fuel assembly inspection results (see item 231.6 above) or prototypical hydraulic flow tests (see item 231.8 below), that fretting wear is not a concern in South Texas.

Response

Westinghouse has provided to the NRC the results of fuel assembly guide thimble tube examinations. This information was made generally available to the NRC in letter NS-TMA-2238, T.M. Anderson to H.R. Denton, dated April 29, 1980. These results show that the design of Westinghouse Reactor Internals, fuel assemblies, and control rod assemblies is such that fretting wear in thimble tubes is not a safety concern in Westinghouse facilities. A thimble wear post irradiation examination was conducted by Westinghouse at Salem Unit l, and there was no evidence of through-wear holes or other indications of excessive wear on the guide thimble tubes.

Question 231.08

WCAP-8278 is used to demonstrate that the design methods for predicting vibration amplitudes and fuel rod fretting wear are conservative, based upon a l000-hour flow loop test of a l2-foot l7xl7 fuel assembly. Similar verification tests for the l4-foot l7xl7 fuel have been previously discussed by Westinghouse and were to be completed in l978. Provide assurance that fuel rod fretting for the l4-foot fuel assembly is not a concern in South Texas.

Response

The hydraulic flow testing of the 17 x 17, 14-ft fuel assembly was completed with no anomalies being observed. The results have been summarized and were transmitted to the NRC in letter ST-HL-AE-1143 dated October 18, 1984.

Question 232.3

Comment on the division of the MTC into density and temperature effects as a function of core lifetime. Recent calculations (to be published) by BNL suggest that the spectral (temperature) component is positive and is a significant portion of the total MTC for cores with large (~10 GWd/t) burnups. Comment on the effect of the use of density only moderator coefficients in the affected transients in Chapter 15, particularly for reload cycles (Ref. letter, Eicheldinger to Ross, dated February 28, 1978, NS-CE-1706).

Response

The referenced letter to the NRC (D. F. Ross) still applies. Transient calculations have been made which include a correction for the deviation between actual moderator temperature to that which is inferred from the transient density at the referenced pressure at which the density coefficients were calculated [Ref. letter, C. E. Eicheldinger (Westinghouse) to D. F. Ross (NRC), dated February 28, 1978, NS-CE-1706].

Question 232.4

State whether the coolant temperature control action is passive (in that the coolant temperature automatically responds to power changes) or whether a deliberate action is undertaken. For example, in a rapid power rise performed as part of a load following procedure, state whether there is an anticipatory increase in moderator temperature in preparation for the power rise.

Response

All changes in reactor temperature are initiated by and follow only changes in turbine loading. No manual anticipatory changes are made in reactor temperature for any power transients.

Question 232.5

Discuss whether the burnable poison rod pattern shown in Figure 4.3-5 of the FSAR is consistent with the power distributions and reactivity coefficients given for the South Texas core.

Response

The burnable poison rod pattern shown in Figure 4.3-5 of the UFSAR is consistent with the power distributions and reactivity coefficients given for the STPEGS reference first core.

Question 232.6

Provide an estimate of the uncertainty in the calculation of the flux at the inner boundary of the pressure vessel. Discuss how the azimuthal peaking factor is obtained. State whether comparisons have been made between calculations and measurements for sample locations. If so, provide this information.

Response

Radial, axial, and azimuthal variations of fast neutron flux within the reactor vessel geometry are generated by means of the two-dimensional discrete ordinates method. Since the azimuthal flux distribution is obtained directly from an R, è computation, no explicit azimuthal peaking factor is applied.

The attached table presents a comparison of measured and calculated neutron flux monitor saturated activities for reactor vessel surveillance capsules removed from two-loop Westinghouse PWRs. Since the monitors listed respond to different portions of the neutron energy spectrum and, further, since the saturated activity is proportional to the neutron flux magnitude, agreement between calculation and measurement provides an indication of the analytical capability to predict both flux level and energy spectrum at the measurement location.

Based on the data included in the attached table, uncertainties in the neutron flux calculation method are estimated to be ±10 percent on a generic basis and ±20 percent on a single measurement, specific plant basis.

TABLE Q232.6-1

NEUTRON FLUX MONITOR SATURATED ACTIVITY AT THE DOSIMETER BLOCK LOCATION FOR SEVEN 2-LOOP PWR POWER PLANTS (NORMALIZED TO 1650 MWT)

SATURATED ACTIVITY (dps/gm)

Capsule	$Fe^{54}(n,p)Mn^{54}$	$Ni^{58}(n,p)Co^{58}$	$Np^{237}(n,f)Cs^{137}$	$U^{238}(n,f)Cs^{137}$
Plant 1	4.92×10^6	5.78×10^7	7.06×10^7	6.71×10^6
Plant 2	6.00×10^6	8.97×10^7	6.96×10^7	9.24×10^6
Plant 3	5.07×10^6	7.42×10^7	6.11×10^7	7.93×10^6
Plant 4	5.09×10^6	7.62×10^7	7.04×10^7	8.19×10^6
Plant 5	5.14×10^6	6.93×10^7	7.37×10^7	8.36×10^6
Plant 6	5.73×10^6	9.41×10^7		6.65×10^6
Plant 7	5.26×10^6	6.80×10^7	7.85×10^7	7.98×10^6
13° Avg	5.32×10^6	7.56×10^7	7.07×10^7	7.89×10^6
13° Calc	5.61×10^6	8.28×10^7	7.11×10^7	7.35×10^6

HISTORICAL INFORMATION

Question 232.9

Discuss the difference in operating strategy, cases calculated, core design, etc., which lead to a peaking factor of 2.50, as compared to the usual value of 2.32 obtained from the constant axial offset control strategy.

Response

STPEGS Units are designed for a peaking factor, F_Q, as presented in the Core Operating Limits Report (COLR). Constant axial offset control strategy with +3, -12 percent axial flux difference limits insure that the $F_Q(Z)$ upper bound times the normalized axial factor, K(Z), is not exceeded during normal operation. The usual value of 2.32 is associated with $\pm$5 percent axial flux difference limits.

The peaking factor, F_Q, and constant axial offset control strategy axial flux difference limits are presented in the COLR.

Question 491.1N

The configuration of control bank D and the discussion in Section 4.3.2.4.13 indicate that you intend to use an "improved load follow" system. For Westinghouse reactors with twelve foot cores this has meant a changed (increased) CAOC offset band and a changed (expanded) set of calculational cases to define the offset limits and F_Q. Furthermore, it has been found necessary to restrict the wider offset band width during the first part of the first cycle. This type of information has not been supplied for the fourteen foot core. If you are using an improved load follow system, provide a discussion of the expanded calculations, the offset band to be used and any necessary restrictions required in its use.

Response

HISTORICAL INFORMATION

The improved load follow Final Acceptance Criteria analysis with the increased Constant Axial Offset Control (CAOC) offset band and expanded set of calculational cases was made to define the offset limits and F_Q for the South Texas (17 x 17 standard), 14-ft core with the light D Bank design. A reduced CAOC offset band (±5%) was used up to 3000 mwd/mtu core average burnup in both units. Beyond 3000 mwd/mtu core average burnup, a +3, -12% band is used. The ±5% axial offset band was removed by Technical Specification Amendment 9 (Unit 1) and 1 (Unit 2).

The CAOC offset band is presented in the Core Operating Limits Repent (COLR).

HISTORICAL INFORMATION

Question 221.6

HISTORICAL INFORMATION

Figure 4.4-11 presents a comparison between a least squares fit through the" THINC-IV calculation and measured values of core ΔT under natural circulation" condition. Provide the calculated ΔT vs. assembly power for each assembly."

Response [HISTORICAL INFORMATION]

The THINC-IV model has been approved as a design tool in an NRC SER, dated April 19, 1978. The values used in Figure 4.4-11 were from a natural circulation THINC-IV generic verification test. This information is included in the STPEGS UFSAR only for background information on the types of testing used. No additional specific analysis was done with this experimental information for the STPEGS UFSAR.

HISTORICAL INFORMATION

HISTORICAL INFORMATION

HISTORICAL INFORMATION

Question 221.7

Equation 4.4-19 (Section 4.4.4.3) presents the design value of enthalpy rise factor:

$$F\frac{N}{\Delta H} = 1.52\left[1 + 0.3\,(1 - P)\right]$$

Both the full power value (1.52) and the assumed power dependence are different from previously approved Westinghouse designs. Provide a plot comparing the design enthalpy rise factor (Equation 4.4-19) and the maximum calculated values of the operating enthalpy rise factor as a function of power level. The calculated enthalpy factors should be based on the proposed rod insertion limits and the constant axial offset control strategy.

Response [HISTORICAL INFORMATION]

Increasing allowable $F^{N}_{\Delta H}$ with decreasing power is permitted by all previously approved Westinghouse designs. This increase is permitted by the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limit as described in Section 4.4.4.3. Equation 4.4-19 (Section 4.4.4.3) presents the design limit of the nuclear enthalpy rise factor for the STPEGS plants:

$$\text{limit } F^{N}_{\Delta H} = 1.52\left[1 + 0.3\left(1 - P\right)\right]$$

where P is the fraction of full power.

The maximum calculated value of the operating nuclear enthalpy rise factor as a function of power level, including uncertainty, does not exceed the design limit at any power level for the STPEGS reference design:

$$\text{operating } F^{N}_{\Delta H}\left(\text{including uncertainty}\right) \leq 1.52\left[1 + 0.3\left(1 - P\right)\right]$$

These calculated nuclear enthalpy rise factors for the reference South Texas Cycle l design are conservatively based on the Technical Specification's rod insertion limits.

As stated in Section 4.3.2.2.6, "Maintenance of constant axial offset control establishes rod positions which are above the allowed rod insertion limits, thus providing increased margin to the $F^{N}_{\Delta H}$ criterion."

Question 221.8

Provide a description of the instrumentation and procedures to be used in implementing the Technical Specification on Reactor Coolant System Flow. Provide a description of the associated uncertainties.

Response

The following is provided for monitoring Reactor Coolant System flow:

1. The STPEGS Technical Specifications will have a requirement to verify Reactor Coolant System (RCS) flow.

2. The operator has at his disposal several methods of detecting significant RCS flow reduction. These are:

   a. Flow meter on each RCS loop.

   b. If operating in an automatic control rod mode (t_{avg} held constant), a reduction in reactor power would be present for significant reductions in RCS flow.

   c. If operating in a manual control rod mode (power held constant), an increase in ΔT across the core would be present for significant reductions in flow.

   d. Local changes in flow could be indicated by incore flux maps (assuming significant changes in local power).

   e. Core exit thermocouple readings.

The Technical Specifications will require that the operator verify flow, perform calorimetric power checks, and incorporate flux maps.

Question 221.9

HISTORICAL INFORMATION

The THINC-IV code assumes a uniform radial pressure distribution at the core exit for all of the South Texas Units 1 and 2 thermal-hydraulic design calculations. The Westinghouse 1/7 scale hydraulic tests at the Forest Hill Facility indicate a significant radial pressure gradient. This gradient has also been observed in recent calculations by staff consultants.

The effects of the expected radial pressure gradient must be accounted for in the South Texas Units l and 2 thermal-hydraulic design. This can be accomplished by:

1. Providing the expected radial pressure gradient; and either

2. a. Providing a revised thermal-hydraulic design calculation which includes the effects of the radial pressure gradient, or

   b. Providing an analysis which demonstrates that the effect is small and that sufficient margin exists in the South Texas conservatism in this area.

Therefore, provide the above information.

Response [HISTORICAL INFORMATION]

As requested in (2), a cosine upper plenum radial pressure gradient with a maximum value of 5 psi at the core center and 0 psi at the core periphery was assumed for four-loop and three-loop operation. The results of this analysis showed that there was no effect on the minimum DNBR (to three significant figures) of this radial pressure gradient on four-loop or three-loop operation.

In performing this analysis, the hot assembly was assumed to be in the center of the core where the greatest flow reduction near the core outlet will occur due to the radial pressure gradient. In addition, an axial power distribution extremely peaked to the top of the core (~+30 percent axial offset) was assumed. This axial power distribution is more severe than would be expected during plant operations.

Thus, the use of a uniform upper plenum pressure distribution in thermal-hydraulic design is acceptable.

HISTORICAL INFORMATION

Question 492.1N

Operating experience on two pressurized water reactors, not of Westinghouse design, indicate that a significant reduction in the core flow rate can occur over a relatively short period as a result of crud deposition on the fuel rods. In establishing the Technical Specifications for South Texas Project, Units 1 & 2, we will require provisions to assure that the minimum design flow rates are achieved. Therefore, provide a description of the flow measurement capability for South Texas Units 1 and 2 as well as a description of the procedure to measure flow.

HISTORICAL INFORMATION

HISTORICAL INFORMATION

Response [HISTORICAL INFORMATION]

Operating experience to date has indicated that a flow resistance-allowance for possible crud deposition is not required. There has been no detectable long-term flow reduction reported at any Westinghouse plant. Inspection of the inside surfaces of steam generator tubes removed from operating plants has confirmed that there is no significant surface deposition that would affect system flow. Although all of the coolant piping surfaces have not been inspected, the small piping friction contribution to the total system resistance and the lack of significant deposition on piping near steam generator nozzles support the conclusion that an allowance for piping deposition is not necessary. The effect of crud enters into the calculation of core pressure drop through the fuel rod frictional component by use of a surface roughness factor. Present analyses utilize a surface roughness value which is a factor of three greater than the best estimate obtained from crud sampling from several operating Westinghouse reactors.

The operator has at his disposal several methods of detecting significant RCS flow reduction, these are:

1. Flow meter on each RCS loop.

2. If operating in an automatic control rod mode (T_c held constant) a reduction in reactor power would be present for significant reductions in RCS flow.

3. If operating in a manual control rod mode (power held constant) an increase in ΔT across the core would be present for significant reductions in flow.

Question 492.2N

Regulatory Guides 1.133, Revision 1 and 1.70, Revision 3 require that FSAR Section 4.4.6 contain a description of the Loose Parts Monitoring System (LPMS) which will be installed at South Texas Units 1 and 2. The information that should be supplied is:

(1) a description of the monitoring equipment including sensor locations;

(2) a description of how alert levels will be determined, including sources of internal and external noise, diagnostic procedures used to confirm the presence of a loose part, and precautions to ensure acquisition of quality data;

(3) a description of the operation program, including signature analysis during startup, normal containment environment operation, the seismic design, and system sensitivity;

(4) a detailed discussion of the operator training program for operation of the LPMS, planned operating procedures, and record keeping procedures;

(5) a report from the applicant which contains an evaluation of the system for conformance to Regulatory Guide 1.133; and,

(6) a commitment from the applicant to supply a report describing operation of the system hardware and implementation of the loose part detection program.

Response

(1) A description of the monitoring equipment is provided in Section 4.4.6.4.

(2) A description of how alert levels will be determined is provided in Section 4.4.6.4.

(3) The LPMS will be used to generate baseline signatures for all channels at center frequencies of 25 HZ, 250 HZ, 2.5 KHZ, and 25 KHZ during initial startup testing, at reactor power levels of approximately 30 percent, 50 percent, 75 percent, and 100 percent. During reactor startups following breaching of the RCS or steam generators and in instances of automatic LPMS actuation additional signatures will be obtained for comparison with previous signatures to detect changes in amplitude or frequency.

Information concerning normal containment environment operation, seismic design, and system sensitivity is provided in Section 4.4.6.4.

(4) Licensed operators (including licensed supervisors) are presented a lecture (approximately 2 hours in length) on the Vibration and Loose Parts Monitoring System.

Response (Continued)

This lecture presents the following:

- Function and description of operation of the system,
- Function and description of major components of the system, and
- Controls and indication associated with the system.

Operators are provided with objectives which identify what knowledge they should gain from the lecture and are examined on these objectives with minimum passing grade of 70 percent.

Following the classrooms training, operators should have sufficient knowledge of the system, and its operation to be able to safely follow plant operating procedures or system operating instructions which address indications or alarms associated with the Vibration and Loose Parts Monitoring System. This training should allow operators to identify when the system has identified a vibration or loose parts problem, relate that indication to other plant indications, initiate actions (if necessary) to maintain the plant in a safe condition, and notify engineering for a detailed analysis of the vibration or loose part.

For practical experience with the operation of the system, current operators will be involved with the plant startup testing and operation during establishment of baseline data for the system. After commercial operation, operators will have practical factors associated with operation of the system during On-The-Job training as necessary to ensure that operators are able to conduct applicable procedures.

(5) The conformance of the Loose Parts Monitoring System (LPMS) to Regulatory Guide (RG) 1.133 has been provided in Section 4.4.6.4 and Table 3.12-1.

(6) The information provided or to be provided in items 1-5 includes sufficient details to satisfy the requirements of RG 1.70 and Standard Review Plan (SRP) 4.4 (Rev. 1 - July 1981). HL&P should not be required to provide an additional report. Note that HL&P will not have any operating experience to report until some time after the STPEGS units are in operation.

Question 492.3N

State your intentions with regard to N-1 loop operation.

Response [HISTORICAL INFORMATION]

At this time, HL&P does not intend to pursue licensing for N-1 loop operation of the STPEGS Units.

HISTORICAL INFORMAITON

HISTORICAL INFORMATION

Question 492.5N

Provide the documentation required by NUREG-0737 Item II.F.2. The responses to the documentation should be given item-by-item.

Response [HISTORICAL INFORMATION]
Instrumentation for the detection of inadequate core cooling is provided per NUREG-0737, Item II.F.2 and RG 1.97 (see Appendix 7A, Item II.F.2). The instrumentation includes reactor vessel water level indication, core exit thermocouples, and a subcooled margin monitor. This instrumentation provides unambiguous, easy-to-interpret indication of inadequate core cooling.

HISTORICAL INFORMAITON

HISTORICAL INFORMATION

HISTORICAL INFORMAITON

Question 492.6N

Do you attribute the vibrational problems at the Paluel Station to the 14 foot core design?

Response [HISTORICAL INFORMATION]

Based on our review of the available information and the operational experience of other Westinghouse 14-ft core plants, we do not believe that the 14-ft core change is responsible for the problems at Paluel. The evidence to date indicates that the use of structurally more flexible thimble tubes in conjunction with the larger flow area around the tubes are the primary causes of the problems at that plant.

For 14-ft core plants the lower support plate has a larger number of smaller diameter flow holes than the plate for 12-ft core plants to allow flow up through the fuel. This results in a larger axial pressure gradient across the lower support plate. Thus the axial flow velocity in the annulus between the thimble and its guide structure increases. This may accentuate the potential for experiencing problems of this nature, particularly when the thimble size is reduced.

Westinghouse and STPEGS are closely monitoring the results of the test programs at EDF and Framatome in order to ensure that STPEGS is not adversely affected by similar problems. In addition, Westinghouse is also closely following the performance of the Westinghouse 14-ft core plants that are in operation.

HISTORICAL INFORMAITON

Question 492.7N

Do you feel the same vibrational problems are possible at STPEGS? If you do, then quantify the safety impact of such a problem. If you do not, then explain any design differences between STPEGS and Paluel that lead to this conclusion.

Response [HISTORICAL INFORMATION]

As was previously noted (letter ST-HL-AE-1334, dated 2/3/86) the vibrational problem experienced at Paluel is the vibration of the BMI thimble, not vibration of the reactor vessel lower internals. The STPEGS Units 1 and 2 use a flux thimble with a nominal outside diameter of .313 inches. The Paluel units (1, 2, 3, 4) are using a thimble with an outside diameter of .295 inches. The STPEGS thimbles also have a slightly thicker wall than the Paluel thimbles. The larger thimble also results in a smaller annular gap between the flux thimble and the inside of the BMI columns. (Unit 1 was modified such that the BMI column gap size is similar to Unit 2.) In conclusion, the stiffer STPEGS thimbles, with the smaller gaps, will perform satisfactorily based on the European plant experience to date.

With respect to the safety aspects of a thimble wear problem if it were to occur, we do not believe the issue to be a safety concern. Previous evaluations have been made by Westinghouse regarding the failure of flux thimble tubes. The evaluation concluded that up to three (3) BMI thimble tubes can fail simultaneously with a complete instantaneous guillotine break, and the coolant loss can be made-up by the output of the on-line charging pump. Since the coolant loss would not exceed the make-up capability of normal charging, no SI (safety injection) signal is generated. The occurrence of a thimble tube leak would be identified by the detectors in the seal table room.

It should be pointed-out that the assumption of three tubes rupturing at the same time is highly conservative. As noted above, even if the tubes ruptured, the plant would easily be able to complete a controlled shutdown so that the leaking thimble could be either isolated or replaced.

HISTORICAL INFORMAITON

Question 492.8N

In light of the Paluel experience, do you still believe that the vessel model flow test which you submitted in your FSAR is still valid?

Response [HISTORICAL INFORMATION]

The purpose of the vessel model flow test is to demonstrate the structural integrity of the reactor vessel system and to provide data regarding the response of the reactor internals during operation. The BMI thimbles were not included in the vessel model flow test. Westinghouse has shown that these model tests are accurate and reliable predictors of plant performance and that these objectives were met. The vibratory levels of the reactor internals have been shown to be negligible and the response of the internals is well behaved. As has been previously discussed in letter ST-HL-AE-1339, dated February 3, 1986, the issue here is vibration of the removable thimble tubes, not vibration of the reactor internals. Hence, the vessel model flow test is still valid. Please note that this issue was discussed by Westinghouse at the MEB review on STPEGS and Westinghouse has provided a vibration assessment report, WCAP-10865, which demonstrates the acceptability of the vibration levels of the STPEGS units.

Question 122.1

Provide a list of all ASME Class 2 and Class 3 components of ferritic steel for each system in South Texas l and 2. Provide the fracture toughness data obtained for the components, and indicate the fracture toughness requirements, specifications, testing procedures, and acceptance standards that were followed to obtain the data.

Response

Fracture toughness testing is required only for the Main Steam and Feedwater Systems; other systems in Safety Classes 2 and 3 are exempted.

The following list is an example of ASME Class 2 and 3 components with ferritic pressure boundary material at STPEGS 1 and 2.

Reactor Containment Fan Coolers
Main Feedwater Isolation Valves
Main Steam Isolation Valves
Main Steam Safety Valves
Main Steam PORVs
Standby Diesel Generator
Standby Diesel Generator Fuel Oil Storage Tank
Auxiliary Feedwater Pumps
CCW Pumps
CCW Surge Tanks
CCW Heat Exchangers
ECW Self Cleaning Strainers
Feedwater Isolation Bypass Valves
Steam Generator Preheater Bypass Valves
Auxiliary Feedwater Pump Discharge Valves
Steam Generator Feedwater Bypass Valves
Auxiliary Feedwater Turbine Steam Isolation Valves
Main Steam Isolation Bypass Valves
Main Steam Vents & Drains Isolation Valves
Auxiliary Feedwater Pump Recirculation Valves
Steam Dump Valves
Feedwater Control Valves
Feedwater Bypass Control Valves

The responses to NRC Questions 122.17 and 122.22 provide the requirements, specifications, testing procedures, and acceptance standards for the fracture toughness.

Question 122.3

State whether any of the following materials that have a yield strength greater than 90,000 psi are being used in the control rod drive mechanisms or in the reactor internals: cold-worked austenitic stainless steels or hardenable martensitic stainless steels. If such materials are employed, identify their usage and provide evidence that stress corrosion cracking will not occur during service life in components fabricated from the materials.

Response

Hardenable martensitic stainless steel Type 403 with a minimum yield strength of 90,000 psi is used for the reactor internals holddown spring. Martensitic stainless steel ASTM A276 Type 410 Condition H with a minimum yield strength of 90,000 psi is used in the following control rod drive mechanism applications: drive rod breech housing, external breech, disconnect rod, positioning nut, and loading bolt.

Stress corrosion cracking in these materials is not anticipated in a Westinghouse PWR with properly controlled reactor coolant chemistry; RCS chemistry specifications and control are discussed in Section 5.2.3.2.1.

Question 122.4

Provide the tempering temperatures of the hardenable martensitic steels and the processing and treatment of other special materials, such as cobalt-base alloys and Inconels, used to fabricate the control rod drive mechanisms and the reactor internals. Provide information regarding the mechanical properties of all of the special materials.

Response

Reactor Internals

The hardenable martensitic stainless steel Type 403 (with minimum yield strength of 90,000 psi) used for the reactor internals holddown spring, identified in the response to Question 122.3, is tempered at 1125^{0}F.

Inconel (Type 600 for clevise and Type 750 for some bolting) with 115,000 psi yield strength is solution treated at 1800^{0}F and double aged at 1350^{0}F and 1150^{0}F.

Stellite is used as a hard surface clad for wear resistance in some areas of the reactor internals (such as the core barrel/vessel keys).

Control Rod Drive Mechanisms

Martensitic stainless steel ASTM A276 Type 410 Condition H (with minimum yield strength of 90,000 psi), which is used in the CRDM applications identified in the response to Question 122.3, is tempered at a minimum temperature of 1050^{0}F.

Inconel X750 Class A with minimum tensile strength of 190,000 psi is used for the latch assembly return springs. This material is temper cold drawn and age hardened at 13500F.

Inconel X750 Class D with minimum tensile strength of 220,000 psi is used for the drive rod locking spring. This material is temper cold drawn and age hardened at 1200^{0}F.

Inconel 718 with minimum Rockwell hardness of 41 is used for the springs on the heavy drive rod. The material is processed by an 18-25 percent cold reduction; it is aged at 1400^{0}F, cooled to 1200^{0}F and held, and cooled to room temperature.

Haynes 25 with minimum yield strength of 120,000 psi and Rockwell hardness of 35-45 is used for the locking button. This material is solution heat treated at 2250^{0}F, quenched or rapidly air cooled, reduced, and aged at 1000^0 to 1050^{0}F. Heat treated and cold worked Haynes 25 with Rockwell hardness of 34-40 is used for the latch pins.

Stellite 6 is used on the latch tips.

<u>Question 122.6</u>

Provide the specific welding materials used for fabricating the ferritic steel components of the reactor coolant pressure boundary (i.e., reactor vessel, pressurizer, steam generator) and give the ASME specifications for the welding materials. Discuss or tabulate separately the specific welding materials used to join parts fabricated from SA 533 Grade A, B, or C, Class 2 and SA 508 Class 2a materials.

<u>Response</u>

Acceptable welding materials for pressure-retaining weldments in the reactor vessel are ASME Specifications SFA 5.5, SFA 5.11, and SFA 5.14, as well as Military Specification MIL-E-18193A.

The welding material used for pressure-retaining weldments in the pressurizer is ASME Specification SFA 5.5. ASME Specification SFA 5.5 welding material is used to join parts fabricated from SA 533 Grade A Class 2 and SA 508 Class 2a materials.

Acceptable welding materials for pressure-retaining weldments in the steam generator (primary side) are ASME Specifications SFA 5.5 and SFA 5.17. ASME Specifications SFA 5.5 and SFA 5.17 welding materials are used to join parts fabricated from SA 533 Grade A, B, or C Class 2 and SA 508 Class 2a materials.

Acceptable welding materials for pressure-retaining weldments in the Model Delta 94 steam generator (primary side) are ASME Specifications SFA 5.5, SFA 5.11 Class ERNiCrFe-7, SFA 5.14 Class ERNiCrFe-7, SFA 5.23. ASME SA 533 Grade A, B, or C Class 2 and SA 508 Class 2a materials are not used in the construction of the Model Delta 94 steam generators.