Department of Energy
Washington, D.C. 20545

APR 29 1982

Docket No. 50-537
HQ:S:82:024

Mr. Paul S. Check, Director
CRBRP Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Check:

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION - CHEMICAL AND MECHANICAL ENGINEERING

References:
1. Letter P. S. Check to J. R. Longenecker, "CRBRP Request For Additional Information," dated February 26, 1982
2. Letter P. S. Check to J. R. Longenecker, "CRBRP Request For Additional Information," dated March 23, 1982
3. Letter J. R. Longenecker to P. S. Check, "Responses to Request For Additional Information - Chemical Engineering," dated April 21, 1982

This letter formally responds to your request for additional information contained in References 1 and 2.

The enclosed responses CS 281.6, CS 281.8, CS 281.9, and CS 281.12, along with the responses previously submitted in Reference 3, complete the Project's response to your questions from Reference 2.

Also enclosed are responses to questions CS 210.2 and CS 210.6. The remaining questions from Reference 1 will be answered as follows: CS 210.1, 3, 4, 5, and 7 by May 14, and CS 210.8 and 9 by May 28, 1982.

The enclosed responses will be incorporated into the PSAR in Amendment 68 scheduled for May 7, 1982.

Sincerely,

John R Longenecker

John R. Longenecker, Manager
Licensing & Environmental
Coordination
Office of Nuclear Energy

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Question CS210.2

Describe methods and procedures used to evaluate the structural integrity at locations having savers thermal stripping effects.

Response

The response to this question is provided in amended Section 4.2.2 by the addition of subsection 4.2.2.6.

4.2.2.6 Thermal Striping Evaluation of Reactor Internals

4.2.2.6.1 General Methodology for Thermal Striping Evaluation

The general method used to evaluate components for thermal striping is illustrated in Figure 4.2-130. The steps involved are as follows:

a) The maximum potential for thermal striping is identified. This potential is the calculated temperature difference in sodium coolant from adjacent streams. Included in the temperature difference are 2σ uncertainties.

b) Scale model tests are run as appropriate and the component thermal striping factors are measured. The measured striping factors are combined with the maximum striping potential to generate a thermal striping temperature time history and define a maximum fluid Δ T.

c) The thermal striping temperature time history is used as the boundary temperature in a transient thermal analysis of selected component. This analysis calculates the magnitude of the fluctuating difference in the surface temperature and the mean temperature (Ts-Tm).

d) The range of Ts-Tm is used to determine a stress or strain range which is used in the fatigue evaluation of the structure.

There are primarily 4 methods used to evaluate structures for thermal striping.

<u>1st Method</u>

This is the most simplified and conservative method used. In this method the maximum fluid ΔT (Hottest fluid Temp. - Coldest Fluid Temp) is compared to an allowable metal ΔT. (Temp. of the metal surface - metal mean temp). The method for determining allowable metal ΔT is described in Section 4.2.2.6.4.

If ΔT fluid < ΔT metal allowable the structure is adequate for striping.

<u>Discussion of Conservatism</u>:

There are several factors which cause this method to be conservative.

1. Using fluid temperatures is conservative, since this assumes an infinite film coefficient and neglects the mean temperature effects on strain. Since striping is a fast transient the mean temperature effect will be small for most cases. However, the film coefficient effect can be significant.

2. Using the observed maximum and minimum as umbrella temperatures is conservative since each cycle would not have this maximum strain range.

3. Assuming a biaxial stress state is conservative for some locations such as at corners, where there is no biaxial stress state.

4. Conservatism incorporated into the design such as factors on design fatigue curves and temperature uncertainties.

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2nd Method

The second method is used when the 1st method does not show the structure adequate. This method is to use the film coefficient and the materials thermal properties to determine the actual metal surface temperature. The metal surface ΔT is then compared to the allowable metal ΔT (Surface-mean) used in method 1. If Δ Tmetal surface < ΔTmetal allowable, the structure is acceptable for striping.

Discussion of Conservatism

Conservatism is the same as for method 1 but excluding item 1.

3rd Method (Detailed Analysis)

This method is used when methods 1 and 2 are not adequate to show that the structure is acceptable and there is reason to believe it can be shown adequate with this method. This method involves the evaluation of actual temperature date from prototypic tests. In this evaluation instead of using the hottest and coldest fluid temperatures, the fluid temperature data is first converted to metal temperatures by thermal solutions. Then the metal temperature peaks are umbrelled i.e., all temperature peaks between specified values are grouped and given the highest absolute temperature within the group. These umbrella temperatures are used to calculate strain ranges for each group.

$$\varepsilon_{r_i} = \frac{\alpha \, \Delta T_{metal}}{(1 - \nu)}$$

α = Instantaneous coefficient of thermal expansion
ν = Poisson's ratio
ε_r = Strain range

Where Tmetal = Difference between the max and min temperature variation of the metal within the group (conservative).

The number of cycles within each group (n_i) is projected throughout the reactor life. The allowable number of cycles (Nd_i) for the strain range in the corresponding group is obtained from the applicable fatigue curve. The total striping fatigue damage is, $D_{fatigue} = \Sigma_1^i \; n_i/Nd_i$

This damage will then be added to the creep-fatigue damage calculated for other transients. If the damage calculated here is within the creep-fatigue damage envelop, Figure 4.2-47A, the striping is considered acceptable.

Conservatism

Items 3) and 4) from Method 1

There is still some conservatism in umbrellaing the temperatures, but not as significant as before.

4th Method (Alternate Detailed Analysis)

Again this method is used only when methods 1, 2, or 3 are not adequate to show the structure acceptable. This method is used when the geometry of the structure is such that the previously discussed methods are overly conservative i.e., thin walled structures where mean temperature effects can be substantial, and corners where biaxiality effects vanish. Either effects tend to reduce the stress and strain.

For this method detailed finite element models are developed to determine the thermal and stress response of the structure for the specific striping environment. The stress and strain ranges determined from the models are then used to evaluate the fatigue damage due to striping.

Conservatism

See Item 4, from Method 1.

4.2.2.6.2 Maximum Potential Fluid ΔTs

To evaluate the effects of thermal striping on reactor internals, a first step is to determine the maximum potential for thermal striping that exists at each selected component. This maximum potential is the maximum temperature difference (ΔT) between the different sources of sodium in each region, including 2σ uncertainties. For example, the striping potential at a lower shroud tube is the ΔT between the control assembly (C/A) 2σ exit temperature and the hottest adjacent fuel assembly (F/A) $+2\sigma$exit temperature. The striping potentials are developed using a conservatively high core ΔT that represents a 115% power condition. This core ΔT is about 60^{O}F higher than expected in the CRBRP. A listing of the maximum potential fluid for different components is included in Table 4.2-68.

4.2.2.6.3 Maximum Fluid ΔTs

The maximum fluid ΔT at a given location is determined empircally as a fraction of the maximum potential fluid ΔT. Water model testing was performed in 1) a full scale mockup of a cluster of seven removable assemblies and in 2) the quarter scale IRFM (Integral Reactor Flow Model) at HEDL. Hot and cold water exiting the core assembly outlet nozzles at appropriate flow rates simulate the assembly to assembly ΔT. Fast response thermocouples near the surfaces of the components models provide the time history of the fluid temperature fluctuations normalized to determine a normalized striping factor.

The results of these tests provide a measure of the maximum fluid ΔT at critical locations.

7-Assembly Tests

The 7-assembly test model is a full scale model of the reactor internals configuration above a cluster of 7 core assemblies including the assembly exit nozzles. Configurations tested include the following components above the central of the 7 core assemblies.

- o Lower shroud tube and primary control rod driveline.
- o Lower shroud tube and secondary control rod driveline.
- o Lower shroud tube at core periphery without control rod driveline.
- o Instrumentation post.

Figure 4.2-131 is a sketch of the 7-assembly test model. The configuration here is with an instrumentation post. In this configuration the thermal striping factors are measured on the instrumentation posts for the different combinations of fuel assemblies and blanket assemblies that are below it.

Integral Reactor Flow Model - Thermal Striping

The integral reactor flow model (IRFM) is a 360 degree 0.248 scale model of the CRBRP outlet plenum and its internal components which models significant hydraulic and vibrational characteristics of the CRBRP. Sodium in CRBRP is modeled by water in IRFM. An overall view is shown in Figure 4.2-132. Fuel, control, and blanket assemblies are represented by tubes orificed at the bottom to give the proper flow distribution. The top ends of all core outlet nozzles are prototypic to produce the proper velocity profile. Four different zones (fuel assemblies, inner blanket and control assemblies, radial blanket assemblies, and removable radial shield assemblies) were provided with different specified flows and temperatures and were varied to model striping conditions at various core life conditions.

The frequency and amplitude of fluid temperature oscillations were monitored on the outlet plenum permanent reactor structures. Thermal striping in these regions are due to temperature differences between the fluids exiting the fuel assemblies, the radial blanket assemblies, the inner blanket assemblies, the control assemblies and the removable radial shield assemblies.

In addition to the striping measurements at the reactor components, measurements were made in the outlet plenum using radial traversing probes. The axial location of these probes is illustrated on Figure 4.2-133.

Test Results

The results of these thermal striping tests are listed on Table 4.2-68 in the form of normalized striping factors. The factors are a fraction of the striping potential with a factor of 1.0 being 100% of potential. Results from the traversing probes in the outlet plenum are illustrated on Figure 4.2-134 in the form of striping ΔT contours. The striping levels on all components outside of this region, such as the reactor vessel thermal liner, the suppressor plate, and the primary piping, are within the range of allowables as shown on Table 4.2-69.

4.2.2.6.4 Striping Limits for Alloy 718 and Stainless Steels 304 and 316

Many reactor internal components experience strain controlled cyclic deformations in excess of 1×10^6 cycles. From the available fatigue data, reference 182, the allowable strain or stress range for 1×10^6 cycles or more approaches the endurance limits of each of the concerned materials. These allowable strain or stress limits can be converted into limits on the metal surface to mean temperature range (ΔT total). The allowable striping ΔT metal can be conservatively determined from the endurance stress or strain ranges for a fully constrained biaxial stress case, e.g. thick flat plate. The following relations are used;

$$\Delta T_{metal} = (1-\nu) \frac{\Delta \sigma_r}{E \alpha_i} \quad \text{(a) (stress formula)}$$

$$\Delta T_{metal} = (1-\nu) \frac{\Delta \varepsilon_r}{\alpha_i} \quad \text{(b) (strain formula)}$$

where ΔT_{metal} - Metal surface temperature minus the mean temperature See Figure 4.2-130, Section C

$\Delta\sigma_r$ = endurance stress range ($2S_a$)

$\Delta\varepsilon_r$ = endurance strain range

α_i = Instantaneous coefficient of thermal expansion

ν = poisson's ratio

E = modulus of elasticity

This limit represents one of the most conservative cases as discussed in Method 2 in 4.2.2.6.1. If the striping ΔT metal on the component is below this limit, no further analysis is necessary.

In some applications the alternating stress or strain is accompanied by mean stress or strain. The mean stress or strain effect is detrimental and the allowables should be adjusted accordingly.

For 304 and 316 stainless steel the design endurance strain ranges can be obtained from Figure 4.2.47B, provided that the metal temperatures do not exceed 1100^{o}F. The striping ΔT_{metal} limits are given in Table 4.2-69 with selected mean strain effects.

The allow 718 fatigue curve is shown in Figure 4.2-48. The striping stress limits are developed from the endurance stress range of the up-to-date data provided in Reference 183. The high cycle (10^8 or more cycles) design fatigue strength curve with no mean stress is shown in Figure 4.2-49. Corrections for mean stress effects should be made by calculating a reduced allowable alternating stress intensity, s_a, as follows:

$$s_a' = s_a \quad \frac{200{,}000 - Smax}{200{,}000 - 2Sa}$$

where

s_a = the lowest allowable alternating stress instensity from Figure 4.2-49 for the metal temperature range of interest, psi

$s_{max.}$ = the maximum calculated stress intensity during the stress cycle, psi

s_y = material yield stress for the metal temperature range of interest, psi

The above equations for s_a' are used with the following restrictions:

A. If $2\ s_a \geq s_y$ or $2\ s_a > s_{max}$, $s_a' = s_a$ and the equation is not used.

B. If $s_{max.} \leq s_y$ and $2\ s_a \leq s_{max.}$, the equation is directly applicable.

C. If $s_{max.} > s_y$ and $2\ s_a < s_y$, use $s_{max.} = s_y$ in the equation.

The striping ΔT metal limits are obtained for any s_a' by applying equation (A). The allowable ΔT metal range for a fully constrained biaxial case with zero mean stress is shown in Figure 4.2-135 for allow 718. To obtain the allowable ΔT metal from Figure 4.2-135 the metal temperature used is the value within the striping range that yields the lowest allowable ΔT metal range. (The data base includes ASTM grain sizes of 5 through 10 with an average grain size of 7)

The effects of the fabrication processes and service environment on the structual integrity of the involved component shall be considered. The effect of grain size on the behavior of Alloy 718 shall also be considered. Some of the UIS liner plates cannot be purchased with an ASTM Grain size finer than 2, therefore a reduction in the allowable stress range is required. The allowable metal ΔT for a required reduction of 32% is shown in Figure 4.2-136 which does not include mean stress effects. When other grain sizes are used the allowable must be adjusted accordingly. Data used to determine the grain size effects on the endurance limit are from References 4.2-186 & 187.

4.2.2.6.5 Summary of Thermal Striping Results

The thermal striping evaluation methods an results are summarized in Table 4.2-68 for the permanent reactor internals. The general conclusion from this table is that all the permanent reactor internals are acceptable for their thermal striping environments. As cited previously and indicated by Figure 4.2-498, the striping values for all reactor components outside the Upper Internal Structure and above the Horizontal Baffle are below the allowable limits for stainless steel.

4.2.3 Reactivity Control Systems

The mechanical designs of the reactivity control systems consist of the primary control rod system (PCRS) and the secondary control rod system (SCRS). Each mechanical system consists of a control rod drive mechanism (PCRDM for primary, SCRDM for secondary) mounted on the top of the reactor vessel closure head; a control rod driveline (PCRD for primary, SCRD for secondary) connecting the mechanism to the absorber in the core region; and a control assembly (PCA for primary, SCA for secondary) located in the core region. The control assembly consists of a movable absorber pin bundle called the control rod and outer duct assembly.

These control rod systems perform the following functions upon signal from the Instrumentation and Control Systems:

1. Provide the primary shutdown (scram) system for the off-normal conditions.

2. Provide normal operational control.

3. Provide normal reactor startup and shutdown reactivity control.

4. Provide additional margin for control in the event of any anticipated reactivity fault.

Secondary Control Rod System

1. Provide the secondary shutdown system for off-normal conditions.

2. Provide reactor shutdown independent of the primary system.

3. Provide additional margin for control in the event of any anticipated reactivity fault.

178. A. L. Ward and L. D. Blackburn, "Ductility and Strength of FFTF/CRBRP Structural Materials Irradiated in Various Spectra, Interim Report", HEDL-TME-78-51, August 1978. (Availability, USDOE Technical Information Center).

179. A. L. Ward and L. D. Blackburn, "Ductility and Strength of FFTF/CRBRP Structural Materials Irradiated in Various Spectra, Second Interim Report", HEDL-TME-79-19, July, 1979. (Availability, USDOE Technical Information Center).

180. L. D. Blackburn, "Strength and Ductility of Fast Reactors Irradiated Austenitic Stainless Steels", HEDL-TME-79-21, August 1979.

181. TID-26666, "Nuclear Systems Materials Handbook", Volume 1, Property Codes 2206 (E-1) and 3304 (E-1), Hanford Engineering Development Laboratory, 1974.

182. A. Batenburg, H. O. Lagally, R. J. Simko, "Coefficient of Dynamic Friction via Analog Computer Data Reduction", ASLE Paper 80-LC-8B-1, August 1980.

183. T. T. Claudson, "Quarterly Progress Report: Irradiation Effects on Reactor Structural Materials, February - April, 1973, "HEDL-TME-73-74, pp. 53-60, May 1973.

184. KEI-64-80, Letter from D. D. Keiser to C. E. Gilmore, DOE/10, "Preliminary Alloy 718 ASME Code Case Package, "May 6, 1980. (Proprietary DOE)

185. R. A. Leisure, "Effect of Carbon and Nitrogen and Sodium Environment on the Mechanical Properties of Austenitic Stainless Steels", WARD-NA-94000-5, December, 1980.

186. C. R. Brinkman, "Material Data Base Used for Design and Elevated Temperatures for Long Term and Cyclic Loading", CRBRP/NRC Meeting on HTS Materials and Structures, April 6-7, 1982.

187. Huntington Alloys, Inconel Alloy 718 product brochure, 10M 10-73, T39, 1973.

*References annotated with an asterisk support conclusions in the section. Other references are provided as background information.

TABLE 4.2-68

REACTOR INTERVALS THERMAL STRIPING POTENTIALS AND STRIPING FACTORS

COMPONENT	LOCATION	MATERIAL	MAX. POTENTIAL FLUID T	SOURCE	NORMALIZED STRIPING FACTOR	TEST BASIS	MAXIMUM FLUID T	MAXIMUM METAL T	BASIS FOR ACCEPTABILITY
UIS	Instrumentation Post	1-718	291°F	1:F/A to B/A	.79	7-Assembly	230	222	Method #2
			343°F	1 above	.58	7-Assembly	199	-	Method #1
	Lower Shroud Tube	I-718	345°F	F/A to C/A	1	7-Assembly	345	270	Method #4
	Upper Shroud Tube	I-718	170°F	Shroud Tube-Outlet Plenum	1	Analysis	170	-	Method #1
	Lower Support Plate Liner	I-718	343°F	1 above	.41	7-Assembly	142	-	Method #1
	Skirt Liner	I-718	343°F	1 above	.41	IRFM	142	-	Method #1
	Skirt Ring Horizontal Liner	II-718	251	2: Radial Blanket Avg. to Peripherial F/A	.62	IRFM	156	-	Method #1
	Chimney Inlet	I-718	343	1 above	.45	IRFM	155	-	Method #1
	Chimney Outlet	I-718	343	2 above	.27	IRFM	93	-	Method #1
	IVTM Port Plug Liner	I-718	251	2 above	.49	IRFM	123	-	Method #1
	IVTM Port Plug Lower end	I-718	251	2 above	.49	IRFM	123	-	Method #1
	IVTM Port Plug Above I-718	316SS	251	2 above	.24	IRFM	60	-	Method #1
	Seismic Keys	316SS	251	2 above	.23	IRFM	58	-	Method #1
	Liner	I-718	251	2 above	.27	IRFM	68	-	Method #1
	Shear Web	316SS	251	2 above	.46	IRFM	115	88	Method #3
	Upper Support	31655	343	1 above	.15	IRFM	51	-	Method #1

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TABLE 4.2-68 (Continued)

COMPONENT	LOCATION	MATERIAL	MAX. POTENTIAL FLUID T	SOURCE	NORMALIZED FACTOR	STRIPING TEST BASIS	MAXIMUM FLUID T	MAXIMUM METAL T	BASIS FOR ACCEPTABILITY
	Upper Support Plate	31655	343	1 above	.15	IRFM	51	-	Method #1
Core Former Structure	Liners	I-718	326	F/A to Core Interstitial Flow	.61	IRFM & Interstitial Flow Test	200	- *	Method #1 *
Horizontal Baffle	Plate	316SS	145	Leakage to Outlet Plenum	.48	IRFM	70	-	Method #1
	FI&SA Nozzle	I-718	300	FI&SA to Outlet Plenum	.83	IRFM	248	192	Method #2

Method #1 - Most conservative, all cycles are umbrellaed under the maximum fluid T, and this fluid T is compared to the allowable metal T.

Method #2 - All cycles are umbrellaed under the maximum metal surface T, and this metal T is compared-to the allowable metal T.

Method #3 - The striping data peaks which are above the endurance strength T are umbrellaed under increments of T ranges, and the fatigue damage for these increments are determined and projected for the reactor life. This fatique damage is added to the creep-fatigue for other transients, then compared to the damage envelope.

Method #4 - A detailed finite element analysis (time history) of the component for its striping environment is performed.

*Evaluation in Process

TABLE 4.2-69

ALLOWABLE STRIPING METAL TEMPERATURE RANGE
FOR 304 AND 316 SS INCLUDING
MEAN STRAIN EFFECTS (< 1100°F).

ε mean (in/in)	$\Delta\varepsilon_r$ (in/in)	Allowable Surface Temp. Range (°F)
no mean	.0014	84
0002	.0013	78
00045	.0012	72
0009	.0011	66
00165	.0010	60
0025	.0009	54

Allowable Striping Metal Temperature Range for 304 and 316 SS Including Mean Strain Effects (< 1100°F).

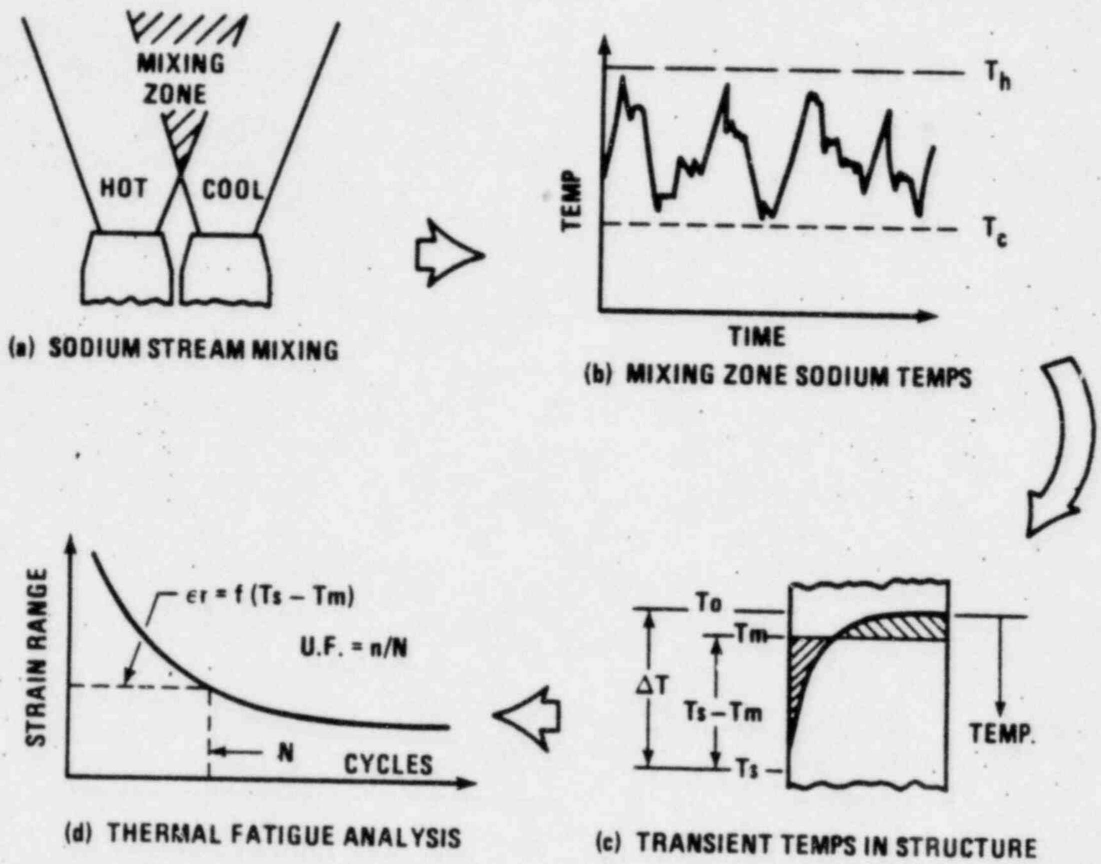

Figure 4.2-130. Thermal Striping Evaluation Steps

7044-1

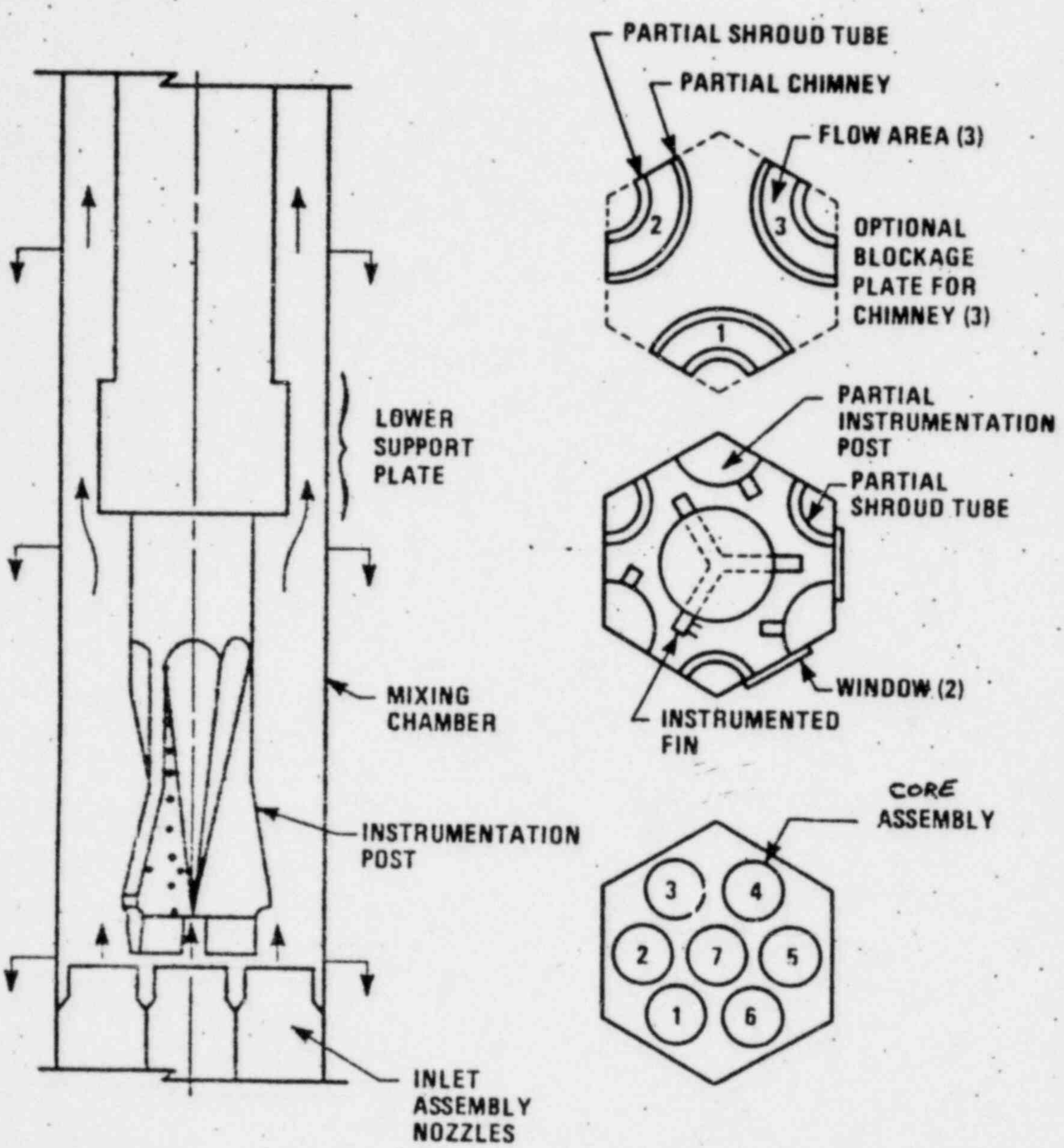

Figure 4.2-131. Instrumentation Post and Chimney Model

7044-2

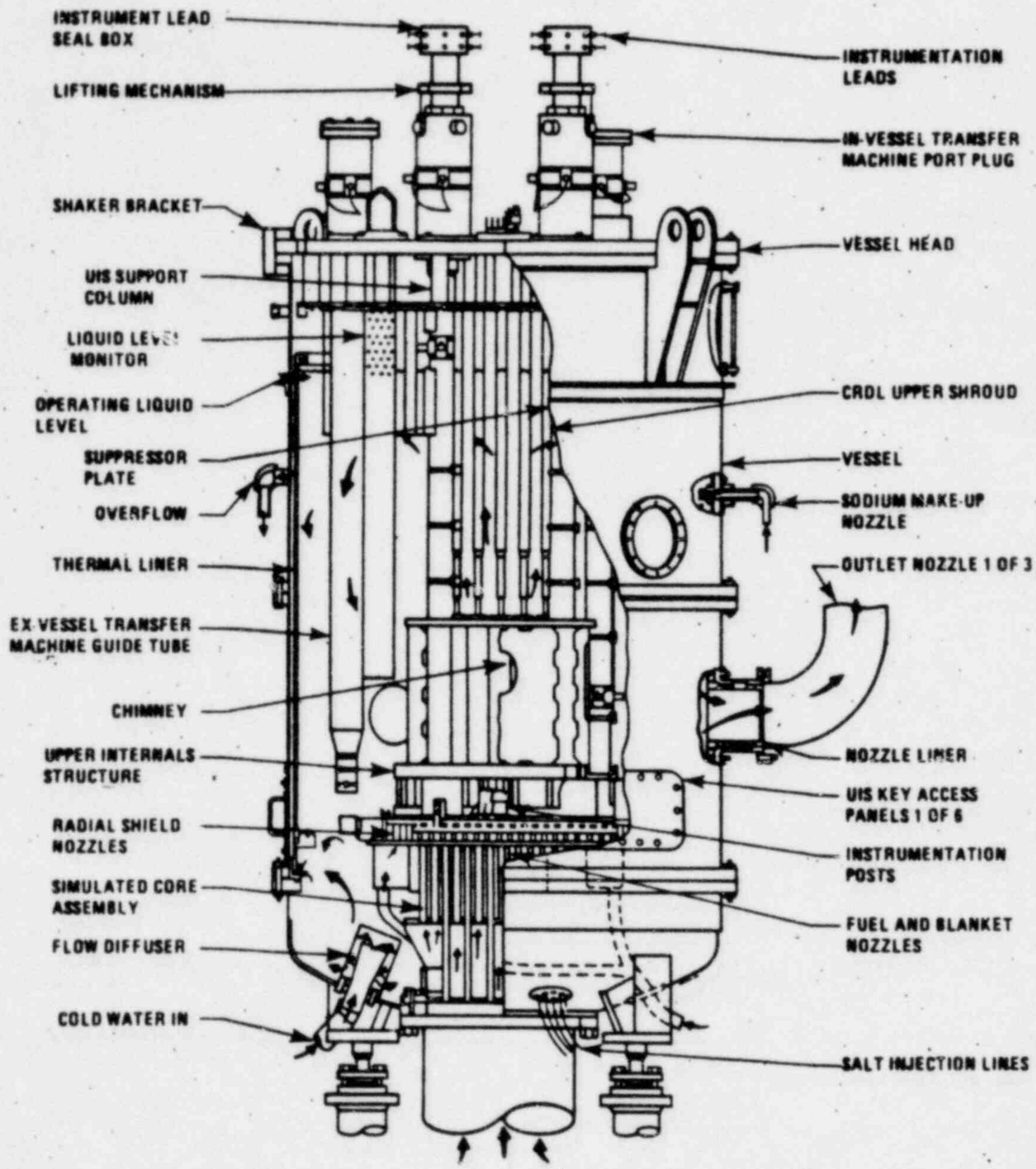

Figure 4.2-132. CRBRP Integral Reactor Flow Model

7044-3

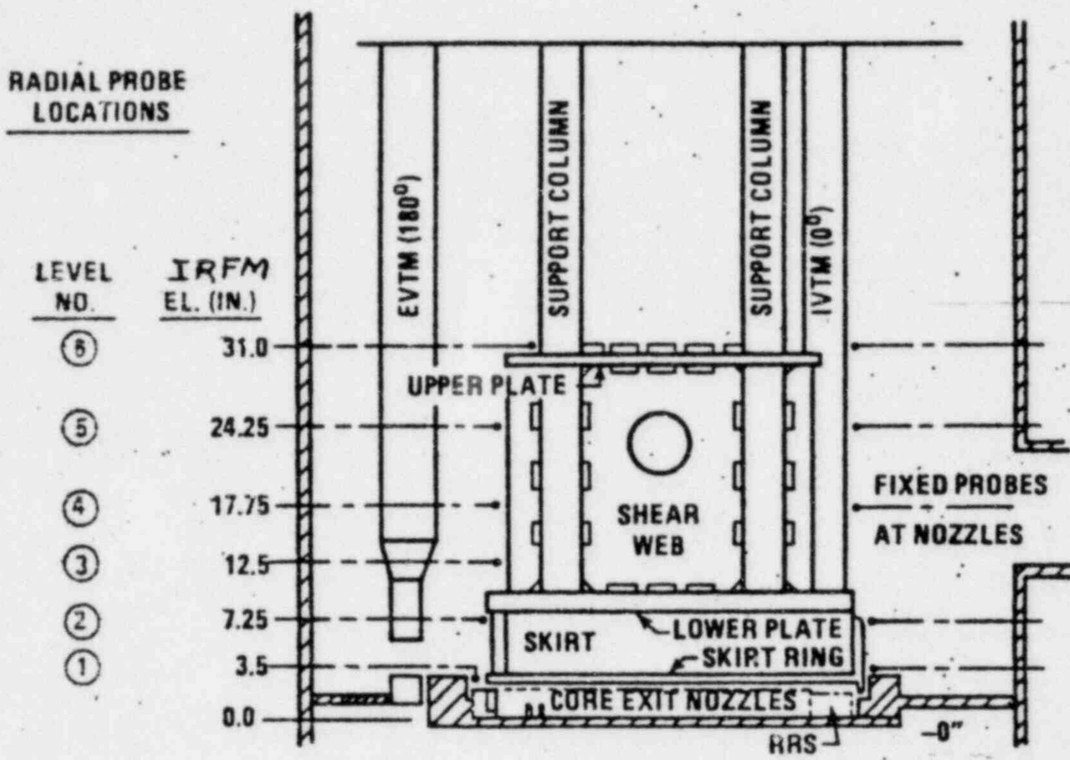

Figure 4.2-133. Radial Traversing Probes Axial Locations

7044-4

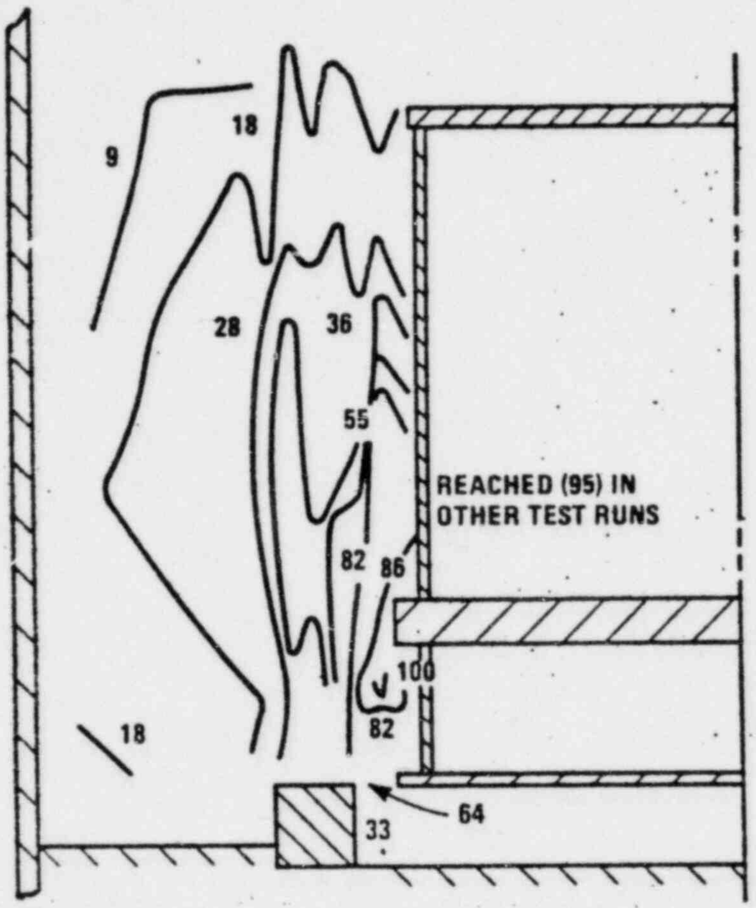

Figure 4.2-134. Striping Contours, °F, in the Outlet Plenum

7044-5

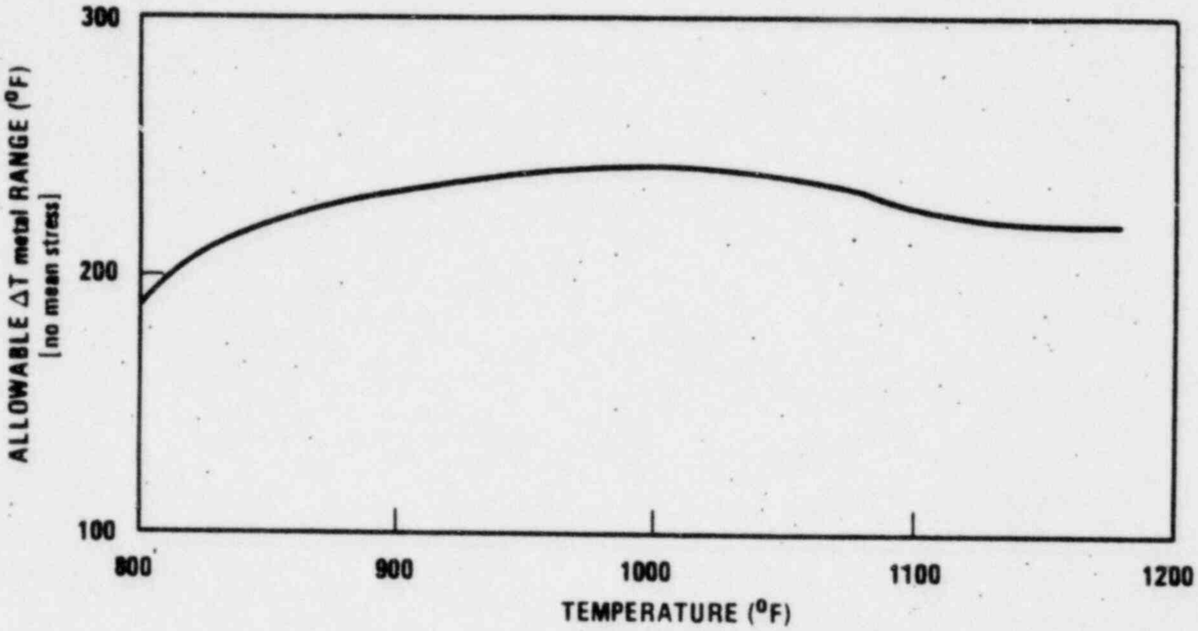

Figure 4.2-135. Allowable ΔT metal Range for Alloy 718 for a Fully Constrained Biaxial Case (Grain Size 5 and Finer)

7044-7

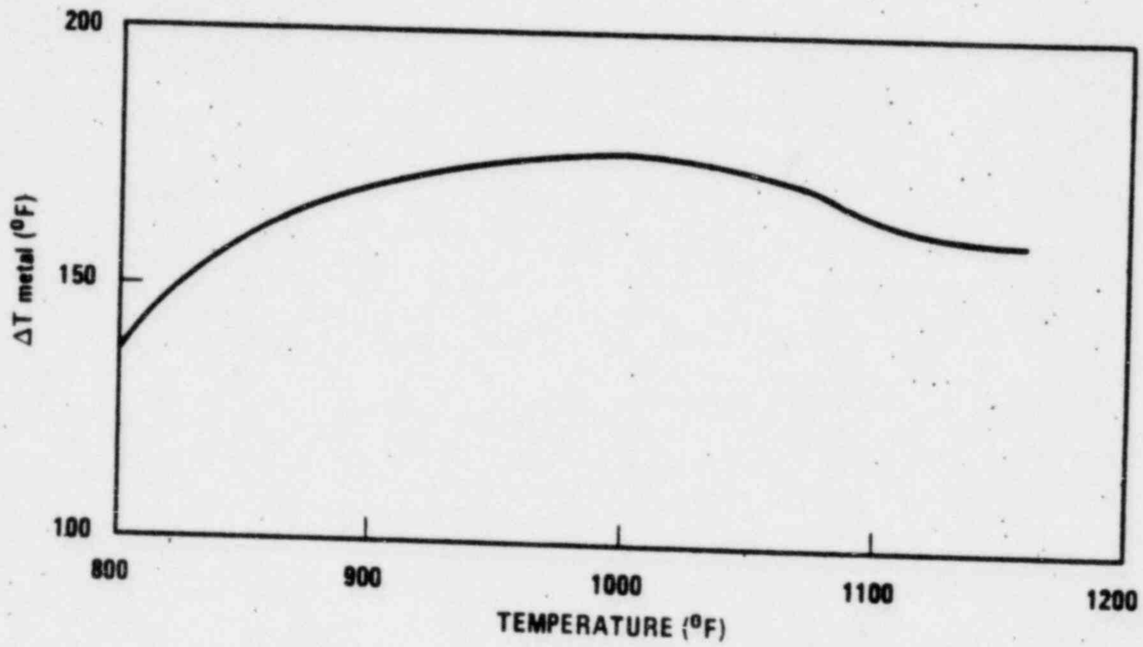

Figure 4.2-13~~1~~6. Allowable ΔT metal Range Vs. Temperature for Alloy 718 (Grain Size of 2) for a Fully Constrained Biaxial Case

Question CS210.6

Describe testing or analysis performed which assures the leakage will not occur at 2-1/2 Cr - 1 Mo threaded closures in steam generator bolted connections during elevated temperature service.

Response

The design of the steam generator modules has been modified to include integral welded steamheads in place of the bolted configuration. Furthermore, the design of the steam generator to piping interface has been changed to an all welded system which deletes the need for bolted flanged spool pieces at the inlet and outlet of both the evaporators and superheaters. Therefore, no additional analysis or testing to assure a leak-free system is required.

The PSAR will be modified to reflect the welded steamhead in a future amendment.

Question CS 281.6

In the CRBR primary and intermediate sodium piping system, Fe, Cr, and Ni are dissolved from the high temperature regions and deposited in the lower temperature regions because of super-saturation. Included in this process of mass transfer is the formation and decomposition of various transition metal and sodium double oxides. Deposition of these mass transfer and corrosion products may cause flow restrictions and loss of heat transport efficiency of heat exchangers. Describe the criteria and bases in your analyses of mass transfer and deposition of corrosion products in the CRBR primary and IHX sodium systems to assure necessary system flow and heat transfer. Include the instrumentation and detection system which will alarm when these limits are exceeded.

Response:

Primary Sodium Piping System

The IHX is designed with an effective tube length of 24.21 feet which is 5.12 ft or 33% greater than required for nominal operation. Included in the 33% excess allowance is a 9% factor for heat transfer degradation due to the deposition of mass transfer products on the primary side of the unit.

Corrosion products go into solution in the core region of the reactor either by direct dissolution or by the formation of soluble oxide complexes. As the coolant flows through the cooler regions of the system it becomes super-saturated with respect to these corrosion products and they precipitate out. Precipitation is expected to occur in the IHX. The deposits will result in some degradation of the overall heat transfer co-efficient. This potential problem was recognized in the early 70's and work was performed to determine the magnitude of this effect. A summary of this work is given below.

Two tube-in-shell heat exchangers were available from an ongoing corrosion program. The first had operated in an all stainless steel system with a hot leg temperature of 1325°F for 0.84 years. The second operated in a T304ss/Incoloy 800 system for 1.5 years with a hot leg temperature of 1100° and a high, (28 ppm by amalgamation) oxygen level. Heat transfer measurements were made on these heat exchangers and compared with similar measurements on new heat exchangers of identical design. The percentage change in heat transfer coefficient was determined for each set of readings.

The exposure conditions experienced by the heat exchangers were excessive in that the first one operated with a high maximum hot leg temperature (1325°F) and the second with a high oxygen level. Equivalent operating times at reactor operating conditions (1100°F mean T_{max}) were calculated at 5.3 years and 11.4 years. It was judged that the deposit thicknesses reached equilibrium. Additional increases in thickness are prevented by flow induced shearing of the friable deposits.

The average deposit heat transfer resistance was calculated as 8.4×10^{-5} h-ft^2 °F/BTU. The overall heat transfer coefficient of the IHX design was 1190 BTU/h-ft^2 °F. Adding the deposit resistance gives a calculated degradation of 9%. This 9% value was used for the FFTF IHX design and in the CRBRP design.

The effect of corrosion products on pressure drop and flow blockage is not addressed in section 5.3 of the PSAR. Flow blockage is addressed in section 15.4.1.3. Flow blockage in the Core Assembly in 15.4.1.3.1. 'Prevention and Detection' C, (Corrosion Products).

Deposition induced pressure drop in the IHX is not considered to be a factor because of the large flow cross sections on the shell side.

The effects of corrosion product deposition in the PHTS will result in a very gradual change, if any, in system performance. The PHTS performance is continuously monitored and critical performance parameters are calculated by the plant computer from temperature, flow rate and pressure drop sensors in the PHTS system.

Monitoring sensors in each loop include resistance temperature detectors at the inlet and outlet of the IHX, pressure sensors in the hot and cold legs, and a flow meter.

The performance evaluation for the PHTS is identified in the system procedures and includes:

Question CS281.8

Provide the design criteria and bases that demonstrate wastage allowance of the CRBR steam generator tubes, caused by sodium-water reaction products, is acceptable. The analysis should include Na water reaction temperature and other major variables in the small water leak situation.

Response

The steam generator tube wastage allowance and provisions to accommodate tube leaks are discussed in the revised PSAR Section 5.5.3.11.4. It should be noted that section 5.5.2.3.4 discusses the function of the tube sheet baffling as wastage baffles. This provides tube protection in the most likely location for leaks.

As a final level or protection against tube leaks in a steam generator, the steam generators and the IHTS are being designed to withstand the effects of a large sodium water reaction (SWR). The ASME Code categories being applied in the design of the steam generators and IHTS piping and components for the large SWR event are given in Table 5.5-10.

The design basis leak (DBL) for the CRBRP was selected based upon examination of the physical processes which exist for leak initiation and growth. The conservatism of this postulated DBL will be confirmed through the LLTR test program (Ref. 12).

Two types of tests have been reported which provide information on the leak growth mechanism - small scale tests which model effects of a SWR on materials, and large scale tests which model a large water leak in a model of a steam generator. Smaller scale sodium-water reaction tests have been done to develop an understanding of the effect of a SWR on neighboring tubes in a steam generator. Three mechanisms have been identified for leak growth: self-wastage, impingement, and overheating (mechanical damage from pipe whip, although extremely unlikely, would be considered another mechanism, as discussed later in this section). Self-wastage has been shown to occur for very small leaks in the range of 10^{-6}-10^{-3} lb/sec (Ref. 13). The process is depicted in Figure 15.3.3.3-1. The result of this process is a leak size of the order of 10^{-3} to 10^{-2} lb/sec. which can produce wastage on another tube in the vicinity of the leaking tube.

Wastage can occur on the outside of a steam generator tube from a leak in another tube in the vicinity. Tests of this mechanism have typically been done by using a water jet directed through sodium to a target material sample. Water injection rates of approximately 10^{-4} lb/sec to 1 lb/sec have been tested. The wastage mechanism results in erosion of the target material at maximum rates of 0.001 to 0.007 inches per second (Ref. 14, 29). The wastage rate is found to be a function of the water injection rate, tube spacing, CRBRP steam generator tube at these rates could cause a secondary water leak from the penetration. However, this would require at least 20 seconds to penetrate the 0.109 inch thick tube wall assuming an initiating leak of the proper characteristics to produce maximum wastage.

The size of a secondary water leak resulting from wastage is difficult to quantify since wastage tests are typically done in materials samples rather than pressurized tubes. The wastage areas observed in tests have ranged form 0.1 in^2 to 1.5 in^2. Failure areas corresponding to the highest observed wastage areas would result in water leak rates corresponding to that of a double-ended guillotine tube failure. However, the entire wastage area would not be expected to blow out. The wasted areas are typically pit-shaped with the area of the pit decreasing with depth. It would be expected that the small area at the bottom of the pit would fail, yielding a return water leak which halts the wastage. Therefore, while the size of a secondary failure caused by wastage is difficult to predict, it is expected to be smaller than the leak rate corresponding to a double-ended guillotine failure.

5.5.3.11.4 Compatibility with Coolants

The decarburization kinetics of 2-1/4 Cr-1 Mo base metal and subsequent strength loss are slow enough so that the selection of 2-1/4 Cr-1 Mo steel as steam generator tubing can be made with only a small design stress adjustment. 2-1/4 Cr-1 Mo steel creep rupture properties are insensitive to carbon content until the level drops below 0.03% carbon. This low carbon level will not be reached during the thirty-year tubing design life. The bulk carbon loss is predicted to be 0.04% at the 965^{o}F design temperature based on experimental decarburization data (Reference 5). Since the initial carbon content is 0.07 $<\% \leq 0.11$, the carbon content will not drop below 0.03%.

A minimum wall thickness of 0.109 inches has been specified for the CRBRP steam generator tubes. Of this wall thickness, a minimum thickness of 0.077 inches has been specified for strength with the balance of 0.032 inches allocated for total corrosion allowance of the sodium and water sides. The steam generator turbine material is 2-1/4 Cr-1Mo steel.

The possibility that wastage from a small sodium water reaction (or, more likely, rapid pressure rupture) would cause propagation of a leak to adjacent tubes has been considered in the definition of the design basis steam generator leak event. (See PSAR Section 15.3.3.3)

A small sodium water reaction event that does not result in replacement of the affected steam generator module may also produce wastage of steam generator tubes. The wastage near the leak site (combined with other corrosion) may be equal to or greater than the 0.32 inch corrosion allowance so that tubes will have to be plugged.

Based on the results of the Large Leak Test Rig (LLTR) tests (Reference 29), this amount of wastage is expected to affect only those eighteen tubes within 2 rows of the leaking tube and only a portion of those tubes. Based on the same test series, wastage of tubes beyond the 2nd row is expected to be significantly less than the 0.032 inch corrosion allowance. This is due to the large excess of sodium present which dilutes the SWR reaction products.

After each LLTR/SWR test the steam generator tube wastage was measured using a boreside ultrasonic testing (UT) device. Following the LLTR Series I program of 5 SWR tests and 1 inert gas test, the test steam generator was destructively examined and the wastage directly measured. Excellent agreement was found between these post-test measurements and the inter-test UT measurements. Similar UT equipment will be employed for CRBRP steam tube inspection and would be sensitive down to 4 mils wastage on a routine basis.

The plant will be shutdown on the basis of a confirmed leak. At shutdown the affected unit will be examined, using helium sniffing techniques to locate the leaking tube(s) and UT techniques to determine amount of wall thinning by wastage in surrounding tubes. Leaky tube(s) will be plugged; plugging of adjacent tubes will depend on the extent of wall thinning as determined by UT. The crib for tube plugging will be specified in the FSAR.

Because of the conclusive test results mentioned above and because tubes will be volumetrically inspected and plugged based on actual wastage, quantitative analysis of wastage as a function of temperature or other variables has not been performed; nor will it be usd as a basis for design or operating specifications.

No specific protection is required for protecting Type 304 SS or 2-1/4 CR-1 Mo steels against intergranular attack, stress-corrosion or general corrosion, provided that specified sodium purity is maintained.

In water or steam, carbon steel and 2-1/4 Cr-1 Mo steel are susceptible to caustic gouging and possibly caustic stress corrosion cracking. Maintaining the feedwater and steam drum purity levels as stated below will prevent these forms of localized attack. For normal operation other than start-up conditions, the feedwater and steam drum purity will be specified as follows:

Feedwater Impurities		Feedwater	Steam Drum
Suspended Solids	PPM	—	0.1
Dissolved Oxygen	PPM	.005	-
Silica	PPM	—	0.1
Iron as Fe	PPM	.01	-
Copper as Cu	PPM	.002	-
Hydrazine	PPM	.005-.015	-
Chlorides	PPM	—	.015
Sodium	PPM	.001	.006
Sulfate	PPM	—	.015
pH @ 77^{O}F		8.8-9.2	8.8-9.2
Conductivity (After Cation Removal) @ 77^{O}F micro-mho/cm		0.2	1.0

Limited duration operation with impurity levels above specified limits is allowable for periods not to exceed 24 hours in special instances. These special instances are defined to include condensate polishing system perturbations, such as those immediately associated with a termination of regeneration.

Corrosion impurities may enter the feedwater system through condenser leakage and/or poor makeup water. To guard against damage from such sources, the feedwater and steam drum water are maintained at levels within stated limits by full flow demineralization and continuous steam drum drainflow (blowdown) at a nominal rate of 10% of full power steam flow (See Section 10.4.7).

27. J. C. Amos, et.al, "Evaluation of LLTR Series II Test A-3 Results," General Electric Advanced Reactor Systems Department, November 1980, Prepared for U.S. Department of Energy under Contract No. DE-ATc3-76SF70030, Work Package AF 15 10 05, WPT No. SG037.

28. J. O. Sterns, "Metallurgical Evaluation of the Modular Steam Generator (MSG) after LLTR Testing," ETEC-78-12, Sept. 1978.

29. D. A. Greene, J. A. Gudahl and P. M. Magee, "Recent Experimental Results on Small Leak Behavior and Interpretation for Leak Detection", CONF-780201, Vol. 1, paper No. 12 (First Joint U.S./Japan LMFBR Steam Generator Seminar), February 1978.

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*References annotated with an asterisk support conclusions in the Section. Other references are provided as background information.

Question CS 281.9

Describe the sample and instrument readings and the frequency of measurements that will be performed to monitor the feed water purity and need for condensate cleanup system demineralizer resins and filter replacement. State the chemical limits and precaution to be taken to protect steam generator tubes against excessive corrosion and deposition. Also, provide the basis of establishing the chemistry limits.

Response:

PSAR Section 5.5.3.11.4 presents the feedwater and steam drum purity established to protect the steam generator tubes against excessive corrosion and deposition. PSAR Section 5.5.3.11.4 also adds additional information relative to monitoring and controls. The following major factors provided the basis for establishing the chemistry limits:

1. Because of the relatively low evaporator recirculation ratio in CRBRP, it was recognized early in the program that the CRBRP water chemistry limits would need to be similar to those limits which extensive experience in the fossil fired boiler and nuclear steam generator industries with once-through designs had shown to be required. Basically, this requires the use of all volitile treatment (AVT) consisting of a pH adjustment agent (typically amonium hydroxide) and an oxygen scavenging agent (typically hydrazine). The concentration of AVT agents is controlled in the feedwater to minimize corrosion in both the feedwater train and in the evaporator recirculation loop. Therefore, the then existing industry AVT chemistry requirements were established as the basis for CRBRP chemistry control.

2. These chemistry requirements were further refined to address the particular needs of CRBRP relative to materials, i.e., because of the 90/10 copper-nickel condensor, a 0.002 ppm copper concentration was specified. This low limit minimizes the potential for transport of copper to the evaporator tube internal surfaces where it would cause excessive tube corrosion.

3. The low recirculation ratio in the evaporators results in DNB in the 2-1/4 Cr-1 Mo evaporator tubes. This requires close control of the sodium ions to prevent stress corrosion cracking problems and close control of the chloride and sulfate ions to prevent "under deposit" corrosion. For example, the feedwater sodium ion concentration is maintained at 0.001 ppm maximum to achieve 0.006 ppm maximum in the recirculation loop. Similarly, the chloride and sulfate ions are maintained at low values in the feedwater to achieve a 0.015 ppm maximum for both species in the recirculation loop.

In order to meet the evaporator water chemistry requirements described above, requirements for condensate system demineralizer resin regeneration and/or replacement, continuous monitoring/recording and grap sampling of the Condensate Polishing effluent have been established as follows:

Max. Allowable Impurities	Design Limit Operation Above 5% Power	Monitoring	Grab Sample
Total Suspended Solids	16 ppb	None	Yes
Silica (SiO_2)	5 ppb	Continuous	Yes
Iron (Fe)	5 ppb	None	Yes
Copper (Cu)	<1.5 ppb	None	Yes
Sodium (Na)	1 ppb	Continuous	Yes
Chloride (Cl)	2.5 ppb	Continuous	Yes
Cation Conductivity at $77^{o}F$	0.2 mmho/cm	Continuous	Yes

Because of the conclusive test results mentioned above and because tubes will be volumetrically inspected and plugged based on actual wastage, quantitative analysis of wastage as a function of temperature or other variables has not been performed; nor will it be usd as a basis for design or operating specifications.

No specific protection is required for protecting Type 304 SS or 2-1/4 CR-1 Mo steels against intergranular attack, stress-corrosion or general corrosion, provided that specified sodium purity is maintained.

In water or steam, carbon steel and 2-1/4 Cr-1 Mo steel are susceptible to caustic gouging and possibly caustic stress corrosion cracking. Maintaining the feedwater and steam drum purity levels as stated below will prevent these forms of localized attack. For normal operation other than start-up conditions, the feedwater and steam drum purity will be specified as follows:

Feedwater Impurities		Feedwater	Steam Drum
Suspended Solids	PPM	--	0.1
Dissolved Oxygen	PPM	.005	-
Silica	PPM	--	0.1
Iron as Fe	PPM	.01	-
Copper as Cu	PPM	.002	-
Hydrazine	PPM	.005-.015	-
Chlorides	PPM	--	.015
Sodium	PPM	.001	.006
Sulfate	PPM	--	.015
pH @ 77°F		8.8-9.2	8.8-9.2
Conductivity (After Cation Removal) @ 77°F micro-mho/cm		0.2	1.0

Limited duration operation with impurity levels above specified limits is allowable for periods not to exceed 24 hours in special instances. These special instances are defired to include condensate polishing system perturbations, such as those immediately associated with a termination of regeneration.

Corrosion impurities may enter the feedwater system through condenser leakage and/or poor makeup water. To guard against damage from such sources, the feedwater and steam drum water are maintained at levels within stated limits by full flow demineralization and continuous steam drum drainflow (blowdown) at a nominal rate of 10% of full power steam flow (See Section 10.4.7).

To determine the feedwater quality continuous analysers with alarms are provided to sample conductivity, dissolved oxygen, hydrazine, turbidity, pH, sodium, chloride and silica. Continuous samples of steam drum downcomer water and periodic samples of drum drain (blowdown) water are monitored for conductivity, sodium, silica and pH. The downcomer continuous sample monitors are also alarmed if out of specification conditions occur. The condenser hot-well is monitored for conductivity and sodium ions to guard against condenser leakage. The demineralizer effluent is guarded against impurities break-through by in-line measurements of silica, conductivity and sodium. Finally, the feedwater train is monitored downstream of the deaerator for pH and oxygen content to prevent potential corrosion of this portion of the steam system. An alarm is coupled with the most critical in-line measurements to signal departure from specified levels.

Question CS 281.12 (9.8)

The Impurity Monitoring and Analysis System consists of primary heat transport system (PHTS), ex-vessel storage tank (EVST), intermediate heat transport system (IHTS) sodium characterization, fuel handling cell (FHC), and IHTS Cover Gas Sampling Systems. Provide the chemical and radiochemical limits for the sodium and the cover gas analyses. In addition, describe the method, sampling procedures, and frequency of sampling.

Response

The chemical and radiochemical limits for sodium and cover gas systems are listed in Table CS 281.12-1. The methods for sodium sampling are discussed in PSAR Section 9.8. For the required sampling frequency for sodium, see the response to Questions CS 281.5 and CS 281.10. Cover gas sampling methods and frequencies have not yet been finalized. The procedures to be utilized for sampling will be developed prior to plant operation and will be described in the FSAR.

TABLE CS 281.12-1

Sodium

Impurity	Primary	Intermediate	EVST
O_2	2 ppm	2 ppm	5 ppm
H_2	0.2 ppm	0.2 ppm	0.4 ppm
Pu, U	10 ppb	10 ppb	10 ppb
C	0.7 ppm	0.7 ppm	0.7 ppm
Other	10 ppm	10 ppm	10 ppm

Cover Gas

Impurity	Primary	Intermediate	EVST	FHC
H_2	10 ppm	8 ppm	8 ppm	-
O_2	2 ppm	10 ppm	10 ppm	25-75 ppm
N_2	100 ppm	15 ppm	-	6% vol
CH_4	4 ppm	25 ppm	25 ppm	-
CO		-	-	-
H_2O	2 ppm	8 ppm	8 ppm	25-75 ppm

Radiochemical limits for the reactor cover gas during continued reactor operation are provided in PSAR Section 16.3.2.3. Cover gas radiochemical limits during refueling operations are provided for the reactor in PSAR Section 16.3.10.3.2 and for the EVST and FHC in PSAR Section 16.3.10.3.1.