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Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION

Reference: Letter, P. S. Check to J. R. Longenecker, "CRBRP Request for Additional Information," dated February 26, March 11, and April 9, 1982

This letter formally responds to your request for additional information contained in the referenced letter.

Enclosed are responses to Questions CS 210.12, CS 270.12, CS 220.34, and CS 421.04. These responses will also be incorporated into the PSAR Amendment 69, scheduled for submittal in July.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of the
Clinch River Breeder Reactor
Plant Project
Office of Nuclear Energy

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Question QCS 210.12

Describe the CRBR prototype scale model testing program and its correlation with the CRBR preoperational vibration testing program for assessing CRBR internals vibration motion. In addition, describe the acceptance criteria for internals vibration tests.

Response

Section 3.9.1.3 has been completely revised to incorporate the current CRBRP vibration test program for reactor internals. The test program including scale model testing is described in Section 3.9.1.3.1.4. The correlations developed from the scale models to predict CRBRP vibration are also given in Section 3.9.1.3.1.4 under the paragraph titled "Model to Prototype Scaling Ratios." Utilization of the scale model test data in the CRBRP vibration testing and considerations for developing the acceptance criteria are discussed in Section 3.9.1.3.5.

3.9.1.3 Dynamic System Analysis Methods for Reactor Internals

3.9.1.3.1 Summary of Overall Flow Induced Vibration Assessment Program for CRBRP

A comprehensive vibration assessment program to assure that excessive flow induced vibratory motion of reactor internals does not exist, is part of the CRBRP program. The program, schematically illustrated in Figure 3.9-4, assures the structural integrity of reactor internals. The program accomplishes this objective by means of the following key actions:

1. Design to avoid flow induced vibration
2. Evaluate susceptibility of each component to flow induced vibration
3. Analyze to provide a model evaluation of each component
4. Test both scale model and full system simulations
5. Factor FFTF operational measurements and experience, as well as relation between FFTF scale model testing and in-reactor data, into CRBRP evaluations.
6. Monitor upper internals structure vibrational response during pre-operational testing to supplement and confirm vibration results from the model test program.

The program integrates the input from these sources to assure that excessive flow induced vibratory motion of reactor internals does not occur.

The overall program has been formulated to meet the intent of NRC Regulatory Guide 1.20. The program relies on highly instrumented model tests to assure that flow induced vibration problems will not exist in the reactor. This approach is an extension of the vibration assessment program developed for FFTF. It utilizes the experience gained from FFTF model testing, where models were used to assess the potential for flow induced vibration, with the

objective of developing correlations which will predict CRBRP prototypic motions using scale model test data. Where scale model tests may not be adequate, such as for long, slender members with flow thru clearances, full scale tests are performed.

Details of the program's key features are provided in the following sections.

3.9.1.3.1.1 Design

CRBRP components are designed to avoid excessive flow induced vibration. Heavy structures are utilized, where possible, to withstand vibratory forces. Generally, structures and components are designed to have natural frequencies as high as practical to avoid coincidence with known forcing mechanisms. Mechanical stresses that are caused by flow induced vibrations that are unavoidable must meet the fatigue requirements of the ASME Boiler and Pressure Vessel, Section III, and Code Case N-47. When combined with the effects of other operating conditions, maximum displacements will be limited, as required, to assure proper functioning of the components.

3.9.1.3.1.2 Component Evaluation

A general survey of all CRBRP reactor internals was made. In this survey the following considerations were given to each component:

- Type of possible component excitation, vortex shedding, turbulence, pump pulsation, jet impingement, jet reaction, gap modulation or other fluid elastic mechanisms.

- Importance Index - based on the relative importance of the component and type of excitation.

- Investigative priority - priority is based on degree of concern and whether the component can be evaluated by state-of-the-art methods.

Table 3.9-9 summarizes this survey and provides the methods selected to address the question of flow induced vibration.

3.9.1.3.1.3 Analysis

Existing analysis methods (ANSYS, etc) are used to compute the natural frequencies and mode shapes. Component response is calculated, assuming a fluid forcing mechanism, and the resulting stress or component motion evaluated. Presently, one of the most effective means of designing against the occurrence of severe flow induced vibration problems is to separate structural natural frequencies from expected excitation frequencies; generally, natural frequencies of interfacing components are separated.

By utilizing existing flow induced vibration analysis and information, the free and forced response of the CRBRP structures are investigated. The prediction of flow induced vibration response is limited to the state-of-the-art in describing the fluid to forcing function associated with fluid excitation mechanisms and damping. Many excitation mechanisms could be responsible for CRBRP component vibration. Some examples are cross-flow vortex shedding, parallel flow boundary layers, fluid borne noise, wakes from adjacent components and fluid elastic coupling between adjacent components. Although all of these mechanisms are of concern, generally characterization is possible for only the first two mechanisms when each is assumed to act alone. Even then the forcing functions are based on assumption. Therefore, experimental evaluation of components using scale models is relied upon. The results from the experimental portion of the program will be factored into the analysis and final assessment of the component acceptability regarding flow induced vibration.

3.9.1.3.1.4 Testing

For many regions of the reactor, the complex interaction of the flow field with the reactor structures precludes the utilization of analytical methods for evaluation of component flow induced vibration. For this reason a comprehensive series of system tests, both scale model and full scale, have been planned for CRBRP. These tests reproduce flow fields closely matching

those which will exist in the actual reactor. The planning of the experimental programs was carried out in consultation with recognized personnel in the field of flow induced vibration from the Argonne National Laboratory, Hanford Engineering Development Laboratory and the Westinghouse Research Laboratories. In addition, during performance of the program, the test methods, similarity requirements, test results, etc. are continually reviewed by these personnel as well as by personnel from other organizations knowledgeable in flow induced vibration. Every effort is made to ensure that the results obtained from the model studies are correct and directly applicable to CRBRP.

The major models used in the experimental portion of the CRBRP comprehensive vibration assessment program consist of the following:

a. Inlet Plenum Feature Model

b. Bench Test Models

c. Selected Full Scale Models

d. Integral Reactor Feature Model (IRFM)

A description of these various models follows. Since the IRFM test program is the most significant part of the total test program this test is discussed in detail in later sections. Results from the various model tests are provided in Section 3.9.1.3.2.

a. <u>Inlet Plenum Feature Model (IPFM)</u>

The IPFM models all hydrodynamically wetted surfaces in a full 360 degree (0.248 scale) sector of the inlet plenum and lower internal components. This test, initiated in 1974, was completed in 1976. Seven of the 61 lower inlet modules (LIM) were dynamically simulated with accelerometers mounted on four of these seven modules to monitor their vibrational response. The LIMs which were dynamically simulated (i.e., equal Strouhal number) are those located nearest to the reactor vessel inlet nozzles, because they are subjected to the highest cross-flow velocity and thus most susceptible to vibration. Results from the IPFM test series are summarized in Section 3.9.1.3.2.

b. Bench Tests

Based on component evaluations, an experimental program to evaluate the flow induced vibration characteristics of selected regions of the Upper Internals Structure was established. Testing was performed by Argonne National Laboratory in a 1/3 scale facility which structurally and hydraulically simulated the instrumentation post and a chimney with a lower shroud tube.

Results of these bench tests are given in Section 3.9.1.3.2.

c. Full Scale Model Testing

It was anticipated that during the performance of the experimental program, the need for additional, selected full scale model tests might be identified. For example, it became necessary to conduct selected full scale tests on portions of the UIS where similarity criteria or Reynolds number similarity could not be satisfied by scale model testing. These full scale tests include the gap modulation flow induced vibration test of the control rod upper to lower shroud tube slip joint, the UIS chimney vibro impact test, and the IVTM port plug gap modulation flow induced vibration test.

In addition to these full scale tests, a special test is being conducted on the IVTM Port Plug in conjunction with the gap modulation flow induced vibration test. During the test it is planned to assess the susceptibility of the IVTM Port Plug to self-excited vibration. If the levels do not increase, it will be a positive indication that the potential for self-excited vibration does not exist.

Finally, full scale testing of a prototype Primary Control Rod System - a Primary Control Rod Drive Mechanism, Primary Control Rod Driveline, and Primary Control Assembly in sodium at 400^{o}F - has been performed. The Primary Control Drive (PCRD) was instrumented with three pairs of

accelermometers to measure its flow induced vibration response. Emphasis on the PCRD was at the dashpot/cup area. The PCA was instrumented with four accelerometers which were positioned at two elevations on the outer duct of the assembly. The in sodium tests were conducted in an aligned and misaligned configuration, the test variables for a given configuration being rod withdrawal height and prototypic sodium flow through the PCA and the shroud tube enveloping the driveline. Results of this testing are given in Section 3.9.1.3.2.

d. Integral Reactor Flow Model (IRFM)

The outlet plenum region of the reactor contains components which could potentially have flow induced vibration problems. Therefore planning for a 1/4 scale flow and vibration system model of this region was initiated in 1975. The objectives of the IRFM test program are as follows:

1. Measurement of the velocity pattern in the outlet plenum and in the vicinity of major outlet plenum structures to provide input for the prediction of flow induced vibration.

2. Evaluation of flow induced vibration characteristics of selected outlet plenum structures.

3. Evaluation of flow induced vibration characteristics of the primary and secondary control rod drivelines.

It was decided that to satisfy these objectives the model test program would be conducted in two phases since the design of all outlet plenum components was not finalized. For the initial phase of IRFM testing, Phase I, approximately 100 accelerometers were located on critical regions of the outlet plenum components as shown in Figure 3.9-5. The location of this instrumentation is based on modal analyses of these components. Table 3.9-10 summaries the instrumentation planned for the Phase II vibration tests in IRFM.

In order to satisfy the objectives of the test program the following tests were performed during Phase I:

1. Velocity Test (outlet plenum for both three-loop and two-loop operation, inter-chimney region, chimney, core assemblies - upper internals).

2. Flow Induced Vibrations (preliminary evaluation of current designs).

A flow step summary of the flow induced vibration test is provided in Table 3.9-6. As the table shows, vibration data was obtained on the model for the full range of anticipated flow including the refueling mode. In addition to the flow induced vibration data, experimental modal analysis (shaker tests) of the UIS and outlet plenum internals was conducted for two conditions: (1) the model dry in air and (2) no flow but model vessel full of water. For the Phase II testing, the flow induced vibration test sequence will be repeated. The shaker tests will also be repeated for the final designs.

Model Description

The IRFM is an approximately 0.248 scale model of the wetted surfaces in the CRBRP reactor outlet plenum. It is a 360° model, including three outlet nozzles as in the CRBRP reactor, and is capable of two or three loop operation. The vessel support is not prototypic, but is provides sufficient frequency separation between support, vessel and components.

Both hydraulic and vibrational testing are performed in IRFM. However, since flow induced vibrations are of concern for only certain outlet plenum structures, only those structures will reflect hydraulic and structural simulation. Table 3.9-8 is provided to define the components which require one or both types of simulation.

In the design of the IRFM, provision was made for simulating the height variation of the reactor assembly exit nozzles. Provisions were also made

for simulating misalignment between the UIS and the core. Two height and two radial alignment variations have been tested. Also, the refueling position mode, in which the upper internals structure is raised 9-1/2" and decoupled from the lower internals, has been tested.

For the Phase I series of tests the following components were modeled in IFRM:

- o Upper Internal . Structure
- o Instrumentation Post
- o Instrumentation Conduit
- o Liquid Level Monitor
- o IVTM Port Plug
- o Ex-vessel Transfer Machine Guide Tube
- o Upper Control Rod Shroud Tube
- o Control Rod Driveline

The second phase of IRFM testing structurally simulates the final, released designs of outlet plenum hardware. Phase II model simulations provide data on prototypically modeled components prior to reactor operation and will be used to predict prototypic vibration of outlet plenum components. To date the IRFM has been modified to include the thermal liner, the outlet nozzle liner, a redesigned Liquid Level Monitor Port Plug (LLMPP), and a dynamic simulation of the vortex suppressor plate. For the final series of IRFM Phase II tests, the simulation of the UIS is being modified to include dynamic simulation of the following: two chimneys, the thermal liner plates in the mixing chamber, and the UIS jacking mechanism method of UIS column support.

In addition, it is planned to locate four biaxial accelerometers on the UIS model to monitor its gross motion during a simulation of the reactor's preoperational test program. The accelerometers are located on the model in the identical positions of the plant accelerometers. By correlating the data from these two sets of accelerometers, predictions on the motion of the prototype unit can be made from data obtained from the highly instrumented IRFM model.

Vibration Modeling Considerations

The similitude requirements for valid flow induced vibration model testing are generally well known (see Refs. 1 and 2).

Similitude ratios which are pertinent to flow induced vibration modeling are summarized below. Subscripts m and p refer to model and prototype, respectively.

$(Re)_m/(Re)_p$, where Re is the Reynolds number (DU/ν),

S_m/S_p, where S is the Strouhal number (fD/U),

$(\rho_s/\rho)_m/(\rho_s/\rho)_p$, where ρ_s is the structure material density and ρ is the fluid density, and

(δ_m/δ_p), where δ is the log decrement.

For a given structure in incompressible flow, and ignoring (a) externally driven boundary motions, (b) externally generated forces (such as pump pulsations), (c) surface wave effects (Froude number), and (d) surface tension effects (Weber number) the dependent parameters of interest are a function of four main independent parameters.

$$y/D = \phi\left[\frac{f_n D}{U}, \frac{\rho_s}{\rho}, \frac{DU}{\nu}, \delta\right]$$

where y = vibration amplitude

D = characteristic length (diameter)

f_n = natural frequency

U = flow velocity

ρ_s = density of structure

ρ = density of fluid

ν = kinematic viscosity

δ = mechanical damping, log decrement

An equivalent set of parameters is:

$$y/D = \phi \left[\frac{K}{U^2 D}, \frac{\rho_s}{\rho}, \frac{DU}{\nu}, \frac{C}{\rho U D^2} \right]$$

where K = spring rate, lb/in
C = mechanical damping constant, lb-sec/in

For geometrically similar structures, the scaling law between model and prototype using subscripts m and p respectively, is

$$\left(\frac{y}{D}\right)_p = \left(\frac{y}{D}\right)_m$$

This relationship will hold when the four similitude parameters are the same in model and prototype.

Table 3.9-7 shows the flow vibration parameters and associated ratios for the model and prototype.

In the following paragraphs, the similitude ratios are discussed as they pertain to the IRFM simulation. The model fluid will be water at 100°F and the prototype fluid is sodium at 950°F. The structural material is stainless steel.

Reduced Frequency and Density

Using stainless steel and a model temperature of 100°F will result in component stiffnesses which are approximately 24% greater than properly scaled values.

Because the stiffness is about 24% higher, the vibration natural frequencies will be about 11.4% higher as a result. To obtain the same values of reduced frequencies $f_n D/U$ in the model as in the prototype, the model flow velocities must be 11.4% higher than the prototype flow velocities. The structural mass will be properly scaled, but the fluid densities will not.

Under these conditions, the fluid elastic parameter $K/\rho U^2D$ will be .83 times the prototype value, which is conservative.

Reynolds Number

A modeling scale D_{model}/D prototype = 0.248 has been chosen. A flow velocity ratio U_{model}/U prototype = 1.1 was chosen, corresponding to an operation condition giving the same reduced frequency of structural vibration, f_nD/U, in model and prototype. Under these conditions the ratio of Reynolds numbers (Re - Du/ν) will be:

$Re_{model}/Re_{prototype}$ =0.1

Reynolds number will not be duplicated in the model. This is not considered to be significant except possibly in a limited flow/vibration regime. This is further discussed below.

Circular cylinders in steady cross flow exhibit regular vortex shedding with well defined shedding frequency in the Reynolds number redime $80 \leq Re \leq 3.5 \times 10^5$ (subcritical) and $Re \geq 3.5 \times 10^6$ (transcritical), where $Re = DU/\nu$ is based on the cylinder diameter D and cross flow velocity U. For $500 \leq Re \leq 3.5 \times 10^5$ the shedding frequency f_s is given with fairly good accuracy by $f_s = .2$ U/D Hz*, whereas for $Re \geq 3.5 \times 10^6$, $f_s = .27$ U/D. In the subcritical and transcritical regimes, coincidence between vortex shedding frequency f_s and structural vibration natural frequency f_n, can give rise to large alternating forces and large lateral vibration amplitudes. In the supercritical regime, $3.5 \times 10^5 \leq Re \leq 3.5 \times 10^6$, any shedding is irregular and the resulting vibration excitation forces are random. Therefore in the supercritical region there is no possibility of driving frequency coincidence with the structural vibration natural frequency.

On account of the foregoing considerations, it is important to identify the Reynolds number regimes of cross flow past cylindrical structural components of CRBRP prototype and flow vibration models.

*In the formulas for shedding frequency, the units of U and D are such that U/D has units of sec^{-1}.

Values of Reynolds numbers in the model and in the prototype, determined for the aforementioned assumed conditions are given in Figure 3.9-3 as functions of the product (prototype cross flow velocity) x (prototype cylindrical-component diameter). Also indicated in Figure 3.9-3 are the limits defining the supercritical regime of random excitation. It can be seen that the model will be conservative for subcritical flows but non-conservative for trans-critical flows. That is, there is a DU regime (11 in-ft/sec $\leq DU \leq$ 107 in-ft/sec) where vortex shedding frequency coincidence that might take place in the model would not take place in the prototype; and there is another DU regime (116 in-ft/sec $\leq DU \leq$ 1200 in-ft/sec) where vortex shedding frequency coincidence that might take place in the prototype would not occur in the model. CRBRP model vibration test results must be viewed with this point in mind. The mismatch in Reynolds numbers emphasizes the need for analytically examining upper internals components for susceptibility to vortex shedding excitation.

Density Ratios

The value of density ratios, $(\rho s/\rho)_m/(\rho s/\rho)_p$, from Table 3.9-7 is 0.851. Thus, the effect of the virtual mass of the fluid is somewhat greater in the model, yielding lower effective model natural frequencies. The fluid excitation of the structure by the fluid is more effective in the model, yielding somewhat greater amplitudes. Both effects are small and considered conservative.

Vibration Displacements

With respect to the modeling based on the requirements that the ratio of model-to-prototype Strauhal number be unity, the previously cited regimes of DU will establish the scalability of model results. In those regimes where the model is conservative, the effect of density ratio damping should be further conservatism. Then the ratio $(y/D)_m = (y/D)_p$ is considered conservative. The model results obtained in the non-conservative regime and the regime wherein $S_m/S_p \neq 1$ are not directly scalable to the prototype, and the results must be further analyzed based upon the test circumstances to establish applicability.

Model to Prototype Scaling Ratios

Based upon the values of Table 1 and the geometric scaling ratio of 0.248, the following are model-to-prototype ratios of measured parameters:

f_m/f_p = 4.432 (frequency)

Δ_m/Δ_p = 0.248 (displacement)

F_m/F_p = 0.076 (force)

$\ddot{x}_m/\ddot{x}_p$ = 4.871 (acceleration)

To aid concentration of the testing on areas of significance, guidelines have been established for vibration levels requiring detailed measurements and assessment. If the vibration measurement for a component show either acceleration levels greater than 0.3g's, occurrance of impacting or a flow rate dependent resonance, the component data is assessed for potential design impact with additional measurements performed if necessary to obtain conclusive data. If neither of these effects occur, the component does not require further vibration assessment.

3.9.1.3.2 Test Results

A. IPFM

Data from the IPFM tests show the maximum prototypic vibrational amplitude of a LIM is 0.2 mils. This value is very low and is structurally acceptable. The tests have also shown that dynamic coupling between the Core Support Structure and the LIMs is neglible.

B. Bench Model Tests

Test results from the 1/3 ANL scale model of an instrumentation post indicate vibrational response is extremely small, less than 0.4 mils in bending. Testing of an outer chimney (chimney located above a blanket region) showed small vibration amplitudes and only slight rattling between the chimney and its support. Testing of a central chimney with a shroud tube (chimney located above an active region of the core) indicated similar results.

C. Full Scale Model Testing

- o The upper to lower shroud tube slip joint test have been successfully completed with no excitation mechanism discovered for simulated prototypic leakage conditions.
- o The UIS chimney vibro-impact test has been completed and the data is being evaluated. Preliminary evaluation of test data indicates that no gross motion of the chimney occurs at flow rates up to 95% of design flow rate. At 95% of design flow rate, signals from displacement transducers indicated fluctuations. It does not appear however that impacting is occurring at the 95% flow condition. Additional data evaluation is being performed and the model will be visually inspected for signs of impacting at disassembly.

Primary Control Rod System Tests

- o Low energy level flow induced vibration occurred across a broad frequency spectrum.

- o Acceleration levels increased according to the square of flow velocity without indicating any resonant peaks or instabilities.
- o Vibration behavior of the system was not significantly affected by either rod withdrawal height or misalignment.
- o Low level flow induced impacting did occur in both the dashpot and control assembly areas. The acceleration magnitude and rate of impacting also increased approximately as the square of the flow rate.

D. IRFM Phast I

Test results of the Phase I tests can be summarized as follows:

- o The measured responses of instrumented components were generally smal and proportional to the flow rate over the flow regime tested.
- o There were no observed unstable dependencies upon flow rate nor abnormally high fluid excited forced components.
- o There were no observed occurrences of vortex shedding synchronous with component resonant frequency over the full flow range tested.
- o The vibrational characteristics of the UIS and instrumented outlet plenum components were essentially independent of core configuration, loop mode operation, and UIS/core alignment.
- o Impacting was observed on the following components, with the location of impacting coinciding with assembly gaps incorporated in the prototype design and scaled in IRFM;

<u>UIS Keys/Core Former Ring</u>

At 110% flow a maximum impacting of approximately 2.1 g's peak-to-peak was measured on an accelerometer mounted on the model core former ring radially outward from the keyways. The cyclic rate of this impact was less than 1 cycle/sec. The corresponding g value in the plant is 0.43 g's.

UIS Upper Shroud Tube/Lower Shroud Tube

At 110% flow a maximum impacting of up to 3.5 g's p-p at a cyclic rate of 4 to 5 impacts/sec was detected. These measurements were recorded from a shroud tube containing a control rod driveline. Another shroud tube without a CRDL exhibited impacting level of 3 g's p-p with a cyclic rate of 2 impacts/sec. These g levels when scaled to the reactor are 0.71 and 0.62 respectively. Based on these results, a full scale model test of the upper to lower shroud tube joint was performed for the final design configuration. As noted in C above, no vibrational problems were found.

Chimney/Spider/Lower Shroud Tube

This region was not directly instrumented for Phase I but responses on neighboring accelerometers indicate impacting was probable. The results of the full scale chimney test are described in C above.

UIS Column/Closure Head

Indications of impacting were detected with maximum levels of response less than 1 g p-p and a cyclic rate less than 1 impact/sec. The corresponding g value for the plant is 0.21 g's.

Control Rod Driveline

The control rod driveline response was acceptable at 100% and higher fluid velocities. Impacting at the dashpot/pision interface was infrequent.

IRFM Phase II

Results from the completed portions of the Phase II testing are as follows:

- The measures response of the Phase II components was generally small.

- o The termal liner, outlet nozzle liner, and the Liquid Level Monitor Port Plug measured responses exhibited no unstable, flow rate dependency nor excessively high fluid excited response over the entire range of flow rates tested. Suppressor plate measured response was generally consistent and small over the range of test flow rates. The measured displacement amplitudes were less than 0.5 mils rms (model scale) at the maximum observed response levels.

- o The gross motion of the UIS during simulated refueling conditions (UIS raised to remove keys from the core former structure) was generally of low amplitude and erratic. The dominant measured lateral frequency was 12 Hz with calculated displacements on the order of 0.1 to 0.2 mils rms (model). Infrequent large amplitude responses were recorded. The maximum peak model displacement at the lower end of the UIS during these large amplitude responses was 16 mils. The motion was very sporadic and random occurring once every two to five seconds on the average.

3.9.1.3.3 Application of FFTF Experience to CRBRP

The CRBRP reactor intervals vibration program is similar to the FFTF vibration program and was formulated to maximize use of FFTF experience. Both programs utilize a combination of analysis, scale model tests, feature tests and selected in-reactor vibration monitors. The FFTF Hydraulic Core Mockup (HCM), a 0.285 scale model, was designed to simulate the vibrational and hydraulic characteristics of the reactor system just as IRFM does for the CRBR. Vibration measurements were obtained in the HCM and found to result in acceptable vibration levels.

FFTF in-reactor vibration monitoring of selected components was performed during pre-operational acceptance testing for confirmation of the scale model test conclusions. Pre-operational, non-nuclear, in-reactor vibration tests included accelerometer instrumentation in the Instrument Tree Guide Tubes (IGTs) and the Vibration Open Test Assembly (VOTA). Upon completion of the non-nuclear acceptance tests, IGT accelerometers were replaced with normal plant instrumentation for nuclear operation while the VOTA instrumentation was retained. Although the HCM and FFTF tests do not have identical instrumentation locations to permit direct one-to-one comparison, the FFTF data permits conclusions on the following points: 1) Overall vibration conclusions from both FFTF and HCM; and 2) Comparisons of HCM scale model test results with FFTF measurements for similar, although not identical, locations.

The overall conclusions from the HCM and FFTF tests are:

- o Overall test results from HCM and FFTF were similar and in fair agreement with respect to frequency and rms for similar instrumented components.
- o No gross vibrational problems such as significant impacting were observed in either HCM or FFTF.
- o The measured responses were stable and a function of increasing flow rate.
- o There is no incidence of syncronous vortex shedding at component natural frequencies.

FFTF in-reactor vibration measurements were obtained from two Instrument Tree Instrument Guide Tubes (IGTs) representing the shortest and longest IGTs. Accelerometers were mounted on a flow and temperature removable instrumentation assembly inserted in the IGTs. The HCM IGT accelerometers were mounted externally on the IGTs. However, the HCM IGTs were not prototypic of the final FFTF design. Figure 3.9-7 compares the first mode rms displacements for the two shortest HCM IGTs with the shortest FFTF IGT. The HCM rms displacement prediction compares well with the FFTF results and are conservative. The HCM frequency prediction is quite good and consistent with the slight length differences. For the longer HCM IGTs and the longest FFTF IGT, the frequency and flow dependence are similar in trend with the HCM rms displacement predictions being non-conservatively smaller (approx. 7 mils max.) than FFTF (approx. 30 mils max.). The non-prototypicality of the HCM IGTs is expected to be more influential for the long IGTs and could be the cause of the non-conservative HCM results in this case.

The FFTF Vibration Open Test Assembly (VOTA) was designed primarily as a vibration sensor and is a 40 foot assembly spring loaded into a core assembly receptacle. The HCM Closed Loop In-Reactor Assembly (CLIRA) is similar to the FFTF VOTA. The HCM CLIRA accelerometer measurements were scaled to FFTF and modified for differences in accelerometer locations based on analytical mode shapes for fully constrained and partially constrained support conditions. The isothermal VOTA tests showed a first mode response of 4 Hz indicating partial constraint. Figure 3.9-8 compares the FFTF VOTA isothermal results with the projected HCM results for partial constraint. The agreement is quite good. During FFTF power ascent, the VOTA first mode shifts from 4 to 10 Hz due to increased core clamping. The VOTA results for this condition compared to the HCM CLIRA projections for full constraint show that the HCM results are conservative (approx. 3 mils rms displacement compared to 1 mil for VOTA).

The FFTF vibration comparisons confirm that the HCM scaling parameters (similar to IRFM vibration movel parameters given in Section 3.9.1.3.1.4) generally result in the expected conservative predictions for the reactor internals. The agreement between FFTF and HCM results is quite good for both overall vibration conclusions and for comparable vibration magnitudes.

3.9.1.3.4 Overall Test Conclusions

The good agreement between FFTF and HCM test results support the validity of well designed scale modal testing for predicting reactor internals vibration response. Acceptable comparisons between FFTF and HCM were obtained for vibration displacements, flow dependence and frequencies with both tests showing low vibration levels for FFTF.

The CRBR reactor internals scale model and full scale test results complated to date show no indication of any significant vibration problems. No unstable dependence upon flow rate nor highly fluid excited forced components have been observed. Vibrationally induced impacting at gaps has been shown to result in low g level impacts judged to be below material damage thresholds. Although some testing remains to be complated for final design configurations, the differences between the preliminary and final test configurations are not likely to cause a major change in the vibrational responses for the outlet plenum components. Where the acceptability of scale model testing might be questionable such as for flow thru small clearances (UIS shroud tube gap, IVTM port plug, primary control rod driveline), full scale tests were performed to further assess vibration potentials.

In general, the test configurations were constructed to maximize the gaps between components. In the CRBR, misalignments from normal manufacturing tolerances (UIS chimneys, shroud tubes and IVTM port plug for example) and thermal gradients (UIS mixing chamber liner plates and the horzontal baffle for example) tend to close gaps and minimize vibration potentials.

Both the FFTF and CRBR pre-reactor operation test programs have shown no significant vibrational problems for the reactor internals. The CRBR plan for in-reactor confirmation of the test results is similar to the FFTF in-reactor program of using in-reactor vibration monitoring of selected components to confirm the pre-reactor operation test and analysis results.

The FFTF VOTA accelerometers in the core regions exhibited excessively large reciprocal of frequency (1/f) noise characteristics starting at very low power levels (<1%). Accelerometers above the core had minor low frequency degradation as a result of elevated temperature exposure (<1000°F). The mechanisms for these effects are not fully understood at this time. FFTF utilized piezoelectric, lithium niobate crystal accelerometers for the in-reactor tests which are the same accelerometers planned for CRBR in-reactor testing. Based on these FFTF results, the accuracy of CRBR measurements under reactor power conditions cannot be determined.

3.9.1.3.5 CRBRP In-Reactor Vibration Monitoring for Reactor Internals

The results of the test and analysis program completed to date have shown the adequacy of the reactor internals for flow induced vibration considerations. To supplement this comprehensive vibration assessment program and to provide confirmation of scale model test results, vibration monitoring of the UIS is planned for CRBRP. While the test results indicate acceptable vibration levels for all reactor internals, the principal area of concern from an inadequacy of the pre-reactor operation test program would be gross vibratory motion of the UIS. Conceptually, gross vibratory motion of the UIS could cause impacting of the keys to the core former structure keyways with resulting fatigue failures at the keys or other gap elements. Based on the vibration test results, other areas of the reactor internals are even less sensitive to vibration than the gross UIS motion.

Four biaxial accelerometers will be located on the UIS as shown in Figure 3.9-6. As can be seen from the accelerometers' orientation, lateral, vertical and torional motions of the UIS or impacting at the UIS keys can be monitored with the accelerometers.

Accelerometer measurements will be recorded and evaluated at varying flow rates during pre-operational testing. Acceptable accelerometer performance at temperatures above about 600°F and at reactor power conditions cannot be assured. If the accelerometers remain operational with acceptable accuracy, data will be obtained and evaluated during the initial power ascent.

Phase II of the IRFM scale model tests has four accelerometers in the UIS at the same location as the CRBRP UIS. These IRFM acceleromaters will be correlated with other extensive instrumentation on the IRFM UIS to provide a reference data base for interpretation of the CRBRP UIS accelerometer measurements. Identification of the peaks and frequencies in the IRFM tests will provide guidance for assessment of the CRBRP measurements.

Vibration data from the final series of IRFM tests will be evaluated to demonstrate component acceptability and to develop acceptance criteria for the CRBRP internals vibration tests. Components with significant vibration in the IRFM tests will be evaluated to show stress, fatigue and wear acceptability including allowances for test uncertainties. These analyses will be used to develop acceptance criteria for the plant tests to assure that UIS structural and functional requirements are satisfied. The UIS measurements will be compared to the acceptance criteria and to the IRFM data. Major differences between the IRFM and plant data will be assessed to assure that the differences are not an indication of an unexpected, significant vibration problem. Acceptance criteria for the UIS accelerometer measurements will be provided in the FSAR.

3.9.1.3.6 Application of Regulatory Guide 1.20

The following comments address application of Regulatory guide 1.20 to CRBRP.

Regulatory Position C.1: Classification of Reactor Internals

The classification provided in this regulatory position will be followed by the CRBRP Project to categorize the reactor internals. It is anticipated that most of the CRBRP internals will be in the category of "Prototype."

Regulatory Position C.2: Vibration Assessment Program for Prototype Internals

Item C.2.1 (Vibration Analysis Program), is applicable to CRBRP. The scope of the analysis for the CRBRP is described in Section 3.9.1.3.1 and Chapter 4 of the PSAR.

Item C.2.2 (Vibration Measurement Program) is applicable to CRBRP. The test operating conditions will be provided when the preoperational and initial startup test details are established.

The provisions as given in Item C.2.3 (Inspection Program) were developed primarily for LWRs. Due to the limitation of the state-of-the-art of inspectability of LMFBRs, the requirements set forth in Item C.2.3 are considered largely not applicable to the CRBRP.

The intent of Item C.2.4 (Documentation of Results) will be met, as applicable, in the context of the program described above.

Item C.2.5 (Schedule) is provided herein. Sub-Item 1 requesting classification of the reactor internals is fulfilled by the above "Prototype" designation. Sub-Item 2 for a commitment on the scope of the vibration assessment program is fulfilled by Section 3.9.1.3. Sub-Items 3 (description of the vibration measurement phase) and 4 (summary of the vibration analysis program) will be provided in the FSAR as a further development of the information given in Section 3.9.1.3. Sub-Item 5 requests preliminary and final reports within 60 and 180 days, respectively, of the completion of CRBRP vibration testing. CRBRP reports will be provided consistent with these requested periods.

Regulatory Position C.3: Vibration Assessment Program for Non-Prototype Reactor Internals

[illegible], no CRBRP reactor internals are expected to be classified as "Non-Prototype." Therefore, no assessment with regard to the applicability of this regulatory position is attempted at this time.

TABLE 3.9-6

FLOW STEP SUMMARY

Test or Test Condition	Flow Rate of Fluid Condition								
	Shaker Test		Water (% of CRBRP Rated Velocity)+						
Three Loop Operation	Still Air	Water	10	33	66	100	110	120*	**
Velocity Test									X
Vibrations									
a) Operating Mode	X	X	X	X	X	X	X	X	X
b) Refueling Mode***	X	X	X						
Two Loop Operation									
Velocity Test						X			X
Flow Vibrations							X	X	X

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* Operating at this flow depends on proximity to vibration limit

** To be determined, vibration limit or test facility limit

\+ For two loop operation, rated flow is approximately 2/3 three loop operation value.

*** Approximately 11 in. gap between top of reactor assemblies and the upper internals structure for the refueling mode.

NOTE: All components will be monitored and only those shown most responsive will be recorded and analyzed. 19

NOT CHANGED

TABLE 3.9-7

Flow Vibration Parameters - CRBRP Model and Prototype

Conditions:

Same structural material in model and prototype (stainless steel)
Model Fluid = water at 100°F
Prototype fluid = liquid sodium at 995°F

Parameter	Model	Prototype	Ratio
Young's Modules of Elastic, E (psi)	$28.2x10^6$	$22.7x10^6$	$\frac{E_m}{E_p} = 1.242$
Fluid Density, ρ (lbs/cu in.)	0.0359	0.0297	$\frac{\rho_m}{\rho_p} = 1.206$
Material Density, ρs (lbs/cu in.)	0.290	0.282	$\frac{(\rho s)_m}{(\rho s)_p} = 1.028$
ρs/ρ	8.078	9.495	$\frac{(\rho s/\rho)_m}{(\rho s/\rho)_p} = 0.851$
Poisson's Ratio, μ	0.282	0.301	$\frac{\mu_m}{\mu_p} = 0.937$
Kinematic Viscosity, ν	$9.0x10^{-6}$	$3.05x10^{-6}$	$\frac{\nu_m}{\nu_p} = 2.951$

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NOT CHANGED

TABLE 3.9-8

Outlet Plenum Components Modeled
In IRFM Test

Component	Hydraulic Simulation	Structural Simulation	FIV Instrumentation
UIS Structure			
Support Columns	Yes	Yes	Yes
Chimneys	Yes	2 (Ph. II)	Yes
Support Plates	Yes	Yes	Yes
Instrument Post	Yes	3	Yes
UIS Lower Shield Plate	Yes	Yes	Yes
Control Rod System			
Control Rod Driveline	1	1	Yes
Shroud	Yes	3 (Ph.I/1 (Ph.II)	Yes
Guide Tube	Yes	3	Yes
Shroud/Guide Gap	2	2	
Instrumentation			
Conduit	2	2	Yes
LLM	Yes	1	Yes
Vessel			
Thermal Liner	Yes	Yes	Yes
Thermal Baffle	Yes	No	No
In-Vessel Storage	1	No	No
Suppressor Plate	Yes	Yes	Yes
Outlet Nozzle Liner	Yes	Yes	Yes
Reactor			
Fuel, Blanket and Radial Shield Assembly	Yes*	Yes**	Yes**
Control Assembly	Yes*	Yes	No
Core Barrel	Yes	No	No
Core Barrel UIS Keys	Yes	Yes	Yes
Miscellaneous			
EVTM Guide Tube	Yes	Yes	Yes
IVTM Port Plug	Yes	Yes	Yes

* Flow Distribution and Assembly Exit Geometry need only be simulated for fuel, blanket and control assemblies. For radial shield assemblies, only outlet flow distribution will be simulated.

**Testing Involving Core Dynamic Simulation may be performed at ANL rather than in the IRFM.

Table 3.9-9
FLOW VIBRATION OF CRBRP INTERNALS
General Survey

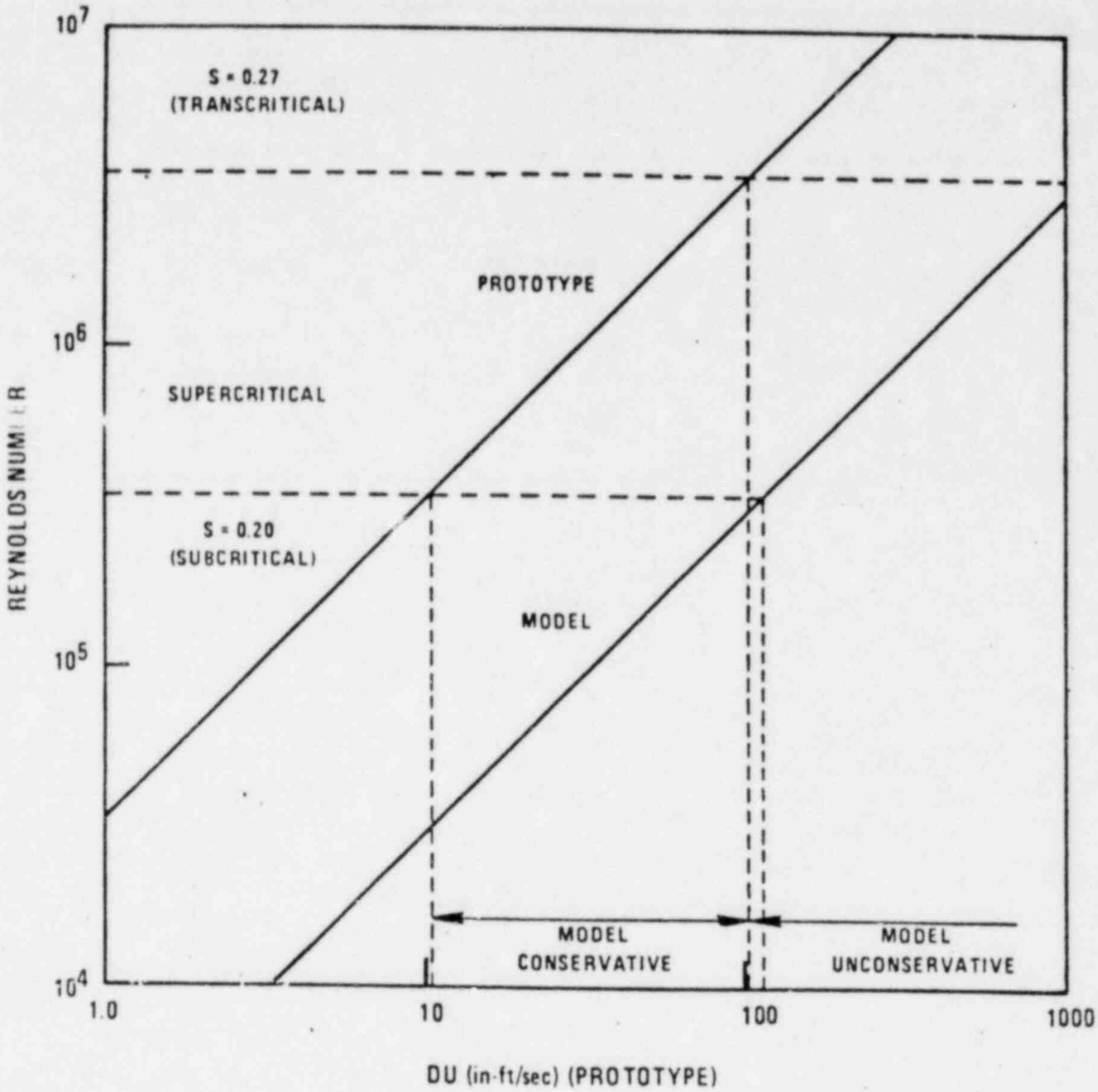
Reactor Internals Components	Type of Excitation	IMPORTANCE INDEX High	Medium	Low	METHODS SELECTED for RESOLUTION Analysis	IRFM Testing	Full Scale Test	Bench Test
1. Entire Upper Internals Structure (cross motion)	Vortex Shedding			X	X	X		
	Jet Impingement		X			X		
	Jet Reaction		X			X		
	Turbulence			X		X		
	Pump Pulsations			X	X	X		
2. UIS Guide Tubes for Control Rods	Vortex Shedding	X			X	X		
	Turbulence	X		X	X	X		
	Pump Pulsations			X	X	X		
	Gap Modulation	X					X	
3. UIS Chimneys	Vortex Shedding	X			X	X		
	Turbulence			X	X	X		
	Pump Pulsations			X	X	X	X	X
	Jet Reaction			X			X	X
4. UIS Support Columns	Vortex Shedding	X			X	X		
	Turbulence			X	X	X		
	Pump Pulsations			X	X	X		
5. UIS Instrumentation Leads	Vortex Shedding	X			X	X		
	Turbulence	X			X	X		
	Pump Pulsations			X	X	X		
6. UIS Support Plates and Plate Liner	Vortex Shedding			X		X		
	Turbulence			X		X		
	Pump Pulsations			X		X		
7. UIS Instrumentation Posts	Vortex Shedding	X				X		X
	Turbulence		X			X		X
	Jet Impingement		X			X		X
8. UIS Cylindrical Liners Sleeve	Vortex Shedding			X				X
	Turbulence			X				X
	"Base Motion"			X				X
9. UIS IVTM Port Plug	Vortex Shedding	X			X	X		
	Turbulence			X	X	X		
	Pump Pulsations			X	X	X		
	Gap Modulation	X			X		X	
10. Control Rod Drivelines	Vortex Shedding	X			X	X	X	
	Turbulence	X			X	X		
	Gap Modulation						X	
11. Core Support Structure Core Barrel	Turbulence			X	X			
	Pump Pulsations			X	X			
12. Core Support Structure Module Liners	Vortex Shedding			X	X	X		
	Turbulence			X	X	X		
	Pump Pulsation			X	X	X		
13. Horizontal Baffle	Vortex Shedding			X	X			
	Turbulence			X	X			
	Pump Pulsation			X	X			
14. EVTM Guide Tube	Vortex Shedding	X			X	X		
	Turbulence			X	X			
	Pump Pulsation			X	X			
15. Suppressor Plate Supporting Hangers	Vortex Shedding	X			X	X		
	Turbulence			X	X	X		
	Pump Pulsation			X	X	X		
16. Suppressor Plate Segments	Vortex Shedding			X	X	X		
	Turbulence			X	X	X		
	Pump Pulsation			X	X	X		
17. Liquid Level Monitor (LLM) Tubes and Support Members	Vortex Shedding	X			X	X		
	Turbulence			X		X		
	Pump Pulsation			X				
18. Vessel Thermal Liner	Turbulence			X	X	X		
	Pump Pulsation			X	X	X		
19. Outlet Nozzle Liner	Turbulence			X	X	X		
	Pump Pulsation			X	X	X		
	Jet Reaction			X	X	X		

Table 3.9-10

IRFM PHASE 2E VIBRATION TEST INSTRUMENTATION SUMMARY

Component	Concern	Data Required	Instrumentation Quantity: Displacement	Instrumentation Quantity: Accelerometers	Instrumentation Quantity: Strain	Instrumentation Location: Displacement	Instrumentation Location: Accelerometers	Instrumentation Location: Strain	Comments
1) UIS Keys (0° & 120° & 240°)	Impact - Wear Stress	Displacement - Acceleration - Force	6	6	8	R-T	R-T	See Comments	Strain gages appropriately located and calibrated - need 4/Key
2) IVTM Port Plug	Impact - Wear	Displacement - Acceleration - Force	3	11 -		R-T and 0°, 180°, 300°	R-T 0°, 90°		Two strings of four accelerometers along length
3) UIS Chimney	Exitation due UIS motion, Structural	Acceleration, Displacement	8	8		R-T Top, Bottom Support	R-T Midspan, Top		Two structurally simulated chimneys
4) UIS Column	Mode shape	Strain		8					Two strings of four accelerometers along length See Note 1
5) UIS Column to Bearing	Impact and Bearing Load	Displacement Force	2		4 (Force Trans)				Force transducers at top of bearing race on one column) See Note 1
6) UIS Column to Structure	Impact - Force	Force			3 (Force Trans)				Force transducers on structure perpendicular to pin axis See Note 1
7) Column to UIS CRDM Plug	Vibration	Acceleration			2		R-T		
8) UIS Upper/ Lower Shroud	Impact - Vibration	Acceleration - Displacement - Mode Shape	2	10		R-T	R-T		Two strings of four accelerometers along length 0°, 90° shroud to include instrumented driveline and dash pot to monitor their response to shroud exitation
9) Instrument Post	Vibration	Acceleration		4			R-T Tip		
10) UIS Top Plate	Vibration	Acceleration		1			V Center		
11) Bottom Plate	Vibration	Acceleration		1			V Center		
12) UIS Thermal Liners	Vibration	Acceleration							A study is required to establish the modeling for this component
13) Prototype Instrumentation		Acceleration		4					Biaxial accelerometers located same as shown on UIS drawing.

Note 1: This instrumentation to be aligned with same axis.



NOT CHANGED

Figure 3.9-3. Reynolds Number for Prototype

9763-1

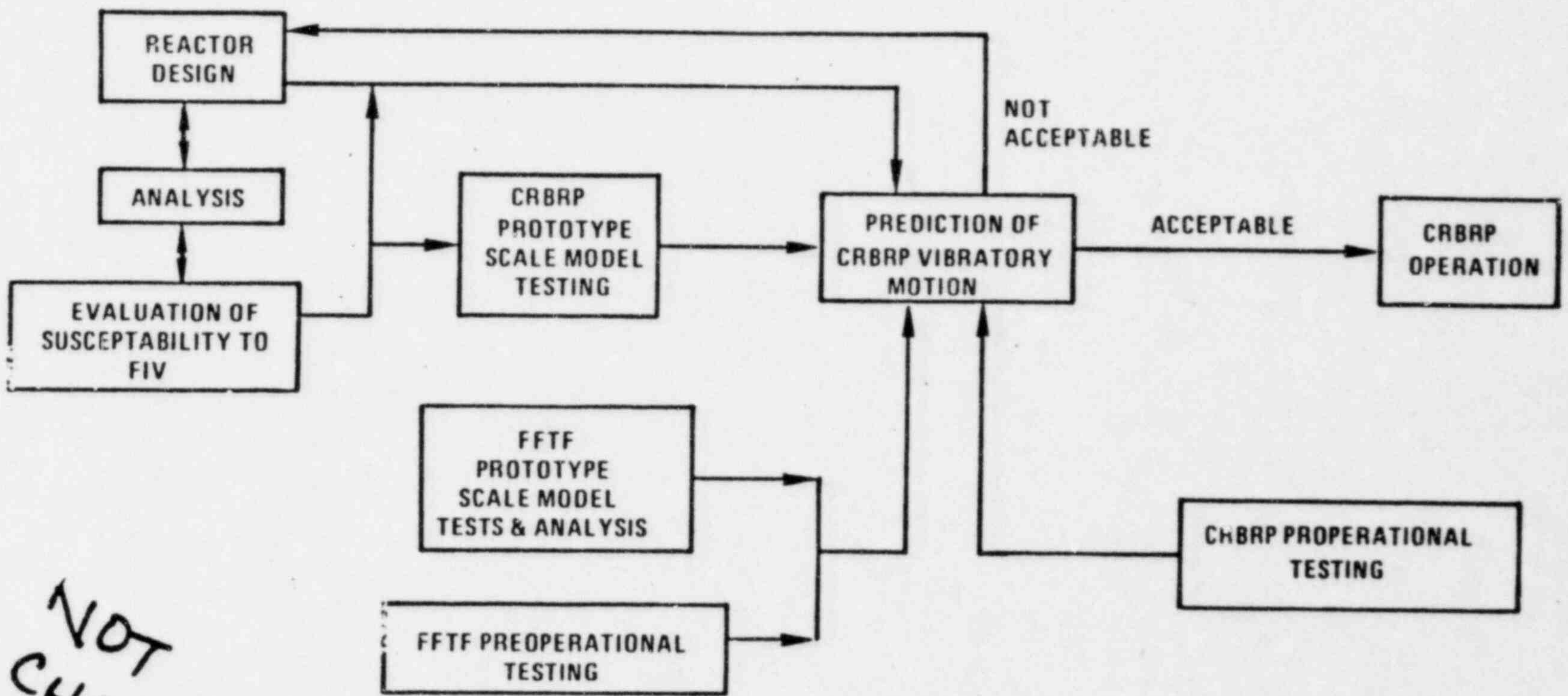


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Figure 3.9-4 Reactor Internals Flow Induced Vibration Program

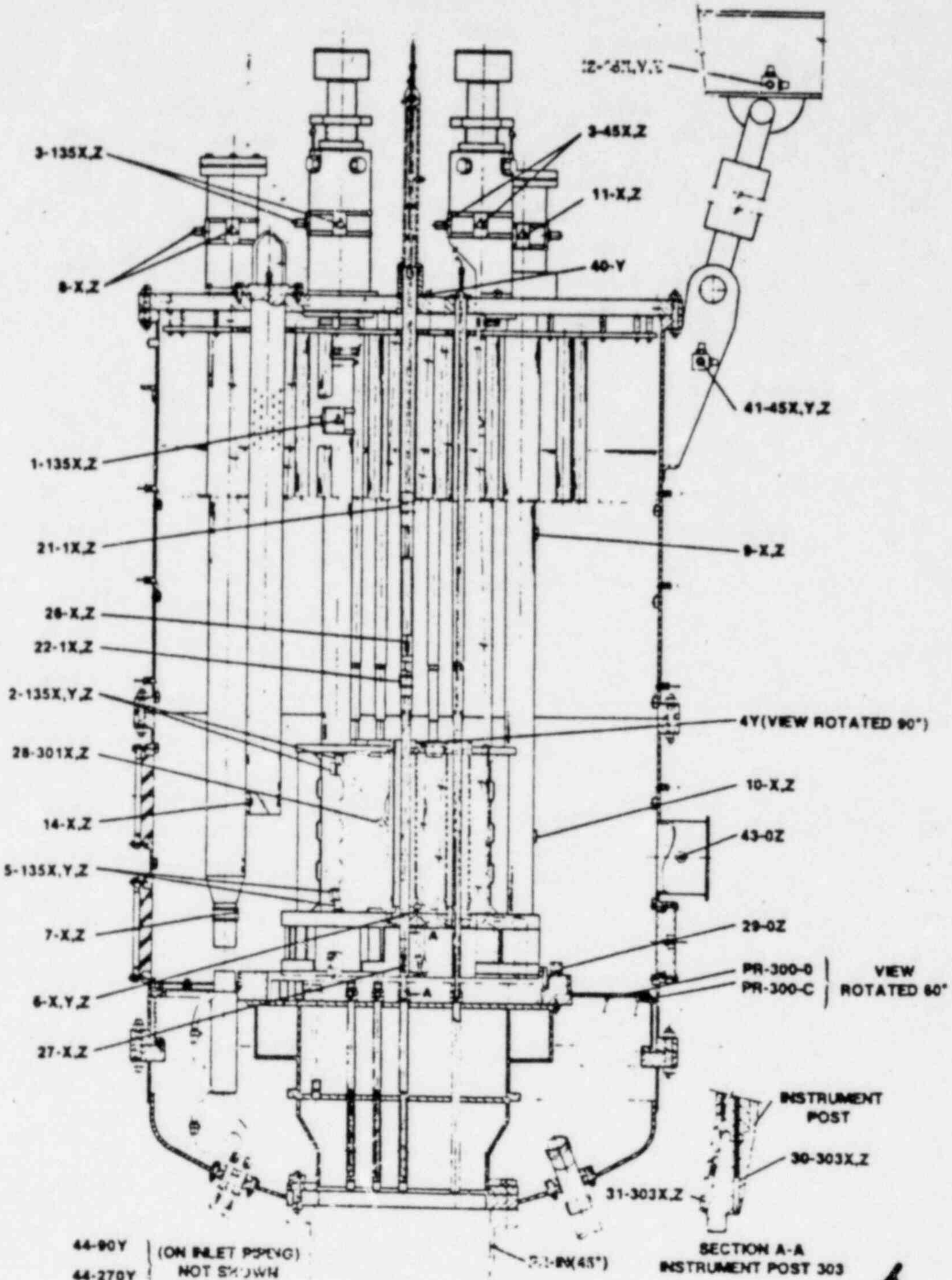

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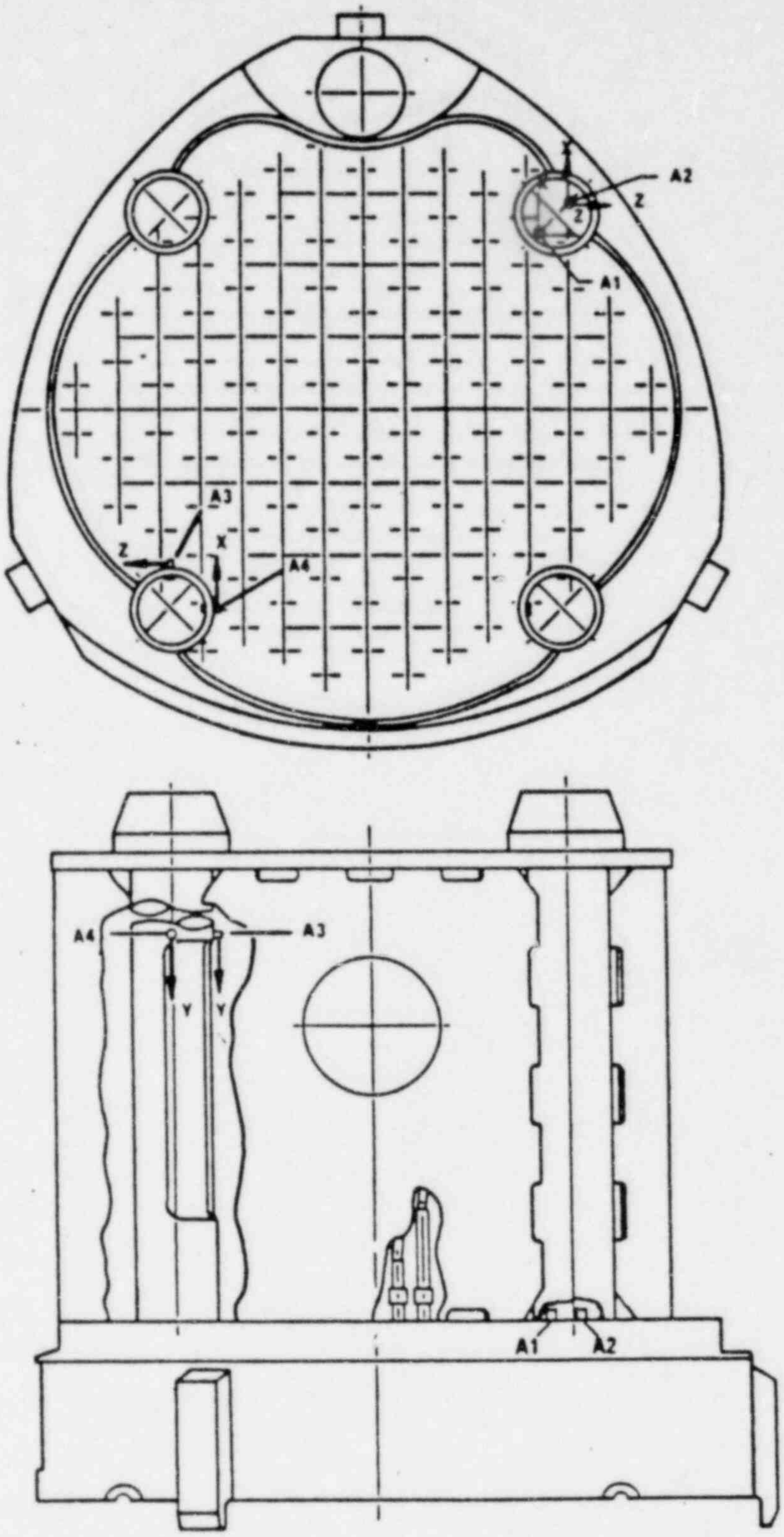
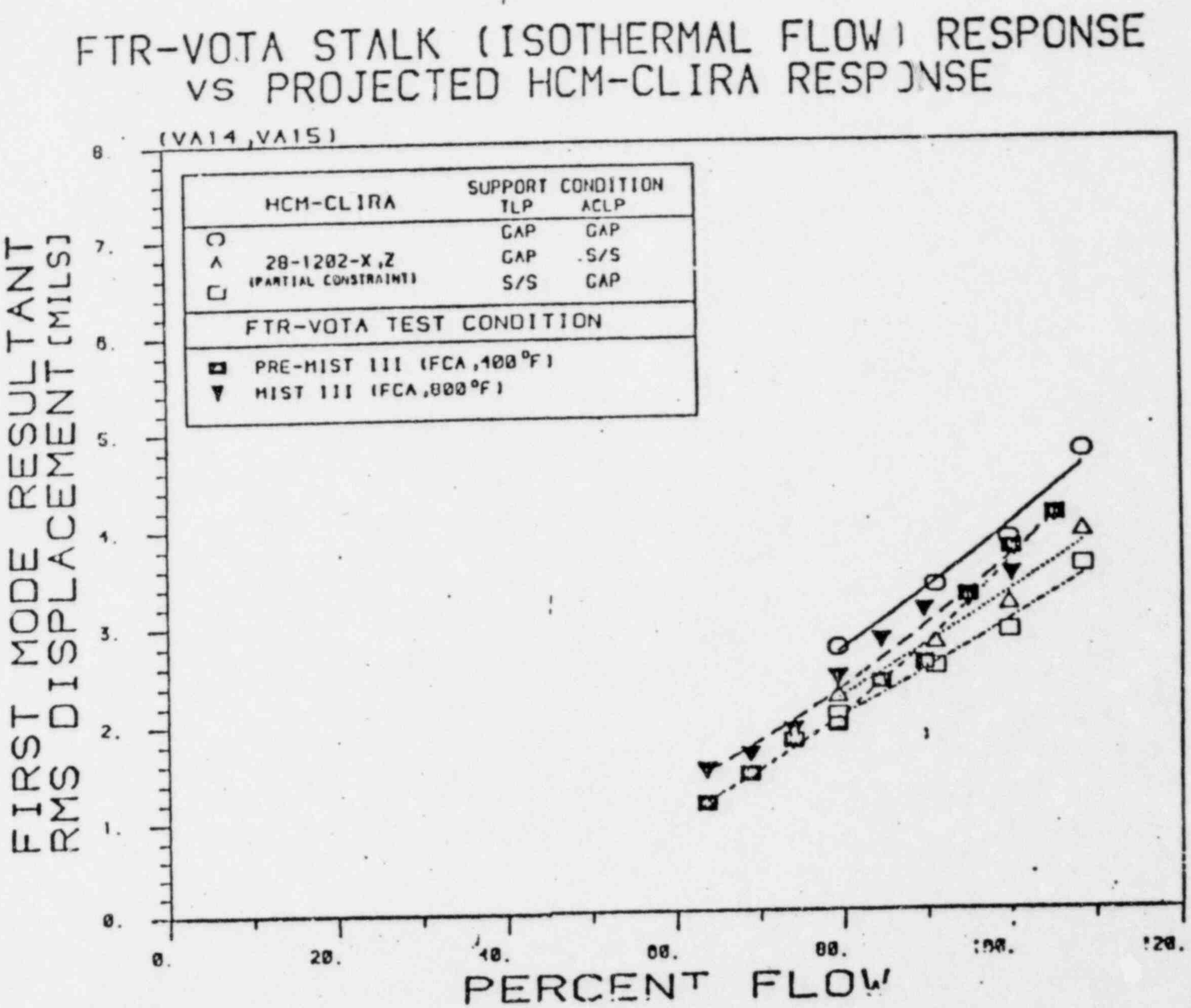
Figure 2.2.5 ... Reactor ... Model and Instrumentation



NOT CHANGED

9082-1 Figure 3.9-6 Location of Accelerometers on Upper Internals Structure



3.4-8
FIGURE 26. FTR-VOTA Stalk (Isothermal Flow) Response vs. Projected HCM-CLIRA Response with Partial Constraint

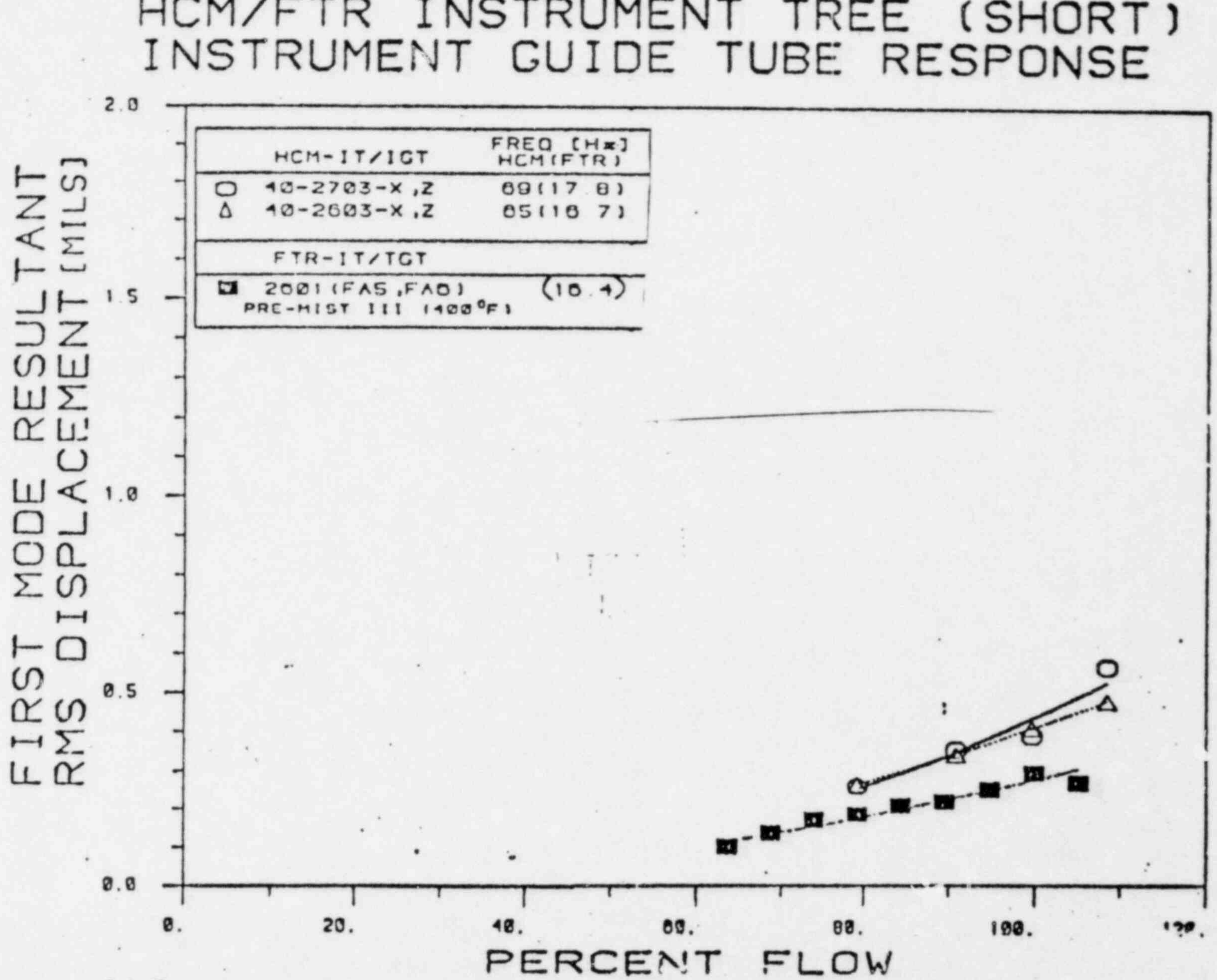


FIGURE 3.9-7. HCM/FTR Instrument Tree (Short) Instrument Guide Tube Response

Question CS270.12

Provide an amendment that discusses the procedure that will be used to combine the equipment loading from earthquake, generally in the 0 to 33 Hz range, and from the dynamic shock wave loading resulting from Na-H_2O reaction with a frequency range of 20 to 100 Hz. Equipment to be qualified by tests should be specifically addressed.

Response

The combination of SSE and OBE loads with events is described in PSAR Section 3.7A which meet the principle and intent of Reg. Guide 1.48.

CRBRP has identified a postulated sodium/water reaction as an emergency event. The Sodium Water Reaction Pressure Relief System (SRWPRS) as been designed for the emergency event. The design utilizes passive devices (i.e. rupture discs, piping and tanks). The rupture discs are designed to remain intact for seismic events, but will rupture under the pressure loading (shock wave) associated with a large sodium/water reaction. Analysis and tests are being performed to assure the discs will rupture only when required. (See Section 5.5, Ref. 26 and 27).

The design of the steam generator system is such that the components will accommodate the earthquake loadings at the end of life without a failure that would lead to a sodium/water reaction. Therefore, there is no technical reason to associate the two separate events with the design base.

Question CS 220.34 (a)

There are a number of misprints, unclear statements and typographical errors which need your correction and/or clarification.

a) Figure 3.8-9 should label elements discussed in the text (3.8.3.1.1). Details of concrete reinforcing are needed to evaluate the support ledge.

Response

Revised PSAR Figure 3.8-9 labels elements discussed in the text (3.8.3.1.1). An additional sketch, showing the reinforcing bars in the reactor cavity at the ledge area has been included.

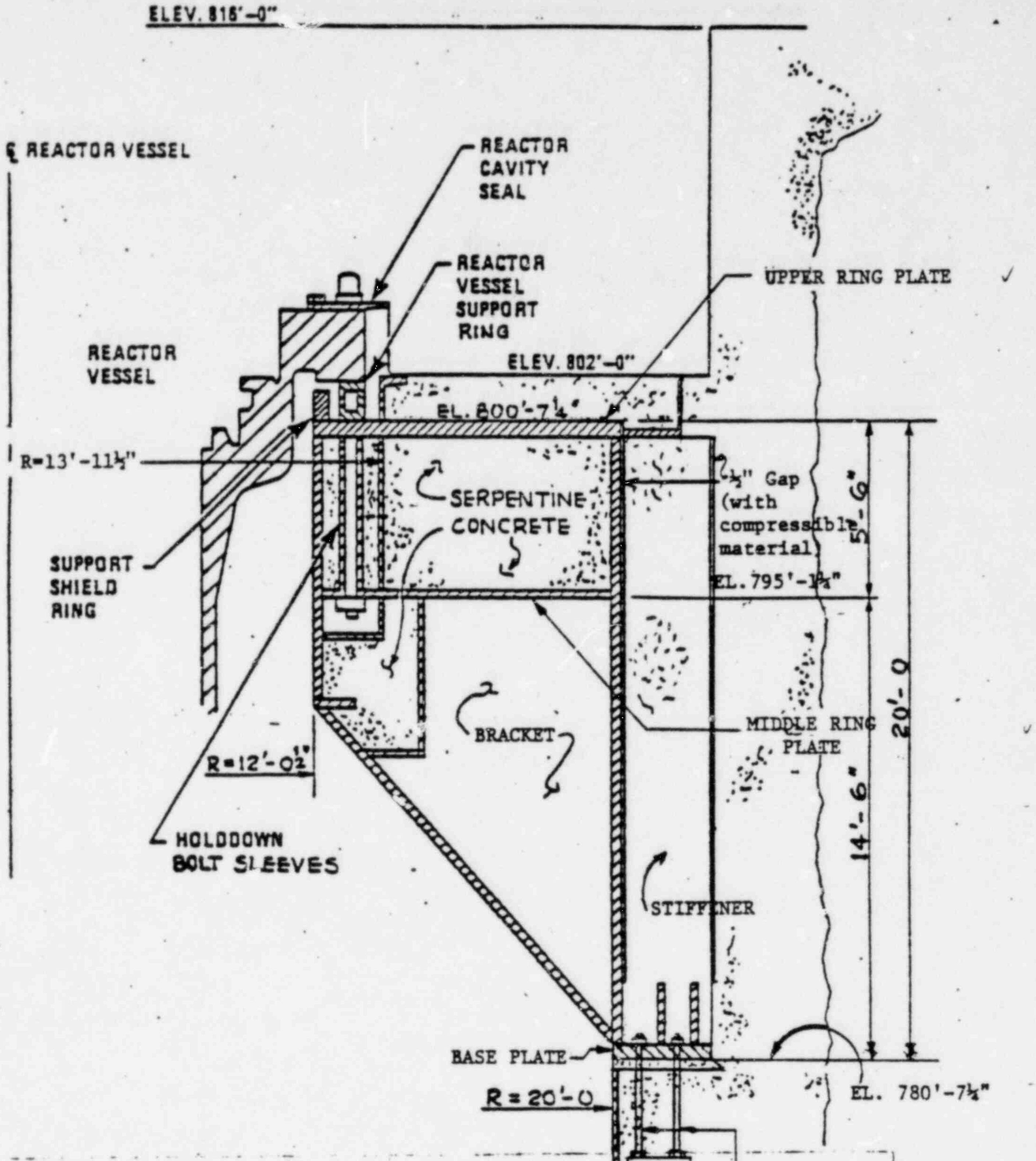


FIGURE 3.8-9 TYPICAL SECTION THROUGH REACTOR VESSEL LEDGE SUPPORT

Sheet 1 of 2

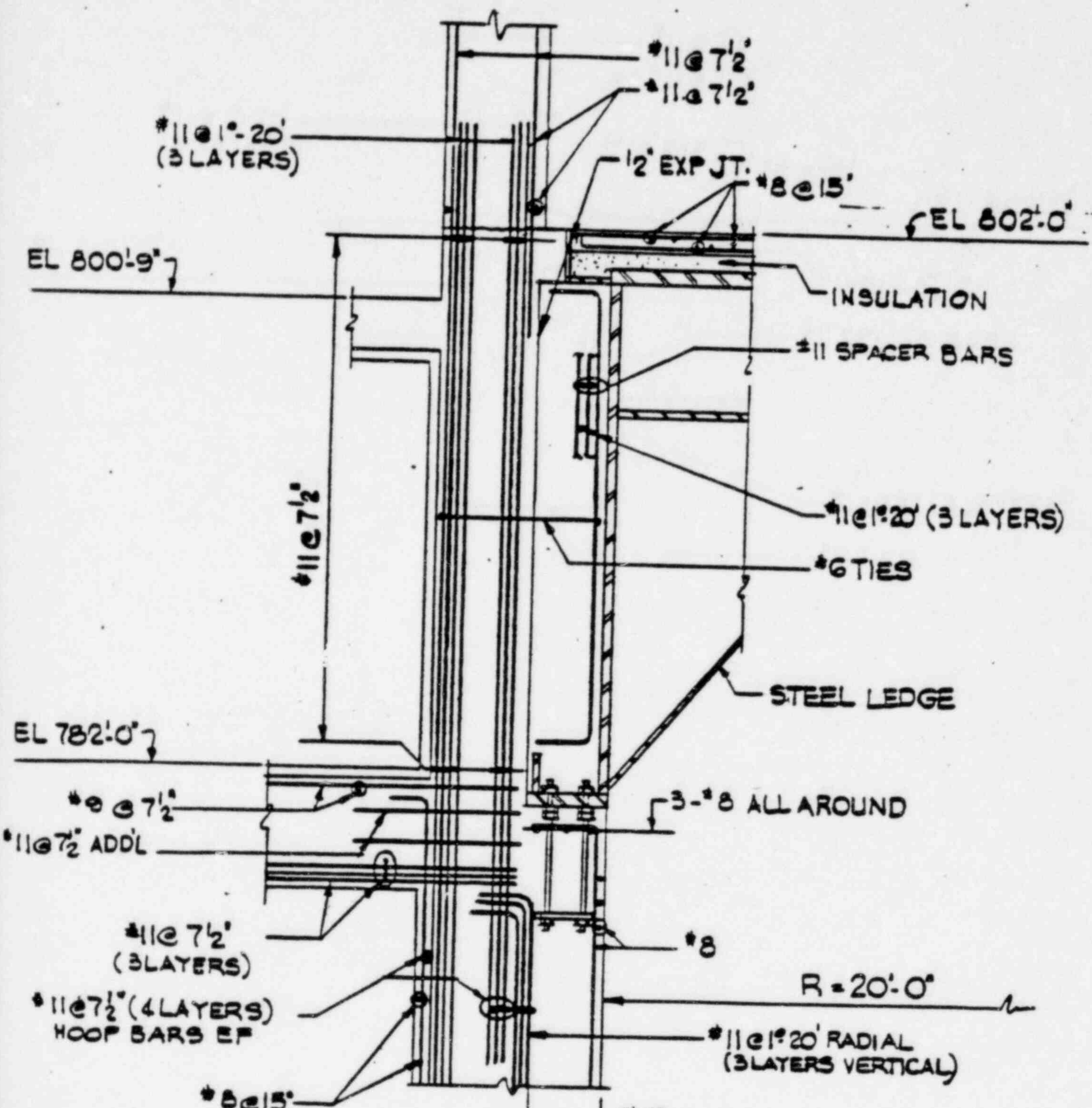


FIGURE 3.8-9 TYPICAL SECTION THROUGH REACTOR VESSEL LEDGE SUPPORT
Sheet 2 of 2

Question CS 220.34 (b)

Is the load of 50,000 kips mentioned in Section 3.8.3.3.4 to be evenly distributed around the support ledge?

Response

The load of 50,000 kips on the Reactor Support Ledge, stated in Section 3.8.3.3.4, has been revised. The revised load is given in terms of time-histories for vertical and toroidal loads. These time-histories are shown in CRBRP-3 ,Section 5.2 (Reference 10a, PSAR Section 1.6).

These loads are axisymmetrical and therefore are applied uniformly around the support ledge.

The 33% increase in allowable stresses for steel due to seismic or wind loadings will not be used.

U - For concrete structures, U is the section strength required to resist design loads and based on methods described in ACI 318-77.

3.8.3.3.2 Internal Structure as Containment

No portion of the internal structure provides a direct containment function. The embedded part of the steel containment is designed such that it can withstand the design pressure without the assistance of the concrete walls.

3.8.3.3.3 Creep, Shrinkage and Local Stresses

No prestressed concrete design is considered for the design of the facility. Therefore, creep and shrinkage loads will be only considered to the extent they are provided in the reference concrete codes or as may be warranted by prudent design approach. The loads transferred from the support structure that generally influence local areas will be checked to insure that the local stresses are within acceptable limits to preclude impairment of the structural function.

3.8.3.3.4 Loads Due to Structural Margin Beyond Design Base (SMBDB)

Refer to CRBRP-3 Section 5.2 (Reference 10a of PSAR Section 1.6).

3.8.3.3.5 Sodium Fire Load

See Table 3.8-2 for the accident pressures and temperature loads.

3.8.3.3.6 Hot Sodium Spill Effect

The portions of the reactor cavity and cells, where exposure to radioactive hot sodium is a design basis accident, are provided with carbon steel liners designed to survive a sodium spill (see Section 3A.8). The liners will not compromise gas tightness of the cell.

3.8.3.3.7 Accident Temperature Load

See Table 3.8-2 for design temperatures.

3.8.3.3.8 Negative Pressure on the Liners

Any negative pressure on the liner will be resisted by a grid of structural anchors embedded in the concrete.

Question CS 220.34 (c)

The text at the top of page 3.8-16 is generally confusing. In particular, the text seems to negate the need for having both cases 10 and 11. The PSAR should delineate how the appropriate dynamic load factor will be determined. Load combination 10 and 11 need more justification for not including T_a unless A is meant to include thermal effects. In that case, the definition of A in 3.8.3.3.1.4 needs to be changed.

Response

For load combinations 6, 7 and 8 in some cases dynamic analysis based on a force time-history input was performed. In other cases, dynamic load factors are calculated using the procedures described in Section 3.5.4.6 of the PSAR which are consistent with the requirements of SRP, Section 3.5.3.II.B.2.

Load combinations 10 and 11 were intended to cover the cases involving margin events beyond the design basis.

Load combination 11 will be deleted since it is already covered in the last paragraph of 3.8.3.3.10.1.B that states: "Both cases of L having its full value of being completely absent will be checked for." Also, the definition of A in 3.8.3.3.1.4 will be revised to state: "A.....force on the structures due to third level design margin requirement (SMBDB)."

Under SMBDB conditions the thermal conditions are represented by To (Ta = To).

The dynamic forces under SMBDB conditions (load "A") are defined as the time histories of the vertical and toroidal moments presented in CRBRP - 3. The effects on the reactor vessel support ledge have been calculated by dynamic analysis.

PSAR Sections 3.8.3.3.1.4 and 3.8.3.3.10.1 will be updated to include the above revised design information.

W_T - Loads generated by the Design Basis Tornado as specified in Section 3.3. They include loads due to the tornado wind pressure, loads due to the tornado-created differential pressures, and loads due to the tornado-generated missiles. (Tornado loads do not apply to internal structures.)

H - Hydrostatic loads due to maximum flood (as defined in Section 3.4).

3.8.3.3.1.4 Abnormal Loads

Abnormal loads are those loads generated by a postulated accident within a building and/or compartment thereof. Included in this category are the following:

P_a - Pressure equivalent static load within or across a compartment and/or building, generated by the postulated accident, and including an appropriate dynamic load factor to account for the dynamic nature of the load,

T_a - Thermal loads under thermal conditions generated by the postulated accident and including T_o.

R_a - Pipe reactions under thermal conditions generated by the postulated accident and including R_o.

A - Force on structure due to third level design margin requirement. (SMBDB)

Y_j - Jet impingement equivalent static load on a structure generated by the postulated accident, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_r - Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated accident, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_m - Missile impact equivalent static load on a structure generated by or during the postulated break, such as pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

In determining an appropriate equivalent static load for Y_r, Y_j and Y_m, elasto-plastic behavior may be assumed with appropriate ductility ratios and as long as excessive deflections will not result in loss of function of any safety-related system.

3.8.3.3.1.5 Other Definitions

S - For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

7) $U = D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ E$

8) $U = D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ E'$

9) $U = D' + L + T_o + R_o + H$ where D' = dead load without hydrostatic load due to normal groundwater

10) $U = D + L + T_o + A$

In combinations (6), (7) and (8), the maximum values of P_a, T_a, R_a, Y_r, Y_j and Y_m, including an appropriate dynamic load factor, will be used unless a time-history analysis is performed to justify otherwise. Combinations (5), (7) and (8) will be satisfied first without the tornado missile load in (5) and without Y_r, Y_j and Y_m in (7) and (8). When considering these loads, however, local section strength capacities may be exceeded under the effect of these concentrated loads, provided there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent will be checked for.

3.8.3.3.10.2 Loading Combinations for Steel Structures

Elastic working stress design methods, as specified in Part I of AISC Specification for the Design, Fabrication and Erection of Structure Steel for Buildings, will be used for design of all Category I steel structures under both Service Load and Factored Load conditions.

A. Load Combination for Service Load Conditions

For Service Loads including earthquake (OBE) and wind loads (if applicable), the following load combinations will be satisfied:

1) $S = D + L$
2) $S = D + L + E$
3) $S = D + L + W$

If thermal stresses due to T_o and R_o are present, the following combinations will also be satisfied:

1a) $1.5\ S = D + L + T_o + R_o$
2a) $1.5\ S = D + L + T_o + R_o + E$
3a) $1.5\ S = D + L + T_o + R_o + W$

Both cases of L having its full value or being completely absent will be checked for.

Question CS 220.34 (d)

In combinations (4) and (8) inclusive in Section 3.8.3.3.10.2.B, are thermal loads to be neglected when it can be shown that they are secondary and self-limiting in nature and or or where the material is ductile?

Response

The second paragraph of Section 3.8.3.3.10.2.B of the PSAR has been updated.

B. Load Combinations for Factored Load Conditions

For Factored Loads including earthquake (OBE or SSE), tornado (if applicable) and pipe break effects, etc., the following load combinations will be satisfied.

4) $1.6\ S = D + L + T_o + R_o + E'$

5) $1.6\ S = D + L + T_o + R_o + W_T$

6) $1.6\ S = D + L + T_a + R_a + P_a$

7) $1.6\ S^* = D + L + T_a + R_a + P_a + 1.0\ (Y_r + Y_j + Y_m) + E$

8) $1.7\ S^* = D + L + T_a + R_a + P_a + 1.0\ (Y_r + Y_j + Y_m) + E'$

9) $1.6\ S = D' + L + T_o + R_o + H$ where D' = dead load without hydrostatic load due to normal groundwater

10) $1.6\ S = D + L + T_o + A$

In combinations (4) to (8) inclusive, thermal loads may be neglected when it can be shown that they are secondary and self-limiting in nature.

In combinations (6), (7) and (8), the maximum values of P_a, T_a, R_a, $Y_r + Y_j + Y_m$, including an appropriate dynamic load factor, will be used unless a time-history analysis is performed to justify.

Combinations (5), (7) and (8) will be first satisfied without the tornado missile load in (5) and without Y_r, Y_j and Y_m in (7) and (8). When considering these loads, however, local section strengths may be exceeded under the effect of these concentrated loads, provided there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent will be checked for.

In loading combinations, no load factors of less than unity will be used in design or analysis.

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*For these two combinations, (7) and (8), in computing the required section strength, S, the plastic section modulus of steel shapes may be used.

Question CS 220.34(e)

In Section 3.8.3.4, on page 3.8-18, the figure numbers referenced are incorrect.

Response

The referenced figure numbers in PSAR Section 3.8.3.4, page 3.8-18, have been revised.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 General Analysis Procedures

Structural analysis for each cell (except the reactor cavity) will be performed by considering one-foot wide vertical and horizontal strips through the structure. These strips, in essence, constitute structural segments which allow analysis by conventional methods.

The analysis of each cell will be performed by choosing one or more sections of one-foot width in both vertical and horizontal directions. The number of strips taken depends upon the configuration of each cell. In the case of analyzing a PHTS Cell, one typical vertical cross section (Figure 3.8-5) and two horizontal sections (Figures 3.8-6 and 3.8-7) are chosen for an illustration.

A rigid frame method of analysis will be used for determining the moments and shears in all members of a frame for each of the following anticipated loadings:

a) Dead and Live Loads,

b) Seismic Conditions,

c) Pressure Loads, and

d) Thermal Loads.

Loading combinations, using the factored loads as indicated in Section 3.8.3.3.10.1 will be used to establish the most critical design stresses for each component of the frame.

Vertical and horizontal strips thus analyzed will provide the necessary steel reinforcement for the vertical and horizontal directions of the particular cell analyzed.

Question CS 220.34(f)

In Section 3.8.3.5.2, more description is needed of the "energy absorption check."

Response

Ductility ratio, μ, is a measure of the capacity of a structure to absorb energy in the plastic range. Energy absorption is satisfied when the calculated value of the required ductility ratio is less than the allowable ductility ratio for the material under a specific loading condition.

PSAR Section 3.8.3.5.2 will be updated to include the above design information.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Stress

Structures designed by the stress limitation methods will be considered acceptable, when design stresses for the most severe combination of loads are within the limits prescribed by the appropriate codes and standards noted in Section 3.8.3.2.

3.8.3.5.2 Strain

Since the design of the reinforced concrete structures will be governed by ACI-318, a strain limit of 0.003 used as a basis of this code will be inherently provided in the design. For the steel structures, a maximum stress limit of 0.9 Fy is established. Therefore, strains cannot exceed ninety percent of the yield strain. In specific instances, where plastic behavior of the steel will be a design basis, an energy absorption check will insure that the functional requirements of the structure are not impaired. Ductility ratio, μ, is a measure of the capacity of a structure to absorb energy in the plastic range. Energy absorption is satisfied when the calculated value of the required ductility ratio is less than the allowable ductility rario for the material under a specific loading condition. For discussion on liner strain, see Section 3.8.3.4.

3.8.3.5.3 Gross Deformation

The load combinations and stresses noted in Section 3.8.3.3 will insure that the deformation of a structure will be no greater than that ordinarily permitted for structures of this type. Reinforced concrete structures subject to loads combined with SSE will be nearly stressed to their ultimate capacity. However, several relieving features such as strength gain with age, relief of thermal stresses due to cracking, etc., will preclude any excessive deformation. A design check will be performed to insure that the deformation resulting from SSE and other loads will in no way impair proper functioning of the critical systems or components.

3.8.3.5.4 Factor of Safety

The factor of safety for the working stress design will be in accordance with the limits noted in Section 3.8.3.3. For the ultimate load design, load factors will be in accordance with the combinations noted in Section 3.8.3.3.

3.8.3.5.5 Shear Response

The shear response will be established upon the basis of the classical relationship between the Young's modulus and the shear modulus. The assumed value of the Poisson's ratio and concrete moduli will be checked against the properties of concrete as determined through tests of design mixes to be used in the plant construction.

Question CS 220.34(g)

In Section 3.8.3.7, if internal structures that are designed to hold more than 10 psig pressure are not to be tested at 1.15 times their design pressure, provide justification for not performing such tests.

Response

Pressure testing of the lined cells is not a requirement since the cells do not perform a containment function. A sodium spill accident within a cell will cause heating of the cell atmosphere and structure and a pressure buildup, but there are no requirements restricting cell atmospheric leakage. The structural design of the concrete walls of the cell, per ACI 349, ensures structural integrity under the sodium spill accident conditions.

The maximum pressure on the containment vessel under the most severe accident condition will not exceed 10 psi. A detailed description of the containment System is provided in PSAR Section 6.2.

Question CS 220.34 (h)

In Section 3.8.4.4.1, how are equivalent static loads obtained?

Response

Equivalent static loads are obtained by equation

$$P = C\,q\,A$$

where P = equivalent static load in lbs

C = drag or lift coefficient

q = dynamic pressure in lb/ft^2

A = exposed area in sq. ft.

The procedure in ANSI A58.1 is followed. PSAR Section 3.8.4.4.1 will be revised to include the above design information.

3.8.4.3.2 Creep, Shrinkage and Local Stress

Pre-stressed concrete design is not adopted for the design of the facility. Therefore, creep and shrinkage loads will be only considered to the extent they are provided in the referenced codes or as may be warranted by prudent design approach.

3.8.4.3.3 Sodium Fire Load

The cells, pipeways, and buildings where sodium fire is a postulated design basis accident, will be designed to withstand accident pressure and the associated temperature effects.

3.8.4.3.4 Hot Sodium Spill Effect

The protective devices such as steel catch pans or steel plate liners will be provided in floor areas subject to sodium spills to prevent concrete-sodium reaction.

3.8.4.3.5 Loading Combinations

All other Category I structures will be designed and analyzed for the loading combinations listed in Subsection 3.8.3.3.10.

3.8.4.4 Design and Analysis Procedures

3.8.4.4.1 Analysis Procedures

Classical theory, equations and numerical methods will be used as necessary in the analysis of the structures. Classical methods used in the analysis will be in accordance with standard textbooks, handbooks, and papers as used in engineering practice. The following computer programs will be used in the static analysis:

1. NASTRAN
2. MARC CDC
3. MRI - STARDYNE
4. GT STRUDL
5. Other in-house computer programs

Loads and loading combinations as delineated in Section 3.8.4.3 will be considered. For dead loads, live loads, wind loads, tornado loads and accident loads, all of the methods listed above will be used. Wind loads, tornado loads and accident loads are converted to equivalent static loads and will be applied to the structure as uniform or concentrated loads.

Equivalent static loads are obtained by equation

$$P = C\,q\,A$$

where P = equivalent static load in lbs.

C = drag or lift coefficient

q = dynamic pressure in lb/ft^2

A = exposed area in sq. ft.

The procedure in ANSI A58.1 is followed

Wind and tornado loadings, flood loadings and missile loading applied on structures are discussed in Sections 3.3, 3.4 and 3.5.

Question CS 220.34 (I)

The Section 3.8.2.6 and 3.8.2.7 are missing and should be provided. It appears that your Section 3.8.2.5 should be revised.

Response

PSAR Section 3.8.2.5 has been totally revised and PSAR Sections 3.8.2.6 and 3.8.2.7 have been added.

The seismic analysis will include the local effects of the air locks vibrating as independent systems. The seismic effects of this independent vibration will be added directly to all other seismic effects.

The Equipment/Personnel airlock will be supported entirely by the containment vessel shell.

3.8.2.5 Structural Acceptance Criteria

The structural acceptance criteria is based on full compliance with the requirements of ASME B&PV Code, Section III, Divisions 1 (subsection NE) and 2 as applicable. The requirements of Regulatory Guide 1.57 are included in the structural design criteria. Table 3.8.2-1 from NUREG-75/087 was adopted to define the stress limits for the different load combinatins. Buckling requirements are defined in the "Buckling Criteria" given in PSAR Section 3.8.A.

Applicable codes and standards used in formulating the structural criteria are described in Section 3.8.2.2.

Load combinations for the Containment Vessel are given on Tables 3.8-1, 3.8-1a and 3.8-1b. Stress limits are given in Table 3.8-3.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

3.8.2.6.1 Materials

3.8.2.6.1.1 Shell Apurtenances and Structural Shapes

The shell is fabricated by the full penetration buttwelding of uniform, steel plates of thicknesses that range from 1" to 1-3/4". The shell plates, the circumferential shell stiffeners, the large penetration assemblies, and the polar crane girder are made of carbon steel having the generic designation SA516, Grade 55. Pipe penetration assemblies are made of carbon steel having a generic designation SA333. Welding electrodes for joining these carbon steels are designated as SFA-5.1 having an E70 or E70XX classification. These materials fully conform to Section III, Division 1 of the ASME B&PV Code.

The designs of structures, not within the scope of the ASME B&PV Code, conform to the requirements of the American Institute of Steel Construction (AISC) Specifications.

3.8.2.6.1.2 Bottom Liner

The Bottom Liner is a 1/4 inch thick steel plate conforming to ASME SA 516 Grade 70. The structural shapes used for anchoring the liner plate into the concrete foundation, leak-chase channels, angles and back-up bars are ASME SA 36 carbon steel.

Steel plates and structural shapes fully conform to Section CC-2500 of the ASME, B&PV Code, Section III, Division 2. Welding materials conform to Section CC-2600 of the same Code.

For corrosion protection see Section 3.8.2.2.2

3.8.2.6.2. Quality Control

General information on Quality Control is given in Chapter 17.

3.8.2.6.2.1 Shell Appurtenances and Structural Shapes

Metallic materials used to fabricate the shell and its parts, appurtenances, and penetrations are required to be in compliance with Subsections NA-1220 and NE-2000 of Section III, Division 1, of the ASME B&PV Code. Metallic materials used to fabricate structures not within the scope of the ASME B&PV Code are required to be in compliance with the American Institute of Steel Construction (AISC) Specifications.

All weld designs are required to be in compliance with Subsection NE-3350 and are to be examined per the requirements of Subsection NE-5000 and as outlined in Section V of the ASME B&PV Code or as outlined in Appendix A of Section III, Division 1, of the ASME B&PV Code as appropriate. In addition, all Category A and B welds below grade will be examined by a Halide test.

Weld procedures are to be qualified by compliance with Section IX of the ASME B&PV Code.

Fabrication is to be in compliance with Subsection NE-4000 of Section III, Division 1 of the ASME B&PV Code.

3.8.2.6.2.2 Bottom Liner

Material manufacturers are required to provide Certified Material Testing Reports (CMTR) for bottom liner material and welding material in accordance with ASME B&PV Code, Section III, Division 2, Section CC-2130. Bottom liner welds are examined by liquid penetrant or magnetic particle examination followed by vacuum box testing for leak tightness. Non-butt and attachment welds are examined by either liquid penetrant or magnetic particle inspection methods.

Liquid penetrant examinatin is in accordance with ASME B&PV Code, Section V, Article 6. Evaluation of penetrant indications and their acceptance standards are in accordance with ASME B&PV Code, Section III, Division 2, paragraph CC-5544. Magnetic particle examination is in accordance with ASME B&PV Code, Section V, Article 7. Indications are evaluated and accepted in accordance with paragraph CC-5545.

Leak testing of liner seam welds is performed by the vacuum box method using at least 5 psi differential with the atmospheric pressure in accordance with ASME B&PV Code, Section III, Division 2, paragraph CC-5535. The channel to liner plate and related pressure barriers are tested for leak tightness in accordance with paragraph CC-5535.2 of the same Code. Indications for vacuum box and leak-chase examinations are evaluated and accepted in accordance with paragraph CC-5546 of the same Code. Material identification is in accordance with ASME B&PV Code, Section III, Division 2, CC-2540. Control of material handling, shipping and storage are in accordance with CA-4450 and CA-4600.

The bottom liner is included in the Code NPT-CC stamp of the containment vessel as a "Part" in accordance with the requirements of NA-8000.

Fabrication tolerances are in accordance with CC-4522 and welding details are in accordance with CC-4542.

3.8.2.7 Testing and Inservice Inspection Requirements

Corrosion protection and protective coatings for the containment are described in Section 3.8.2.2.2.

Testing and inservice inspection requirements of the containment are discussed in Section 6.2. In addition to the testing described in Sections 3.8.2.6 and 3.8.5.6, the following construction stage and preoperational tests will be performed.

3.8.2.7.1 Bottom Liner Plate Test

Before placing concrete over the bottom plate and after the vacuum box test of the welds, the liner welds will be covered by the leak-chase channels. The channel to liner plate and related pressure barrier shall be tested for leaktightness by pressurizing the channels to containment design pressure (10 psi). If any indicated loss of channel test pressure occurs within two hours as evidenced by a test gauge, the channel to liner weld will be soap bubble tested.

3.8.2.7.2 Pressure Tests

A pneumatic pressure test shall be made on the Containment Vessel, airlocks, and equipment hatch at a pressure of 11.5 psig. Both inner and outer doors of the airlocks will be test ed at this pressure. All pneumatic tests shall meet the requirements of Appendix J to 10CFR50, NE-6000 of ASME B&PV Code Section III, Division 1 and applicable parts of Division 2. EXceptions to Appendix J to 10CFR50 are identified in Section 6.2.1.4.

3.8.2.7.3 Leakage Rate Test

Following successful completion of the pressure test, a leakage rate test at 10 psig will be performed with the airlock inner doors closed. The allowable leakage rate of the steel containment shall be 0.1% by volume of the containment in a 24 hour period and shall meet the requirements of 10CFR50 Appendix J.

3.8.2.7.4 Operational Testing

After completion of the airlocks fabrication, including all latching mechanisms, interlocks, etc., each airlock will be given an operational test consisting of repeated operation of each door and mechanism to determine whether all parts are operating smoothly without binding or other defects.

3.8.2.7.5 Leak Testing Airlocks

The airlocks will be pressurized with air to 11.5 psig. All welds and seals will be observed for visual signs of distress or noticeable leakage. The airlock pressure will then be reduced to 10 psig and a thick soap solution will be applied to all welds and seals and observed for bubbles or dry flaking as indications of leaks. All leaks and questionable areas will be repaired. During the overpressure testing the outer door will be locked with hold-down devices if required to prevent upsetting of the seals.

The internal pressure of the airlock will be reduced to atmospheric pressure and all leaks repaired after which the airlock will again be pressurized to 10 psig with air and all areas suspected or known to have leaked during the previous test be retested by above soap bubble technique. This procedure will be repeated until no leaks are discernible by this means of testing.

3.8.3 Concrete and Structural Steel Internal Structures of Steel Containment

3.8.3.1 Description of Internal Structures

The internal structures within the containment principally consist of the cells and other areas as listed in Table 3.8-2 and as shown on the General Arrangement figures of Section 1.2. The internal structures are enclosed by two continuous circular walls located on each face of the steel containment vessel between the foundation mat and operating floor levels. The circular walls act as a radiation shield and pressure boundary in local cell areas, as a support for vertical loads and carry horizontal shears to the foundation mat. The entire steel containment vessel will be designed for the 10 psig internal pressure. The detailed physical description provided herein is limited to those cells which significantly contribute to the structural system. These cells are reinforced concrete structures designed to the requirements as noted in Table 3.8-2.

Table 3.8-1b

Loading Combinations for Airlocks

The loading combinations for which the airlocks shall be designed are as follows:

Testing:	D + Tt + Pt
Normal:	D + L + To + OBE
Accident and Environmental:	D + L + T' + Pi + OBE
	D + L + T' + Pi + SSE
	D + L + T' + Pe + OBE
	D + L + T' + Pe + SSE
	D + L + SSE

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Question QCS 421.4

The applicant should formally submit a diagram of the auxiliary feedwater system showing the division assignments for all valves and safety grade instrumentation and controls. A discussion should be included to indicate the normal position and position upon loss of power of each valve. In your presentation, and in Section 7.4.1.1.6, credit is taken for the feedwater isolation valve to fail safe in the open position upon loss of electrical power. Justify this fail-safe analysis for all incidents (i.e., hot shorts, power supply overvoltage, etc.) that could prevent operation of this isolation valve.

Response

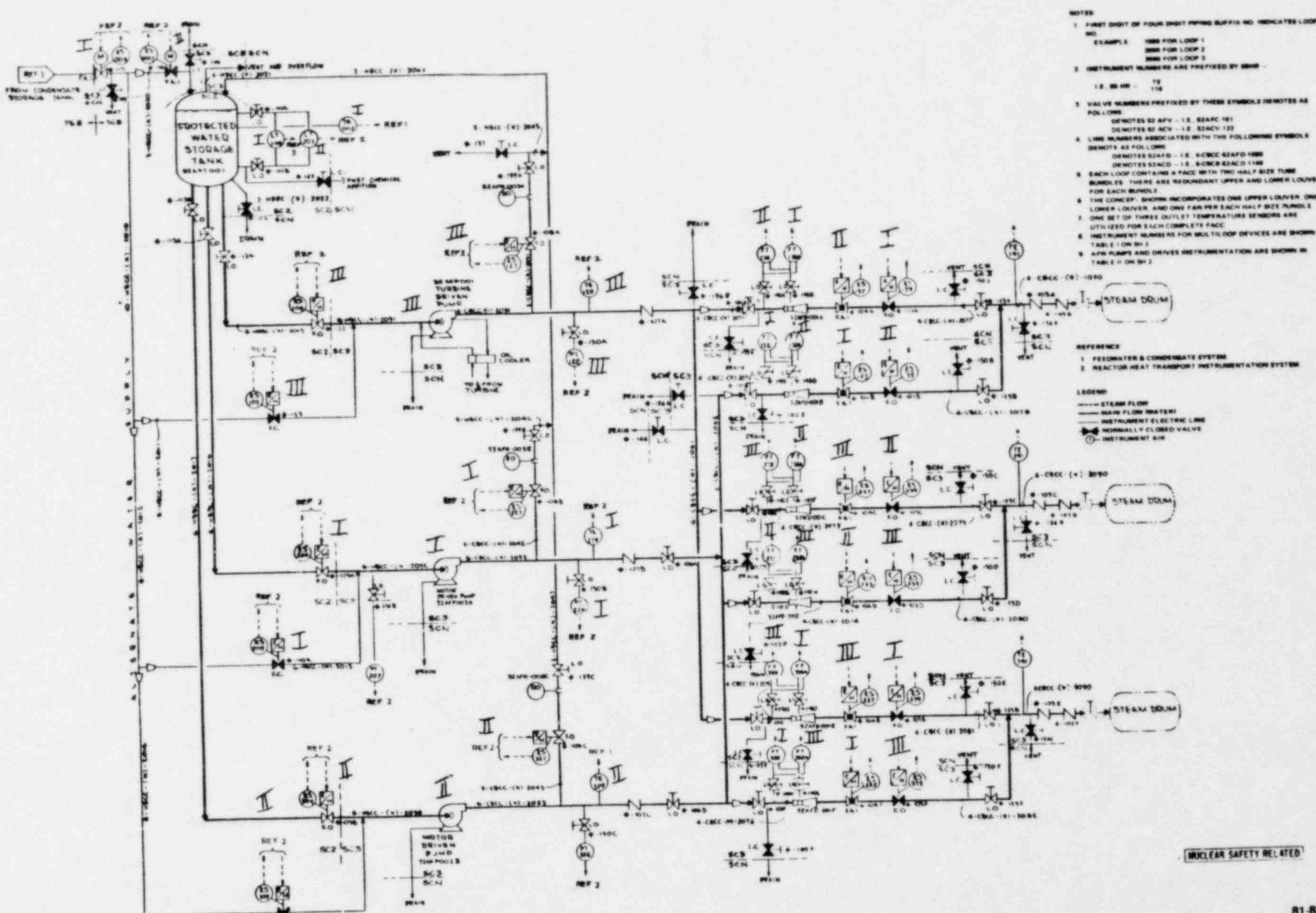
Figures 5.1-5 and 5.1-5a of the PSAR (attached) have been marked up (Figures QCS 421.4-1, 2) to show the power division assignments for all valves and safety grade instrumentation. The controls for the devices are not shown, however, the power assignment is the same as that of the valves, louvers, and fan blade pitch control being actuated. The normal position and failure position of the valves are shown in Figures 5.1-5 and 5.1-5a.

As shown in Figure 5.1-5 there are six isolation valves, two in parallel to each steam drum. One valve supplies water from the electric driven pumps, the other, from the turbine drive pump. Each capable of supplying 100 percent water flow to the steam drum. These two valves are supplied power from separate Class 1E power divisions. The failure of one valve to open on demand would not result in the loss of water to a steam drum.

Failures such as shorts, grounds, power supply opens, etc. which result in the loss of power and thus opening the valves will not be discussed. Even if these occur during reactor power operation they have no effect on power operation or the ability of SGAHRS to function when called upon.

Each isolation valve is supplied from a regulated 120 VAC power supply and is designed to operate with the over voltage transients identified for the power supply.

The wiring and controls for the two valves supplying water to a steam drum are in separate Class 1E power divisions and separation is provided per IEEE 383. Hot shorts between the two valves supplying the same steam drum will not occur. The only hot short which could occur would be with the same power division and between the valve and the controls. This could result in two of the six isolation valves not opening when called upon. Since these two valves supply separate steam drums and there is a 100 percent redundant supply, a failure of two of the six isolation valves would not result in the loss of water to any of the three steam drums.



81-833-05

figure QCS 421, 4-1

~~Figure 5.1.5~~ STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM PIPING AND INSTRUMENTATION DIAGRAM (SHEET 1 OF 2)

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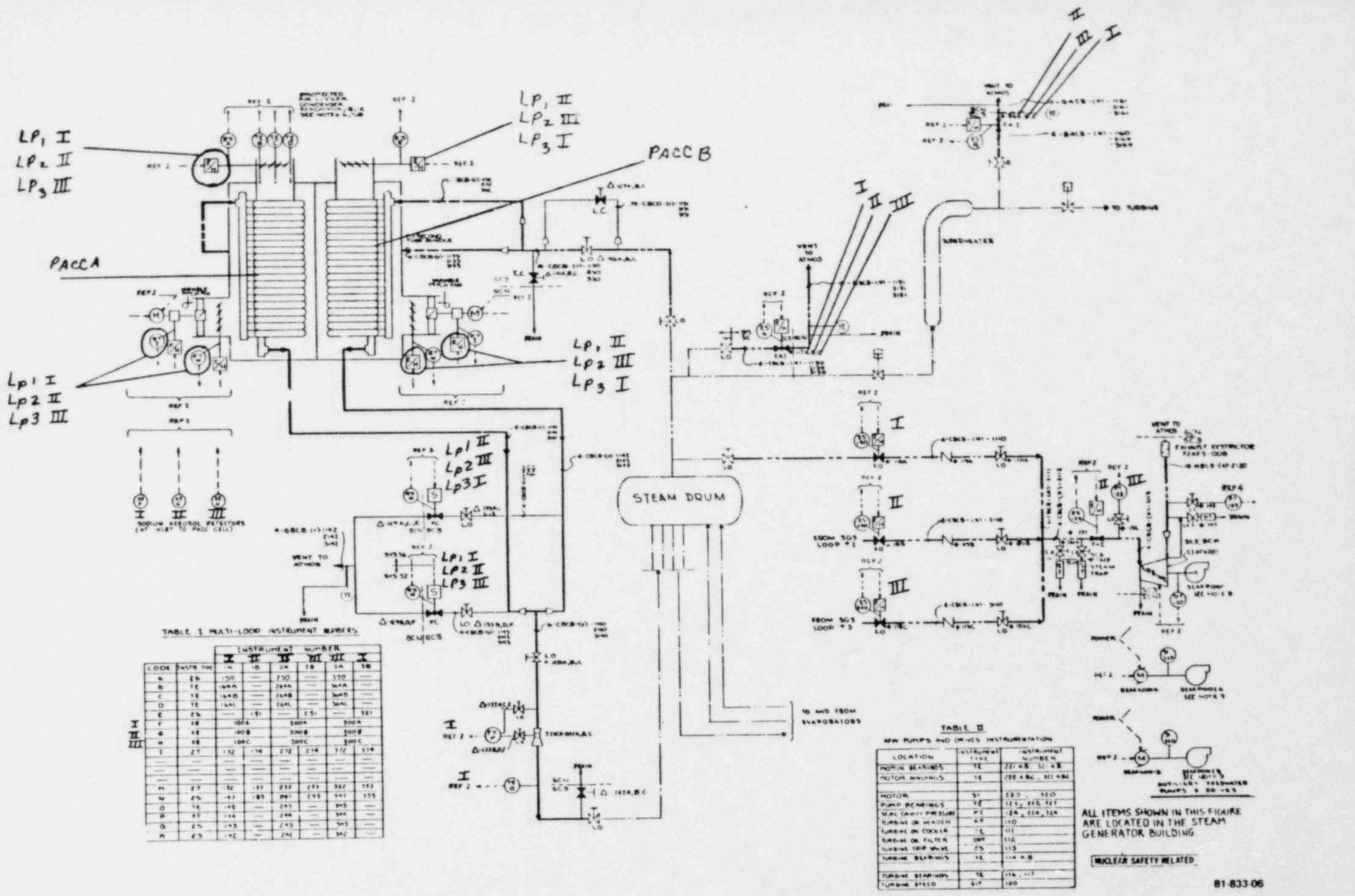

TABLE I MULTI-LOOP INSTRUMENT NUMBERS

CODE	INSTR TYPE	INSTRUMENT NUMBER I 1A	II 1B	II 2A	III 2B	III 3A	I 3B
A	ZS	150	—	250	—	350	—
B	TE	164A	—	264A	—	364A	—
C	TE	164B	—	264B	—	364B	—
D	TE	164C	—	264C	—	364C	—
E	ZS	—	151	—	251	—	351
F	XE	100A		200A		300A	
G	XE	100B		200B		300B	
H	XE	100C		200C		300C	
I	ZT	172	174	272	274	372	374
—	—	—	—	—	—	—	—
—	—	—	—	—	—	—	—
—	—	—	—	—	—	—	—
M	ZT	152	153	252	253	352	353
N	ZS	147	[illegible]	247	255	347	355
O	TE	145	—	245	—	345	—
P	PT	144	—	244	—	344	—
Q	ZS	143	—	243	—	343	—
R	ZS	142	—	242	—	342	—

TABLE II

AFW PUMPS AND DRIVES INSTRUMENTATION

LOCATION	INSTRUMENT TYPE	INSTRUMENT NUMBER
MOTOR BEARINGS	TE	221 A,B, 321 A,B
MOTOR WINDINGS	TE	222 A,B,C, 322 A,B,C
MOTOR	SI	220, 320
PUMP BEARINGS	TE	123, 223, 323
SEAL CAVITY PRESSURE	PT	124, 224, 324
TURBINE OIL HEADER	PT	110
TURBINE OIL COOLER	TE	111
TURBINE OIL FILTER	DPT	112
TURBINE TRIP VALVE	ZS	113
TURBINE BEARINGS	TE	114 A,B
TURBINE BEARINGS	TE	116, 117
TURBINE SPEED	SIT	120

figure QCS 421.4-2 ~~Figure 5.1-5a~~ STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM PIPING AND INSTRUMENTATION DIAGRAM (SHEET 2 OF 2)

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