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DESCRIPTION:
Letter notarized 7-29-75.. trans the following Letter Re. our letter of 9-30-75..

ENCLOSURES:
Enclosure Containing Responses to Request for Additional Information Transmitted by NRC's Letter of 9-30-75.... W/Attached table and figures....

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PLANT NAME: H.B. Robinson # 2

A-3

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October 17, 1975

FILE: NG-3514 (R)

SERIAL NO. NG-75-1686

DOCKETED USNRC OCT 20 1975

RECEIVED OCT 20 1975 U.S. NUCLEAR REGULATORY COMMISSION

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

RE: H. B. ROBINSON UNIT NO. 2
DOCKET NO. 50-261
FACILITY OPERATING LICENSE NO. DPR-23

Dear Mr. Rusche:

In response to Mr. Robert W. Reid's letter of September 30, 1975, to Mr. J. A. Jones requesting additional information in support of our Cycle 4 reload application, Carolina Power & Light Company submits three signed originals and 37 copies of this letter and enclosure. The enclosure contains responses to the requests for additional information enclosed in Mr. Reid's letter.

In response to informal requests for additional information regarding Westinghouse fuel in Cycle 4, the Westinghouse ECCS analysis performed for fuel resident in the core during Cycle 3 is conservative with regard to the Westinghouse fuel which will be recycled into the core in Cycle 4. This is shown by reference to Section 12.0 of WCAP-8341, "Westinghouse Emergency Core Cooling System Evaluation Model - Sensitivity Studies," which demonstrates that the peak clad temperature occurs at the point when densification is complete, or about 1300 MWD/MTU fuel rod burnup in the Robinson Plant.

The fuel that will be cycled into Cycle 4 will have a burnup of at least 5,000 MWD/MTU. With this burnup, Table 12-1 of WCAP-8341 shows that there will be a significant margin in peak clad temperature

12142

with respect to the conservative burnup at which the analysis was performed. Thus the present analysis for Westinghouse fuel is conservative and applicable to Cycle 4 and subsequent cycles employing the fuel.

Yours very truly,

E. E. Utley
Vice President
Bulk Power Supply

EEU/tl

Enclosure

cc: Messrs. H. R. Banks
N. B. Bessac
R. E. Jones
P. W. Howe
J. B. McGirt
R. L. Sanders
D. B. Waters

Sworn to and subscribed before me this 17th day of October, 1975.

Nancy A. Stephens (Yancey)
Notary Public

My commission expires: June 29, 1976

NANCY A. STEPHENS
NOTARY PUBLIC
WAKE COUNTY, N.C.

Enclosure

H. B. Robinson, Unit 2

Responses to Request for
Additional Information Transmitted by NRC
Letter of September 30, 1975

Question 1

Provide an analysis of the consequences of the RCCA drop presented in Section 3.2 of XN-75-14 if automatic turbine cutback does not occur. Also, what happens if the system is under manual control, and control rods are withdrawn to maintain power?

Response

An RCCA drop occurring through failure of an RCCA drive mechanism results in an immediate decrease in power. The power level is prevented from return to power by two actions: automatic turbine runback and rod blocking, both of which are redundantly actuated.

The results of a transient analysis with both of the above actions occurring are depicted in Figures 1-1 through 1-4. The reactor does not return to power. The peak heat flux reaches 89% of rated, and MDNBR does not fall below its initial value.

The transient responses with the assumption that no turbine runback occurs are depicted in Figures 1-5 through 1-8. In this case, the reactor returns to 98.3% of rated power at 20 seconds. The heat flux peaks at 97.7% of rated, but MDNBR does not drop below its initial value. New steady-state values are reached after 40 seconds at power level of 97.7%.

If the system is under manual control, operating procedures require the turbine be removed from load control and manual RCCA bank insertion is initiated to match the turbine load cutback as required. Withdrawal of any RCCA before retrieval of the dropped RCCA is not permitted by the operating procedures. If RCCA withdrawal were to occur, the reactor would be scrammed by its overpower or overtemperature ΔT protection system due to the large change in T_{ave} which is induced by the transient.

Question 2

What fuel surveillance program is planned for these initial Exxon Fuel Assemblies in HBR?

Response

ENC has an on-going fuel surveillance program for all its existing fuel reload contracts. The surveillance program on PWR fuel has only involved Ginna fuel to date but preliminary analysis indicates that the ENC fuel has performed satisfactorily.

CP&L has conducted detailed visual inspection programs on H. B. Robinson fuel during the Cycle 1-2 and Cycle 2-3 refueling outages and plans to perform a limited inspection during the Cycle 3-4 outage. Representative ENC fuel assemblies will be inspected during future refueling outages at H. B. Robinson to verify that the fuel is performing satisfactorily.

Question 3

The effects of a combined seismic and LOCA accident are not addressed in the HBR reload submittal. Will this be included in the seismic analysis to be submitted? If not, state what commitments will be made with regard to analyzing the effects of this accident.

Response

The ENC fuel assembly should be similar in axial stiffness and strength to a Westinghouse assembly design which has been shown to be adequate during a combined seismic and LOCA event. The design is termed adequate if the fuel assembly remains in a coolable geometry and resulting deformations do not prevent control rod insertion. The thicker ENC cladding will result in a stronger fuel assembly. On this basis, it is concluded that the ENC assembly design is adequate during the maximum assumed seismic and LOCA events. ENC has plans for a generic analysis of this subject to be performed during the first half of 1976. CP&L will reference this analysis following its submittal.

Question 4

Provide a drawing of the spacer grid assembly design which in particular shows the details of the Zr-4 grid strips and Inconel spring strips. What other differences are there between the Exxon and Westinghouse spacer grid assembly designs?

Response

Details of the Ziraloy-4 grid strips and Inconel springs are shown in Figure 4-1. The basic differences between ENC and Westinghouse grid spacer designs are as follows:

ENC design:

1) has greater depth
2) is welded versus brazed
3) has one versus two springs per cell
4) bi-metallic design

Question 5

What method of attachment between the spacer grids and the guide tubes is used?

Response

Grid spacers are capacitance resistant welded to the guide tubes. Four weld tabs are provided on the spacer at each guide tube location. Details are shown in Figure 5-1.

Question 6

Provide a more detailed description, or a report if available, of the DLITE code which is used to analyze circumferential strain as a function of burnup and to design the fuel pellet and fuel cladding gap.

Response

As previously submitted to and approved by the NRC in conjunction with Oyster Creek licensing, details of the DLITE code are as follows:

This code incorporates a modified version of a model published by Geithoff, et al., and is used to design the fuel pellet and fuel-cladding gap to achieve definite fuel exposure without exceeding cladding strain limits.

The DLITE model states that for a given cladding strain the maximum allowable fuel exposure (burnup) is a function of (1) the available voidage in the fuel and fuel-cladding gap, (2) the swelling rate, thermal expansion, and distortion of the fuel, and (3) fuel temperature.

The total voidage in the fuel is derived from fabricated porosity in the fuel and pellet dish volume. The amount of fabricated porosity to accommodate fuel swelling is a function of the fuel plasticity (fuel temperature). The swelling model thus divides the fuel into three temperature regions. For each region, a different fraction of the porosity is assumed to be available to accommodate fuel swelling. The fraction increases with increasing temperature because the fuel is more plastic at the higher temperatures.

- The fuel volume is divided into a plastic zone with temperatures above 1700°C, a creep zone with temperatures between 1300-1700°C where rapid creep and diffusion occur, and a low temperature zone below 1300°C with visco-elastic behavior.

- The volume portions of the porosity actually available for volume expansion due to swelling processes are 80 vol. % above 1700°C, 50 vol. % at 1300-1700°C, and not more than 30 vol. % below 1300°C. The last value may rise somewhat during long time operation as the low temperature fuel becomes more plastic.

Some fuel swelling may also be accommodated by the dish at the ends of the dished pellets. The effectiveness of the dish volume for accommodating fuel swelling is related to fuel centerline temperature. The higher the temperature, the greater the plasticity of the fuel and hence a greater flow of fuel material into the dish volume.

The following assumptions were made on the fraction of pellet dish volume available to accommodate swelling:

- For fuel centerline temperatures $\leq$ 1390°C (2534°F), 25% is available.

- For fuel centerline temperatures $\geq$ 1750°C (3182°F), 80% (rather than 100%) is available.

- For fuel centerline temperatures over 1390°C and under 1750°C, a linear extrapolation between 24% at 1390°C and 80% at 1750°C is used. This should yield a conservative answer and yet credit is given for the plastic properties of the fuel at the higher temperatures.

The fuel-cladding gap is used principally to accommodate fuel volume increases caused by thermal expansion and thermal distortion. The gap is sized to prevent the combination of thermal expansion and thermal distortion of the fuel from straining the cladding.

After all available voidage in the fuel-cladding gap, pellet dish, and pellet porosity is used by the swelling and expanded fuel, the cladding is assumed to expand to its limit without restraint to accommodate the swelling fuel. Calculations are continued until the specified cladding strain limit is reached.

Question 7

What were the types of thermal cycles and the number of cycles of each type considered to evaluate the effects of cyclic stresses in the cladding? Were both normal operational and abnormal (upset) transients considered in this evaluation?

Response

Details of thermal cycles used in cladding stress analysis are given in Section 4.4 of the generic fuel design report, Reference 1.

Question 8

As presented in Table 4.5, what is the difference between mechanical wear and fretting corrosion? What was the cause of the 0.8 and 1.0 mil depth wear at two locations attributed to mechanical wear in these fretting corrosion tests?

Response

Fretting corrosion is metal removal which arises when two surfaces in contact and normally at rest with respect to each other experience slight periodic relative motion in a corrosion inducing medium. This implies a vibratory motion such as might result from loose or damaged spacers or from excess clearances between components in a vibrating mode. Mechanical wear is metal removal, resulting from low repetitious relative motion of the two surfaces, such as the differential thermal expansion and contraction associated with reactor heatup and cooldown periods, typically occurring a relatively small number of times but with somewhat greater forces in action. The two reported relatively deep marks, i.e., 0.8 and 1.0 mil depth, were due to mechanical wear. This mechanical wear was attributed to the axial motion of the fuel rods relative to spacers caused by differential heatup rates of guide tubes and fuel rods. There was no evidence of active fretting.

Question 9

Provide an analysis or thermal hydraulic test data which shows the design adequacy of the holddown forces provided by the four leaf springs located in the upper grid plate in preventing fuel assembly liftoff.

Response

The estimated lift force on the assembly, based upon ENC prototype ΔP tests and information furnished by the reactor supplier, is illustrated in Figure 9.1 and is a function of operating temperature.

(1) ENC Report XN-75-39, "Generic Fuel Design for 15 x 15 Reload Assemblies for Westinghouse Plants."

The weight of an immersed fuel assembly is:

@ room temperature > 1250 lbs
@ normal reactor operating temperature > 1300 lbs

The spring constant of the holddown spring system is 450 lbs/in cold and 420 lbs/in hot as determined by test.

At room temperature, the minimum spring compression = minimum assembly height - maximum spacing between core plates = 161.23 - 160.49 = 0.74 inch.

At normal operating temperatures, the minimum spring compression = 0.68 inch - maximum differential thermal expansion between stainless steel core support structure and the zircaloy fuel assembly structure = 0.74 - 0.47 = 0.27 inch.

Figure 9-1 shows the total holddown load (spring load plus immersed assembly weight) for the improbable case illustrated above where tolerances are stacked up in the most conservative way and also for the more probable case where the tolerances are stacked up on a statistical basis.

The holddown springs are sufficiently far from the active core that no significance in-reactor spring relaxation is expected. Therefore, the holddown loads are more sufficient over the total operating range. This is confirmed by the fact that there was no evidence of assembly lift during prototype assembly flow tests over a large temperature range at flow rates above design conditions.

Question 10

In Section 4.2.2, which is titled "Fuel Temperature Analysis," values of UO_2 thermal conductivity, the UO_2 thermal conductivity integral, the fuel densification model and gap closure model used in the Exxon fuel thermal performance model are presented. Why are XN-209 the Exxon densification report, in which these models are described in detail, and the NRC staff safety evaluation on this report not referenced?

Response

In the future all data or models previously reviewed by the NRC will be referenced in subsequent documents. XN-209 and the NRC staff safety evaluation are hereby incorporated by reference in the Cycle 4 reload application as applicable.

Question 11

Provide a reference for the Hanevik data used to develop the densification rate expressions.

Response

The data is taken from: A. Hanevik, et al., "In-Reactor Measurements of Fuel Stack Shortening." This reference is provided in Supplement 1 to XN-174, "Densification Effects on EXXON Nuclear Boiling Water Reactor Fuel." Supplement 1 to XN-174 was incorporated into XN-209, "Densification Effects on EXXON Nuclear Pressurized Water Reactor Fuel" as Supplement 1 to XN-209.

Question 12

Describe the testing and inspections to be performed to verify the design characteristics of the fuel components including cladding integrity, verification of fuel enrichment, burnable poison concentration, fuel pellet characteristics, radiographic inspections, destructive tests, fuel assembly dimensional checks and the program for inspection of new fuel assemblies to assure mechanical integrity after shipment.

Response

Testing and inspection details are described in Section 5.0 of ENC's generic fuel design document, Reference 1.

Question 13

For the control rod ejection accident analyzed for H. B. Robinson Unit 2 in Exxon Report XN-75-44, what were the maximum cladding strains calculated for the cases analyzed?

Response

The maximum cladding strain for the rod ejection accident was calculated to be 0.8%. This occurred at the end of Cycle 4 from full power operation. Mechanical strain at the beginning of Cycle 4 was zero due to initial fuel to clad gap which is large enough that pellet-clad contact does not occur during the accident.

Question 14

The consequences of refueling accident does not appear to be addressed in the reload submittal. A reference to a previous analysis or an analysis for this accident should be provided.

Response

The utilization of Exxon fuel does not alter the analysis of the refueling accident contained in the FSAR, Section 14.2.1.

Question 15

Provide a drawing which shows the method of attachment of the control rod guide tubes to the upper and lower grid plates and the method of attachment of the spacer grids to the control rod guide tubes. Describe any differences between the Exxon design and the Westinghouse assemblies currently in HBR in these areas.

Response

Details of attachment of guide tubes to the upper tie plate, lower tie plate and grid spacers are shown in Figure 5-1.

Question 16

Is the material on page 5.9 of XN-75-38, and associated figures being offered as an alternate or adjunct to your present power distribution control and monitoring method? It is our understanding that you plan to continue use of the presently adopted Constant Axial Offset Control (CAOC) System. Please advise us of your plans for power distribution control and monitoring and your schedule for submittal of the necessary supporting analysis for the reload core. Under separate cover we are providing for your information a recently established NRC position on CAOC.

Response

The material on page 5.9 of ENC Report XN-75-38 and associated figures were presented to demonstrate the margin to limiting values for the variation of the overpower and overtemperature setpoints as a function of axial offset, and were not intended to be a justification of a new power distribution control procedure. Carolina Power & Light Company still intends to employ Constant Axial Offset Control procedures to ensure the maintenance of limiting peaking factors and to ensure margin to DNB limits during anticipated transients.

Question 17

More information is required regarding the following tests included in Attachment A to CP&L letter of August 3, 1975:

1) Initial Criticality CPL-R-6.1
2) Design Check Test CPL-R-6.2
3) Boron Dilution CPL-R-6.3
4) Boron Addition CPL-R-6.4
5) Power Distribution Maps CPL-R-9.4

In particular, the acceptance criteria for these tests should be specified and related to values of physics parameters used in your accident analysis.

Response

(1) Initial Criticality - CPL-R-6.1

a) Acceptance of this test depends on the following:

1) Criticality shall be achieved within 1/2% $\Delta\rho$ of the analytically predicted value. The approach technique using 1/M procedures provides accurate monitoring of this criteria prior to achieving criticality.

2) The conduct of zero power tests at neutron flux levels which precludes doppler influence is assured by determining levels at which doppler influence is first observed and setting the zero power level two decades below this level.

3) The reactivity computer is verified as being properly calibrated when it is capable of solving the in-hour equation accurately, as verified by achieving agreement between the computer and the fuel vendor supplied in-hour equation solution, using reactivity insertion and observing the associated periods.

(2) Design Check Test - CPL-R-6.2

Acceptance criteria for this test are:

a) Difference between end point boron concentrations, measured and predicted, shall not exceed the reactivity equivalence of 1/2% $\Delta\rho$ for all rods out condition.

b) Isothermal temperature coefficients shall be determined to be negative.

(3) Boron Dilution - CPL-R-6.3

Test is acceptable if sufficient data were obtained to establish differential and integral rod worth curves and that the integral worths are within $\pm$10% for individual bank worths and + 10%, - 5% for combined bank worths as predicted by the fuel vendor.

(4) Boron Addition - CPL-R-6.4

The acceptance criteria for this test are the same as that for item 3. The worths in this case are determined in overlap during rod withdrawal. No requirements are placed on worths in this test.

(5) Power Distribution Maps - CPL-R-9.4

This test is acceptable if data necessary for determination of F_q^n, $F_{\Delta H}$, and quadrant power tilts is obtained. For zero power maps, the tolerance of $F_{\Delta H}$ is +8% for predicted values $\geq$1.0 and +15% for predicted values <1.0. For maps taken at power, the values of F_q^n, and $F_{\Delta H}$ must be equal to or less than those allowable by Technical Specifications. Quadrant power tilts at zero power must be equal to or less than 2%; at power, the tilts must comply to Technical Specifications.

Question 18.a

1.a *Describe the manner by which T_f is determined from the fuel model shown on Figure 2.1.*

Response

Since the four fuel nodes are equal volume, the average fuel temperature used for calculating Doppler feedback is given by the equation:

$$T_f = \sum_{i=1}^{4} \frac{T_{fi}}{4}$$

Question 18.b

Pg. 6, Eqns. 2.1.5 to 2.1.10

Discuss the considerations for axially weighted Doppler feedback reactivity in this program, and the lack of fuel temperature dependence for the Doppler coefficient α_D.

Response

The purpose of the PTS-PWR is to analyze abnormal operating transients to establish the adequacy of control settings and engineered safety equipment (relief and safety values, etc.). To do this, the reactivity feedback coefficients for both Doppler and moderator are generally set at extreme values. High or low values are used depending on which is conservative for determining plant response to the transient in question. Under these circumstances, the temperature dependence of the reactivity coefficients is not desirable because it will cause a deviation from the desired conservatism.

Question 19

Pg. 7, Eqn. 2.1.14

Provide an assessment of the approximation in reactor power transients when using this equation in place of equation 2.1.1 for a +30¢ input

reactivity step, and a 5¢/sec ramp up to +60¢ including crossover to equation 2.1.1 at +40¢ in the ramp transient.

Response

Due to the extremely high gain associated with the differential equation 2.1.1, the algebraic approximation given by equation 2.1.14 actually provides a more accurate calculation of power level than Euler integration of 2.1.1. If the instantaneous time Constant (τ_p) of 2.1.1

$$\left(\tau_p = \frac{\ell}{\beta\left[1 - \frac{\delta k}{\beta}(1-\beta)\right]}\right)$$

becomes less than twice the value of the integration step size (Δt, normally 0.005 sec), Euler integration of 2.1.1 begins to rapidly lose accuracy and may even become unstable. For typical values of β and ℓ, the reactivity value at which $\tau_p = \Delta t$ is about +40¢. For this reason, the approximation of equation 2.1.14, obtained by setting 2.1.1 equal to zero and solving for P, is used for values of $\delta k/_\beta$ less than +40¢. The effectiveness of the approximation was assessed by making two runs of a transient involving a +30¢ reactivity step followed by a 5¢/sec ramp to 60¢. The first run was made with the crossover point at its normal value of +40¢. For the second run, the crossover point was set at -20¢ so that Eqn. 2.1.1 was used for the entire transient. The results are shown in Figures 19-1 and 19-2. Tabular values of total reactivity versus time are compared in Table 19-1. As seen by this comparison, there is very little change when the approximation equation is used.

Question 20

Pg. 15, Eqn. 2.2.18

Discuss the rationale for coolant temperature summation over only 9 nodes in the 10 node model of Figure 2.2 to obtain core average moderator temperature.

TABLE 19-1

REACTOR POWER TRANSIENT FOR

REACTIVITY EQUATION ASSESSMENT

Time (Sec)	Case 1 Eq. 2.1.1 Used After + 0.40$ Power (MW)	Case 2 Eq. 2.1.1 Used During Entire Transient Power (MW)
.000	.23000+04	.23000+04
.250	.33819+04	.33821+04
.500	.35595+04	.35599+04
.750	.37421+04	.37426+04
1.000	.39341+04	.39347+04
1.250	.41387+04	.41395+04
1.500	.43587+04	.43597+04
1.750	.45970+04	.45983+04
2.000	.48563+04	.48579+04
2.250	.51398+04	.51404+04
2.500	.54511+04	.54517+04
2.750	.57941+04	.57947+04
3.000	.61736+04	.61742+04
3.250	.65949+04	.65955+04
3.500	.70646+04	.70652+04
3.750	.75900+04	.75907+04
4.000	.81804+04	.81811+04
4.250	.88464+04	.88472+04
4.500	.96012+04	.96020+04
4.750	.10461+05	.10461+05
5.000	.11444+05	.11445+05
5.250	.12575+05	.12576+05
5.500	.13883+05	.13884+05
5.750	.15405+05	.15406+05
6.000	.17186+05	.17188+05
6.250	.18649+05	.18651+05
6.500	.20193+05	.20194+05
6.750	.21839+05	.21841+05
7.000	.23600+05	.23601+05
7.250	.25486+05	.25488+05
7.500	.27509+05	.27511+05
7.750	.29681+05	.29684+05
8.000	.32014+05	.32016+05

TABLE 19–1

Continued

Time (Sec)	Case 1 Power (MW)	Case 2 Power (MW)
8.250	.34520+05	.34523+05
8.500	.37214+05	.37217+05
8.750	.40109+05	.40112+05
9.000	.43222+05	.43226+05
9.250	.46570+05	.46573+05
9.500	.50170+05	.50174+05
9.750	.54042+05	.54046+05
9.905	.56588+05	.56592+05

Response

A term was left out of the Eqn. 2.2.18. It should be:

$$Tca = \left[\sum_{i=1}^{9} Tc(i) + \frac{Tc(10) + Tc(0)}{2} \right] \Big/ 10.$$

The coolant temperatures resulting from Eqn. 2.2.16 are at the node exits, hence, one half the 10th node temperature and inlet temperature are added to the other nine temperatures in the averaging summation.

Question 21

Pg. 16, Eqn. 2.2.20

Identify the initial condition recommended for determining h_o, and provide an assessment of the transient error in h resulting from neglect of Nusselt number variation for the more severe coolant temperature excursions computed with this program.

Response

The value for h_o is determined, generally, from full power operating parameters. If more appropriate, other steady-state conditions can be used. It is an input parameter.

An increase in core average coolant temperature will increase the Nusselt number and, thus, the heat transfer coefficient. For conservatism in calculating clad and fuel temperatures, this effect is neglected.

Question 22

Pg. 22, 2nd Par.

Describe the manner in which the variable time delay FDELAY is determined in PTS-PWR.

The flow-variable time delay FDELAY is a subroutine which computes the variation over time of a property (e.g., temperature, enthalpy,

impurity concentration) of an element of fluid leaving the outlet of a length of piping. Inputs required at each calculation step are: the inlet property value, the flow rate, the calculation time step size, the maximum expected flow rate, and the transport delay time required for an element of fluid to pass completely through the pipe when the flow rate is at its maximum value. It is assumed that there is no heat transfer into or out of the fluid in the pipe, and that there is no longitudinal mixing. (A second version of FDELAY which provides some mixing is now optionally available.)

The computational technique requires a division of the pipe length into an integer number of equal-volume nodes. The subroutine stores the property values at each node in an array in memory. At each time step, the flow value is added to a flow accumulator, and this value is compared with W_n, the rate of flow which would exactly displace the fluid in one node in a single time step. When the flow accumulator exceeds this value, an array pointer is moved which, in effect, moves all of the property values down one node toward the outlet of the pipe. The inlet node is then set to the current inlet property value. The outlet node is also updated, taking on the value that was previously held by the next-to-last node. The actual output of FDELAY uses linear interpolation between this last nodal value and the next-to-last node, based on the ratio of accumulated flow to W_n, to avoid discontinuities in the output value. Finally, when nodal updating occurs, W_n is subtracted from the flow accumulator, and the remainder is left in the flow accumulator.

Question 23

Pg. 26, 1st Par.

Describe the three point differencing technique used to compute primary system fluid mass changes.

Response

The three-point differencing technique mentioned in this paragraph was used to approximate the total primary system fluid mass derivative $\frac{d(M_{p\ell})}{dt}$ as required by Eqn. 2.3.31:

$$\left.\frac{d(M_{p\ell})}{dt}\right|_{t=t_i} \approx \frac{3M_{p\ell}(t_i) - 4M_{p\ell}(t_{i-1}) + M_{p\ell}(t_{i-2})}{2\Delta t}$$

where Δt is the calculation step size. This calculation has been replaced with the following equation:

$$\left.\frac{d(M_{p\ell})}{dt}\right|_{t=t_i} \approx \frac{\sum_{n=1}^{N} V_n\left(\frac{\partial \rho_w}{\partial T}\right)\left[T_n(t_i) - T_n(t_i-1)\right]}{\Delta t}$$

where $\frac{\partial \rho_w}{\partial T}$, the rate of change of water density with temperature, is determined from a function table look-up as a function of $T_n(t_i) + T_n(t_{i-1})$ /2; the summation over n indicates summation of each of the N primary loop nodes; and V_n is the volume of node n.

Question 24

Pgs. 31, 33, and 34, Eqns. 3.1.1, 3.1.3, and 3.1.4

Describe the manner by which U_{SUBC}, U_{BOIL}, and U_{STEAM} are determined in these equations.

Response

Eqn. 3.1.15 pg. 41 is the generalized overall conductance equation that is used to determine U_{SUBC}, U_{BOIL}, and U_{STEAM}. The only difference in the three cases is the last term, $1/h_oA_o$. In the case of boiling heat transfer, the overall conductance is controlled by the first three terms in the

equation because of the typically high heat transfer coefficient during boiling (i.e., h_o = 20,000 B/hr ft^2 ^oF). In the case of high quality, the heat transfer coefficient is so low (h_o = 400 B/hr^2 ^oF) that the last term controls the overall conductance. In this case, the heat transferred is small enough that it can be ignored for purposes of determining plant transient response. In the subcooled heat transfer case, all the terms in the conductance equation are used. The term, h_o, is calculated by the modified Dittus-Boelter correlation, Eqn. 3.1.16. The calculations are performed per unit area, so changing heat transfer areas can be factored in as shown on pgs. 31, 33 and 34.

Question 25

Pg. 33, 1st Eqn.

Provide justification for use of this expression for ΔTMI.

General Form: $DTlM_{ij} = (\Delta T_i * \Delta T_j)^{1/2}$

Response

The square-root approximation for the mean temperature difference $DTlM_{ij}$ between the primary and secondary sides of the steam generator, over the distance between nodes i and j, is used to estimate the effective heat transfer temperature difference. Over a majority of the range of temperature differences encountered in a typical steam generator, the square-root approximation is numerically nearly equal to the log-mean temperature difference approximation given by:

$$DTlM_{ij} = \frac{\Delta T_i - \Delta T_j}{\ln\left(\Delta T_i / \Delta T_j\right)}$$

where $0 < \Delta T_j < \Delta T_i$.

The log-mean temperature difference is recognized as a standard approximation for heat transfer calculations. The square-root approximation was chosen for PTS-PWR because it is a simpler, faster calculation and because it avoids the problem of division by zero when $\Delta T_i = \Delta T_j$.

Question 26

Pg. 34, 1st Eqn.

Provide a quantitive assessment of the error introduced into the determination of TIP3 from the assumption of negligible heat transfer when steam generator secondary quality is less than 0.89. Discuss the conditions resulting in reversal of primary loop flow.

Response

There is no error introduced by this statement. Referring to Figure 3.1 pg. 32, the statement in question is true only if LNB1 + LB1 > LTOP. In other words, if the exit quality of the steam generator is less than 0.89. In the steam generator model, the primary node temperatures are calculated at different points depending on the heat transfer mode on the outside of the tube bundle. The statement T1P3 = T1P2 is just the mathematical way to indicate that a steam phase does not exist.

The statements on pg. 34 relating to reversed primary loop flow, indicate how this is handled in the steam generator heat transfer model. The flow reversals would be predicted by the primary loop hydrodynamics model (i.e., integration of Eqns. 2.3.20 and 2.3.31). It does occur in pump coastdown and pump stall transients.

Question 27

Pg. 35, 2nd Eqn.

Identify the parameter Alpslp in this equation.

Response

Alpslp is the heat transfer area in the steam generator exposed to subcooled water on the secondary side. Similarly, Alpsil is the heat

transfer area in the steam generator exposed to boiling water on the secondary side and Alps23 is the heat transfer area exposed to steam on the secondary side.

Question 28

Pg. 35, Last Eqn.

Describe the steam generator operating condition resulting in the equality given by this equation, and describe the derivation of this equality from equation 3.1.4.

Response

The equation in question, as well as the 3rd equation on pg. 35 are how the calculations are performed for reversed primary loop flow. The equality is not derived from 3.1.4, it is switched to a different calculation to reflect the different flow direction. (The last line on pg. 35 should read: $d\frac{(T1P3)}{dt}$ = right side of Eqn. 3.1.5).

Question 29

Pg. 36, 1st and 2nd Eqns.

Discuss the steam generator operating conditions with Q1ps34 = 0 for both directions of primary flow.

Response

If Q1ps34 = 0 it means there is no boiling in the steam generator. Referring to Figure 3.1, this condition results when the LNB1 calculated by Eqn. 3.1.9, pg. 38 is greater than the total length of the steam generator, LTOT.

Question 30

General Suggestions

All parameters shown in the equations have not been identified in the list of nomenclature. Review of the report would be considerably expedited by use of conventional engineering nomenclature in the equations.

Response

ENC will consider the NRC suggestion in future documents prepared for submittal.

Question 31

The nuclear power transient was calculated with the XTRAN code which has not been reviewed, therefore, questions on the rod ejection analysis may be incomplete.

Response

The NRC comment is noted.

Question 32

Describe the DNB correlation used.

Response

The DNB correlations used in the XTHETA[2] calculations was the Babcock and Wilcox B&W-2 correlation as described in the XTHETA document, Section 4.2.

Question 33

Describe the calculational methods used to predict DNB and the number of fuel rods experiencing clad failure.

Response

The calculation model for predicting DNB was the XTHETA computer code. The maximum cladding temperature calculated was less than 1000°F. Clad failure is not expected to occur at less than 1500°F. Therefore, no rods are predicted to experience clad failure.

Question 34

Discuss the reason why the total peaking factors after ejection are significantly lower than those usually predicted for similar plants.

Response

The total peaking factors were calculated using the PDQ-7[3] code. No credit was taken for the power flattening effects of Doppler or moderator feedback in these calculations. The PDQ-7 calculations were two-dimensional (X-Y) with appropriate axial buckling correction terms. The total peaking factors were determined as the product of the radial peaking factor (calculated using PDQ-7) and conservatively high axial peaking factors of 1.40 and 1.30 for

beginning and end of cycle, respectively. The calculated axial peaking factor at the beginning of Cycle 4 with the control rod configuration as described for rod ejection was 1.14. Therefore, we conclude that peaking factors used in this analysis were the result of conservative calculations.

It is our opinion that the previous total peaking factors are the result of generic analysis performed for similar plants and as such are representative of maximum allowable total peaking factors rather than those which might be experienced by H. B. Robinson during Cycle 4.

Question 35

Discuss the pressure surge calculation and show the variation of reactor pressure with time.

Response

The maximum fuel pellet deposited enthalpy calculated by this analysis was 117 calories/gram. Since this maximum value was so small, it was concluded that a pressure surge calculation would not yield significant results.

Question 36

Present representative values of the Doppler and moderator feedback coefficients used in the calculations.

Response

The results of the analysis made to construct Figure 6.8 of Reference 4 and Figures 5.10 and 5.11 of Reference 5 were used in this analysis. For conservation, the Doppler coefficients were reduced by about 5%, and the soluble boron content was increased (making the moderator coefficient more positive).

Question 37

Evaluate the conservatism of the models and codes used by comparison with experiments, as available, and with more sophisticated spacial kinetics codes. In particular, the importance of 2 or 3-D flux characteristics and changes in flux shapes used for reactivity input and feedback, peak energy deposition, total energy, and gross heat transfer to the coolant should be evaluated.

Response

The calculated results of reactor power, fuel temperature, and deposited enthalpy as obtained using XTRAN have been compared to the corresponding results obtained when the code WIGL-2 was used as the calculational tool for rapid reactor transients. Some of these comparisons are shown in the XTRAN[5] document. Since WIGL-2 is a one-dimensional only code, these comparisons were made by making XTRAN calculations in a single dimensional mode. These comparisons show that the results obtained by using XTRAN are virtually equivalent to those obtained when WIGL-2 is used.

The importance of multidimensional calculations for determining reactivity feedback effects and reactor power as a function of time, is discussed at some length in the XTRAN document. Multidimensional calculations allow the transient to be analyzed with a single calculation rather than the synthesis of several one-dimensional calculations. The results of the XTRAN calculations were used primarily to provide the reactor power as a function of time utilized as input to XTHETA. The reactor power vs. time was conservative because of the conservative heat transfer, Doppler and moderator coefficients used in the XTRAN calculation.

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(2) F. D. Lang, L. H. Steves, L. C. Worley, T. A. Bjornard, "XTHETA: Multi-Node Heatup Code for Single Channel Transient Analysis," XN-74-21, Rev. 2, Exxon Nuclear Company, April, 1975.

(3) W. R. Cadwell, "PDQ-7 Reference Manual, "WAPD-TM-678 Westinghouse Electric Corporation, January, 1965.

(4) F. B. Skogen, "H. B. Robinson Fuel Design Report Volume 2, Neutronic Design for Cycle 4," XN-75-25, Vol. 2, Exxon Nuclear Company, June, 1975.

(5) F. B. Skogen, W. C. Gallaugher, "H. B. Robinson Unit 2, Cycle 4 Reload Fuel Licensing Data Submittal," XN-75-38, Exxon Nuclear Company, August, 1975.

(6) J. N. Morgan, "XTRAN-PWR: A Computer Code for the Calculation of Rapid Transients in Pressurized Water Reactors with Moderator and Fuel Temperature Feedback," XN-CC-32, Exxon Nuclear Company, August, 1975.

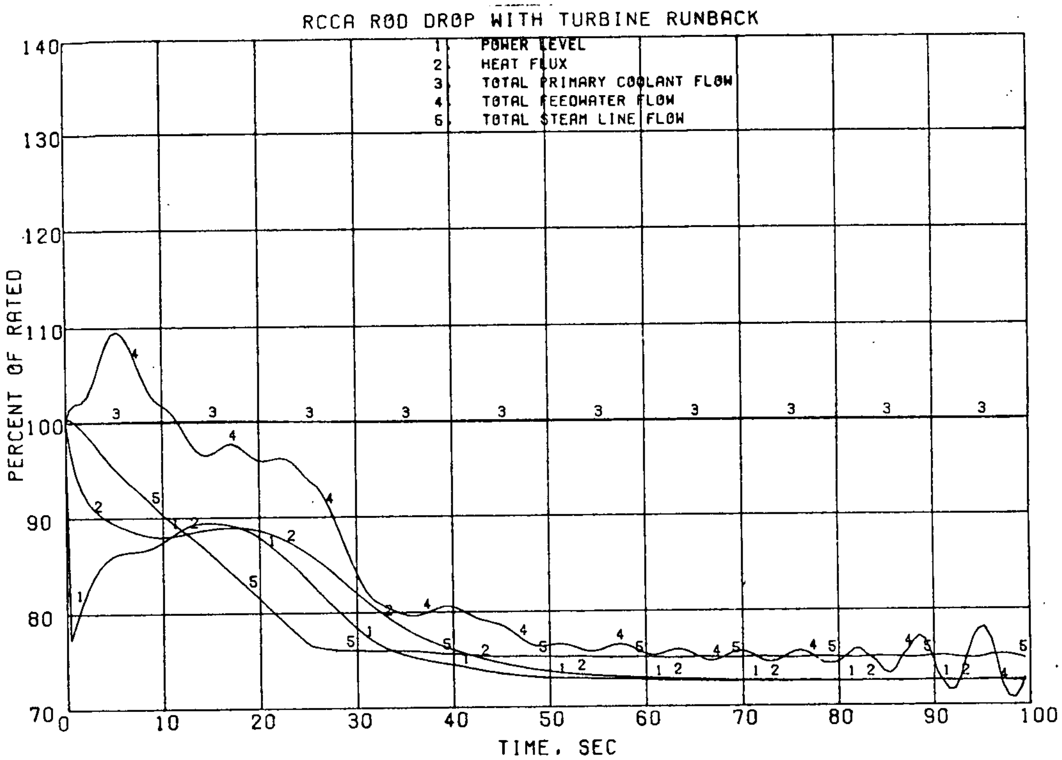


Figure 1—1

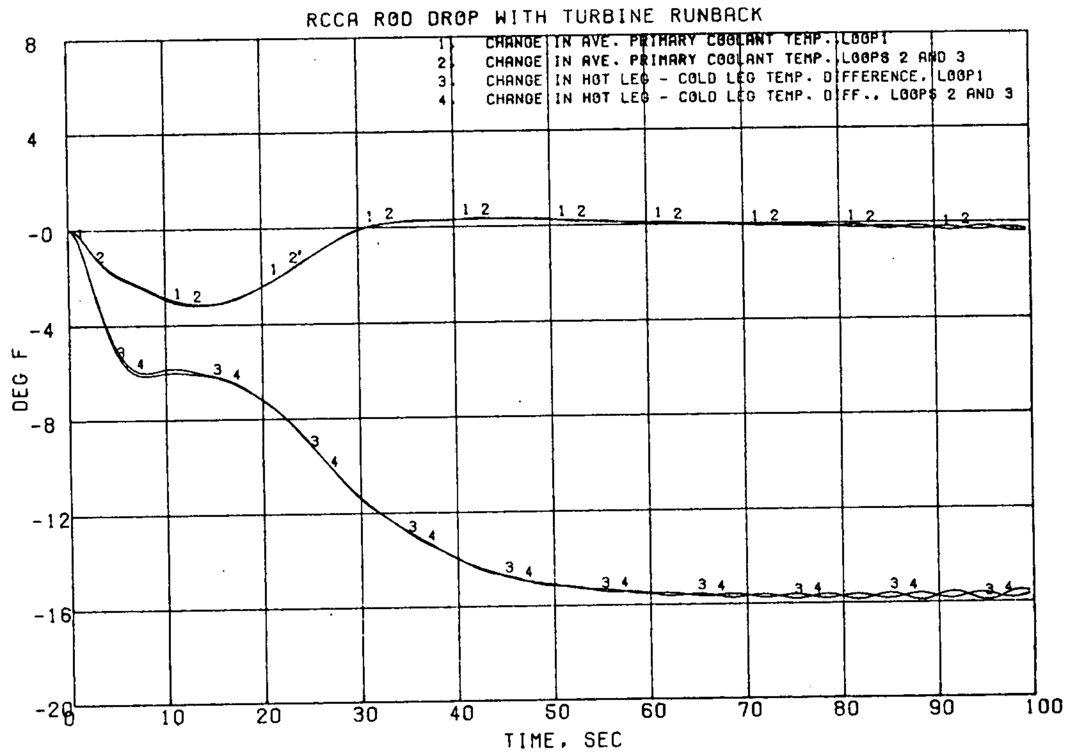


Figure 1—2

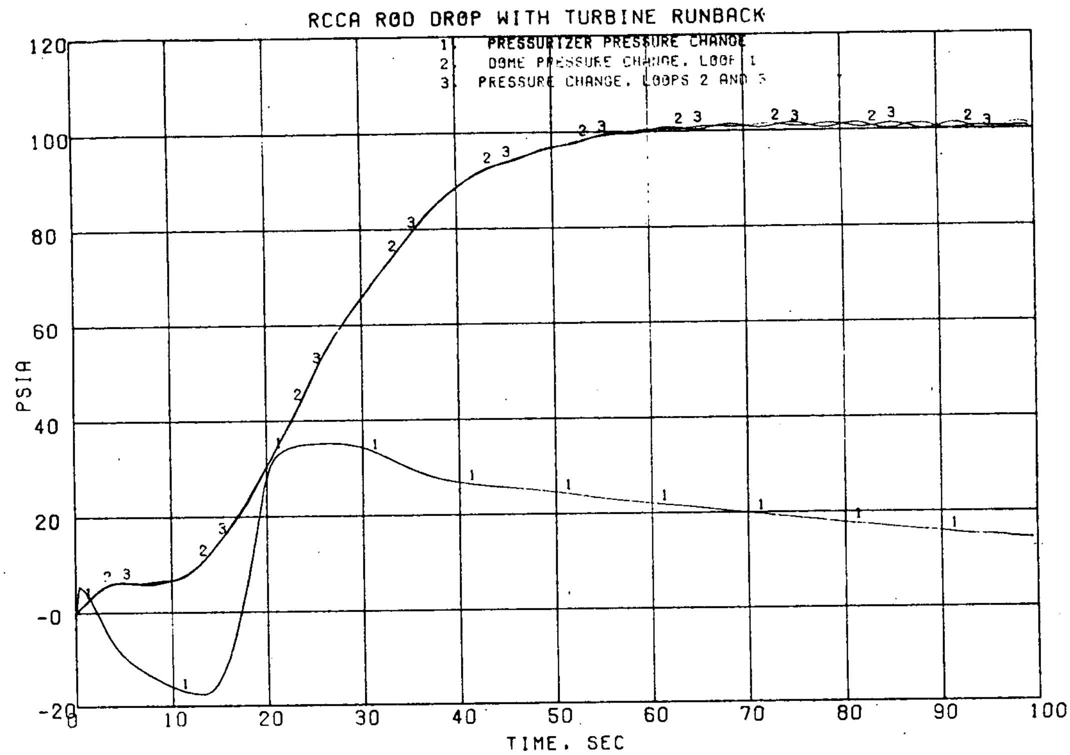
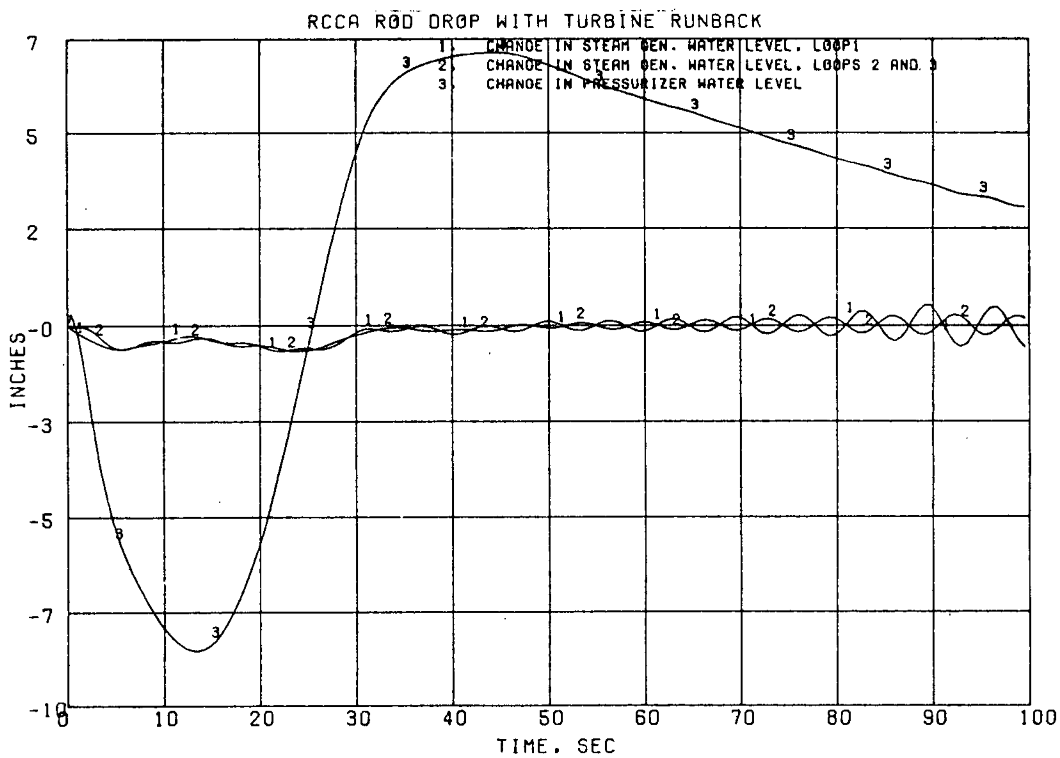


Figure 1–3



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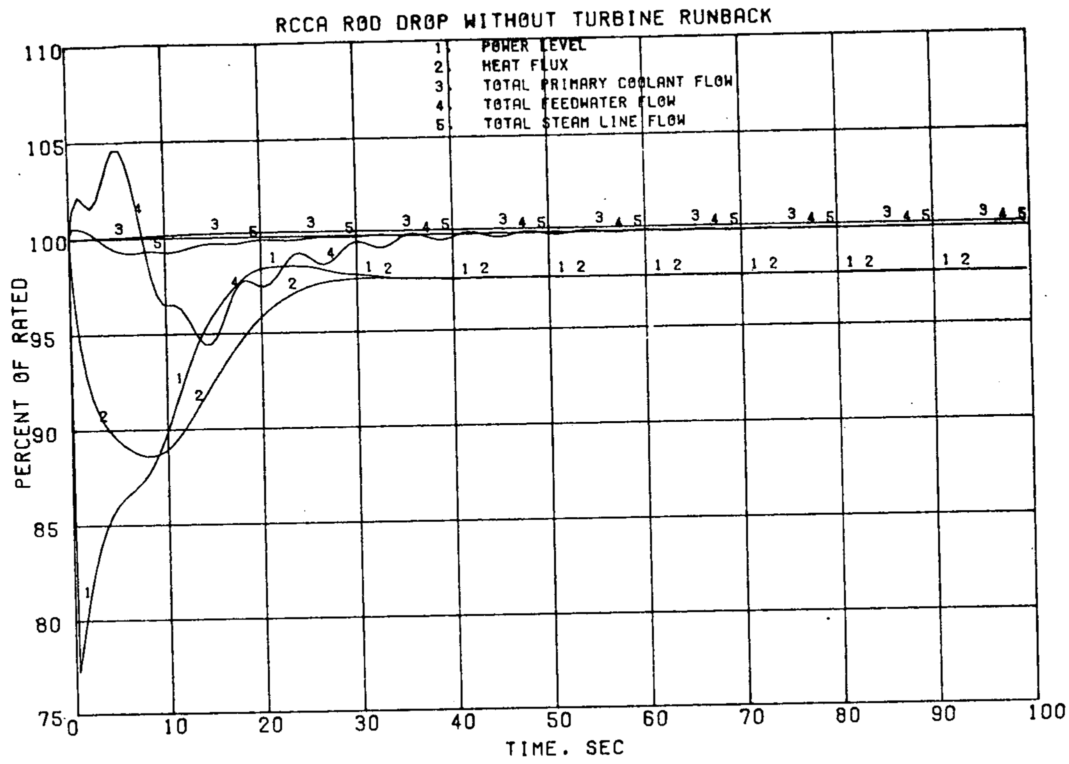

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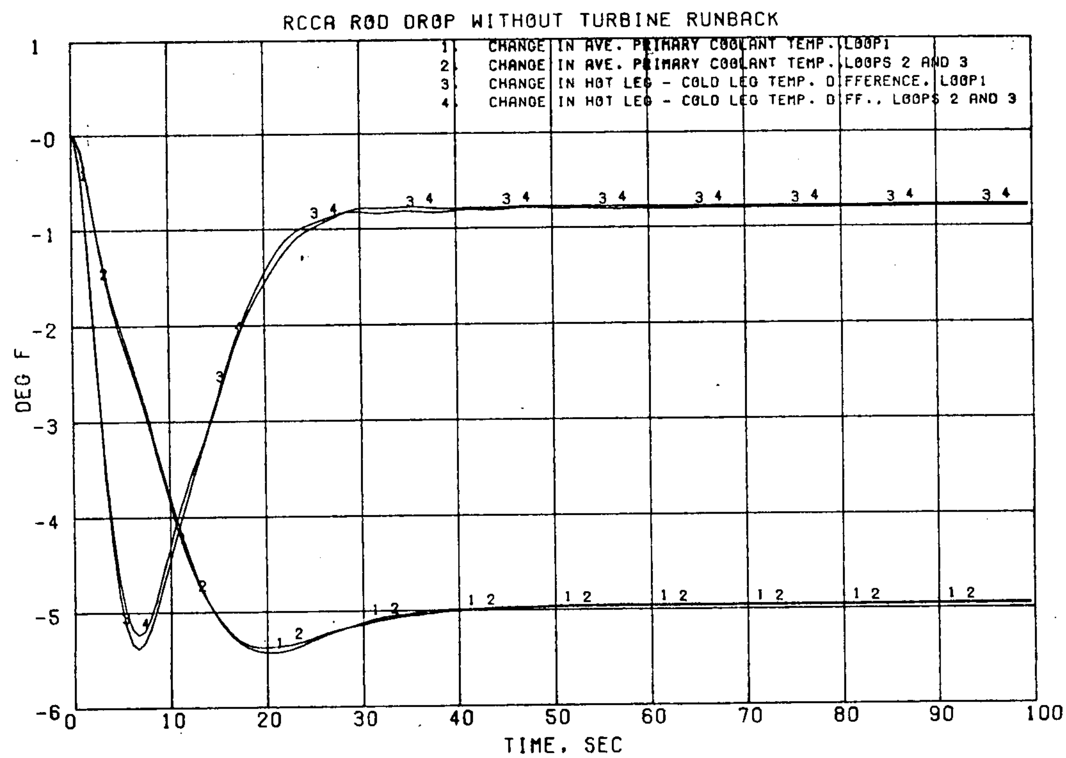

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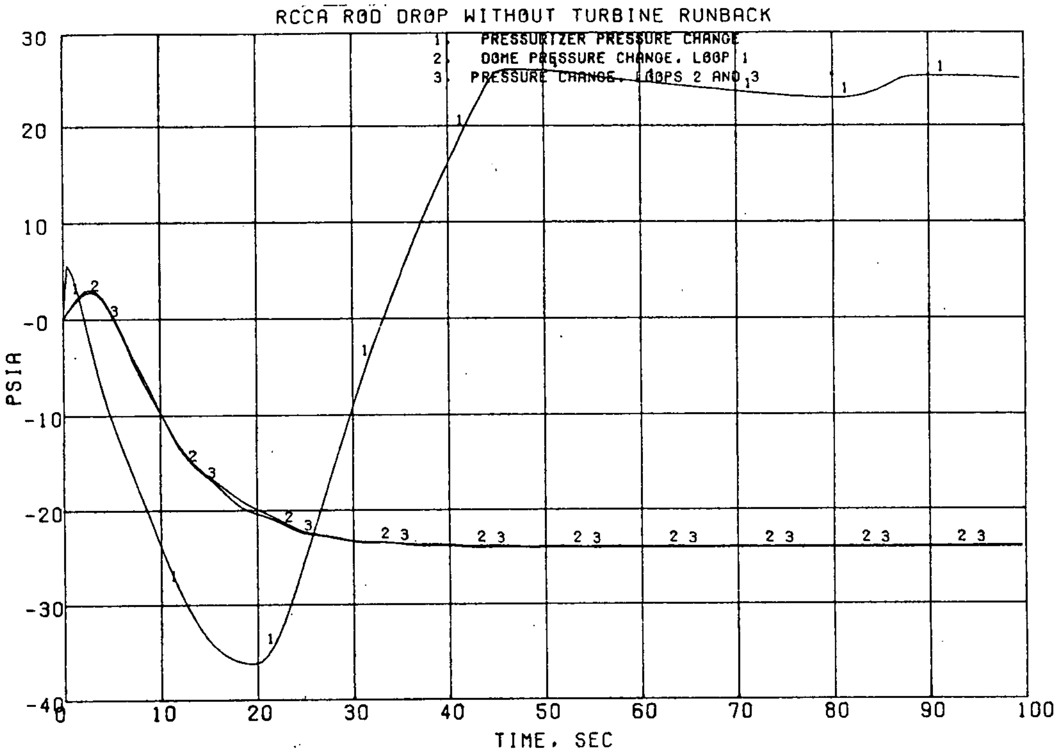
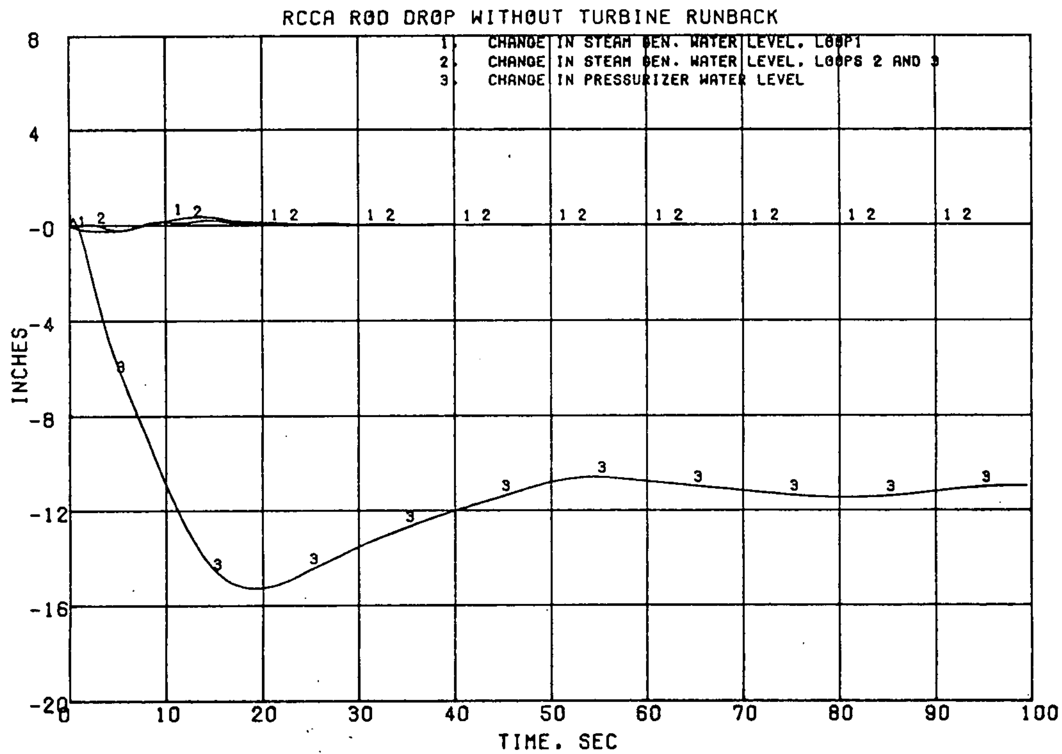


Figure 1—7



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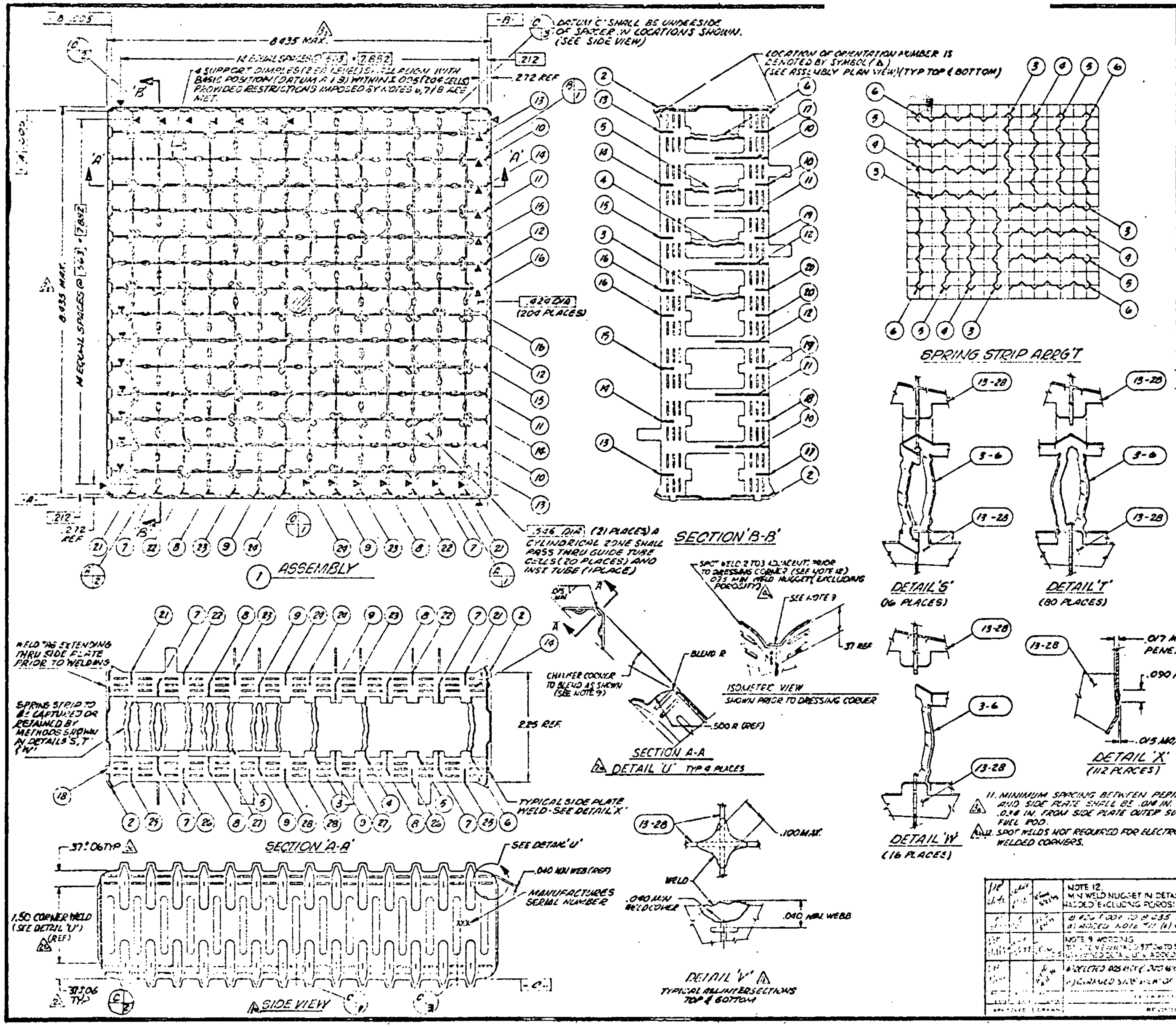


QTY.	PT. NO	DESCRIPTION	REMARKS
▼	1	ASSEMBLY	–
4	2	SIDE PLATE	301,735-OPT 2
4	3	SPRING STRIP	301,736 -1
4	4	SPRING STRIP	-2
4	5	SPRING STRIP	-3
4	6	SPRING STRIP	-4
2	7	STRUCTURAL LADDER	301,737- 7
2	8	STRUCTURAL LADDER	- 8
2	9	STRUCTURAL LADDER	- 9
2	10	STRUCTURAL LADDER	-10
2	11	STRUCTURAL LADDER	- 11
2	12	STRUCTURAL LADDER	- 12
2	13	INSIDE SPRING RETAINING STRIP	301,738 - 9
2	14	INSIDE SPRING RETAINING STRIP	- 10
2	15	INSIDE SPRING RETAINING STRIP	- 11
2	16	INSIDE SPRING RETAINING STRIP	- 12
2	17	INSIDE SPRING RETAINING STRIP	- 5
2	18	INSIDE SPRING RETAINING STRIP	- 6
2	19	INSIDE SPRING RETAINING STRIP	7
2	20	INSIDE SPRING RETAINING STRIP	8
2	21	OUTSIDE SPRING RETAINING STRIP	301,739 - 9
2	22	OUTSIDE SPRING RETAINING STRIP	- 10
2	23	OUTSIDE SPRING RETAINING STRIP	- 11
2	24	OUTSIDE SPRING RETAINING STRIP	- 12
2	25	OUTSIDE SPRING RETAINING STRIP	- 5
2	26	OUTSIDE SPRING RETAINING STRIP	- 6
2	27	OUTSIDE SPRING RETAINING STRIP	- 7
2	28	OUTSIDE SPRING RETAINING STRIP	- 8

NOTES (UNLESS OTHERWISE SPECIFIED)

1. TOLERANCES: .XX±.015 .XXX±.005
2. ALL SURFACE FINISH SHALL BE 63/AA IN ACCORDANCE WITH ANSI B46.1, 1962.
3. REMOVE ALL BURRS & BREAK SHARP EDGES
4. DIMENSIONS SHOWN SHALL APPLY AFTER ASSEMBLY.
5. DIMENSIONS AND TOLERANCES SHALL BE IN ACCORDANCE WITH ANSI Y 14.5, 1966
6. WITH NOMINAL ROD (.424), AGAINST SUPPORT DIMPLES, SPRINGS SHALL HAVE [illegible]
7. VERTICAL ALIGNMENT OF DIMPLES SHALL BE WITHIN .006
8. A CYLINDRICAL ZONE BOUNDED BY 2 SUPPORT DIMPLES & 1 BACK UP LOBE ON SPRING SHALL CLEAR A .432 MIN. DIA. ROD. ROD SHALL NOT BE INSERTED FULL LENGTH
9. AFTER FINAL ASSEMBLY DRESS CORNER REMOVE LEDGE AND BLEND SHARP CORNER
10. FUNCTIONAL INSPECTION PROCEDURES APPROVED BY PRODUCT DESIGN MAY BE USED ON ASSEMBLY TO DEMONSTRATE COMPLIANCE WITH NOTE 7, ASSEMBLY ENVELOPE & TRUE POSITION OF CELLS
11. MINIMUM SPACING BETWEEN PERIPHERAL ROD AND SIDE PLATE SHALL BE .014 IN. OR A NOM. .034 IN. FROM SIDE PLATE OUTER SURFACE TO FUEL ROD.
12. SPOT WELDS NOT REQUIRED FOR ELECTRON BEAM WELDED CORNERS.

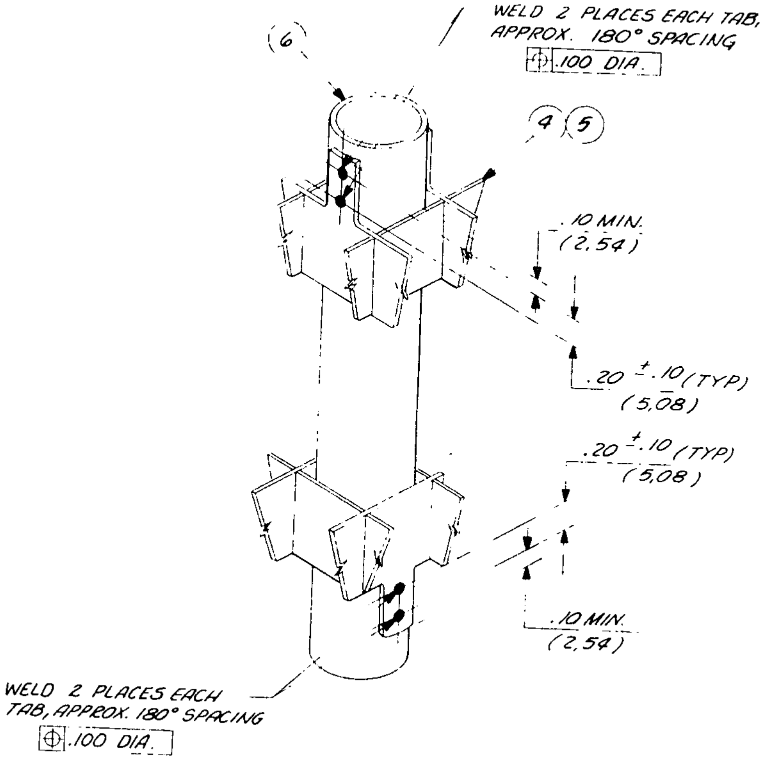
NOTE 12. MIN WELD NUGGET IN DETAIL U. (b) ADDED "EXCLUDING POROSITY" TO .035	4
[illegible] .035 MAX. (b) ADDED NOTE [illegible] (c) CHG DIM. FROM	3
NOTE 9 WORDING [illegible]	2
[illegible] .035 MIN [illegible]	1

EXXON NUCLEAR COMPANY, Inc.

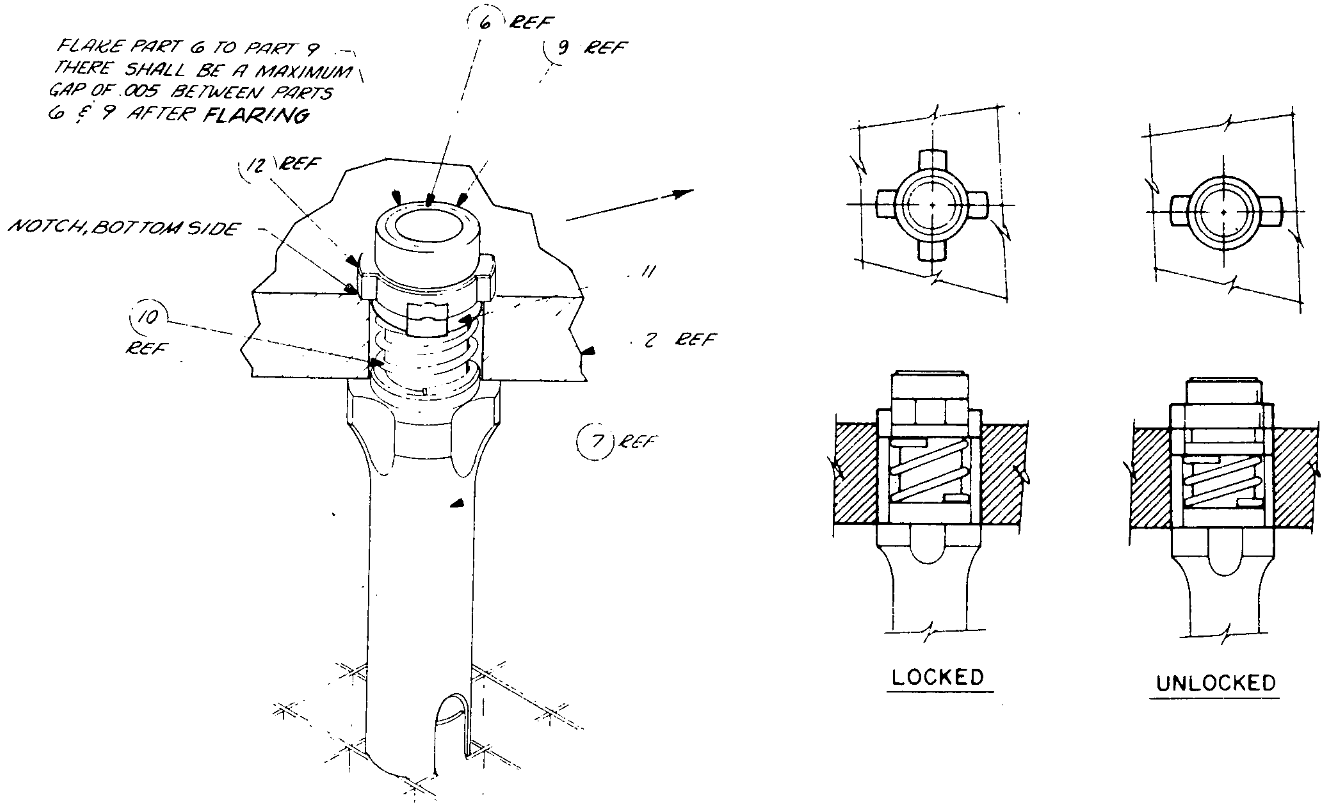
SPACER ASSEMBLY – TYPE A (NON VANED)

FULL SIZE XN-301,706

Figure 4—1

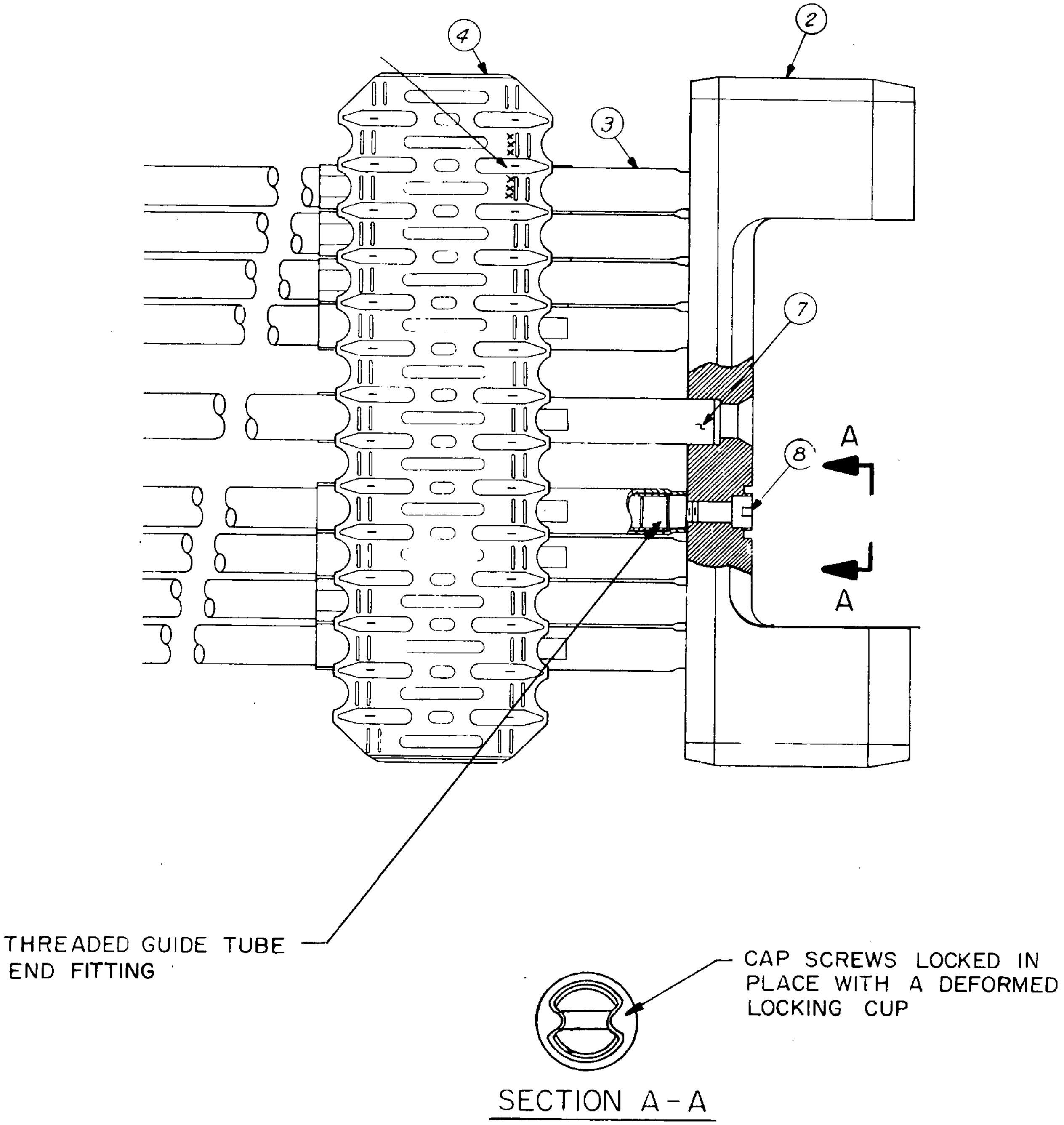


GUIDE TUBE TO SPACER ATTACHMENT



GUIDE TUBE TO UPPER TIE PLATE ATTACHMENT

Figure 5–1 (Sheet 1)



GUIDE TUBE TO LOWER TIE PLATE ATTACHMENT

Figure 5–1 (Sheet 2)

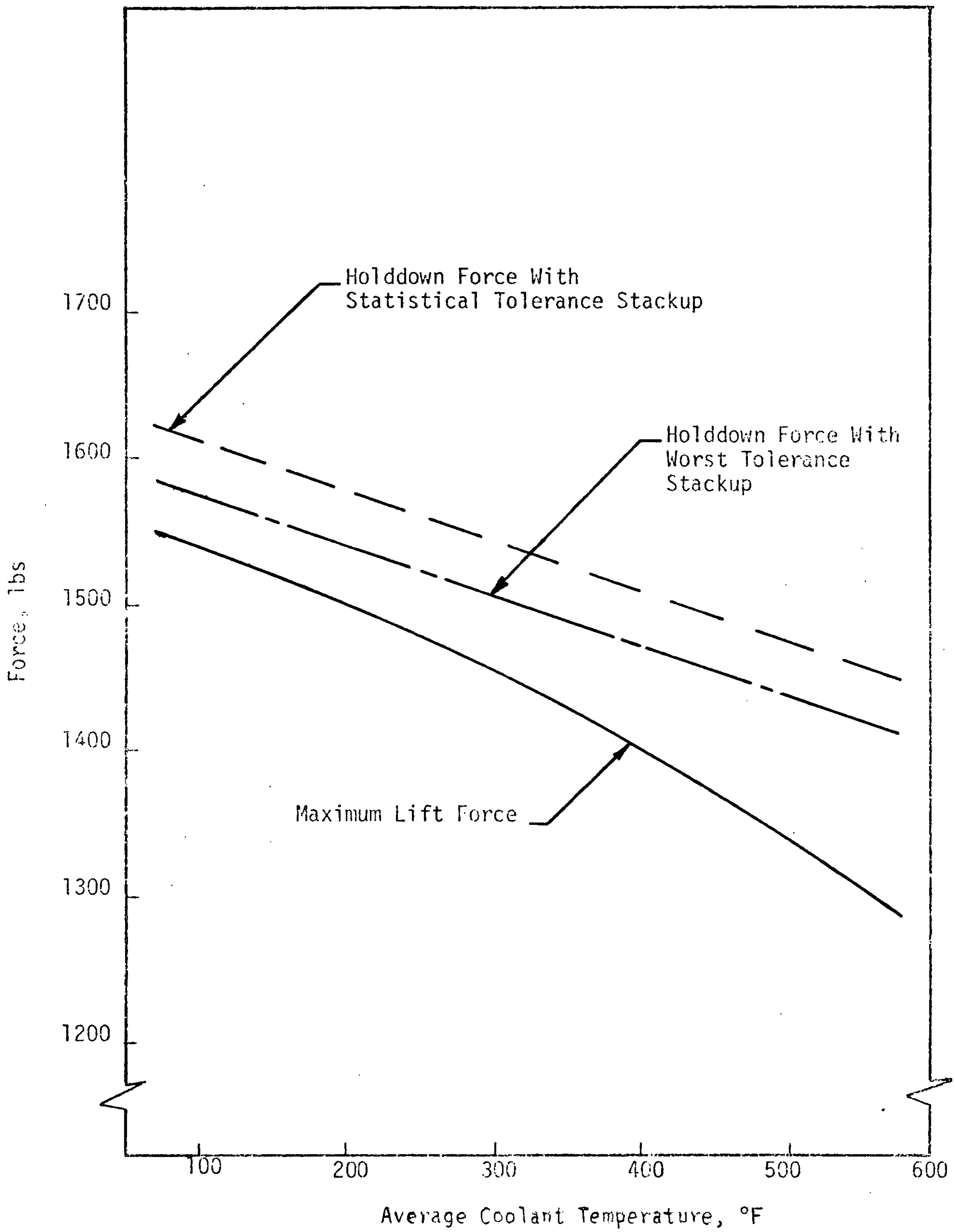

Holddown Force With
Statistical Tolerance Stackup
Holddown Force With
Worst Tolerance
Stackup
Maximum Lift Force
1700
1600
1500
1400
1300
1200
Force, lbs
100
200
300
400
500
600
Average Coolant Temperature, °F

Figure 9—1

Figure 19—1 PTSPWR---REACTIVITY INSERTION---DKSW = 0.40

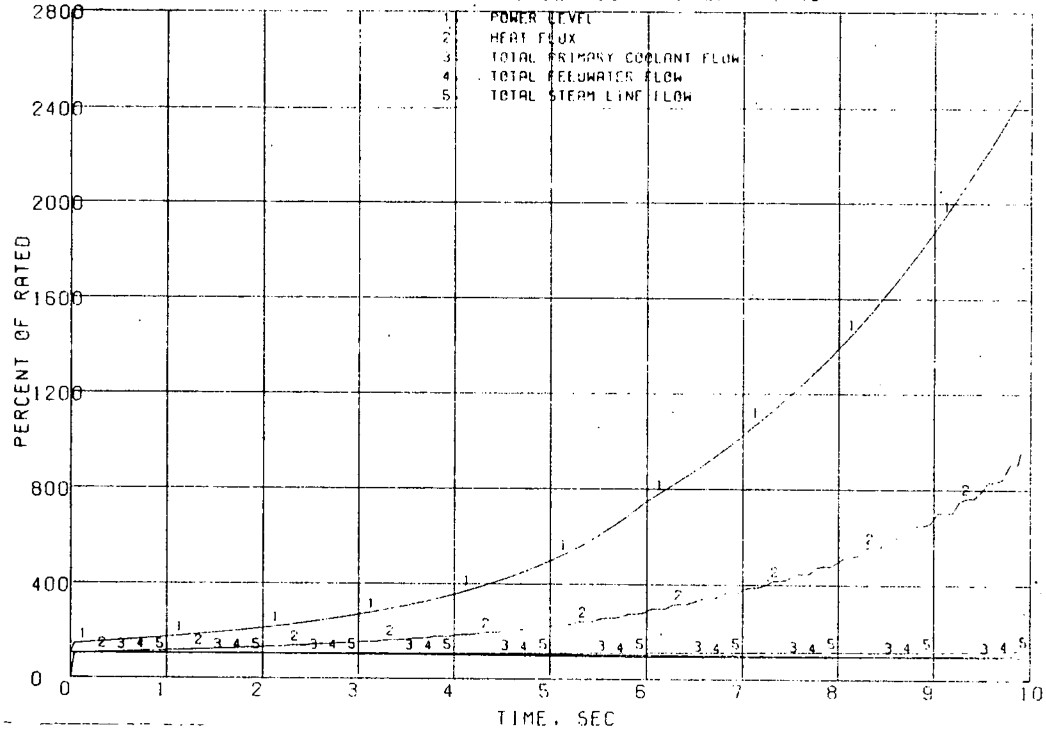


Figure 19—2 PTSPWR---REACTIVITY INSERTION---DKSW = -0.20