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INTERNAL CORE SPRAY LINE FLAW EVALUATION HANDBOOK FOR BROWNS FERRY UNITS 2 AND 3

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Prepared for

Tennessee Valley Authority

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1.Purpose/Objective

The objective of this report is to document the results of fracture mechanics evaluation of the core spray internal piping at Browns Nuclear Plant Units 2 & 3. The outcome of the fracture mechanics evaluation is a set of allowable flaw lengths at key locations in the core spray system. The evaluation also includes leak rate calculations for postulated through-wall indications and an assessment of it in relation to LOCA analysis assumptions. The overall package constitutes a flaw handbook that could be used to disposition any indications that may be detected during the inspection of core spray system piping at Units 2 &3.

The objective of Revision 1 of this report is to reevaluate the allowable flaw sizes using a less conservative method for determining the fluid drag loading during LOCA.

2. Methods

1. Reviewed the reference drawings, the seismic analyses and other information which indicated that the core spray system configurations for the Units 2 & 3 are essentially identical and the loadings are the same. The dimensional tolerances specified on the reference drawings are such that any variations within those values will have insignificant impact on the calculated stress values. One notable difference between the Unit 2 and 3 internal core spray line configurations is the presence of a repair in Unit 3 consisting of a set of installed stainless steel reinforcement brackets across the upper T-box on both core spray loops. A stress analysis and a safety evaluation of this repair was documented in References 1 and 2, respectively. The stress analysis demonstrated the adequacy of the repair. Therefore, a single stress analysis was conducted which is applicable to both the units. The only difference will be in the leak rate calculation where the leakage from the T-box indication will have to be included in the calculations for Unit 3.

2. Create an ANSYS (Reference 3) model for the core spray line. Determine the membrane and bending stresses considering the loadings identified in Section 4.

3. Determine applied stresses at several key locations in the piping system and use the limit load methods of Paragraph IWB-3640, Section XI, ASME Code (see References 4 and 5) as a guide to determine the allowable flaw lengths. It is GE's understanding that the Code of record for the Unit 3 is (or will be) 1989 Edition and that for Unit 2 is 1986 Edition. The internal core spray line is not a part of the

reactor pressure boundary and the Section C.4.4 of BFNP UFSAR states that the reactor internals were designed with Section III of the Code as a guide. Thus, the Section XI evaluation procedures are used as a guide. The 1986 Edition of the Code Section XI did not have the rules addressing flaws in both the flux and non-flux austenitic welds. Therefore, the rules of 1989 Edition of Section XI are used as a guide in determining the allowable flaw lengths.

4. Conduct leak rate evaluations.

3. Assumptions

1. The piping system geometry is as described in the reference drawings. The dimensional tolerances specified on the reference drawings are such that any variations within those values will have insignificant impact on the calculated stress values. It was also judged that any deviations between the as-built geometry and the geometry indicated in the reference drawings would not be significant in terms of stress analysis and the allowable flaw calculations.

2. The seismic inertia loads and anchor displacements are as determined in the reference report.

3. Any other assumptions are stated in the body of the report.

4. Design Inputs

The internal core spray piping is 6-inch schedule 40 and the material is Type 304 stainless steel per applicable GE specification 21A1056. Figure 1 shows a schematic of one of the loops of the internal core spray line. For the convenience of identification, the welds in Figure 1 have been arbitrarily numbered from 1 through 21. A finite element model consisting of one loop of the internal core spray piping was developed to determine the stresses from various design loads. Figure 2 shows a line plot of the finite element model.

The design inputs in this evaluation consisted of: (1) the geometry of the internal core spray line, (2) the applied loads. The geometry of the internal core spray line was obtained from the drawings listed in Reference 6. The applied loads on the core spray line consist of the following: deadweight, seismic inertia, seismic anchor displacements, fluid drag, loads due to flow initiation and thermal (and internal pressure) anchor displacements. Each of these loads are briefly discussed next.

4.1. Deadweight (DW)

The deadweight loading consists of the weight of the core spray pipe and the weight of the entrapped water. The metal weight was determined as 18.9 lbs/ft and the weight of the entrapped water as 12.5 lbs/ft. The stresses for this loading was calculated by applying one 'g' vertical acceleration in the finite element model of the core spray system. For flaw evaluation purposes, the stress from this loading is treated as primary.

4.2. Seismic Inertia

The seismic inertia loading consists of horizontal and vertical inertia forces acting on the entire core spray line due to seismic excitation of the RPV and the core shroud. The locations where the seismic excitation is imparted to the core spray line are the vessel nozzle, the support brackets and the points where it is attached to the shroud. Response spectrum method was used in applying the seismic inertia loading.

The information for the seismic spectra was obtained from References 7 and 8. These seismic reports used a damping of 0.5% for both the OBE and SSE cases. The ratio of SSE to OBE accelerations was specified as 2.0. Figure 3 shows the lumped-mass horizontal model of RPV and its internals. Node 83 on the RPV corresponds approximately to the core spray nozzle elevation. Similarly, node 67 on the shroud corresponds approximately to the location where the core spray system penetrates the shroud. The OBE horizontal response spectrum information in this case was available only at node 83 and is shown in Figure 4.

The spectrum in Figure 4 is at the core spray nozzle. The spectrum information at the shroud attachment point of the internal core spray line is unavailable. However, the seismic analysis report (Reference 7) gives the zero period acceleration (ZPA) at the shroud attachment point as 0.47g versus a ZPA value of 0.35 at the RPV nozzle. This means that the peak spectrum acceleration levels at shroud attachments points would be higher than those at the nozzle. Therefore, it was assumed that the spectrum at the shroud attachment points would be essentially the same as at the RPV nozzle, except that all of the acceleration values would be higher by a factor of (0.47/0.35) or 1.34. Therefore, all of the acceleration values in Figure 4 were multiplied by 1.34 and the resulting values were used in the horizontal OBE inertia analysis. The acceleration values were assumed the same for both the East-West and the North-South directions.

Figure 5 shows the lumped-mass vertical model of RPV and its internals (Reference 8). In this model, node 53 on the RPV corresponds approximately to the core spray nozzle elevation and node 50 on the shroud corresponds approximately to the location where the core spray system penetrates the shroud. In this case also, the OBE spectrum information is available only at node 53 and is shown in Figure 6. It was judged that the vertical spectrum for nozzle in Figure 6 can also be used for the shroud attachment points.

For the purpose of specifying load combinations, the following designations are used:

Operating Basis Earthquake Inertia - X Direction: OBEIX
Operating Basis Earthquake Inertia - Z Direction: OBEIZ
Operating Basis Earthquake Inertia Vertical - Y Direction: OBEIY
Safe Shutdown (or design Basis) Earthquake Inertia - X Direction: SSEIX
Safe Shutdown (or design Basis) Earthquake Inertia - Z Direction: SSEIZ
Safe Shutdown (or design Basis) Earthquake Inertia - Y (Vertical) Direction: SSEIY

For the flaw evaluation purposes, the stresses from the seismic inertia loading are treated as primary.

4.3. Seismic Anchor Displacement

Seismic anchor displacements are applied at the attachment points of the core spray lines at the RPV and the shroud. The following OBE condition anchor motions were obtained from the horizontal seismic analysis report (Reference 7):

RPV Nozzle(in.)	Shroud(in.)
0.1	0.13

The RPV and shroud displacements were applied in opposite directions to obtain conservative stress results. The above displacements were doubled for the SSE case. For flaw evaluation purposes, the stresses from the seismic anchor displacement loading are treated as secondary. The load case designations used are the following:

Operating Basis Earthquake Displacement - X Direction: OBEDX
Operating Basis Earthquake Displacement - Z Direction: OBEDZ
Design Basis Earthquake Displacement - X Direction: SSEDX
Design Basis Earthquake Displacement - Z Direction: SSEDZ

To address the question as to the effect of shroud cracking on the above stated seismic anchor displacements, the information provided in Reference 9 was reviewed. This review indicated that the cracking was minor (less than 10 inches of total length at any horizontal weld) in both the Unit 2 and 3

shrouds except at weld H5 of Unit 3 where the total length of cracking was 81.7 inches. Given that stress analysis is treating the RPV and shroud anchor motions in a conservative manner (i.e., the two displacements are applied in the opposite direction) and that the shroud cracking is relatively minor, it was judged that there is no need to modify the values of seismic anchor motions obtained from Reference 7.

4.4. Fluid Drag

The drag loads consist of the forces resulting from the fluid flow past the core spray line. The flow in the annulus region during the normal operation exerts some downward drag force on the core spray piping. The magnitude of this loading was determined to be approx. 9.5 lbs/ft assuming a conservative value of 5 ft/second for the fluid velocity in the vessel annulus region. During the Upset condition, core spray operation is assumed (no feedwater flow) and, therefore the drag loads are insignificant. During a postulated double-ended break of either the recirculation line or the main steam line, the drag loads on the core spray line were determined to be significant. Reference 10 states that the fluid drag loading on the core spray line during the recirculation line LOCA (limiting case), can be simulated by applying a 0.5 psi downward pressure on the upper curved section and 5.0 psi downward pressure on the lower horizontal section (near shroud penetration). Therefore, this loading was used to calculate the LOCA drag loads. The drag loads are treated as primary loads for the flaw evaluation purposes and are designated as follows:

Drag Load During Normal Operation: DRG1
Drag Loads During LOCA Condition: DRG2

4.5. Core Spray Injection Loading (CSIN):

Two types of loads result when the core spray flow is initiated: internal pressure and the axial loads due to flow. During normal operation, the pressure differential between the inside and the outside of the core spray line is essentially negligible. A bounding internal pressure value of 150 psi was assumed. This is the pressure applied by the core spray pumps in the 'runout' flow condition. The membrane stress due to this internal pressure was calculated using the strength of material formulas.

The thermal sleeve at Browns Ferry 2/3 is slip fit and thus initiation of core spray flow is expected to produce axial loading at the T-box. The internal pressure used for this calculation is that during the full flow of the system and this value is obtained from the Browns Ferry specific process diagram (Reference 11). A pressure value of 79 psi was derived from the ΔP head information on the process diagram.

Deviations between the as-built geometry and the geometry indicated in Reference 10 would not be significant. No consideration was given to post fabrication modifications.

Stresses due to water hammer loads are insignificant since the core spray inlet valve ramps open over a period of time upon system actuation. Additionally, the piping is full of water during actuation due to the presence of the vent hole on the top of the T-box.

4.6. Thermal loads

The two anchor points of the internal core spray line (the core spray nozzle and the brackets on the vessel at one end and the shroud attachment points at the other end) grow vertically and horizontally at different rates due to differences in the materials (low alloy steel for the vessel versus stainless steel for the shroud). Also, these displacements are expected to vary during certain transients due to the differences in temperatures between the vessel and the shroud. The loads produced by these thermal anchor displacements and thermal expansion are treated as secondary. The internal pressure in the vessel also produces vertical direction (Y) anchor motion at the nozzle and the brackets. This displacement was included along with the thermal anchor displacements. The following thermal load cases need to be considered:

Thermal displacements during Normal Steady State Operation: THN
Thermal displacements during Loss of Feedwater Pump transient: LFWP
Thermal displacements during LOCA: LOCAD

The LOCA thermal displacements may consist of several sub-cases. One of the case when the core spray is just initiated following the LOCA event. Another sub-case may be several hours following the LOCA event. The only difference between the various LOCA sub-cases would be the assumed temperatures for the vessel, the shroud, the annulus region and the core spray piping.

The RPVs of BF2 & 3 have stilts as part of the core support structure. The stilt material is nickel-chrome-iron. The stilt length figures into the thermal displacement calculations. The calculated values of differential thermal displacements for the various transient conditions are the following:

Operat. Cond./ Transient	Temperatures (°F)			RPV Press. (psi)	Displacements (in.)		Pipe Temp (°F)
	RPV	Shroud	Stilts		Horz.	Vert.	
THN	522	522	522	1000	0.502	0.065	522
LFWP	300	400	100	665	0.270	0.143	300
LOCA1	522	534	522	35	0.407	0.065	201
LOCA2	522	281	281	35	0.407	0.732	201

The temperatures and pressures stated in the above table are derived from the information contained in the RPV thermal cycle drawing (Reference 12).

5. Load Combinations & Stress Levels

This section describes the manner in which the various loads were combined for the purpose of obtaining stress levels for flaw evaluation. The load combinations used are consistent with Section C.4.4 of the Browns Ferry UFSAR on Reactor Vessel Internals. The limiting stress levels in three critical areas are then summarized.

5.1. Load Combinations

Although the internal core spray piping is not a part of reactor pressure boundary, in many of the later generation BWRs it was designed in accordance with USAS B31.1 Piping Code rules and supplemental GE specifications. The B31.1 Code does not per say identify 'primary' and 'secondary' loads or stresses but the flaw evaluation methodology to be used makes the distinction between these two categories by specifying different safety factors. Also, the flaw evaluation methodology makes the distinction between the normal/upset (Level A/B) condition loads, for which the factor of safety is 2.8, and the emergency/faulted (Level C/D) condition loads, for which the safety factor is 1.4.

The following set of load combinations were considered for the evaluation of normal/upset condition:

(1) DW(P) +DRG1(P) + THN(S)
(2) DW(P) + DRG1(P) + CSIN(P)
(3) DW(P) + DRG1(P) + LFWP(S)
(4) DW(P) + DRG1(P) + OBEIX(P) + OBEDX(S) + OBEIY(P)

(5) DW(P) + DRG1(P) + OBEIZ(P) + OBEDZ(S) + OBEIY(P)
(6) DW(P) + LOCAD(S)

Note that the letter in the parenthesis indicates whether a load is primary or secondary as defined by the ASME Code. Set of load combinations used for the Emergency/Faulted conditions are the following:

(1) DW(P) + DRG1(P) + SSEIX(P) + SSEIY(P) + SSEDX(S)
(2) DW(P) + DRG1(P) + SSEIZ(P) + SSEIY(P) + SSEDZ(S)
(3) DW(P) + DRG2(P)
(4) DW(P) + CSIN(P) + LOCAD(S) + SSEIX(P) + SSEIY(P) + SSEDX(S)
(5) DW(P) + CSIN(P) + LOCAD(S) + SSEIZ(P) + SSEIY(P) + SSEDZ(S)

The LOCAD loads need not be included in the emergency/faulted combination no. 3 since these displacements-controlled loadings develop much later in time when the drag loads due to LOCA have been reduced to insignificant level.

5.2. Calculated Stress Levels

The forces and moments at various nodes in the model for all of the load sources were calculated using the ANSYS finite element code [Reference 3]. These forces and moments were then combined to obtain the total forces and moments for a given load combination. Thus, for each load combination and each node, a set of forces and moments were obtained. Furthermore, within each set, the forces and moments from the displacement-controlled loadings were tabulated separately for the calculation of P_e stress. As described later, the flaw evaluation methodology uses the primary membrane (P_m), primary bending (P_b) and the expansion stress (P_e).

Based on a review of the stress levels throughout the piping system, all of the welds in the system were grouped in five categories for the purpose of allowable flaw evaluations (see Figure 1): welds 10 and 12, (2) welds 9 and 13, (3) welds 8 and 14, (4) welds 4, 5, 6, 7, 15, 16, 17 and 18, and (5) welds 1, 3, 19 and 21. The calculated envelope values of P_m, P_b and P_e stress levels at these five locations are summarized below for the governing normal/upset and emergency/faulted condition load combinations:

Weld ID (Figure 1)	Governing Oper. Cond.	Gov. Load Comb.	P_m (psi)	P_b (psi)	P_e (psi)
10,12	Emer./Fault.	Note 1	1125	12680	4455
9,13	Emer./Fault.	Note 1	930	3070	4320
8,14	Emer./Fault.	Note 1	1030	2925	4360
4,5,6,7, 15,16,17,18	Emer./Fault.	Note 1	1055	4660	1320
1,3,19,21	Emer./Fault.	Note 1	980	6890	180

Note 1: load comb. DW(P) + CSIN(P) + LOCAD(S) + SSEIZ(P) + SSEIY(P) + SSEDZ(S)

The stress levels in the preceding table were used in the allowable flaw evaluations as described in the next section.

6. FRACTURE MECHANICS EVALUATION

The limit load methodology was used in calculating the allowable flaw lengths. This methodology is first described followed by the results of allowable flaw evaluations.

6.1. Limit Load Methodology

Consider a circumferential crack of length, $l = 2R\alpha$ and constant depth, d. In order to determine the point at which limit load is achieved, it is necessary to apply the equations of equilibrium assuming that the cracked section behaves like a hinge. For this condition, the assumed stress state at the cracked section is as shown in Figure 7 where the maximum stress is the flow stress of the material, σ_f. Equilibrium of longitudinal forces and moments about the axis gives the following equations:

$$\beta = [(\pi - \alpha d/t) - (P_m/\sigma_f)\pi]/2 \qquad (1)$$

$$P_b' = (2\sigma_f/\pi)(2 \sin \beta - d/t \sin \alpha) \qquad (2)$$

Where, t = pipe thickness, inches
α = crack half-angle as shown in Figure 6
β = angle that defines the location of the neutral axis
Z= Weld type factor
P_e= Piping expansion stress
P_m = Primary membrane stress

P_b = Primary bending stress
P_b^* = Failure bending stress

The safety factor is then incorporated as follows:

$$P_b^* = Z*SF\,(P_m + P_b + P_e/SF) - P_m \qquad (3)$$

The P_m and P_b are primary stresses. P_e is secondary stress and includes stresses from all displacement-controlled loadings such as thermal expansion, seismic anchor motion, etc. All three quantities are calculated from the analysis of applied loading. The safety factor value is 2.77 for normal/upset conditions and 1.39 for emergency/faulted conditions.

Z Factor

The test data considered by the ASME Code indicated that the welds produced by a process without using a flux had fracture toughness as good or better than the base metal. However, the welds produced by a process using the flux had lower toughness. To account for the reduced toughness of the flux welds (as compared to non-flux welds) the Section XI procedures prescribe a penalty factor, called a 'Z' factor. The examples of flux welds are submerged arc welds (SAW) and shielded metal arc welds (SMAW). Gas metal-arc welds (GMAW) and gas tungsten-arc welds (GTAW) are examples of non-flux welds. Figure IWB-3641-1 may be used to define weld-base metal interface. The expressions for the value of Z factor in Appendix C are given as the following:

$$Z = 1.15\,[1 + 0.013(OD-4)] \text{ for SMAW}$$
$$= 1.30\,[1 + 0.010(OD-4)] \text{ for SAW}$$

where, OD is the nominal pipe size (NPS) in inches. The procedures of Appendix C recommend the use of OD = 24 for pipe sizes less than 24-inches. This approach is very conservative and, therefore, the use of actual NPS was made in calculating the 'Z' factor. This approach is considered reasonable as recent discussions in the Section XI Code Working Group on Pipe Flaw Evaluation indicate that for small diameter pipes, such as the 6-inch diameter core spray piping, the Z-factor may be close to or less than 1.0. Nevertheless, the allowable flaw size calculations were conducted using both the approaches (i.e., assuming a nominal diameter of 6-inches and 24-inches).

The flux-type welding process used in the field was shielded metal arc type (SMAW). therefore, the Z-factors by the two approaches are:

$Z_{6\text{-inch}} = 1.15[1 + 0.013(6-4)] = 1.18$

$Z_{24\text{-inch}} = 1.15[1 + 0.013(24-4)] = 1.45$

If the indication is located in the base metal or near the non-flux weld, Z is assumed as 1.0 and the P_e stresses are not used in the calculation, consistent with Appendix C guidelines.

Typically a search of the QC records is required to ascertain whether a weld at any location in the core spray line was completed using a flux or non-flux process. A search for Browns Ferry Units conducted by GENE produced the following information. The vendor for the internal core spray line (piping between the thermal sleeve and the sliding sleeve in the vertical riser) was Murdock. The weld procedure submitted by vendor (as required by GE specification 21A1056) is contained in VPF2670-1-1 and is a manual TIG non-flux weld process. Since this was the only procedure submitted by the vendor, it is reasonable to conclude that all of the shop welds in the internal core spray lines (piping between the thermal sleeve and the sliding sleeve in the vertical riser) were completed using this weld procedure.

A review of the Browns Ferry Construction Records conducted by TVA engineering indicated that both Gas Tungsten Arc Welding (nonflux) and/or Shielded Metal Arc Welding (flux) procedures were used in the field welds of the internal core spray piping.

For welds in the portion of the internal core spray line that is part of the shroud assembly, no definitive information was available to determine whether a flux or nonflux type weld procedure was used. For information purposes, all of the allowable flaw calculations were conducted considering both the flux and non-flux cases.

6.2. Allowable Flaw Length Calculation

The stresses from the table in the preceding section were utilized to determine the acceptable through-wall flaw lengths. The acceptable flaw size was determined by requiring a safety factor. The flow stress was taken as $3S_m$ (S_m=16.9 ksi for Type 304 stainless steel at 550°F). As specified in Reference 4, safety factors of 2.8 for the normal/upset conditions and 1.4 for the emergency/faulted conditions, respectively, were used. Since the stress analysis considered the welds at the coupling to be intact, the allowable flaw lengths were independent of the engagement length. The calculated values of the allowable flaw lengths are tabulated next.

Weld ID (Figure 1)	Weld Type	Total Allowable Effective Crack w/o Crack Growth Added		Total Allowable Effective Crack with One Cycle of Crack Growth	
		Angle (deg.)	Length (in.)	Angle (deg.)	Length (in.)
10,12	*non-flux*	150.0	8.7	130.0	7.5
	flux, Z= 1.18	119.0	6.9	99.0	5.7
	flux, Z= 1.45	98.0	5.7	78.0	4.5
9,13	*non-flux*	224.0	12.9	202.0	11.7
	flux, Z= 1.18	186.0	10.7	164.0	9.5
	flux, Z= 1.45	173.0	10.0	152.0	8.8
8,14	*non-flux*	224.0	12.9	202.0	11.7
	flux, Z= 1.18	186.0	10.7	164.0	9.5
	flux, Z= 1.45	173.0	10.0	152.0	8.8
4,5,6, 7,15,16, 17,18	*non-flux*	207.0	12.0	187.0	10.8
	flux, Z= 1.18	190.0	11.0	169.0	9.8
	flux, Z= 1.45	177.0	10.2	156.0	9.0
1,3,19,21 (including sleeve to pipe welds)	*non-flux*	190.0	11.0	169.0	9.8
	flux, Z= 1.18	179.0	10.3	157.0	9.1
	flux, Z= 1.45	165.0	9.5	144.0	8.3

The allowable crack lengths given in column 4 do not include the expected crack growth during the next inspection interval. If a crack growth of 1.2 inches is used based on the reinspection at the end of a 18-month fuel cycle, the resulting allowable crack lengths are tabulated in the last column. The basis for the 1.2 inch crack growth is discussed in the next Subsection.

6.3. Crack Growth Evaluation

Prior crack growth analyses performed for BWR shroud indications have conservatively used a crack growth rate of 5×10^{-5} inch/hot hour.

The stresses induced in the core spray line are very low, as evidenced by the stress results presented in the previous section. Those stress results also conservatively include the effects of seismic and core spray

injection loads, which are not typically present. Therefore, the applied stress intensity factor is low, and the corresponding crack growth rate would be significantly below the upper bound value of $5x10^{-5}$ inch/ hot hour used here.

Pre-operational testing of BWR internals has demonstrated that high cycle fatigue resulting from flow induced vibration is not a concern for the core spray piping. Additionally, low cycle fatigue caused by assumed thermal transients which could be potentially imposed by cold fluid injections through the feedwater spargers located directly above the core spray lines have been found to be insignificant. Therefore, fatigue crack propagation of indications in the core spray lines is concluded to be negligible, and is not considered to be a further contributor to the crack growth values discussed here.

Thus, a conservative crack growth rate of $5x10^{-5}$ in/ hot hr can be used in the flaw evaluations. This crack growth rate translates into a crack length increase of (8000 hrs per year $x1.5x5x10^{-5}$) or 0.6 inch at each end of an indication assuming a 18-month fuel cycle. Thus, the projected length, l_f of any indication whose current length at the time of inspection is, l_p, would be (l_p +0.6x2) inches. A factor of 2 in the preceding parenthesis is to account for the growth at each end of the indication.

7. Leakage Evaluation

7.1. Leak Rate Calculation Methodology

The leakage from the core spray line into the RPV annulus could come from a number of sources such as through the 1/4 inch vent hole at the top of the T-box, through the gap between the sleeve and the nozzle ID, and through the presence of any through-wall cracks in the piping.. The leakage rate through the vent hole was estimated assuming incompressible Bernoulli flow through the hole:

$$Q = CA\sqrt{2g_c\Delta P/\rho} \qquad (4)$$

where, Q = Leakage

C = flow coefficient (assumed to be 0.6 for an abrupt contraction as in the case of vent hole)

A = area

ρ = mass density of fluid

ΔP = pressure difference across the pipe/vent

A ΔP value of 79 psi based on Reference 11 was used. Essentially a similar approach was used in Reference 13 to calculate the leakage from the slip-fit region of the thermal sleeve during core spray operation. The calculated leak rate in Reference 13 was 148 gpm considering the worst case combination of slip-fit tolerances.

Leak rate from the through-wall indications in the core spray line can also be estimated using the preceding equation with the value of flow coefficient, C, assumed as 1.0. A key input needed is the crack opening area, A.

The approach used in this evaluation to calculate the value of A, was to assume a conservative value of crack opening displacement, δ, and assume the crack opening configuration to be like a rectangular slot with one side being the crack length, 2a, and the other side as the crack opening displacement. A value of 0.01 inch was assumed for δ. Linear elastic fracture mechanics calculations indicated this assumed value of δ to be conservative. The crack opening area is then simply:

$$A = 2a(\delta) \qquad (5)$$

7.2. Overall Leak Rate Calculation

The leak rate was calculated using an internal pressure of 79.0 psi, which is the upper bound value of steady state pressure during the core spray operation for this plant. Using this value as the ΔP and equation (4) gave a 11.7 gpm leak rate from the vent hole.

The leak rates from any indications would be a function of the detected number and lengths of the indications which will be known only after an examination of the internal core spray piping has been conducted. To facilitate this calculation after the examination results are in, leak rate per inch of crack length is provided herein. This leak rate was calculated as 4.0 gpm per inch of crack length. It should be noted that the crack length to be used for leak rate calculation should include expected crack growth during the next inspection interval.

For Unit 2, additional leakage could result if an access hole cover repair is installed. The leakage during the post-LOCA condition from such a repair was calculated to be 78 gpm (Reference 14). Leakage from other sources such as future shroud repairs should be accounted for if implemented at a later date.

7.3. Impact Assessment on LOCA Analysis

The current SAFER/GESTR LOCA analysis for BF 2 and 3 plants considered a 10% (or 650 gpm) reduction in the core spray flow and showed that peak clad temperature (PCT) limits were satisfied (Reference 15). A fraction or all of this reduction may be available to offset the calculated leakage in the preceding sub-section. Calculations indicate that the Browns Ferry Units are "spray pattern tolerant", thus, minor leaks from the sparger piping will not invalidate the conclusions of the SAFER/GESTR analysis. The fuel types considered in the SAFER/GESTR analysis were: BP8x8R/P8x8R, GE11 and GE8x8NB.

8. References

[1] "Core Spray Line T-Box Modification Design Stress Report for Browns Ferry 3", GE Report No. RDE-50-0692, Rev. 0, DRF No. B13-01653, July 1992.

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[3] DeSalvo, G.J., Ph.D. and Swanson, J. A., Ph.D., ANSYS Engineering Analysis System User's Manual, Revision 4.4, Swanson Analysis Systems, Inc., Houston, PA, May 1, 1989.

[4] ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components, American Society of Mechanical Engineers, 1989 Edition, Paragraph IWB 3640.

[5] Ranganath, S. and Mehta, H. S., "Engineering Methods for the Assessment of Ductile Fracture Margin in Nuclear Power Plant Piping," Elastic-Plastic Fracture: Second Symposium, Volume II - Fracture Resistance Curves and Engineering Applications, ASTM STP 803, C.F. Shih and J. P. Gudas, Eds., American Society for Testing and Materials, 1983, pp. II-309 - II-330.

[6] Browns Ferry Shroud Drawings: (a) Reactor Drawing No. 104R935, (b) Reactor Vessel Purchase Part drawing No. 886D499, (c) Core Spray Line Drawing No. 920D824, (d) Shroud Drawing No. 729E458, (e) Nozzle Thermal Cycles Drawing No. 135B9990.

[7] Summary Report on Small task G056.212: Recalculation of Seismic Responses for Reactor Building Drywell and Internals of Browns Ferry Nuclear power Plant Using El Centro Time History Input (Revised Damping Values, Prepared for TVA by Bechtel, August 1989.

[8] "Master Acceleration Response Spectra (MARS) Report for Seismic Class I Structures," TVA Report No. CEB 88-05-C R1, Volume 5.

[9] "Browns Ferry Nuclear plants 1,2, and 3 Safety Evaluation of Response to generic Letter 94-03," Letter from J.F. Williams of USNRC to O.D. Kingsley, Jr. of TVA, dated January 13, 1995.

[10] GE Procedure No.Y1003B1017, Rev. 0, BWR/4&5 Core Spray Line New Loads Calculation Methods.

[11] "Process Diagram - Core Spray System," GE Drawing No. 161F257, Rev. 9.

[12] "Reactor Thermal Cycles," GE Drawing No. 729E762.

[13] Stress Report VPF-6033-003-(4), Browns Ferry Core Spray Thermal Sleeve/ Safe End Replacement, 1977.

[14] "BF2 Replacement Access Hole Cove Stress Analysis," GE Report No. GE-NE-B1301795-01, December 1995.

[15] "Browns Ferry Nuclear Plant, Units 1,2 and 3 - Safer/GESTR-LOCA Loss-of-Coolant Accident Analysis," GE Report No. NEDC-3248P, February 1996.

9. Units

English units (inches, ksi, psi, ksi√in) are used.

10. Summary & Conclusions

A flaw evaluation, consisting of stress and fracture mechanics analyses of the internal core spray piping of Browns Ferry Units 2 & 3 was conducted with a view to develop a flaw disposition handbook. The procedures of Paragraph IWB-3640, ASME Section XI, were used as a guide in determining the allowable flaw lengths. Allowable flaw lengths were calculated at five critical locations and a leak rate calculation methodology was presented. The methodology presented in this report can be used to disposition any indications detected during the future inspections of the internal core spray lines at Units 2 & 3.

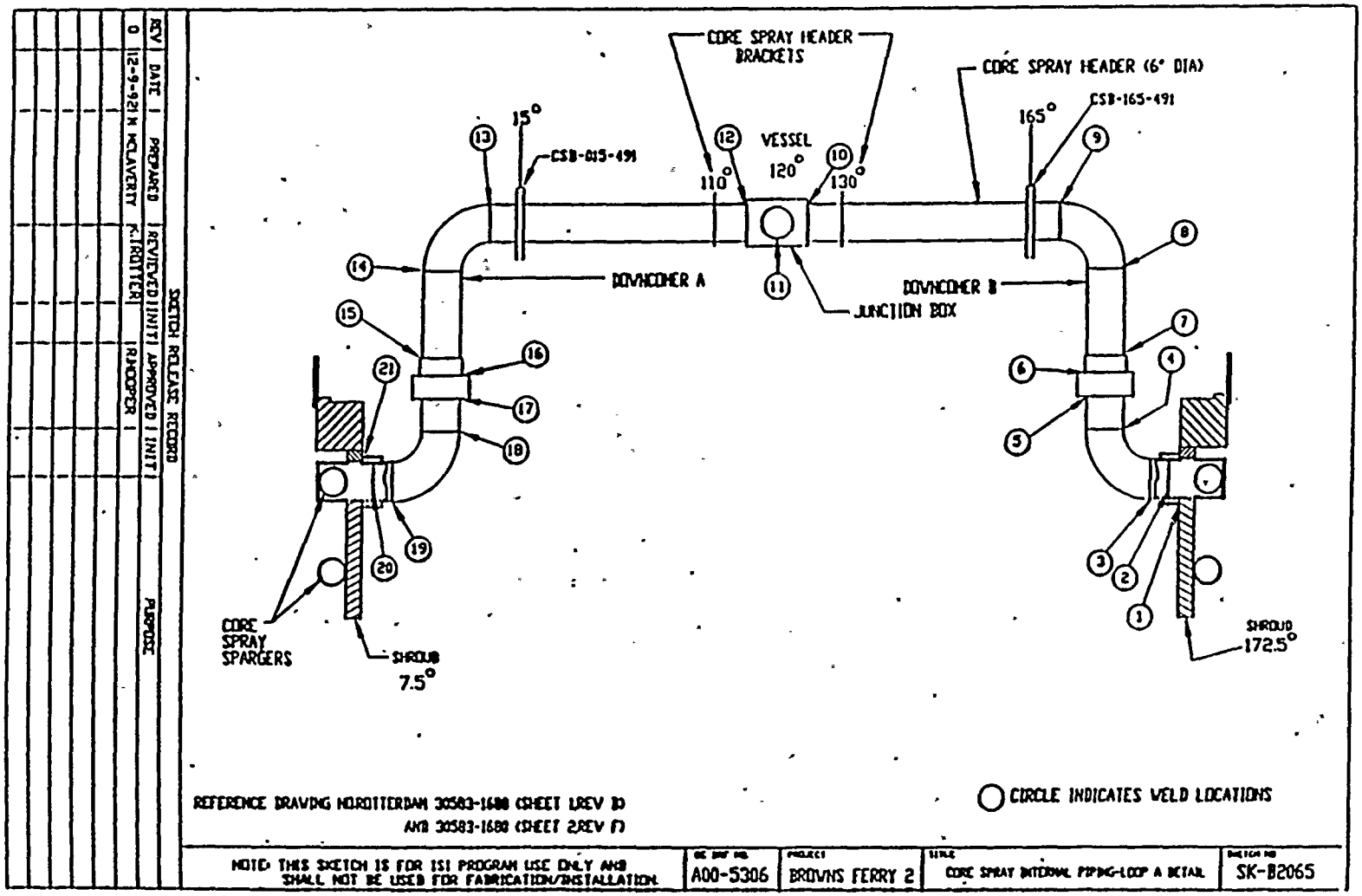


Figure 1 A Schematic of BF 2/3 Internal Core Spray Piping Configuration

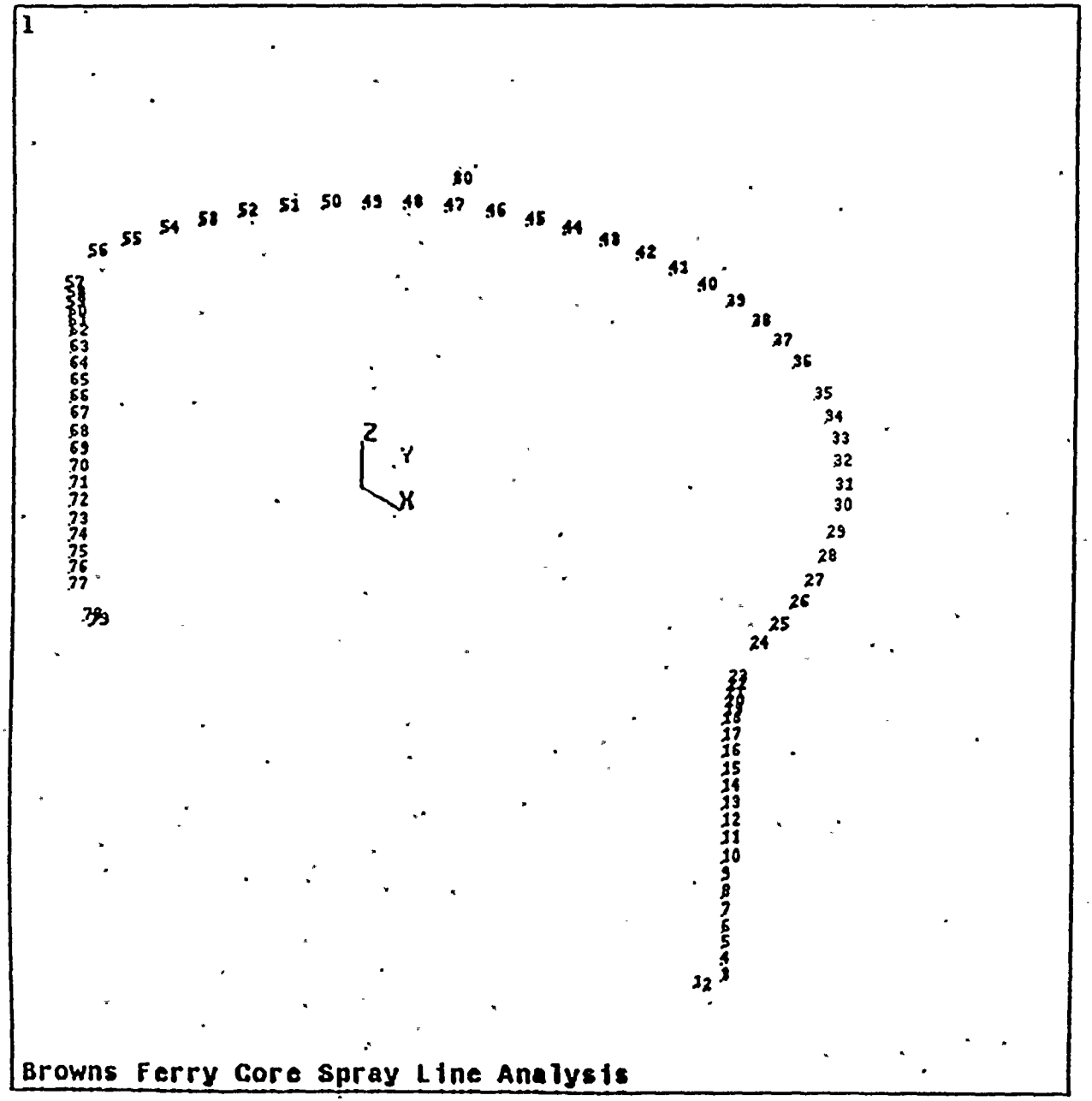


Figure 2 ANSYS Model of the Browns Ferry 2/3 Core Spray Line

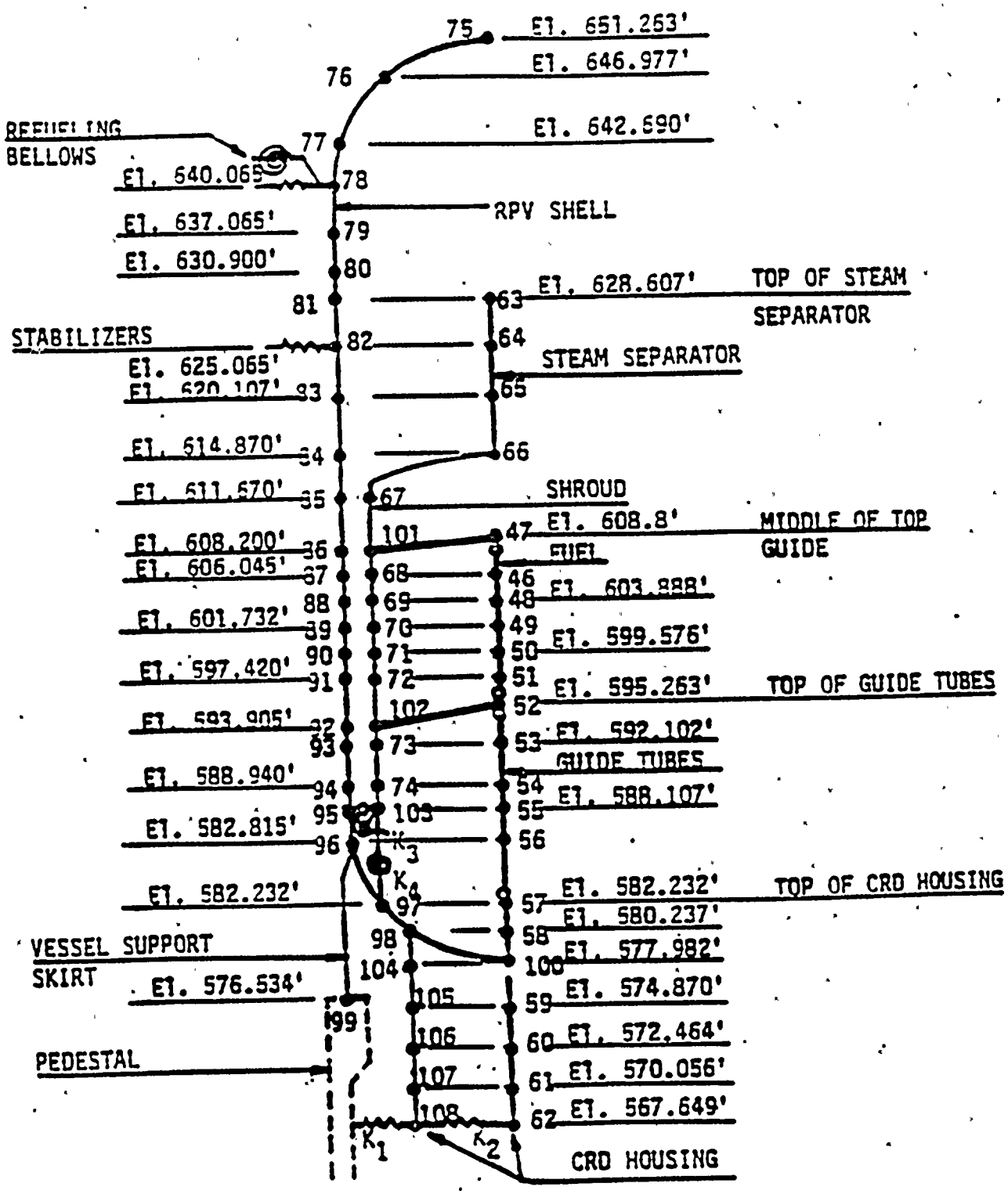
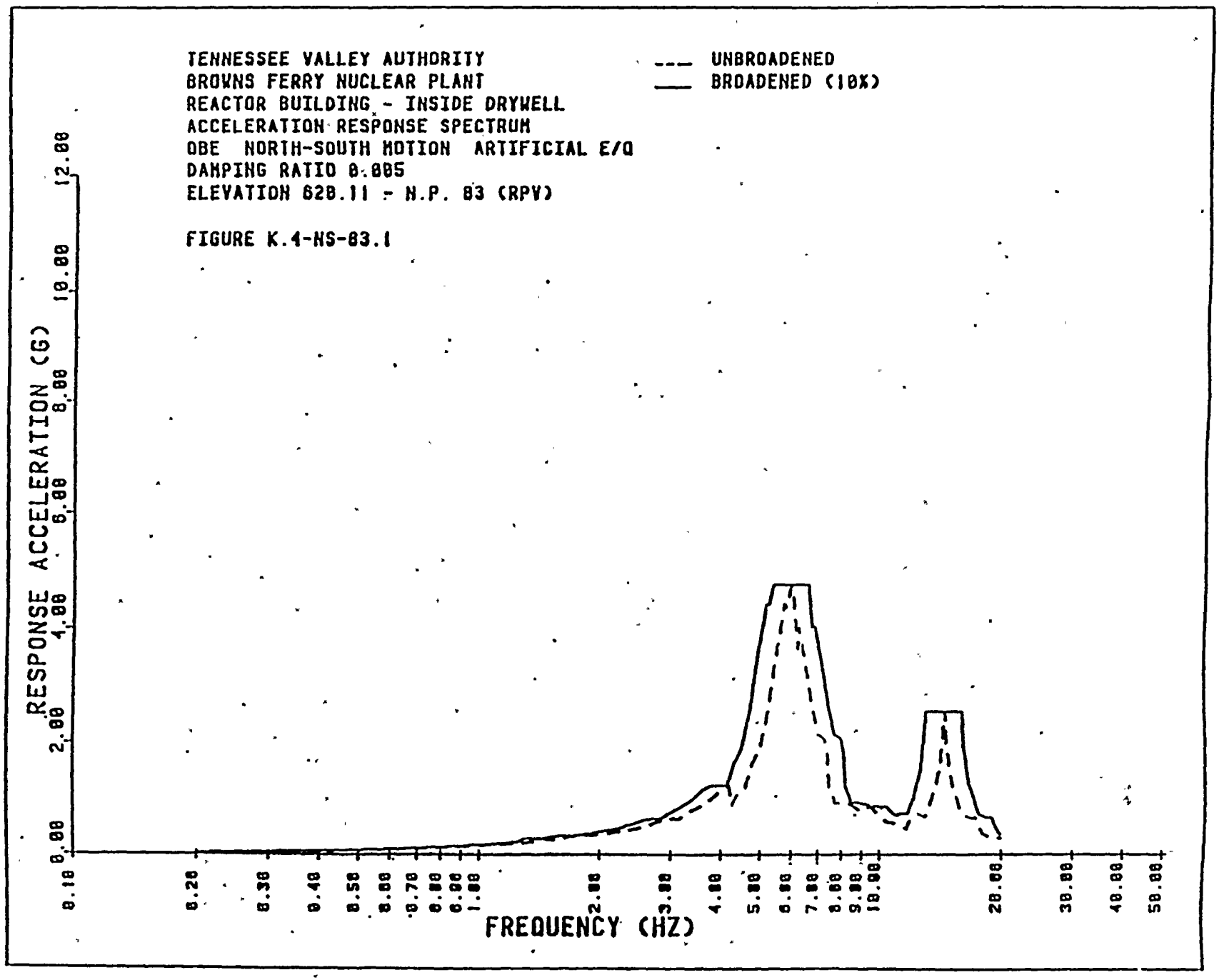


Figure 3 Horizontal Model of BF 2/3 RPV and Its Internals



Acceleration Spectrum at Node 83 of Figure 3

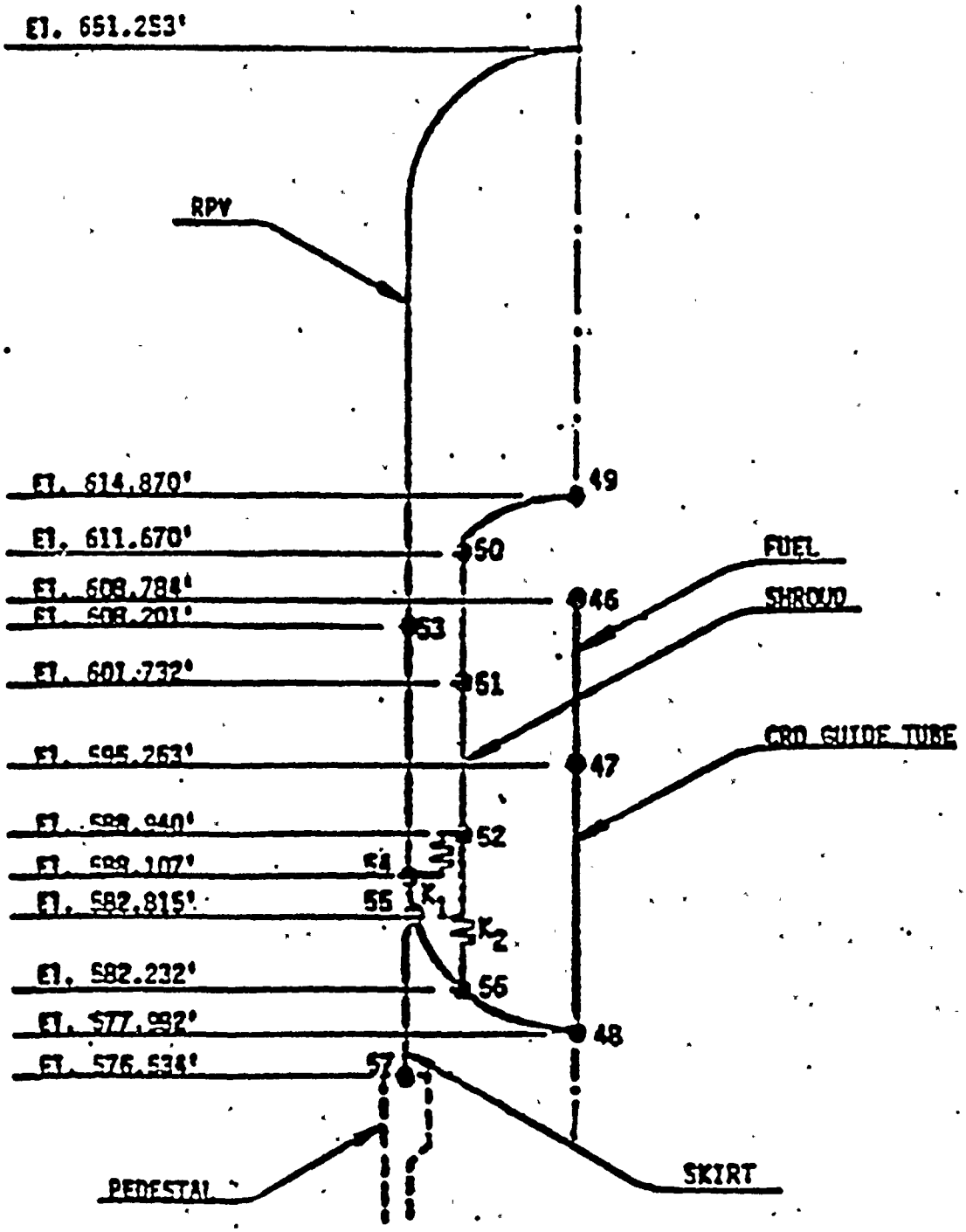


Figure 5 Vertical Model of BF 2/3 RPV and Its Internals

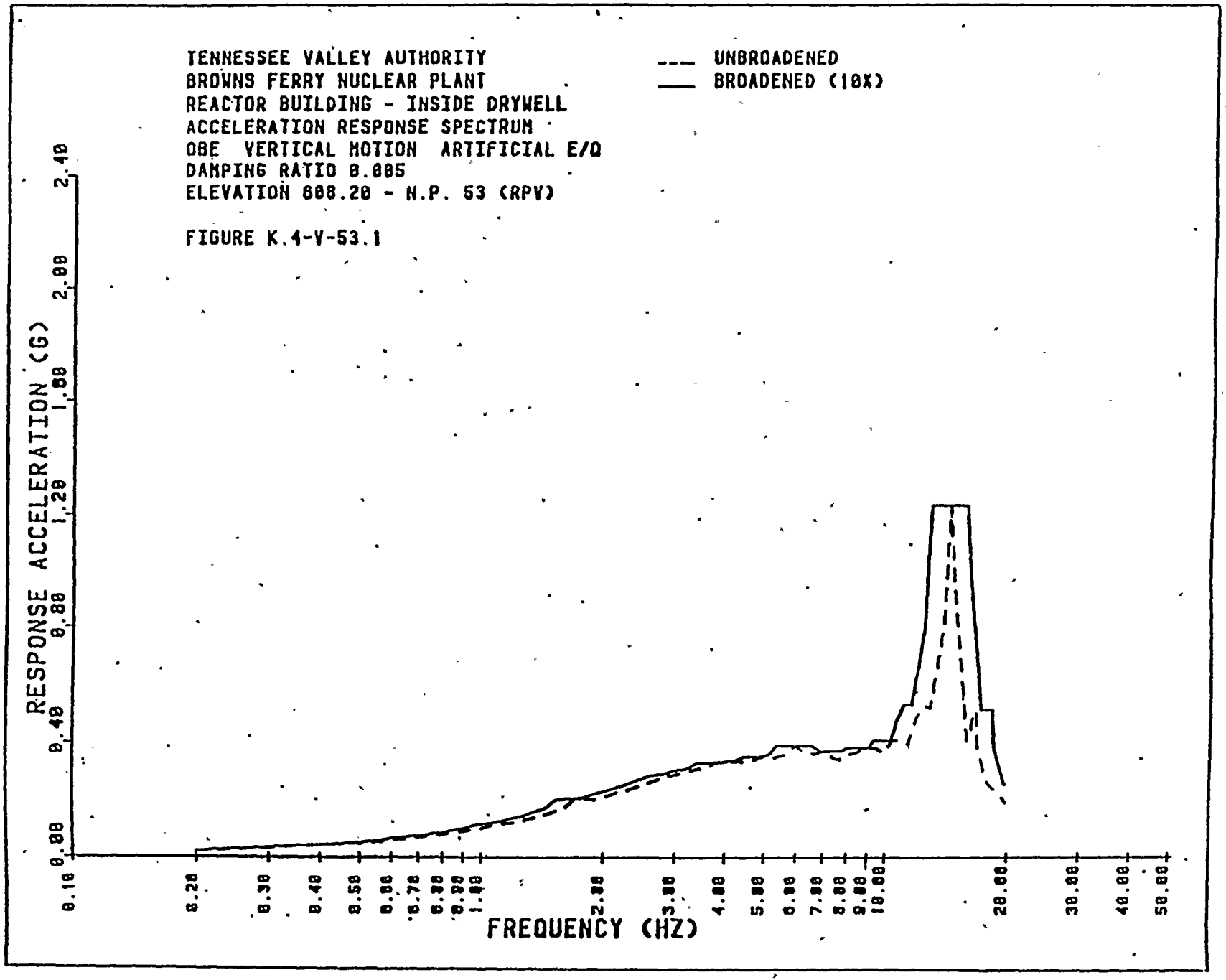


Figure 6 Vertical Acceleration Spectrum at Node 53 of Figure 5

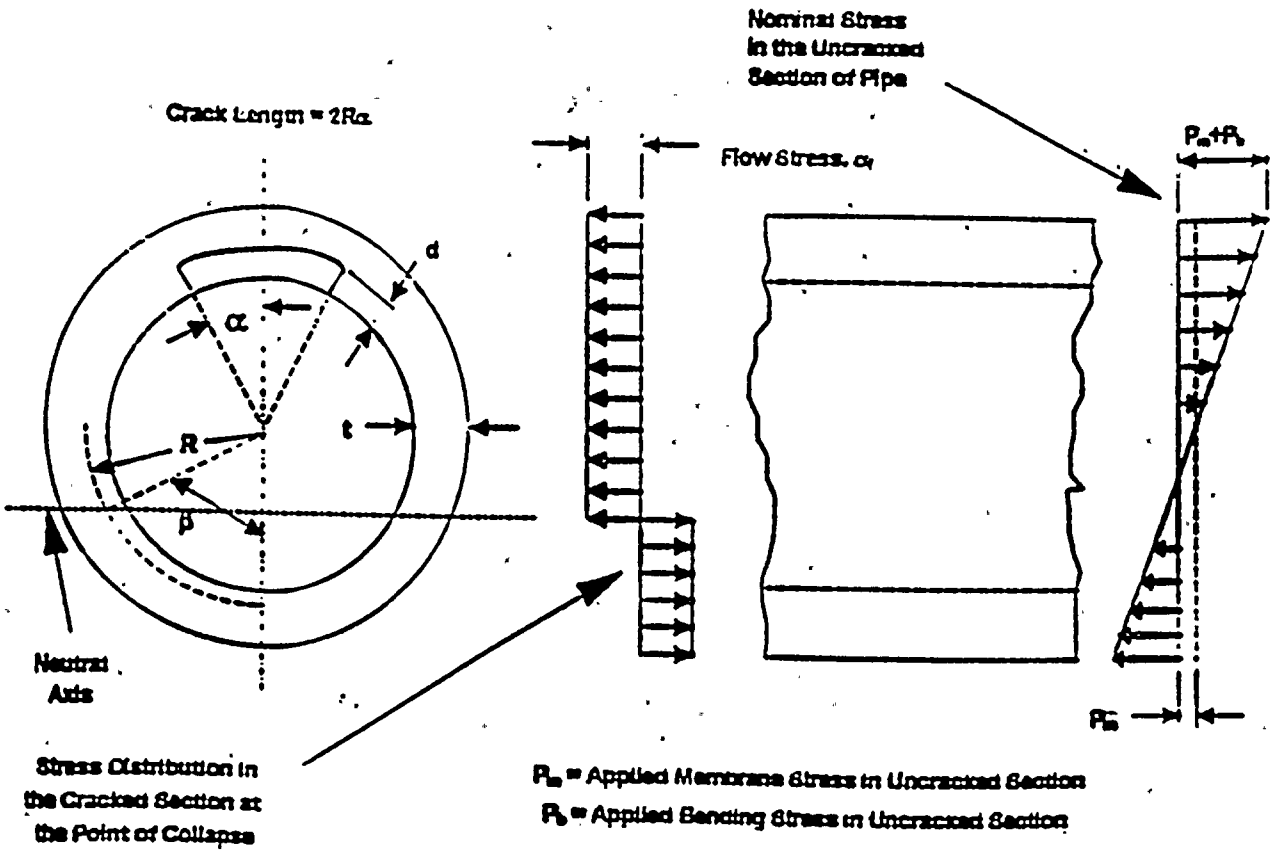
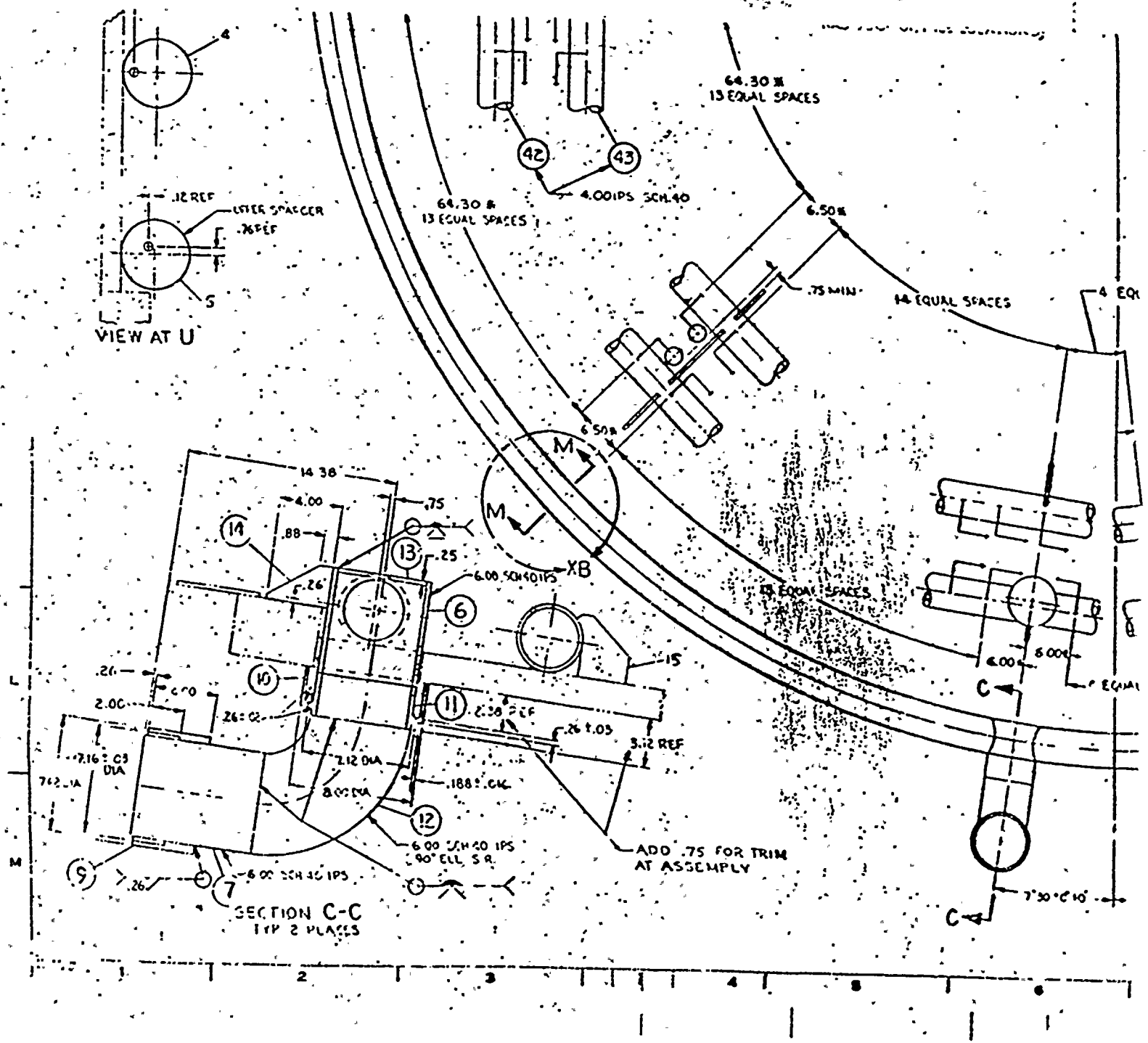


Figure 7 Stress Distribution in a Cracked Pipe at the Point of Collapse

Item Rept. 3


VIEW AT U
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6.00

R12 940817 189

GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

OG94-600-01
August 6, 1994

ACTION REQUESTED
BY TUESDAY, AUGUST 9

To: BWR Owners' Group Primary Representatives

Subject: **REVISION 1 OF BWROG SHROUD DOCUMENT**

References: BWROG-94089 and BWROG-94093, "Response to NRC Request for Shroud Information," July 14, 1994 (transmittal of GENE-523-A107P-0794, "BWR Shroud Cracking Generic Safety Assessment," July 1994)

Enclosure: Appendix A, "Shroud Cracking Safety Assessment," of GENE-523-A107P-0794, "BWR Shroud Cracking Generic Safety Assessment," Revision 1, August 1994

The reference letters provided a response from the BWR Owners' Group to an NRC request for additional shroud information (the response was initially submitted via BWROG-94089, and clarification of proprietary information was submitted via BWROG-94093). When the response was submitted, it was explained that because of the time constraint placed on providing the requested information, the response had not been reviewed and endorsed by the BWROG. Since then, the BWROG Primary Representatives completed their review of the response and voted to endorse it.

While the Primary Representative review was being conducted, GE reevaluated the consequences of 360 degree through-wall cracking for normal, transient and faulted conditions using more realistic (lighter) weights and (smaller) shroud loads. The reevaluation resulted in some changes to Appendix A of the reference, which will be subsequently reissued as Revision 1 to incorporate the changes. However, the conclusions reached in Revision 0 of the document are not changed by the reevaluation, although the changes do result in increased likelihood of detection of 360 degree through-wall cracking during normal operation at some weld locations (namely, the H4, H5 and H6A welds). Other substantive changes resulting from this reevaluation and included in Revision 1 are identified in the attachment to this letter. Also attached is Appendix A of Revision 1.

Bob Pinelli (BWROG Chairman) has directed that the Primary Representatives be provided with a brief opportunity to look at the changes to Appendix A before Revision 1 is submitted to the NRC, because portions of the document are different from the version provided for vote during the endorsement process.

Ref. 6

Please review Appendix A Revision 1 and contact Steve Stark (GE, 408-925-1822) or Robin Dyle (205-877-7121) by 1:00 p.m. (Pacific time), Tuesday, August 9, if you do not believe it should be submitted to the NRC. To facilitate your review, the changes corresponding to the reevaluation are indicated by a line in the margin.

Regards,

WA Zarbis

William A. Zarbis
BWR Owners' Group Projects
Mail Code 482
Tel: 408-925-5070
Fax: 408-925-2476

cc: R. A. Pinelli, BWROG Chairman
K. P. Donovan, BWROG Vice Chairman
L. A. England, EOI
S. J. Stark, GE
R. L. Dyle, SNC

Attachment

SUMMARY OF CHANGES TO APPENDIX A

1. Page A-1, Section A.1.1; Page A-13, Section A.5.2; Page A-14, Section A.6: The lighter weights used in the reevaluation resulted in greater lift during normal operation. Before, the lift was calculated to be less than two inches and therefore ECCS injection lines would not be affected. Now, the lift is calculated to be eight inches in the limiting case and ECCS injection lines may therefore be affected. However, note that this is for the case of normal operation and therefore ECCS availability is not a factor.

2. Pages A-2 and A-3, Section A.1.2: Before, significant lift was only expected at the H3 weld location during normal operation. At the H4, H5 and H6A welds, cracks were expected to be too tight to cause leakage significant enough to be detected during normal operation. As a result of the reevaluation, greater lift is expected at the H3 weld (but not enough to for the top guide to clear the top of the fuel bundles), and lift at the H4, H5 and H6A welds is now calculated to be significant enough to allow detection during normal operation for most plants.

3. Page A-8, Section A.3.1.1: The basis for the smaller shroud loads is the TRAC code.

4. Page A-8, Section A.3.1.1: Before, the amount of contact between the top guide and fuel bundles was expected to be on the order of two inches. As a result of the reevaluation, the amount of contact will be less than two inches but will still be maintained.

Appendix A
SHROUD CRACKING SAFETY ASSESSMENT

Section 4 summarizes the very low likelihood of a 360°, >90% deep crack existing in conjunction with a design basis LOCA event. A discussion of the safety consequences associated with 360° through-wall cracking is, therefore, not necessary. However, for information the safety consequences associated with 360° through-wall cracking are evaluated in this Appendix for normal, transient and faulted conditions.

A.1 NORMAL OPERATION

If it is postulated that separation of the shroud assembly along a horizontal weld did occur during normal operation, there could be some upward displacement, depending on the postulated crack location, operating conditions and plant type. This displacement is calculated such that sufficient flow is allowed to the outside shroud region for the inside pressure to equalize to the upper shroud weight. The maximum displacement for the various weld locations is indicated below and is not sufficient to disengage the top guide from the fuel channels. This displacement is 13 to 15 inches for most plants, and six (6) inches for BWR/6 designs. The top guide maintains the top of the fuel assemblies properly spaced. Therefore, the core arrangement and fuel bundle orientation would be held intact. Also, any separation (other than a tight crack) will result in significant flow through the gap.

In cases where separation would occur, a significant amount of flow to the outer shroud region would likely be detected during normal operation by the reactor operator using available instrumentation for monitoring reactor performance, as described below. Additional training or special procedures may be needed to facilitate identification of characteristics that would lead to detection. After detecting and confirming an anomaly, a normal shutdown is expected to be initiated until the cause of the anomaly is found and corrected.

A.1.1 H1 and H2 Welds (Above the Top Guide)

If 360 degree through-wall cracking at either the H1 or H2 weld locations did occur during normal operation, the upward displacement of the shroud above the crack is calculated to be less than eight (8) inches. This maximum displacement is expected for plants with high shroud pressure difference when operating near rated power and flow conditions. Most plants will experience smaller displacement. However, all plants will experience some displacement. At these weld locations, flow through the resulting gap would be detected during normal operation by the reactor operator using available instrumentation for monitoring reactor performance, as described below.

The hot coolant escaping to the outer shroud region will increase the temperature of the coolant entering the core. The lower inlet subcooling will result in a large core reactivity and thermal power reduction. For a two (2) inch gap, the leakage flow is estimated to exceed 20% of rated. The resulting thermal power loss would also exceed 20% of rated. Even for gaps as small as a one-quarter inch, the flow is sufficient to affect core power by twice the normal instrumentation power uncertainty of 2%. For the larger gap size, the reduction in recirculation flow resistance would also be significant. Additionally, those plants with recirculation loop cavitation monitoring instruments will indicate low subcooling of recirculation loop fluid, while all plants should indicate higher than normal recirculation loop temperature(s). If the crack and leakage occurred on one side of the shroud only, the indications would be asymmetrical which would facilitate detection.

After detecting such an anomaly, a normal shutdown is expected to be initiated until the cause of the anomaly is found and corrected.

Analogous situations have previously been observed in BWRs. In 1984, a plant began startup with shroud head bolts improperly engaged, resulting in bypass flow paths similar to those that would result from through-wall cracking of the shroud. A similar situation also occurred at a different plant in 1991. In both cases, anomalies such as those described above were detected and the operators shut the plant down.

A.1.2 H3, H4, H5 and H6A Welds (Welds Above the Core Plate)

If 360 degree through-wall cracking at either the H3, H4, H5 or H6A weld locations did occur during normal operation, the upward displacement of the shroud above the crack would be small, such that the top guide would not clear the top of the fuel channel, and therefore fuel bundle orientation and control rod insertability are maintained.

A maximum displacement of 6 inches is calculated for the H3 weld location, for those plants with high shroud pressure differential when operating near rated power and flow conditions. This type of gap will be detectable as discussed below. Through-wall 360° cracking at the H4, H5 and H6A welds would result in smaller displacements (less than three (3) inches), and detectability depends on the actual displacement. The weight of the shroud and steam separator assembly above the H4, H5 and H6A weld locations may be sufficiently high to hold the shroud assembly in place during all normal operating conditions (for plants with low shroud pressure differential) such that only a tight gap is expected. For this case, the flow would occur through a gap of less than 0.002 inches. The estimated flow through such a gap would typically be about 0.05% of total core flow (based on a 0.002 inch gap around the shroud's entire circumference and a typical differential pressure of eight

2.

pounds per square inch). Flow of this magnitude will have no impact on plant operation, and will not be detectable.

At the H3 weld and for the case where displacement larger than a tight crack occurs at the H4, H5 or H6A locations, the flow through the gap may be a detectable amount of the total core flow. This flow would come from the subcooled liquid in the core bypass region and will not increase the recirculation flow temperature significantly. This magnitude of flow loss in this region will cause the remaining flow in the bypass region to void, leading to reduced core reactivity. A 2 inch gap will result in seven (7) percent core flow loss and about five (5) percent lower thermal power. This amount of power loss is 2 to 3 times the normal power measurement uncertainty, and would be readily detectable. Power anomalies of less than 2% (corresponding to one-quarter inch gap) are not expected to be detectable. 2.

After detecting a possible anomaly as described above, a normal shutdown is expected to be initiated until the cause of the anomaly is found and corrected. For those plants whose characteristics do not lead to displacement, the leakage will be limited to that of a tight crack, and be undetectable.

A.1.3 H6B, H7, H8 and H9 Welds (Welds Below the Core Plate)

If 360 degree through-wall cracking at either the H6B, H7, H8 or H9 weld locations did occur during normal operation, the upward displacement of the shroud above the crack would be a half inch. This displacement is limited by the contact of the core support plate with the fuel support structures. This gap is expected for near rated core power and flow operating conditions for any plant and at any weld location. This type of gap will be detectable as discussed below. At low operating power and flow conditions, a tight gap would exist with flow through a gap of less than 0.002 inches. The estimated flow through such a gap would typically be about 0.1% of rated core (based on a 0.002 inch gap around the shroud's entire circumference and a differential pressure of 15 pounds per square inch [psi]). Flow of this magnitude will have no impact on plant operation, and will not be detectable.

For the case where maximum displacement may occur (a gap on the order of one-half inch over most of the circumference), the estimated flow would be about 15% of total core flow. This flow would come from below the core support plate region. This magnitude of flow loss in this region will cause a significant reduction in core reactivity and about 10% lower thermal power. Additionally, significantly lower pressure difference across the core will be measured. A decrease of about 7 psi is estimated, which is approximately one-third of the total normal measurement. Finally, abnormal relationships of recirculation pump flow to core flow, and core power to core flow, will be evident. All of these anomalies will be readily detectable.

After detecting such an anomaly, a normal shutdown is expected to be initiated until the cause of the anomaly is found and corrected.

At the rare condition of an intermediate power and/or flow condition when some upward displacement occurs, but the gap has not yet fully developed (about 0.1 inch), the flow through the gap would be about 3% of rated core flow. This case would probably not be detectable, as the power loss is about 2% of rated and will be masked by instrument uncertainties. The undetected loss of flow would result in a one (1) percent non-conservative minimum critical power ratio (MCPR) calculation by the monitoring computer. This loss is not significant because large MCPR margins exist at intermediate power/flow conditions.

A.1.4 Additional Normal Operating Considerations

The effect of a through-wall crack on water level indication has been considered. If a significant crack developed abruptly at high power, steady state conditions, the rapid creation of leakage flow into the outer shroud region of the vessel will cause an observable increase in water level. The changes in core power and flow resulting from the leakage flow will vary depending on the location and magnitude of the leakage. If the water level control system is not able to maintain level, an automatic scram on high or low water level will occur. However, in the more probable case of slow crack growth, the water level transient is not enough to reach the trip setpoints, and water level would be subsequently restored to the proper level.

A.2 ANTICIPATED OPERATIONAL EVENTS

Assuming there are no indications of shroud leakage, this section discusses anticipated operational occurrences that could increase shroud loads above those experienced during normal operation: pressure regulator failure - open, recirculation flow control failure - increasing to maximum flow, and inadvertent actuation of the Automatic Depressurization System (ADS). Other anticipated operation events that are typically considered during evaluations of fuel thermal margins and vessel overpressure (these events include turbine trip with turbine bypass failure, control rod withdrawal error, and main steamline isolation valve [MSIV] closure) do not cause increased lift forces on the shroud. Therefore, their consequences are not affected by the condition of the shroud.

The consequences of these events are additional upward loads on the shroud welds. For a 360 degree through-wall crack, these loads may lead to complete weld separation and/or result in higher upward displacements than during normal operation. It is concluded that as a result of these events all appropriate criteria (MCPR Safety Limit, Low Water Level, and Reactor Overpressure Limit) are not violated.

If complete shroud separation occurred during the event, it would likely be undetected. If, following such an event, the plant attempted a normal startup,

operating anomalies such as those discussed in Section A.1 would be detected when the plant achieves high power and flow condition, causing the operators to initiate a normal shutdown.

A.2.1 Pressure Regulator Failure - Open

This postulated Safety Analysis Report (SAR) event involves a failure in the pressure controls such that the turbine control valves and the turbine bypass valves are opened as far as the maximum combined steam flow limit allows. For units with standard bypass capacity (about 25% of rated steam flow), the worst case involves inadvertently increasing the steam flow to about 130% of rated. This is also true for units with larger bypass capacity if the steam flow limit is set at 130% or less. A depressurization and cooldown occurs which is isolated by MSIV closure. This steam flow increase leads to increased lifting force on the shroud head. This increased steam flow is of short duration, about 3 seconds. Depending on the particular plant characteristics, the maximum separation distance for mid-shroud welds (H3 through H6A) will range from none to less than that required to clear the top guide from the fuel channels (as the distance is bounded by the Main Steam Line Break). Because the fuel remains properly aligned, core geometry is maintained and successful scram assured. The loads on the weld locations below the core support plate (H6B through H9) are only increased by about 5% and do not result in much different consequences than at normal operation. The consequences of this event meet the applicable licensing criteria.

A.2.2 Recirculation Flow Control Failure

This postulated SAR event involves a recirculation control failure that causes all recirculation loops to increase to maximum flow. In this type of case, the upward pressure will change from a part-load condition to the high/maximum system flow capability condition over a time period of about 30 seconds. The increased lifting forces are bounded by the Pressure Regulator Failure discussed in Section A.2.1. However, because this event results in increasing core power (instead of decreasing power as for the Pressure Regulator Failure event), fuel thermal overpowers are affected. Shroud separation at the upper welds will decrease overpowers because the hotter coolant will limit core power increase. Shroud separation at the middle and lower welds will not affect the event as core power will correspond to the core flow which successfully enters the core and increases reactivity. The consequences of this event meet the applicable licensing criteria.

A.2.3 Inadvertent Actuation of ADS

Inadvertent actuation of the ADS valves is another postulated SAR event that increases load on the shroud. The maximum steam flow and the depressurization rate are significantly smaller than for the postulated main steamline break, causing a short-term increase in steam flow of about 50% of rated steam flow (plant

dependent). The increase in the shroud ΔP resulting from the opening of the ADS valves would occur over a period of about one second, spreading the effect of the change in load. This event is very similar to the Pressure Regulator Failure (Section A.2.1), except that the loads are somewhat larger but of shorter duration, and are also bounded by the Main Steam Line Break. However, the conclusions of Section A.2.1 are the same.

A.3 DESIGN BASIS ACCIDENTS

The combined probability of an accident occurring simultaneously with a seismic event is extremely low. Plant-specific probabilities for a seismic induced LOCA typically range in the order of 1E-4 to 1E-9 per year. The combined probability of a seismic induced LOCA event occurring when a severe (360 degree circumferential through-wall) undetected shroud crack exists is even lower. However, such a postulated event is addressed in this section.

Two Design Basis Accidents (DBAs) are evaluated: a Main Steam Line Break, and a Recirculation Line Break. The following discussion addresses the impact of an undetected crack as it affects the key issues of control rod insertability, coolable geometry, ECCS performance, and SLCS effectiveness.

The primary factor in this determination is the expected movement of the shroud during the postulated accident. Control rod insertability will be assured if the fuel remains properly arranged in the core, and the guide tubes and shroud remain aligned. A coolable geometry will be maintained if the fuel and vessel internals do not impede the normal flow of coolant to the fuel. For vessel pipe break locations above the Top of Active Fuel (TAF), short and long term cooling is accomplished by ECCS injection anywhere in the reactor vessel in a flow amount equal to the steaming rate of the core. For vessel pipe break locations below the TAF, short term cooling is accomplished by ECCS injection inside the shroud, over the fuel and elsewhere. Long term cooling is accomplished by maintaining the level inside the shroud, to the jet pump level, by ECCS injection anywhere inside the shroud.

Shroud cracks impact the plant's response to either the main steamline or recirculation breaks only to the extent that added ECCS flow is needed to overcome the leakage through the crack. Proper ECCS performance is achieved if the ECCS coolant is available when and where needed. All BWRs are equipped with Core Spray (CS) systems. Some BWRs, depending on type, are also equipped with Low Pressure Coolant Injection (LPCI), High Pressure Core Spray (HPCS), and High Pressure Core Injection (HPCI) systems. Except for the HPCI, each of these systems inject inside the shroud through upper penetrations and/or jet-pumps. Shroud cracks can only affect these systems if severe (greater than 2 inches) shroud displacement occurs.

SLCS effectiveness is assumed if injection is possible. The SLCS is used to shut down the reactor if control rods are not adequately inserted (for a recirculation line break, the nature of the accident may limit the effectiveness of SLCS by draining the boron from the core). The SLCS injects to the lower plenum for some BWR designs, and through the core spray lines for the remaining BWR designs. For a shroud separation above TAF, shroud cracks and displacement will not prevent SLCS from performing its intended function for either the recirculation line break or the main steamline break. For a shroud separation below TAF, shroud cracks and displacement do not affect SLCS performance for the main steamline break, and SLCS effectiveness for the recirculation line break is also not affected within the limitation discussed above.

Shroud movement is increased if a seismic event is considered coincident with the DBA. For this evaluation the accident without seismic considerations is addressed first, and then the added effect of the seismic loads from a safe shutdown earthquake (SSE) is examined.

The Main Steamline Break Accident imposes the largest lifting loads on the shroud head and lower shroud welds, and has the greater potential to defeat the shroud functions. The Recirculation Line Break Accident does not impose large pressure drops on the shroud, and in fact the shroud pressure drop decreases from its initial value. However, this break has the greater potential to cause excessive fuel damage.

A.3.1 Main Steamline Break

A.3.1.1 Main Steamline Break - No Seismic Event

The main steamline break inside primary containment is the postulated worst case because it results in the most severe depressurization. During this event, the reactor is rapidly depressurized as a result of a postulated instantaneous, double-ended break of the largest steamline. Thus a maximum pressure difference develops across the shroud as fluid flow is drawn from the core region toward the break.

The shroud head pressure drop characteristics calculated for the instantaneous, double-ended steamline break accident were evaluated for various BWR sizes. The initial shroud head pressure drop loading is a result of the depressurization of the steam dome region which reduces system pressure overall, but which increases differential pressure across the shroud in the short term. This pressure loading increase is short-lived (less than 2 seconds) and decreases to below normal steady state loads.

The increased loads on upper welds (H1 through H6A) will result in greater upward separation than for normal operation and those events discussed in Section A.2. The separation at the uppermost weld (H1) location will not impact any of the

key functions identified in Section A.3, and therefore the displacement magnitude is not of concern. The separation at the H2 through H6A welds can impair ECCS lines if displacement exceeds two inches; however, because the core remains covered for this accident, only injection inside the reactor vessel is needed for normal cooling. Therefore, although a key function is affected in this scenario; no impact on ECCS performance exists.

Assuming separation at a mid-shroud weld (H3 through H6A), the possibility of the separated upper shroud lifting until the top guide lifts above the fuel channels must be evaluated. The amount of lift for this condition has been evaluated for various BWR sizes, and these evaluations have determined that the maximum attainable lift is less than that required to clear the fuel channels for BWR/3-5 plants. This evaluation used the TRACG model with standard conservative design basis analysis assumptions. Using these conservative design basis assumptions, acceptable results are not shown for the BWR/2 and BWR/6 designs. For these plant types, it is expected that further evaluations with less bounding assumptions would give acceptable results. This is demonstrated by a RELAP5 evaluation (Reference by GPUN) performed for a BWR/2 plant. This calculation shows a decrease in peak shroud loads of about 30% when compared to a conservative design basis assumptions. For BWR/3-5, contact between the top guide and fuel channels is maintained. Because the fuel geometry remains unchanged, the control rods can be inserted and no impact on performance exists. 3. 4.

The increased loads on the lower welds (H6B through H9) will not result in additional separation. The magnitude of the separation (one-half inch) is limited by the clearance between the core support plate and the fuel support structures. The weights of the fuel, fuel support, and control rod guide tubes are sufficient to maintain the core plate, and thus the shroud, in place. Therefore, there is no impact on performance.

A.3.1.2 Main Steamline Break Plus Seismic Event

If a main steamline break is postulated to occur simultaneously with the design basis earthquake and a 360 degree through-wall crack is also postulated, the added load of the earthquake can result in greater shroud displacement than that described above. Additional separation at the H1 and H2 welds will not lead to greater impact than that described in Section A.3.1.1. The important parameters become the amount of separation for mid-shroud (H3 through H6A) welds, impacting the top guide/fuel channel interaction, and the fuel lift margin for lower (H6B through H9) welds.

H3, H4, H5 and H6A Welds (Welds Above the Core Plate)

The effect of seismic loads is estimated to be less than 1 inch displacement in both the vertical and lateral directions. This magnitude is based on specific

plant evaluations, and engineering judgment as to the expected variation for other plant configurations. Rotation (or tipping) of the shroud assembly due to lateral seismic motion is also calculated to be less than 1 inch, and will only be momentary as the shroud will fall back in place. These amounts of displacement will not significantly alter the conclusions of Section A.3.1.1. With the same conditions discussed in A.3.1.1, the top guide will still maintain proper fuel assembly alignment, and small lateral, momentary movement or tipping will not prevent control rod insertion, so safety functions will not be affected.

Furthermore, this postulated event does not impact the SLCS function to inject in the reactor vessel and accomplish the shutdown, should control rod insertion be assumed to fail due to shroud movement or an independent CRD-related failure.

As discussed in Section A.3.1.1, ECCS injection may be somewhat impaired. However, because the core remains covered for this accident, coolant inventory makeup anywhere in the reactor vessel is sufficient to assure proper short and long-term cooling.

H6B, H7, H8 and H9 Welds (Welds Below the Core Plate)

The effect of seismic loads is estimated to be less than 1 inch displacement in the lateral direction. Rotation (or tipping) of the shroud assembly is limited to one-half inch also because of the core support plate to fuel support structures clearance. This amount of displacement will not alter the conclusions of Section A.3.1.1 for these welds either. Additional lift forces will reduce the core weight downward forces; however, the guide tubes will continue to limit the vertical displacement to one-half inch. Additionally, a less than 1 inch seismic induced lateral movement will not prevent the control rod insertion. The control rod insertion will occur as seismic oscillations will realign the guide tubes and fuel support structures.

It should be mentioned that for a lower shroud weld, vertical separation is limited to one-half inch and, as such, ECCS injection will not be impaired.

Again, this postulated event does not impact the SLCS function to inject in the reactor vessel and accomplish the shutdown, should the control rod insertion be assumed to fail.

A.3.2 Recirculation Line Break

A.3.2.1 Recirculation Line Break - No Seismic Event

For the design basis recirculation line break, the differential pressure across the shroud decreases from the initial value as the reactor depressurizes, upward forces are reduced, and thus there is no significant threat to core shroud integrity. Any initial shroud separation along a particular location will be limited to a tight crack because any significant separation prior to the accident would be detected during normal operation as discussed in Section A.1. With shroud integrity maintained, a floodable core region and a coolable geometry are preserved. The additional leakage through any tight weld cracks will be minimal and can be easily made up by the operating ECCS. Also, it is noteworthy that because upward shroud displacement is non-existent for this accident, ECCS injection is not impaired. Therefore, the recirculation line break analysis results are unaffected by shroud cracking.

Lateral forces for weld locations in the beginning of a recirculation line break are large acoustic forces of short duration (on the order of a few milliseconds) followed by smaller blowdown forces for several seconds. Lateral motion is not expected because of the resistance of the irregular crack surface to lateral motion without lifting. If sufficient lifting occurs prior to the accident, it will be detected during normal operation as discussed in Section A.1. Tipping (i.e. rotation) is not expected from these forces as the acoustic loading is of very short duration, and the blowdown loads are not large enough to overcome the restoring moment by the shroud downward forces. The acoustic load calculated for the shroud is conservatively applied to the rotation. Also, GE is recalculating the blowdown force for a typical jet pump plant using detailed three dimensional thermal hydraulic models. The recalculation results currently obtained show that the blowdown force is larger than previous calculations. However, the overturning moment caused by the blowdown force is still bounded by the restoring moment of the shroud weight. Therefore, the recirculation line break analysis results are unchanged.

A.3.2.2 Recirculation Line Break Plus Seismic Event

For the design basis recirculation line break simultaneous with a seismic event, additional vertical and lateral forces will exist. As discussed above, the forces in the core region resulting from the recirculation line break exert an almost instantaneous downward pull on the shroud and will prevent vertical and lateral displacement along a weld location. The lateral seismic loads, combined with the asymmetric blowdown loads may lead to small (less than 3/4 inch) momentary tipping for some welds (such as H6A). However, the restoring moment of the shroud weight will prevent permanent displacement. As stated above, displacement for lower welds (H6B to H9) is limited by fuel support structures.

A.4 EMERGENCY OPERATOR ACTIONS

The Emergency Procedure Guidelines (EPGs)(Reference A1) are the basis for plant-specific Emergency Operating Procedures (EOPs). The EPGs are symptomatic in that they respond to detected symptoms and do not require diagnosis of the event by the operator. They address a very wide range of events, both less severe and more severe than design basis accidents.

Because the EPGs do not depend upon the successful performance assumed in design basis analyses, they do not lose effectiveness if actual events vary from the design basis assumptions. In addition, operator training is not limited to events which follow specific scenarios, but include events with degraded conditions which fully exercise the operator actions specified in plant-specific EOPs.

The EPGs provide instructions for reactor pressure, water level, and power control, as well as for control of key primary containment parameters. The actions which operators should take are described below for postulated events of a main steamline break and a recirculation suction line break. Even though automatic control rod insertion is expected to occur for these events as discussed in Section A.3, operator actions are also described if no control rod insertion is postulated.

A.4.1 Main Steamline Break

A postulated main steamline break occurs high in the reactor vessel and results in rapid reactor vessel depressurization. ECCS will automatically initiate and quickly restore water level above the TAF. In fact, the injection flow required to maintain this water level is much less than the full complement of ECCS. Operator actions will ensure the reactor is shutdown and water level restored. The operator will take actions to restrict flow to control water level in the normal control band.

Shroud cracking and/or displacement, as described in Section A.3, will not restrict the ability to recover the water level to above TAF for this event. In addition, the displacement is not expected to prevent control rods from inserting even if the break occurs in combination with a seismic event. However, if the control rods failed to insert, appropriate operator actions are already included in the EPGs. In the unlikely condition that the event progresses to the point that boron injection is required to shutdown the reactor, shroud cracking or displacement will not prevent boron injection or boron mixing.

In summary, existing symptomatic EOPs and operator training are fully adequate to accommodate the potential consequences of shroud cracking and displacement for a main steamline break.

A.4.2 Recirculation Suction Line Break

A postulated recirculation suction line break occurs low in the vessel and also results in rapid reactor vessel depressurization. ECCS will automatically initiate and quickly restore collapsed water level to the top of the jet pumps. Short term two-phase water level in the core will be above TAF, but as the core is cooled and the voids collapse, water level settles out at the top of the jet pumps; and the excess injected water spills out the break into the drywell. The injection flow required to maintain water level at this height is much less than the full complement of ECCS. The integrity of all ECCS lines is maintained for a recirculation line break.

Operator actions ensure the reactor is shutdown and attempt to restore water level to above TAF. In the most limiting case, this cannot be accomplished by vessel injection alone due to the break size and location, so primary containment flooding is undertaken. When the primary containment water level is raised to above the TAF elevation, reactor water level may be restored and maintained above TAF.

Shroud displacement is not expected to occur as a result of a recirculation line break, as described in Section A.3, even if the event occurs in combination with a seismic event. However, even if displacement did occur, there would be no change in the operators inability to recover level above TAF, and primary containment flooding will still be required.

In the highly unlikely condition that the control rods failed to insert, appropriate operator actions are already included in the EOPs.

In summary, existing symptomatic EOPs and operator training are fully adequate to accommodate the potential consequences of shroud cracking and displacement for a recirculation line break.

A.5 ADDITIONAL CONSIDERATIONS FOR NON-JET PUMP (BWR/2) PLANTS

The following sections discuss the various differences between a Non-Jet Pump BWR and other BWRs for a 360 degree through-wall shroud crack. While many differences exist between these plants, the important factors for a Non-Jet Pump BWR are as follows: (a) recirculation flow enters the reactor vessel from the bottom, (b) ECCS for large breaks are 2 redundant, double capacity, core sprays, (c) short and long term cooling responses for large recirculation line breaks rely on core spray, as the vessel will not flood, (d) failure of the H8 weld (shroud support ring to shroud support plate weld) results in the shroud having little vertical support, (e) the reactor vessel has 2, instead of four [4] steamlines, and (f) accident plus seismic is not a design or licensing basis.

A.5.1 Normal Operation

For normal operation, Non-Jet Pump plants will experience smaller shroud loads than other plants. This is the result of lower power density, lower core flow, and generally smaller plant sizes. Separation can occur at the H1 and H2 welds and at the welds below the core support plate. Flow through these gaps will be significant at near rated power and flow operation such that anomalous core characteristics will be detected and normal shutdown carried out.

The special case of a 360 degree through-wall crack at the H8 weld for this shroud configuration will likely result in upward displacement (up to one-half inch) at high power and flow operation. Downward displacement is not expected under normal operation conditions. Again, any displacement will result in sufficient leakage flow from the lower plenum to the outer shroud region resulting in large power anomalies. These anomalies will likely be detected and normal shutdown carried out.

A.5.2 Anticipated Operational Events

For anticipated operational events, Non-Jet Pump plants will not experience as large a load increase as other plants. This is the result of smaller steam flow, core flow, and ADS capacities. Separation at the H1 and H2 welds will be higher than during normal operation.

A.5.3 Design Basis Accidents

For the Main Steamline Break Accident, the upward shroud displacement on Non-Jet Pump plants is expected to be somewhat larger than that for other plants. As discussed above, characteristics of this plant type will result in generally smaller loads; however, the limitation of 2 steamlines results in larger design basis loads, and consequently greater lift. However, as stated in Section A.3.1.1, detailed accident analyses would show that maximum lift would be limited such that the top guide and fuel channels remain aligned (e.g. less than 14 inches of lift). While it is expected that core geometry will be maintained, core spray lines are expected to be damaged by the possible shroud displacements (e.g. greater than 2 inches). However, the break is above the TAF, so ECCS injection inside the reactor vessel at or above the core steaming rate will assure short and long term cooling.

For the Recirculation Line Break Accident, shroud loads will be larger for Non-Jet Pump plants than for other plants. This is the result of the different location of the recirculation lines (vessel bottom) and the larger size of these lines. No displacement is expected at any but the vertically unsupported weld (H8). This displacement, however, is expected to damage the core spray lines and result in impaired core spray cooling. The degree of any resulting cooling deficiency depends on the final condition of the core spray system after a downward shroud displacement. Long term cooling is unchanged as containment flooding is

unaffected. It is important to note that failure of this weld (H8) during normal operation (as discussed above in Section A.5.1) is detectable. An undetectable degraded weld condition is highly unlikely to fail during accident conditions because of the low additional stresses of this event on the weld.

A.6 CONCLUSIONS

The conclusions drawn from the safety assessment, assuming 360° through-wall shroud cracking, are as follows:

1. If separation of the shroud assembly along a horizontal weld did occur during normal operation, the resulting displacement would range from 8 inches to a few mills, depending on the postulated crack location, and plant characteristics. This displacement will allow the core assembly and fuel bundle orientation to be held intact. Displacements larger than one-quarter inch will result in sufficient flow through the gap resulting in an anomalous core power and flow characteristic. This anomaly will likely be detected during normal operation using available instrumentation and the operator would then initiate a normal shutdown.

2. The consequences of anticipated operational events are not increased by the assumption of 360 degree through-wall shroud cracking.

3. The combined probability of a design basis accident occurring simultaneously with a seismic event is extremely low. This criterion is also beyond the licensing basis of several BWRs, such as BWR/2s. The combined probability of an accident plus seismic event occurring when a 360 degree circumferential through-wall undetected crack exists is even lower. However, such a postulated event has been evaluated considering either a Main Steam Line Break or a Recirculation Line Break. Although the hypothetical through-wall cracked shroud region is expected to separate to some degree, the accident consequences are not adversely affected by shroud separation.

4. A through-wall crack along the entire shroud weld circumference will lead only to possible upward displacement and/or possible lateral displacement. Any upward displacement would be momentary and any lateral displacement would be temporary (except under accident plus seismic conditions), so that control rods will insert (or SLCS will properly function if control rods are posulated to not insert). Downward displacement is impeded by remaining shroud and/or additional shroud support structures for all BWRs except for the H8 weld in a BWR/2.

A.7 REFERENCES

[A1] NEDO-31331, Revision 4, "Emergency Procedure Guidelines", March 1987.

Ref. 7

Cracking in core spray piping

SIL No. 289
Revision 1
Supplement 2

January 5, 1996

SIL No. 289 (superseded by Revision 1), issued February 1, 1979, and SIL No. 289 Revision 1, issued May 2, 1980 discussed cracking in the core spray sparger arms at two operating BWRs. SIL No. 289 Revision 1 Supplement 1, issued February 23, 1989 (superseded by Revision 1), and SIL No. 289 Revision 1 Supplement 1 Revision 1, issued March 15, 1989, identified two other locations in the core spray sparger that are susceptible to cracking. This SIL No. 289 Revision 1 Supplement 2 advises that additional core spray piping welds within the reactor vessel have been identified as being susceptible to cracking and provides additional recommendations pertaining to these findings. This SIL supersedes and closes RICSIL No. 074.

Discussion:

The core spray supply piping within the reactor vessel (internal core spray piping) is accessible for routine visual inspection. NRC I&E Bulletin No. 80–13, issued May 12, 1980, requires visual inspection of the core spray spargers and associated piping at every refueling outage. These inspections are routinely performed through utilities' augmented inspection programs.

Cracking has been observed in core spray piping at a number of US BWRs. The observed cracking is in the thermal sleeve collar, the downcomer slip joint sleeve (a creviced weld), and the downcomer piping elbow weld. The following discussion provides details on the cracking.

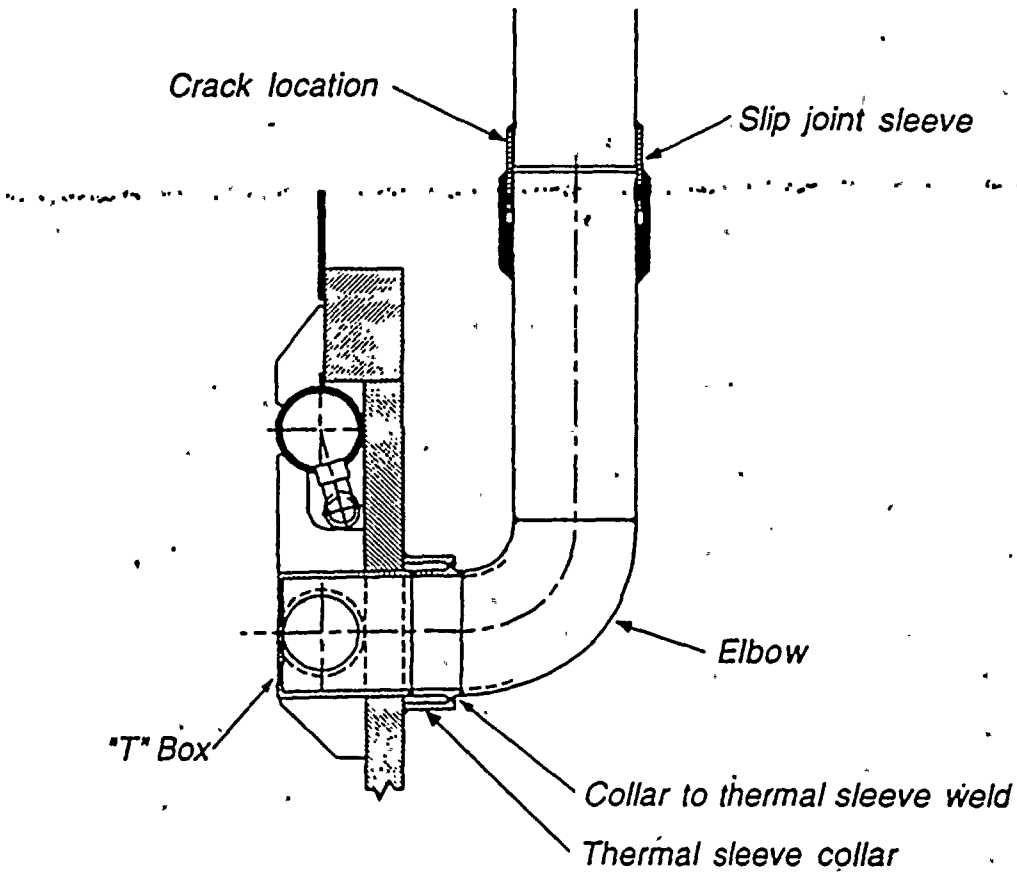
At a BWR/3 located in the United States, an enhanced (0.5–mil wire resolution) visual examination of internal core spray header and downcomer piping was performed during a routine refueling outage. Cracks were identified in three locations. See Figure 1 for the location of these cracks. The cracks were described as very tight and required visual examination at very close camera–to–subject distances (1 – 3 inches). A supplemental ultrasonic test (UT) was performed to characterize the visually detected indications. RICSIL 074 provided details on the observed cracking.

Cleaning was performed on all cracks to aid in resolution and evaluation of the cracking. In most cases, the supplemental UT confirmed the location and length of the visually detected cracking. Engineering analysis of all the cracks determined that the affected core spray piping was acceptable for continued service. Sufficient ligament remained in the affected welds and no immediate repairs were necessary.

At a BWR/4 located in the United States, a routine visual examination of the core spray header and downcomer piping was performed in accordance with NRC I&E Bulletin 80–13 (1–mil wire resolution). This plant had identified a 3–inch crack in the upper heat–affected zone of the sleeve connecting the downcomer to the sparger inlet pipe during the previous refueling outage. At that time, engineering analysis had determined that the affected core spray piping was acceptable for continued service.

The visual examination performed during this recent outage showed that the previously identified crack on the downcomer sleeve had grown. In addition, cracks in the upper heat affected zone of two other downcomer sleeves were detected. The cracks were described as very tight and required visual examination at very close camera–to– subject distances (1 – 3 inches). A supplemental UT was performed on all four downcomers to characterize the visually detected indications. The locations of these indications are shown in Figure 2.

Figure 2 - Core spray cracking - BWR/4



- At a BWR/3, an indication was detected visually at one of the sparger to T-box welds. In addition, an indication was detected visually on a downcomer sleeve weld. Supplemental UT on the sleeve weld did not verify the presence of the indication.
- At a BWR/4, three indications were detected by visual inspection at various core spray piping locations, including two at thermal sleeve collars. Supplemental UT verified the presence of cracking at the thermal sleeve collar locations.

Applicability

This SIL is generally applicable to all GE BWRs, however, there are some differences among the product lines which should be evaluated on a plant specific basis.

All BWRs/3/4 and /5 have a thermal sleeve collar to shroud weld similar to that found in the BWR/3 described above. In BWR/2s the thermal sleeve is welded directly to the