CEOG COMBUSTION ENGINEERING OWNERS GROUP

CEN-614

SAFETY EVALUATION OF THE POTENTIAL FOR AND CONSEQUENCE OF REACTOR VESSEL HEAD PENETRATION ALLOY 600 OD-INITIATED NOZZLE CRACKING

Prepared for the
C-E OWNERS GROUP
December 1993

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ABB

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SUMMARY

This report supplements CEOG report CEN-607, "*Safety Evaluation of the Potential for and Consequence of Reactor Vessel Head Penetration Alloy 600 ID-Initiated Nozzle Cracking.*" CEN-607 provided an evaluation of the technical and regulatory basis concluding that the potential for *ID-initiated* Primary Water Stress Corrosion Cracking (PWSCC) is not an immediate safety concern in CEOG plants. This report provides the basis to support the conclusion that *OD-initiated* PWSCC is also not an immediate safety concern.

PWSCC was found in the vicinity of the weld that attaches the Alloy 600 control rod drive mechanism nozzles to the reactor vessel head of the Bugey-3 PWR plant in France. A review of the European data in CEN-607 shows that the nozzle materials used in CEOG plants are the same type as that which experienced cracking in Europe. A technical evaluation, however, indicated that CEOG plants are less susceptible to similar cracking. Nonetheless, because PWSCC in Bugey-3 resulted in throughwall cracking, this report considers the potential for and consequence of OD-initiated PWSCC in the presence of a postulated throughwall ID-initiated crack.

A stress field analysis of the CEDM nozzle in CEOG plants shows that in the region above the weld the compressive zones do not extended uniformly around the OD of the nozzle. Thus, tensile loads are present and consideration must be given to the potential for PWSCC on the OD if an ID-initiated throughwall crack is postulated. An environmental chemistry assessment, however, indicates that OD-initiated PWSCC or caustic-induced SCC is not expected to occur.

Nonetheless, if OD-initiated cracking is postulated, crack propagation calculations for the most limiting CEDM in CEOG plants demonstrate that it would take more than 91 years operation (well beyond the current license period) before catastrophic failure would be possible. A determination of the safety significance of OD-initiated PWSCC indicated that because catastrophic rupture is not credible during the current licensed period, there is no safety significance associated with a circumferential crack on the OD of the CEDM. Thus, a 10 CFR 50.59 evaluation concluded that the potential for or consequence of Alloy 600 reactor vessel head penetration nozzle cracking does not involve an unreviewed safety question and is not an immediate safety concern.

Based on the evaluation summarized in this report, the CEOG considers that there is no new safety significance associated with the potential for OD-initiated PWSCC. The safety implications of this phenomena are fully addressed by current practices, precautions, and procedures in CEOG plants.

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1.0 INTRODUCTION

In 1991, Primary Water Stress Corrosion Cracking (PWSCC) was discovered in the vicinity of the J-groove partial penetration welds in Control Rod Drive Mechanism (CRDM) adapter penetration nozzles in Bugey-3, a PWR plant in France (see Figure 1.1). These penetration nozzles are made of Alloy 600. At the time that the cracks were discovered, Bugey-3 had operated for only 12.5 calendar years (or an equivalent of 72,000 Effective Full Power Hours (EFPH)). One inside diameter (ID) crack in the Bugey 3 plant was above the weld and extended throughwall, thereby breaching the pressure boundary. The throughwall crack was detected when primary coolant leaked through the crack during a hydrostatic pressure test that was part of the 10-year In-Service Inspection program.

The Control Element Drive Mechanism (CEDM) penetration nozzles in CEOG plants are also made of Alloy 600. These nozzles are installed in essentially the same manner as the CRDM nozzles in Bugey-3 (i.e., the Alloy 600 nozzle is placed into the head with an interference fit and then welded in place). Both European and CEOG reactor vessel heads are fabricated from low alloy steels. Hence, based on the similitude of (i) materials used, (ii) fabrication processes employed, and (iii) operating conditions to which the nozzle are exposed, there may be a potential for PWSCC in CEOG plants.

A detailed technical evaluation was performed and documented (Reference 1) to assess the potential for and consequence of ID-initiated PWSCC in CEDM nozzles. Based on that evaluation, the CEOG concluded that:

- Catastrophic failure of the reactor vessel head adaptor tubes would not occur because 1) ID circumferential cracking is not expected, and 2) axial cracks that initiate will not propagate rapidly through the nozzle wall.

- It is extremely unlikely that boric acid corrosion following leakage from a throughwall crack could continue undetected. Walkdown inspections required by Generic Letter 88-05 would reveal evidence of leakage before ASME code structural limits are challenged due to material loss from wastage.

- A 50.59 evaluation shows that an Unreviewed Safety Question is not created.

The NRC agreed with this position in Reference 6.

As of this writing, over 2500 CRDM nozzles have been inspected around the world (mostly in Europe). All cracking attributed to PWSCC has had a predominantly axial orientation and only one Bugey-3 nozzle had throughwall cracking. In addition, European reactor vessel head penetrations fabricated by ABB-CE have not encountered cracking. Nonetheless, the existence of this one case of undetected throughwall axial cracking is sufficient to raise a concern regarding the potential for and consequence of reactor vessel head penetration Alloy 600 outside diameter (OD)-initiated nozzle cracking.

To evaluate the technical and safety aspects of OD-initiated nozzle cracking, this report postulates the existence of an undetected throughwall crack from ID-initiated cracking. It should be emphasized that there is no evidence or indication that suggests that such a crack currently exists in any CEOG plant.

The evaluation that follows includes the following elements:

- A detailed stress field analysis of the OD of the thinnest (i.e., most limiting) CEDM nozzle

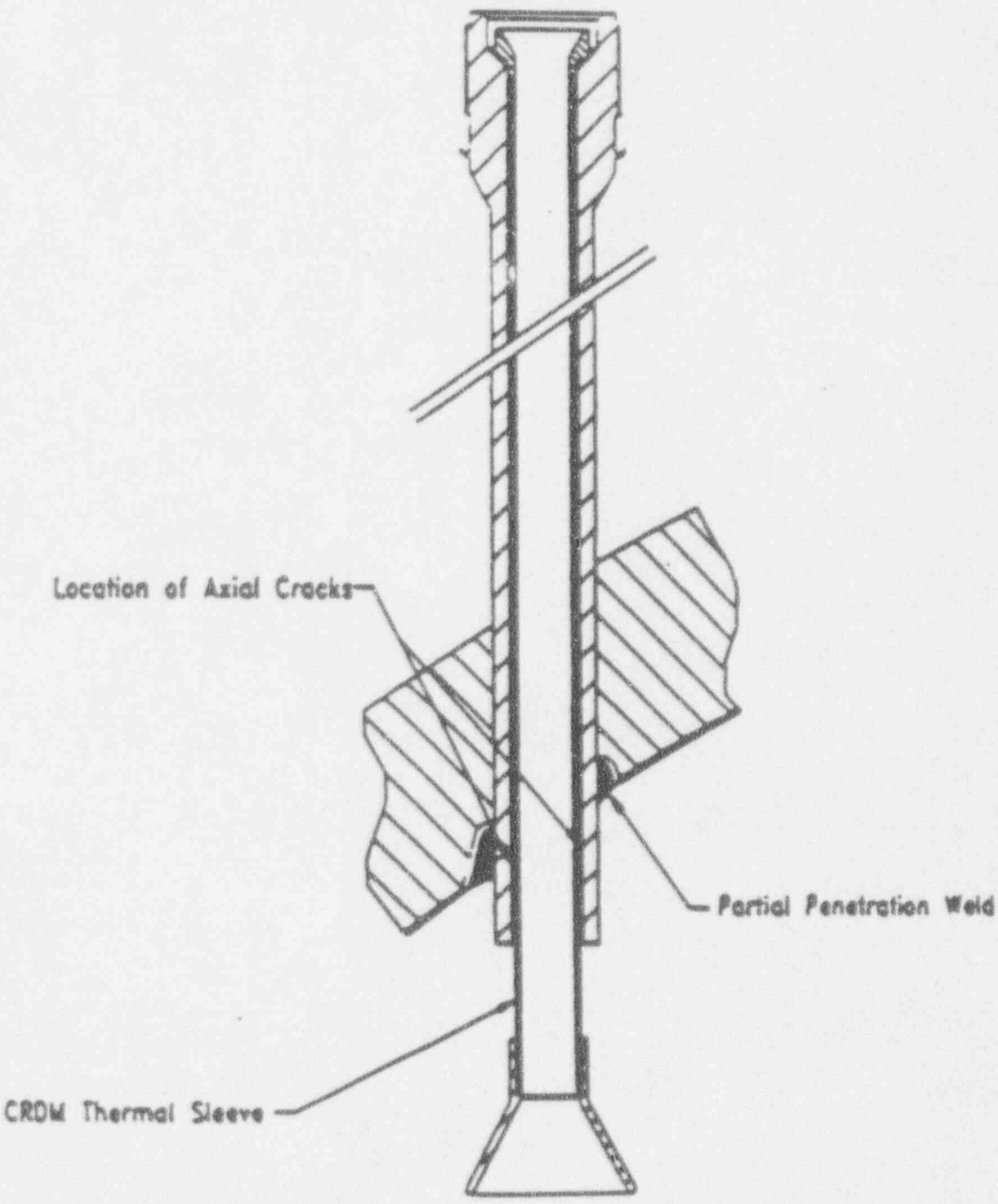
- A qualitative environmental chemistry assessment based on primary coolant leaking into the crevice between the head and the CEDM nozzle

- A conservative crack propagation calculation for a circumferentially oriented throughwall cracks initiated at the OD of the thinnest CEDM nozzle.

Based on the results of the above elements, a determination of the safety significance of OD-initiated nozzle cracking is presented. This is followed by an evaluation to determine if the potential for or consequence of OD-initiate PWSCC creates an Unreviewed Safety Question.

Figure 1.1

Schematic of a CRDM Nozzle Configuration in Bugey-3



2.0 EVALUATION OF EXISTING CONDITION

2.1 STRESS FIELD ANALYSIS

Based on the results of a sensitivity study, the stress analysis of the OD of the CEDM in this report uses 20-noded elements with an explicit heat-transfer method for application of the J-groove weld. This modelling technique was applied to a nominally thick-walled CEDM (0.662 inches) and the thinnest-walled CEDM (0.332 inches) present in CEOG plants.

2.1.1 Description of Model

The finite element model in the original CEOG CEDM analysis for consideration of the potential of ID-initiated cracking used 8-noded isoparametric thermal and structural elements. These elements did not include mid-side nodes. The original CEOG CEDM analysis used STIF 70 elements in the thermal analysis (refer to Figure 2.2.1). STIF 70 is a 3-D Solid Thermal-Isoparametric Finite element with 8 nodes that is used with the ANSYS 4.4A1 code. Each node has temperature as one additional degree of freedom. Since this is a lower-order element, linear shape functions are used to represent the temperature distribution across the element. This type of heat transfer element has a 2x2x2 lattice of integration points that are evaluated using a Gaussian integration procedure.

The general heat transfer equation that applies is:

$$\rho c_p \frac{\partial T}{\partial t} = \frac{\partial}{\partial x}\left(k_{xx}\frac{\partial T}{\partial x}\right) + \frac{\partial}{\partial y}\left(k_{yy}\frac{\partial T}{\partial y}\right) + \frac{\partial}{\partial z}\left(k_{zz}\frac{\partial T}{\partial z}\right) + \bar{\bar{q}}$$

where, ρ = density
c_p = specific heat
k_{xx} = thermal conductivity
$\bar{\bar{q}}$ = internal heat generation rate

Typically, the STIF 70 elements are acceptable for large models that require extensive computer run time. They do not, however, provide the higher level of refinement available from higher-order elements. Use of the higher-order elements involves higher-order polynomials formulations with mid-side nodes and additional integration points. Although longer run times are required, the 20-noded elements provide more precise results than the "stiffer" lower-order 8-noded elements.

The original CEOG CEDM analysis was modified to include 20-noded solid elements. The STIF 90, which is also a 3-D Solid Thermal-Isoparametric Finite element, was used. STIF 90 elements have mid-side nodes and use a quadratic polynomials in the element formulations instead of a linear expression. In addition, STIF 90 has a 3x3x3 lattice of integration points. The quadratic polynomials of the higher order element allow more "flexible" polynomials to be used in the overall element formulations.

In the structural analysis, the temperature information is read in from the thermal analysis and the associated thermal stresses are then computed. The structural element used in the original CEOG CEDM analysis was STIF 45, a 3-D isoparametric solid element (refer to Figure 2.2.1). As with the STIF 70 elements, the STIF 45 element has 8 nodal points with three degrees of freedom at each node corresponding to translations in the x, y, and z directions. Integration points are specified in a 2x2x2 lattice within the element and use a numerical (Gaussian) integration procedure. The integration point values are then extrapolated to the nodes.

The original CEOG CEDM structural analysis was modified to include 20-noded solid elements. STIF 95, a higher-order element compared to STIF 45, is also a 3-D isoparametric solid used in structural analyses, but has mid-side nodes activated. Each node has three degrees of freedom corresponding to translations in the x, y, and z directions. This element also has a plasticity feature and that is well-suited to model curved boundaries such as the reactor vessel head. The integration points in this element include 8 integration points at the corners and 6 integration points located near the centers of each element face for a total of 14 integration points. Again, the integration point values, are evaluated using a Gaussian integration procedure and extrapolated to the nodes. These additional integration points provide a finer distribution of results when compared to a lower-order element.

2.1.2 Description of Loading

2.1.2.1 Welding Process

The original CEOG analysis explicitly modeled the heat transfer characteristics of the CEDM nozzle and the adjacent reactor vessel head during the welding process. Specifically, a two-pass J-groove weld buildup model is simulated. This method uses first principle heat transfer equations to generate the temperature distribution in the components (See Figure 2.2.2). Engineering judgement, as confirmed by numerical analysis, continues to suggest that the primary contributor to stresses in the J-groove region is the residual stresses introduced during the fabrication process. Thus, the selection of an accurate simulation of the welding process is considered to be important.

The heat transfer model consists of a thermal analysis that is applied to structural analysis. In the thermal analysis, the first weld pass is heated to approximately 3000°F and then allowed to cool to an intermediate temperature before a second weld pass reheats the region to approximately 3000°F. Both weld passes are given the same heat inputs and are allowed to cool to ambient temperature. The structural analysis uses the temperature information from the thermal analysis and the temperature-dependent weld material properties which increase in strength as the weld material cools to ambient temperature. The outer shell nodes and symmetric locations are restrained from moving.

2.1.2.2 Hydrostatic Testing

After the residual stress loading is applied by the heat transfer model, a hydrostatic test load is applied. To do this, the pressure in the Alloy 600 nozzle is gradually increased. Specifically, the hydrostatic test begins at 610 psia and reaches a value of 3110 psia before it is gradually decreased to approximately zero pressure. This sequence is performed twice at ambient temperature.

2.1.2.3 Normal In-Service Conditions

The normal operating pressure is first cycled through a controlled heatup and cooldown transient. The first operating conditions cycle begins at 100°F and 203 psia. Both the temperature and pressure are gradually increased to 600°F and 2235 psia and then gradually reduced to 70°F and zero psia. During the second operating conditions cycle, the initial conditions are again 100°F and 203 psia and the final vessel temperature and pressure are increased to 600°F and 2235 psia. This time, however, the model is held at these conditions to simulate operation of the power plant.

2.1.3 Results

The region of most concern in this analysis is the two element rings above the weld region in the Alloy 600 tube; nodes C59-C1315 and B69-B1325 (cf., Figure 2.2.3). These two elements are important since they have the highest stress to cause a crack to initiate and subsequently propagate. The weld region, itself, has relatively large compressive regions.

Table 2.1.1 provides results from node locations above the weld for the 20-noded heat transfer finite element analysis of the thin-walled and nominal-walled CEDM in CEOG plants. The table shows that the stresses for the thin wall case are more adverse than those for the nominal case with hoop stresses predominating in magnitude over axial stress, for both cases.

Color stress contour plots (Figures 2.2.4 through 2.2.6) for a ring of elements above the weld are provided for the thin-wall case. Figure 2.2.4 show that compressive zones do not extended uniformly around the OD of the nozzle. Thus, in the presence of tensile loading consideration is given to the potential for PWSCC on the OD if an ID-initiated throughwall crack is postulated.

A linear plot (Figure 2.2.7) provides additional information regarding the magnitude and behavior of axial and hoop stresses in the thin-wall nozzle. This detailed information is input to the crack propagation analysis described in Section 2.3.

Table 2.1.1

Selected Results of CEDM Structural Field Analysis

20-noded Heat Transfer 3-D Model	Thinnest CEDM	Nominal CEDM
OD (inches)	4.047	4.047
ID (inches)	3.382	2.719
t (inches)	0.332	0.664
OD/t	12.2	6.1
Max OD Hoop Stress (ksi) --Downhill	67	-7.5
--Uphill	70	29
Max OD Axial Stress (ksi) --Downhill	20	4.0
--Uphill	36	15

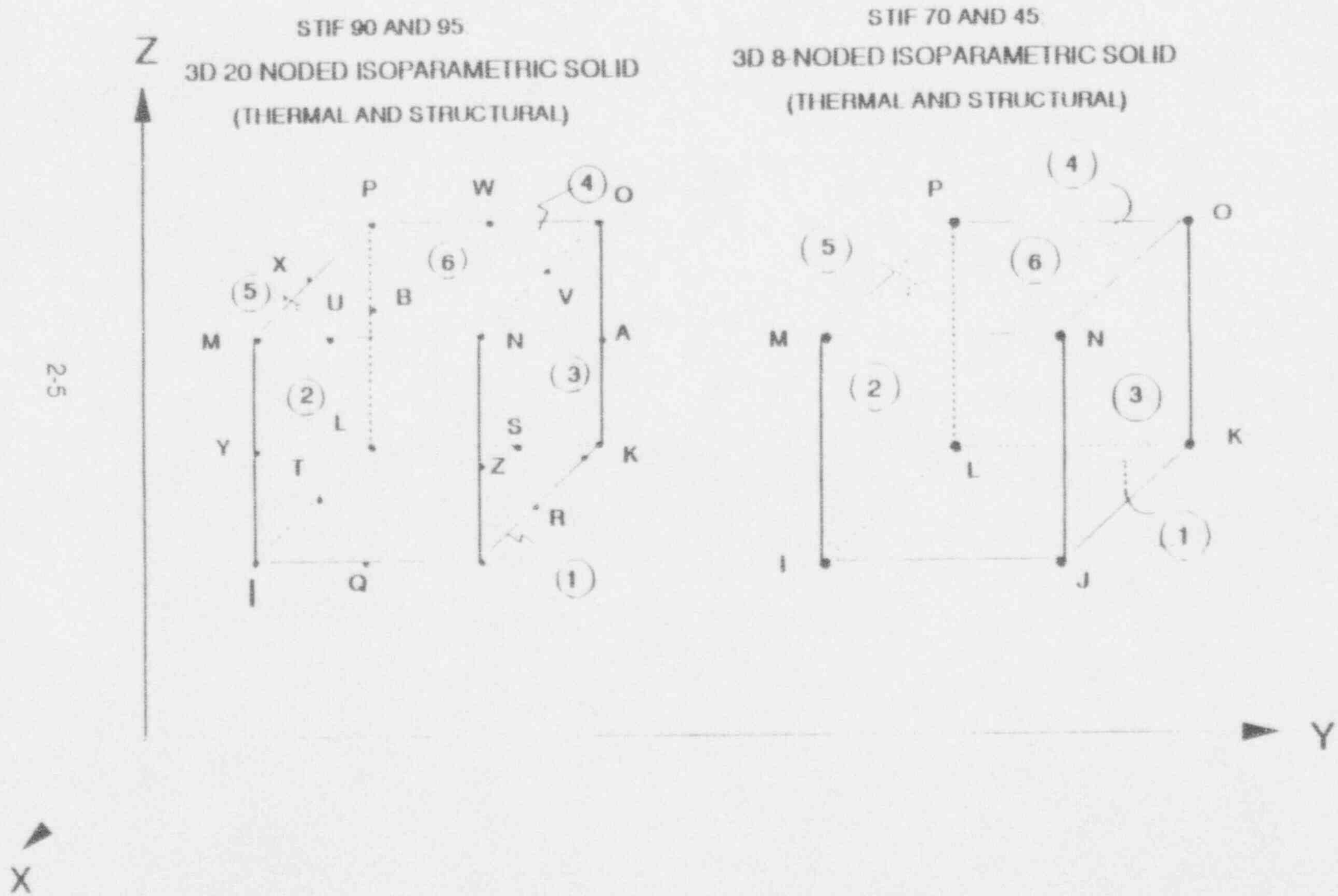


Figure 2.2.1
Finite Element Model Nodalizations

Figure 2.2.2
Modelling of Welding Process

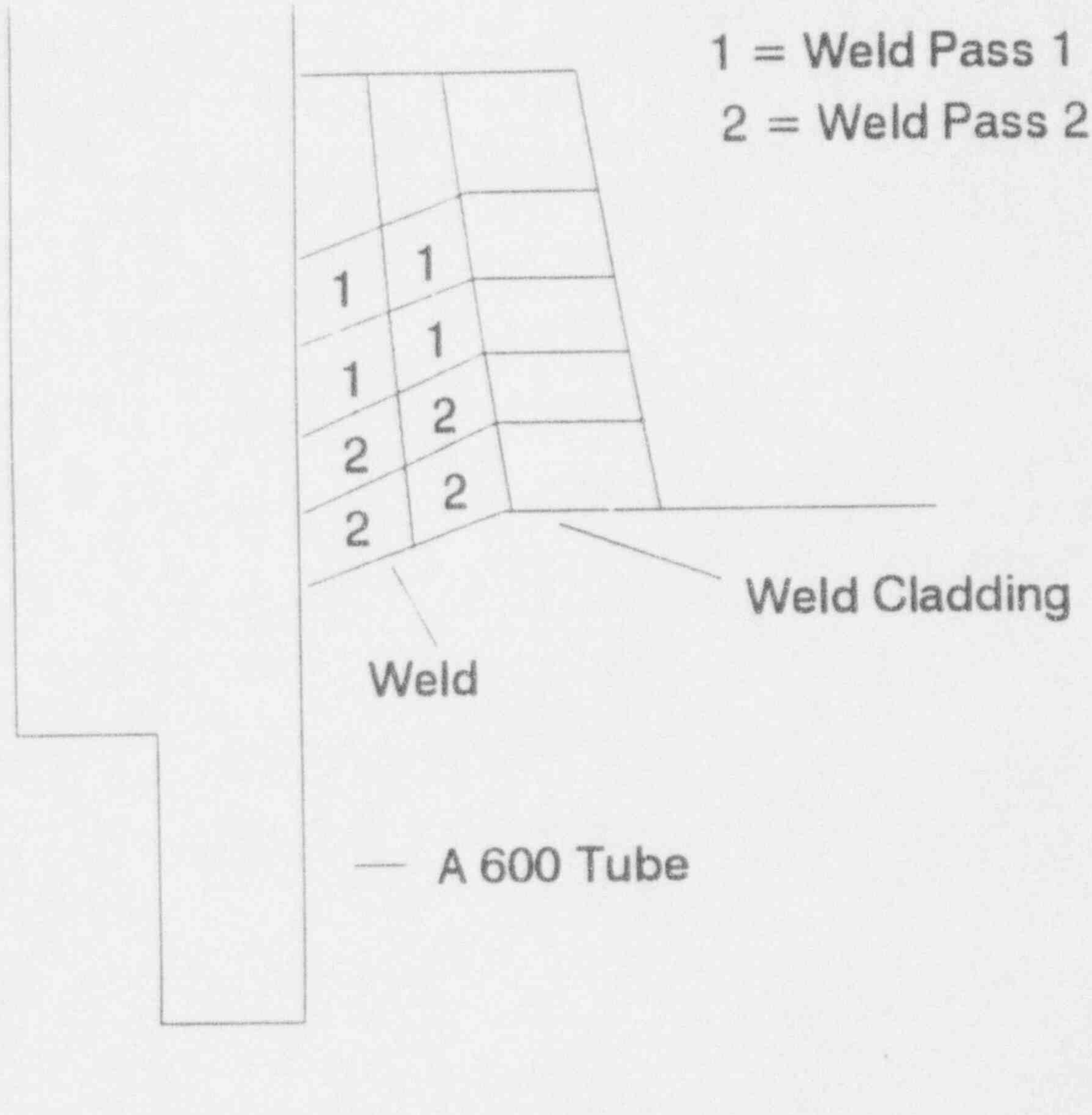
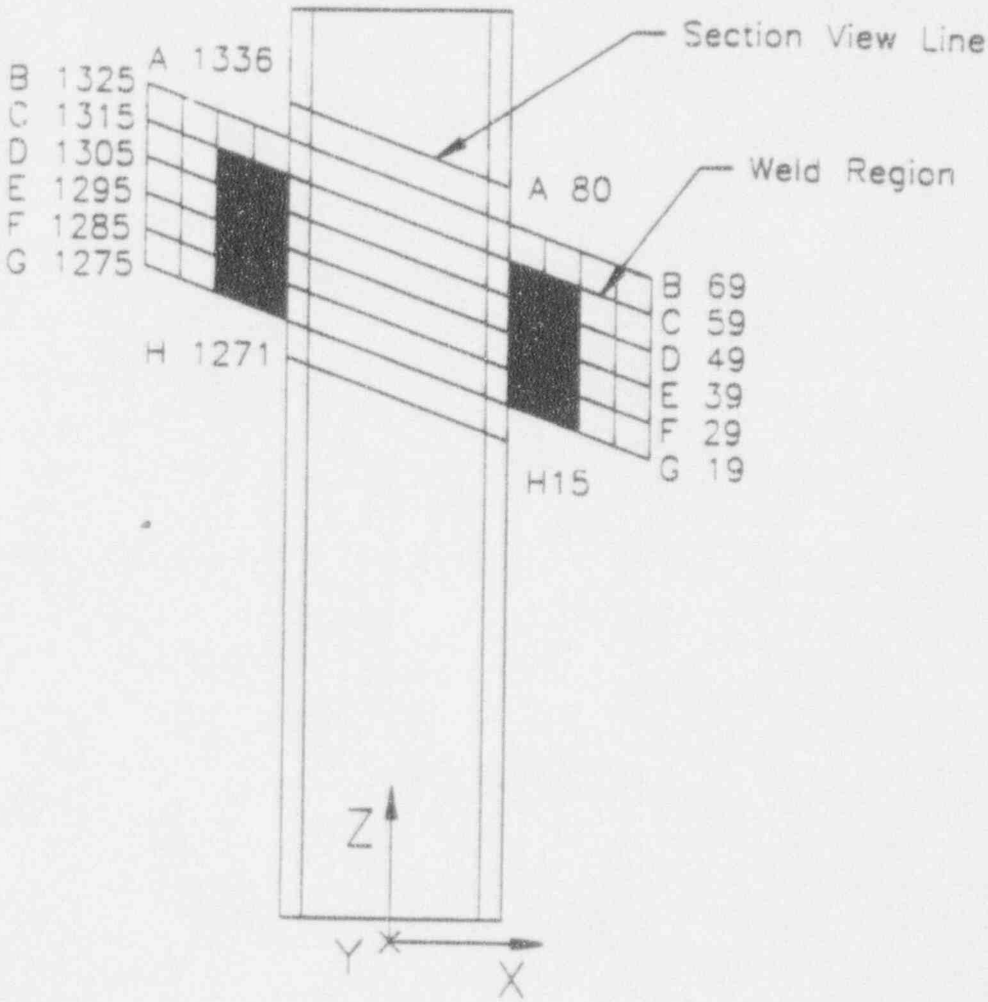


Figure 2.2.3
Modelling of CEDM Nozzle Configuration



SZ = Axial Stress
SY = Hoop Stress

Figure 2.2.4
Thin-walled CEDM Axial Stress Pattern
[Color Plot]

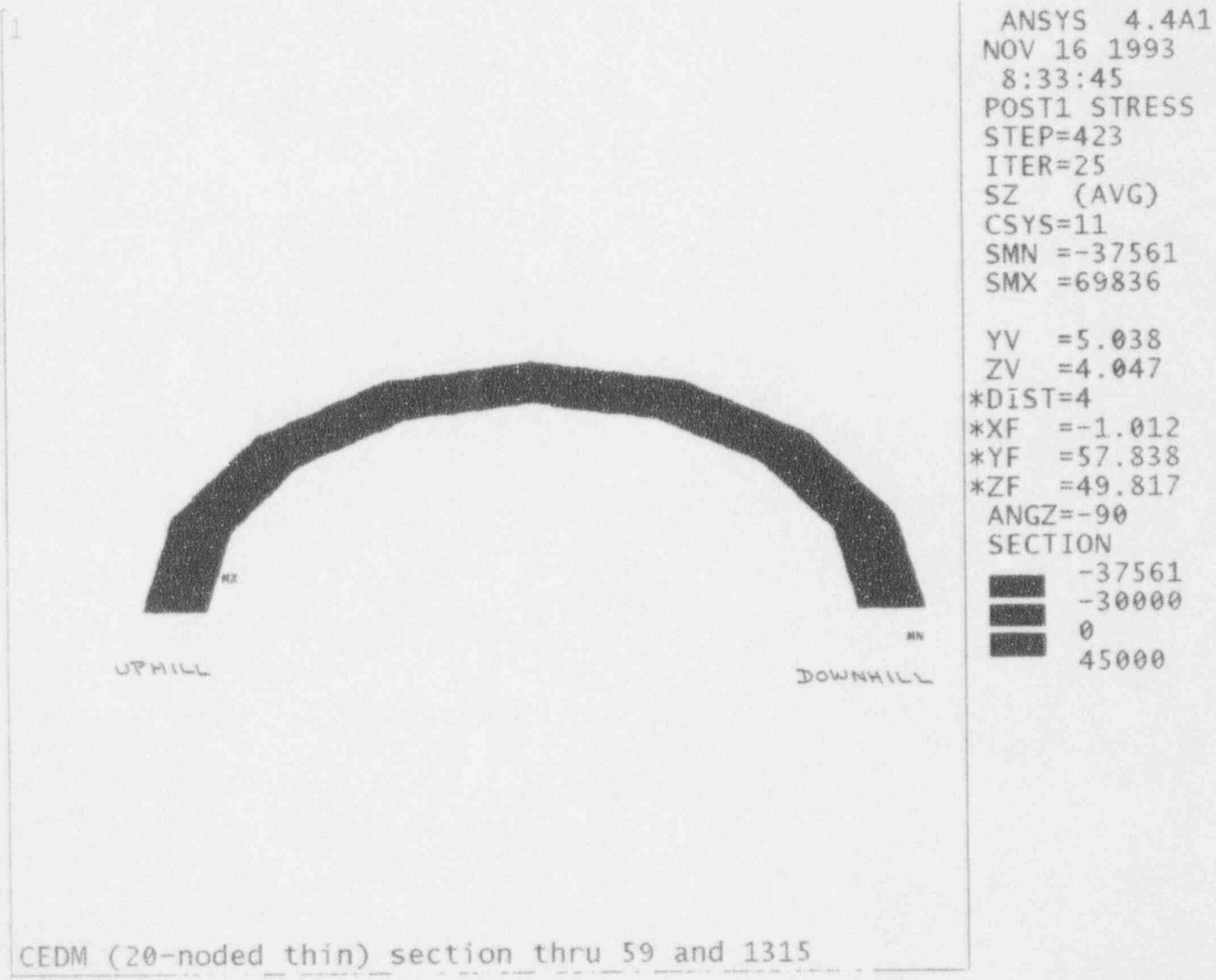


Figure 2.2.5
Thin-walled CEDM Detailed Axial Stress Pattern
[Color Plot]

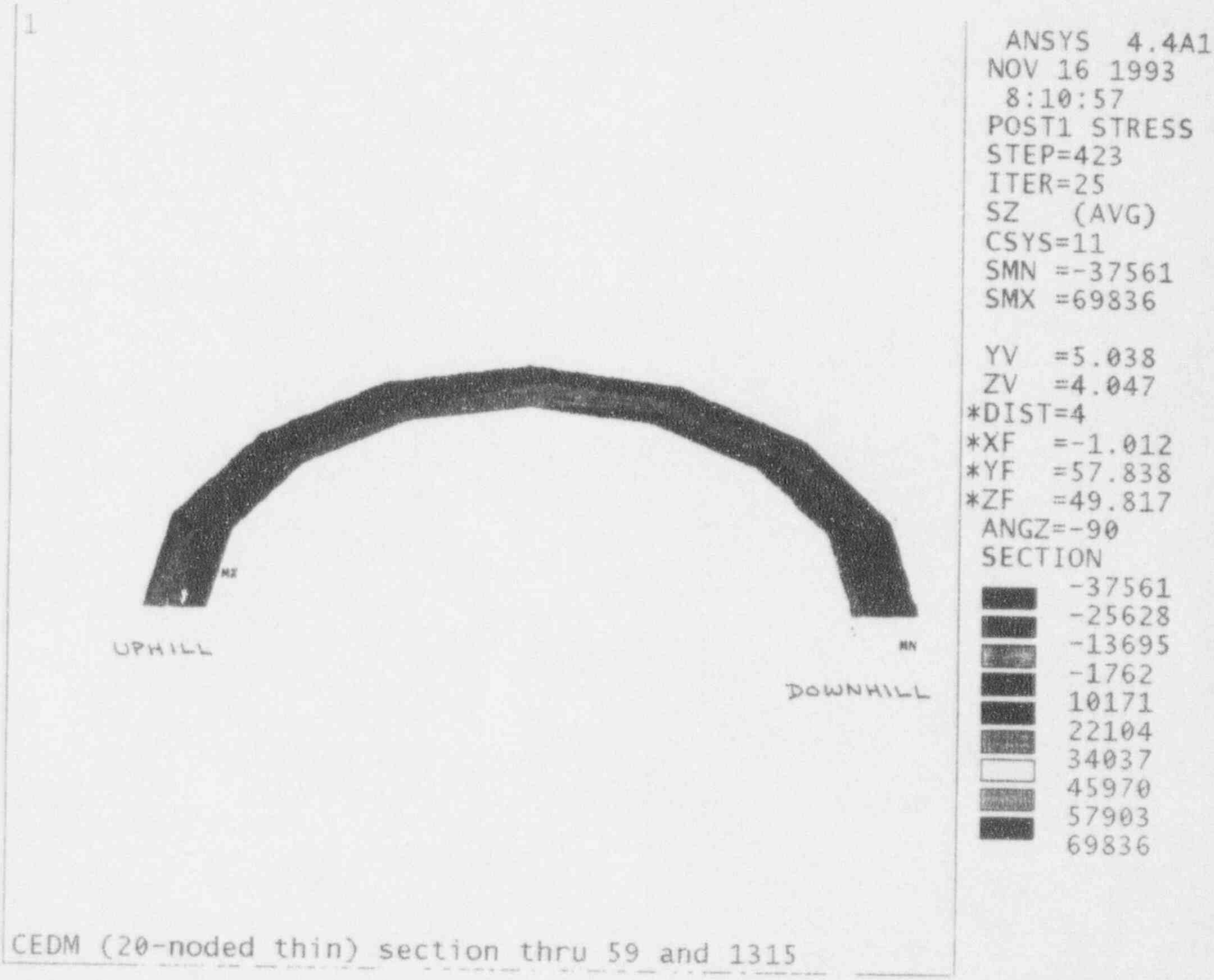


Figure 2.2.6
Thin-walled CEDM Detailed Hoop Stress Pattern
[Color Plot]

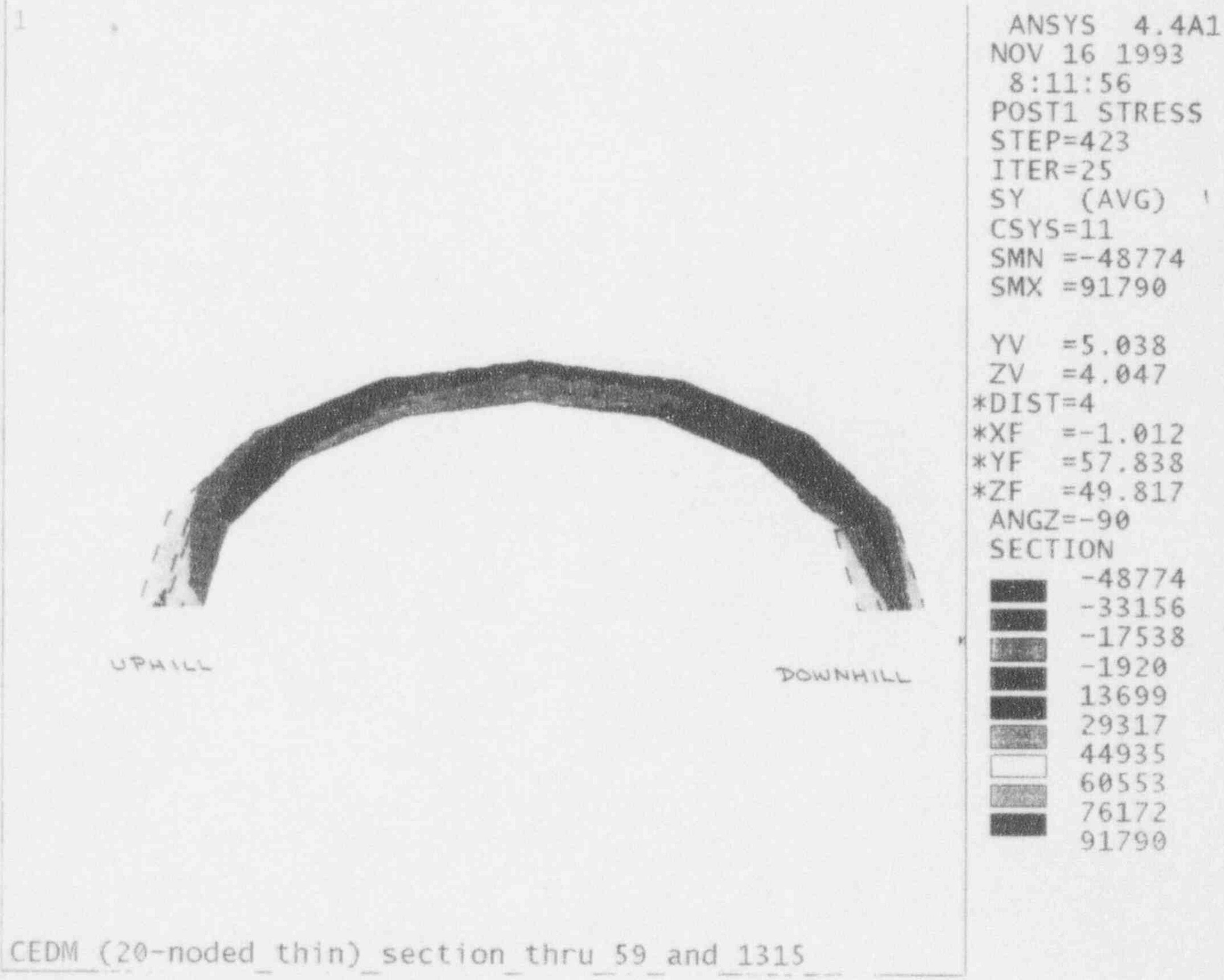


Figure 2.2.7
Nominal-walled CEDM Detailed Axial Stress Pattern
[Color Plot]

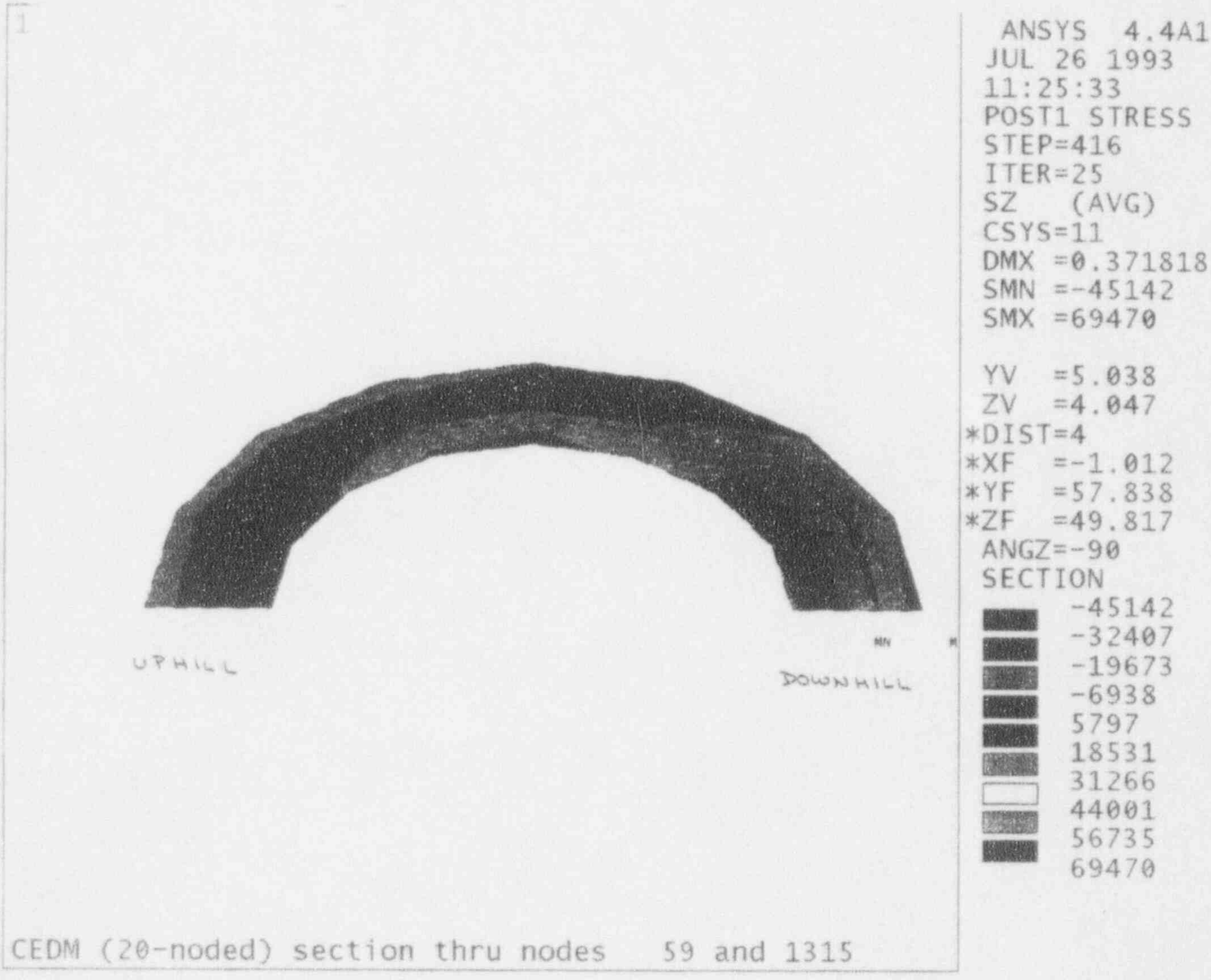
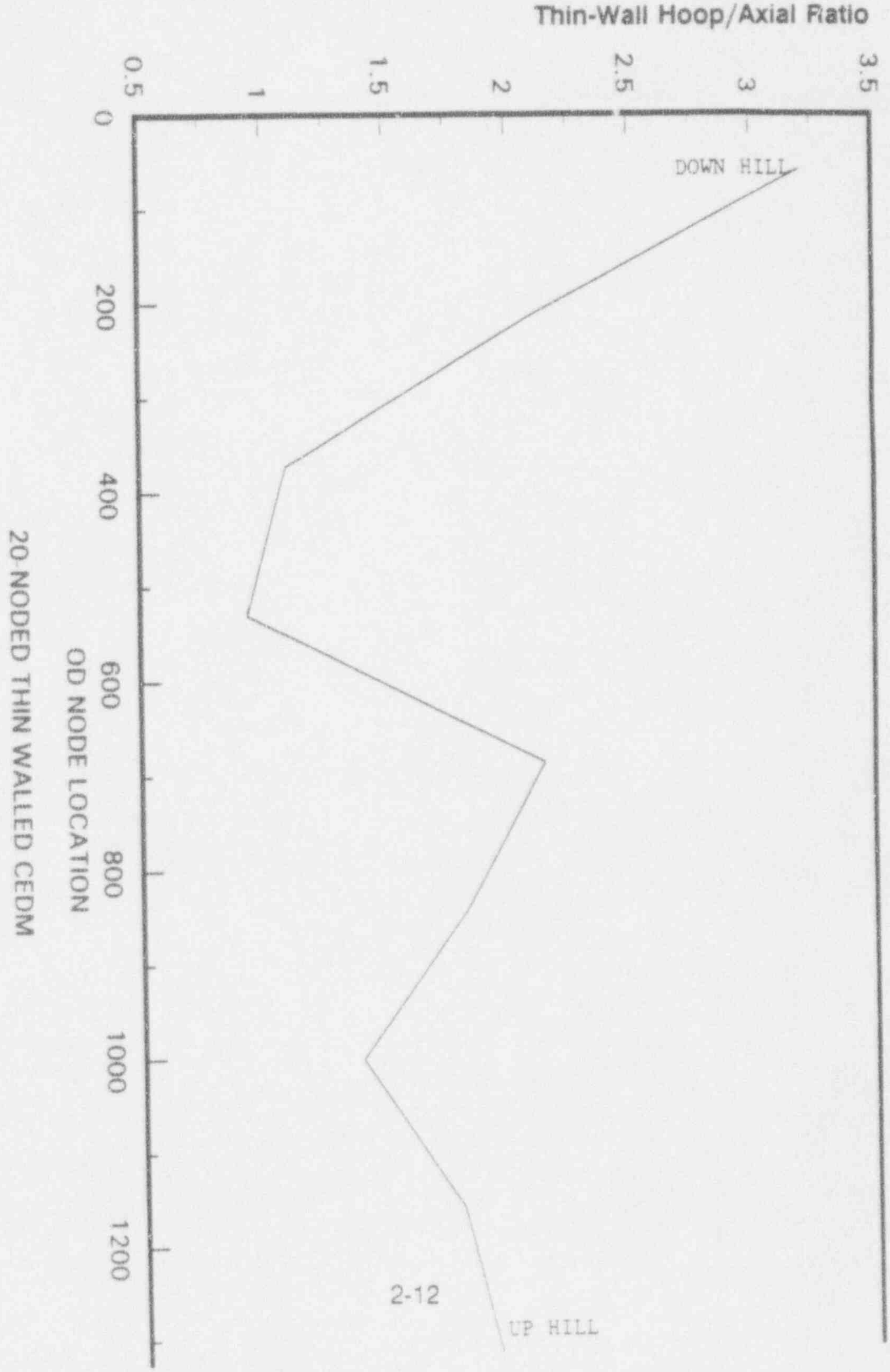


Figure 2.2.8
Thin-Wall Hoop/Axial Ratio



2.2 ENVIRONMENTAL CHEMISTRY ASSESSMENT

Before discussing crack propagation, a discussion of the potential for OD crack initiation is relevant. Assuming a throughwall ID-initiated crack, similar cracking on the OD of the nozzle can only occur if coolant can enter into the crevice between the nozzle and reactor vessel head. Presumably, this is how the circumferential crack found on the OD of the nozzle in Bugey-3 was formed. It is important to note, however, that while the small OD circumferential crack in the leaking Bugey-3 nozzle is proposed to have been initiated by PWSCC, the laboratory report of this crack stated that it was not possible to determine if the crack initiated on the outside surface or was actually a component of the ID initiated throughwall crack (Reference 2). This uncertainty is significant in the context of the evaluation below since OD-initiated PWSCC has never before been observed in any operating plant.

The EdF technical evaluation concluded that the small size of the OD crack in the leaking Bugey 3 nozzle was not a safety problem. EdF is continuing to assess the potential for crack growth on the OD of the nozzle. The French work is focusing on the environmental effects on crack growth rate. In particular, Framatome is conducting crack growth rate tests using Alloy 600 tubing material in highly concentrated lithium/boric acid environments. Preliminary Framatone results show no cracking after 500 hours. The EdF program for evaluating crack growth rate also includes testing of forged material like that used to fabricate the reactor vessel head CRDM nozzles.

The evaluation that follows focuses on the potential for OD-initiated PWSCC in CEDM nozzles from an environmental chemistry perspective. It is assumed that the other conditions necessary for PWSCC are present; namely, high tensile stress and a susceptible material. The potential for caustic induced SCC, resulting from the concentration of Lithium, is also evaluated.

The approach used to analyze whether primary coolant leakage from a leaking CEDM nozzle could initiate or sustain OD cracking considers two different scenarios. Each scenario is distinguished by the nature of the crevice between the nozzle and the reactor vessel head as described below.

The "Sealed" Crevice - This scenario has a crevice that is effectively sealed from the containment environment by a tight, interference fit between the nozzle and the reactor vessel head. Such a crevice provides a chamber or series of chambers (voids) into which primary coolant would enter following throughwall PWSCC of the nozzle (ID-to-OD). The sealed crevice scenario is possible given the lack of evidence of nozzle leakage at Bugey-3 during operation.

The "Vented" Crevice - This scenario is defined by assuming that there is direct pathway through which primary coolant leaking into the crevice exits to the contaminant environment.

2.2.1 Sealed Crevice Scenario

In the first case, the environmental conditions in the crevice are expected to be quite similar to those in the circulating coolant. The coolant chemistry in the crevice will be the same as the primary coolant at the time the throughwall crack opens sufficiently to allow coolant to enter the crevice (assuming zero solution replenishment). Depending on when in the operating cycle the failure occurs, the lithium concentration will vary. The pH, however, is expected to remain in the typical operating range of 6.9-7.4. The temperature of the coolant within the crevice (reactor vessel head region) is estimated to be about 4°F cooler than the hot leg temperature. Thus, the chemical and temperature environment in the crevice is essentially that of circulating primary coolant and the likelihood for PWSCC on the OD of the nozzle is also essentially the same as that for the ID of the nozzle.

The expectation that OD cracks can occur in a sealed crevice, however, is considered to be overly conservative. The tight fit of the nozzle in the reactor vessel head could minimize or even prevent the opening of any throughwall crack, thereby, eliminating primary water leakage through the crack. Any residual oxygen in the primary coolant will react with the low alloy steel or be trapped in the crevice. Thus, an oxidizing environment will not be present in the crevice. An oxidizing environment could be detrimental to PWSCC initiation and growth.

2.2.2 Vented Crevice Scenario

In a vented situation, there is a potential for lithium hydroxide to concentrate in the crevice and significantly increase the likelihood for caustic-induced OD stress corrosion cracking of the Alloy 600 nozzle. In the vented crevice scenario, however, large quantities of boric acid should be plainly visible on the reactor vessel head as primary water evaporates, carrying boric acid off in the escaping steam even with very small (0.01 gpm) leak rates. While a mechanism for concentrating lithium hydroxide (and boric acid) is established, evidence of a CEDM leak would be manifested long before a critical caustic environment could be developed in the crevice.

In addition, boric acid will concentrate simultaneous with lithium hydroxide. Since the boric acid concentration in the crevice solution (coolant) is much greater than that of lithium hydroxide, a significant neutralizing effect is expected. Although boric acid has a greater volatility relative to lithium, the major portion of boric acid is expected to remain in the crevice.

Empirical evidence from experiments conducted by the Japanese suggests that boric acid mitigates caustic SCC propagation (References 3 and 4). The inhibitory effect of boric acid appears to be correlated with the boron content incorporated into the oxide film at the crack tip which increases with increasing solution pH. The molar ratio of boron-to-sodium in these referenced studies was approximately 7:1. The molar ratio of boron-to-lithium in the reactor coolant will always exceed 7:1 for the entire fuel cycle. Hence, there will exist an excess of boric acid relative to lithium hydroxide in the vented crevice environment. Thus, if CEDM leakage was to go undetected for a period of time, it is unlikely that even concentrated crevice solutions could sustain crack propagation to a point of concern.

Based on the above, it is concluded that CEDM leakage into an unsealed (i.e., opened) crevice would be detected in advance of the initiation and propagation of caustic-induced OD stress corrosion cracking.

2.2.3 Boric Acid Corrosion

The effects of borated water on the low alloy steel head material was evaluated in detail in Reference 1. This evaluation indicated that low level undetected leakage could persist for 8.8 years without degrading the integrity of the head. Furthermore, borated water trapped within a seal crevice would result in only minor corrosion of the low alloy steel.

2.2.4 Results

Depending upon whether a sealed or vented crevice exists, one of two chemical environments could arise on the OD surface of a CEDM nozzle. An assessment shows that neither environment can cause initiation of PWSCC or caustic-induced SCC. Although cracking on the OD is not expected, the next section calculates the crack growth behavior for a postulated circumferential crack on the OD surface.

2.3 CRACK PROPAGATION CALCULATIONS

This section describes the fracture mechanics analysis performed to determine the time required for a postulated OD surface crack to propagate first through the CEDM wall and then around the CEDM circumference. Fracture mechanics analyses were performed on both a thin wall and a nominal wall CEDM penetration.

2.3.1 Methodology and Assumptions

The crack propagation evaluation is based on several assumptions. First, it is assumed that an ID-initiated axial flaw propagates through the CEDM nozzle above the weld. Although not expected, it is further assumed that a surface flaw is initiated on the OD of the CEDM nozzle as a result of prolonged exposure to reactor coolant. Based on this postulated OD flaw, the CEDM OD cracking potential is evaluated.

The OD circumferential flaw is assumed to have an aspect ratio of 3:1 - crack length to the crack depth. The time to propagate this OD circumferential flaw through to the ID is calculated. The resulting throughwall crack is then propagated around the periphery of the CEDM tube. The time for the crack to propagate to the minimum ligament length is then determined. The minimum ligament length is defined as the smallest uncracked arc length required for the CEDM nozzle to support the axial force on the CEDM end-cap resulting from operating pressure. The time required for the OD circumferential flaw to propagate throughwall and the time required for this throughwall crack to reach the minimum ligament length are added. This total time provides a conservative estimate of the minimum time for which the structural integrity of the CEDM is assured.

The K_I, stress intensity factor, used in the crack growth calculation is calculated using standard LEFM (Linear Elastic Fracture Mechanics) techniques. The average throughwall stress distribution in the CEDM is used to develop the stress intensity factors for both the thin wall and nominal wall CEDM nozzles. This data was developed in the finite element analysis described in Section 2.1 and shown in Figures 2.2.5 and 2.2.7. The crack propagation rates were found by integrating a K_I verse crack growth curve. The crack growth rate curve was developed from References 7 and 8.

The thin and nominal wall CEDM have dimensions shown in Table 2.3.1 below. For both cases, an initial OD surface flaw with a depth (a) of 0.0032 inch is assumed (cf., Figure 2.3.1). As previously mentioned, the crack is assumed to have an aspect ratio of 3:1. This ratio remains constant throughout the throughwall propagation. The temperature of the CEDM at the location of cracking is conservatively assumed to be 600°F for the thin wall analysis and 620°F for the nominal wall analysis.

Table 2.3.1
CEDM Dimensions

	Thin Wall CEDM (inch)	Nominal Wall CEDM (inch)
Inner Radius (R_i)	1.69	1.36
Mean Radius (R_m)	1.86	1.69
Outer Radius (R_o)	2.02	2.02
Thickness (t)	0.33	0.66

It is conservatively assumed that the OD surface crack propagates through the CEDM wall at the worst average throughwall stress location. In addition, the stress field driving the crack is for the uncracked configuration. That is, the axial stress distribution of the uncracked structure is conservatively applied to the cracked structure. It should be noted that any loads applied to the CEDM nozzle external to the vessel head, such as straightening loads during fabrication or seismic loads during operation, are not transmitted to the weld region. These loads are resisted by the shrink fit region of the CEDM in the vessel head and are, therefore, by design not transmitted to the partial penetration weld.

After the flaw is allowed to propagate throughwall, a second propagation model is employed to extend the crack around the nozzle (cf., Figure 2.3.2). Since global bending of the CEDM is prevented by the shrink-fit region, the crack is modelled by a center crack propagating in a finite plate (cf., Figure 2.3.3). The width of the plate corresponds to the average circumference of the CEDM. The initial length (l) of the crack for this portion of the analysis is based on the dimension of the throughwall crack at the mid-wall location (cf., Figure 2.3.2). This corresponds to a crack length of 0.86 inch for the thin wall analysis and 1.73 inches for the nominal wall analysis.

As with the throughwall crack propagation model, the stress field driving the crack is assumed to be the average stress acting upon the crack face, and is linearly interpolated from the stress distribution developed in the finite element analysis (see Figures 2.2.5 and 2.2.7). The axial stress distribution of the uncracked structure is conservatively applied to the cracked structure.

The crack propagates to a final length (l) of about 10.7 inches in the thin wall CEDM in about 81 years. This is the minimum ligament length required to support the axial load caused by internal pressure. The nominal-wall analysis shows that a crack length (l) of 3.26 inch is reached after about 100 years of simulated crack growth. In the nominal wall CEDM nozzle, the crack propagates very slowly. As a result, the analysis was terminated before the minimum ligament length was reached.

2.3.3 Results

The results of the thin wall CEDM crack growth analysis are presented Table 2.3.2. This table shows that it takes a postulated crack oriented in the circumferential direction 10 years to propagate through the wall and 81 years to propagate around wall before the minimum ligament length is reached. The crack has an arc length of 10.7 inches at this point and it extends about 330° around the CEDM. These results are presented graphically in Figures 2.3.4 and 2.3.5.

Table 2.3.2
Thin Wall Analysis Results

Analysis	Initial Crack Size [in]	Final Crack Size [in]	Propagation Time [yrs]
Throughwall	0.0032	0.33	10
Around Wall	C.43	10.7	81
Total Time	NA	NA	91

The results of the nominal wall CEDM crack growth analysis are presented Table 2.3.3. For the nominal wall case, it takes a postulated circumferential crack 11 years to go through the wall and 100 years to reach a size of 3.26 inches (which extends approximately 110° around the CEDM). To reach the minimum ligament length, the crack would have to extend almost 350° around the nozzle. These results are presented graphically in Figures 2.3.6 and 2.3.7.

Table 2.3.3
Nominal Wall Analysis Results

Analysis	Initial Crack Size [in]	Final Crack Size [in]	Propagation Time [yrs]
Throughwall	0.0032	0.66	11
Around Wall	0.86	3.26	100*
Total Time	NA	NA	111

* analysis intentionally terminated at this time

Figure 2.3.1
Through-wall Crack Propagation Model

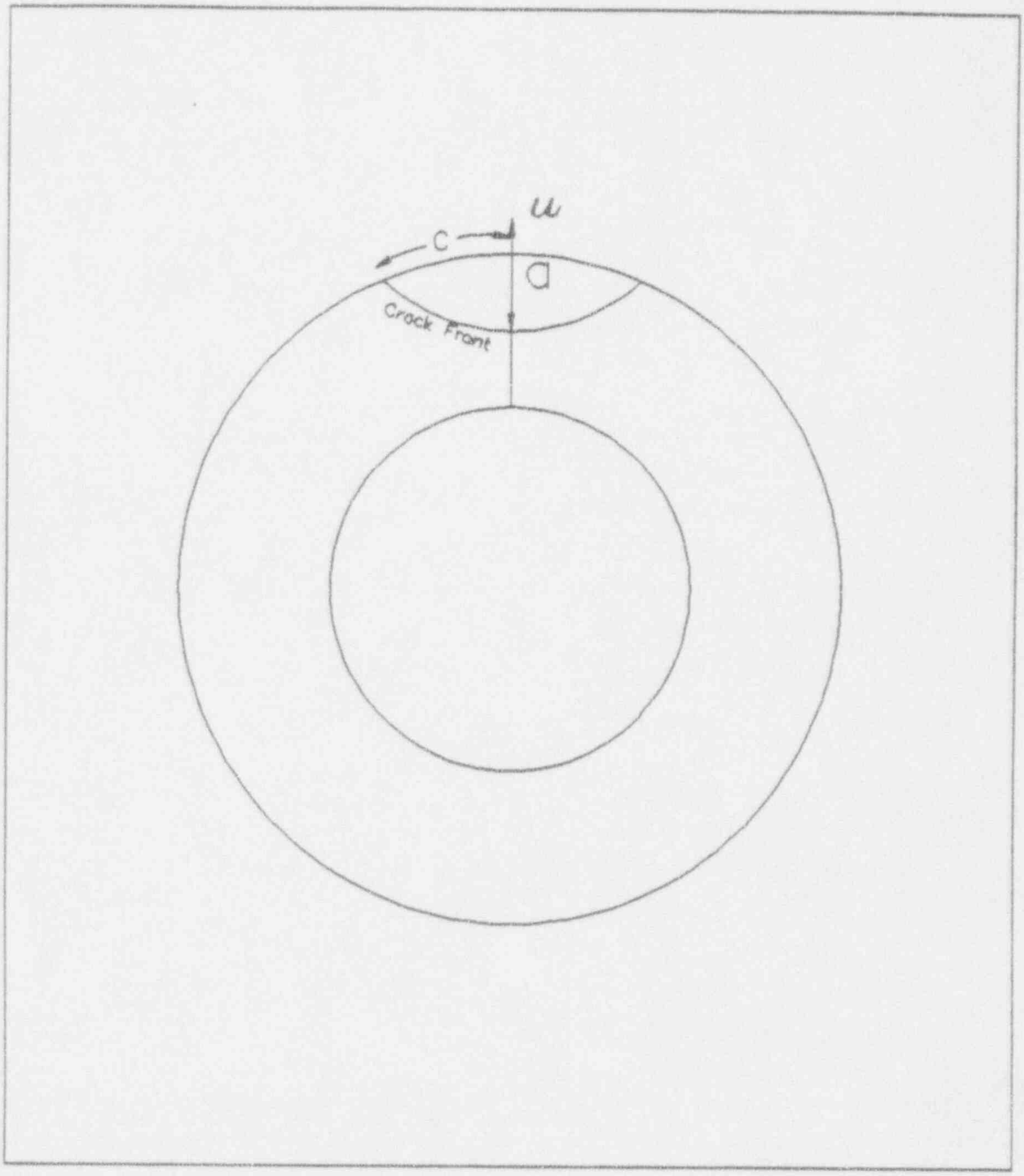


Figure 2.3.2
Around-wall Crack Propagation Model

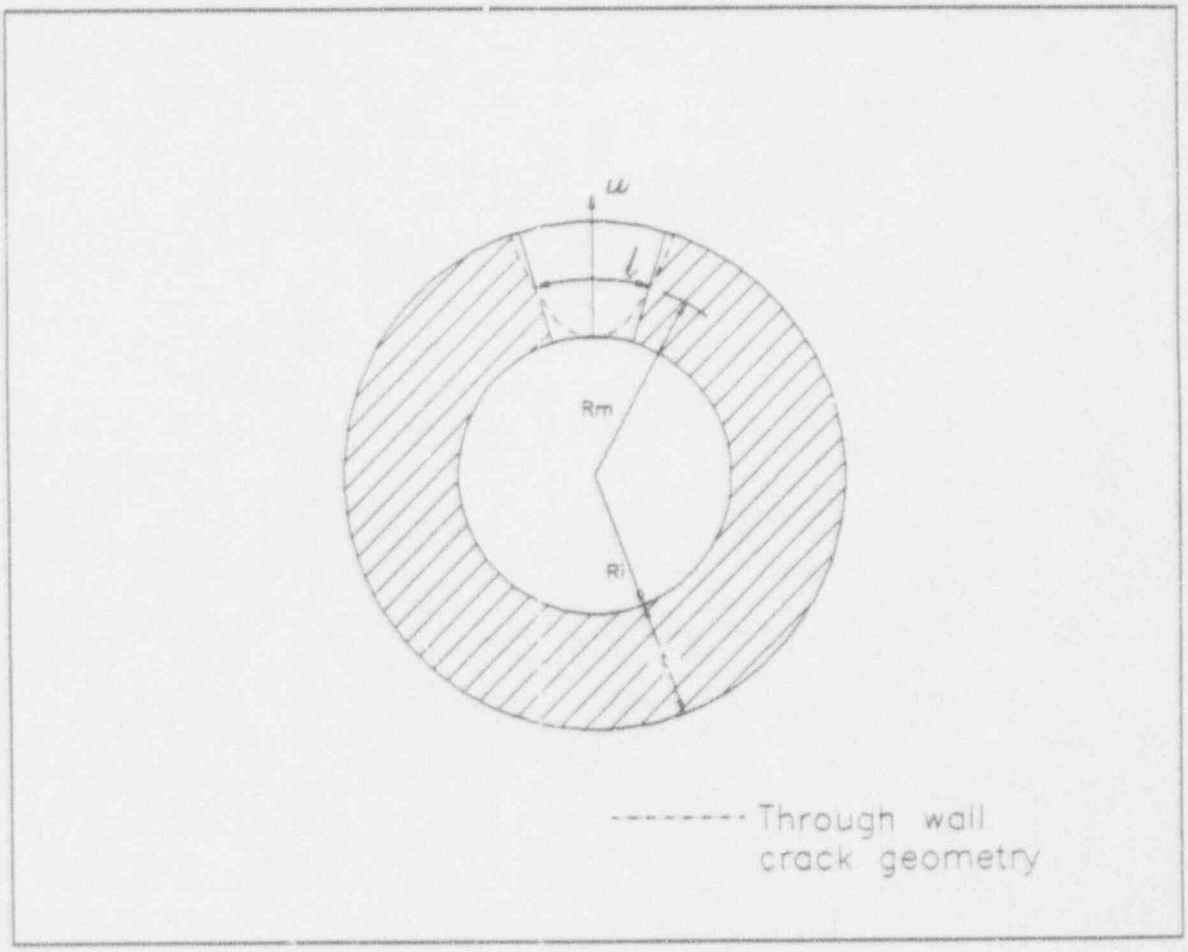
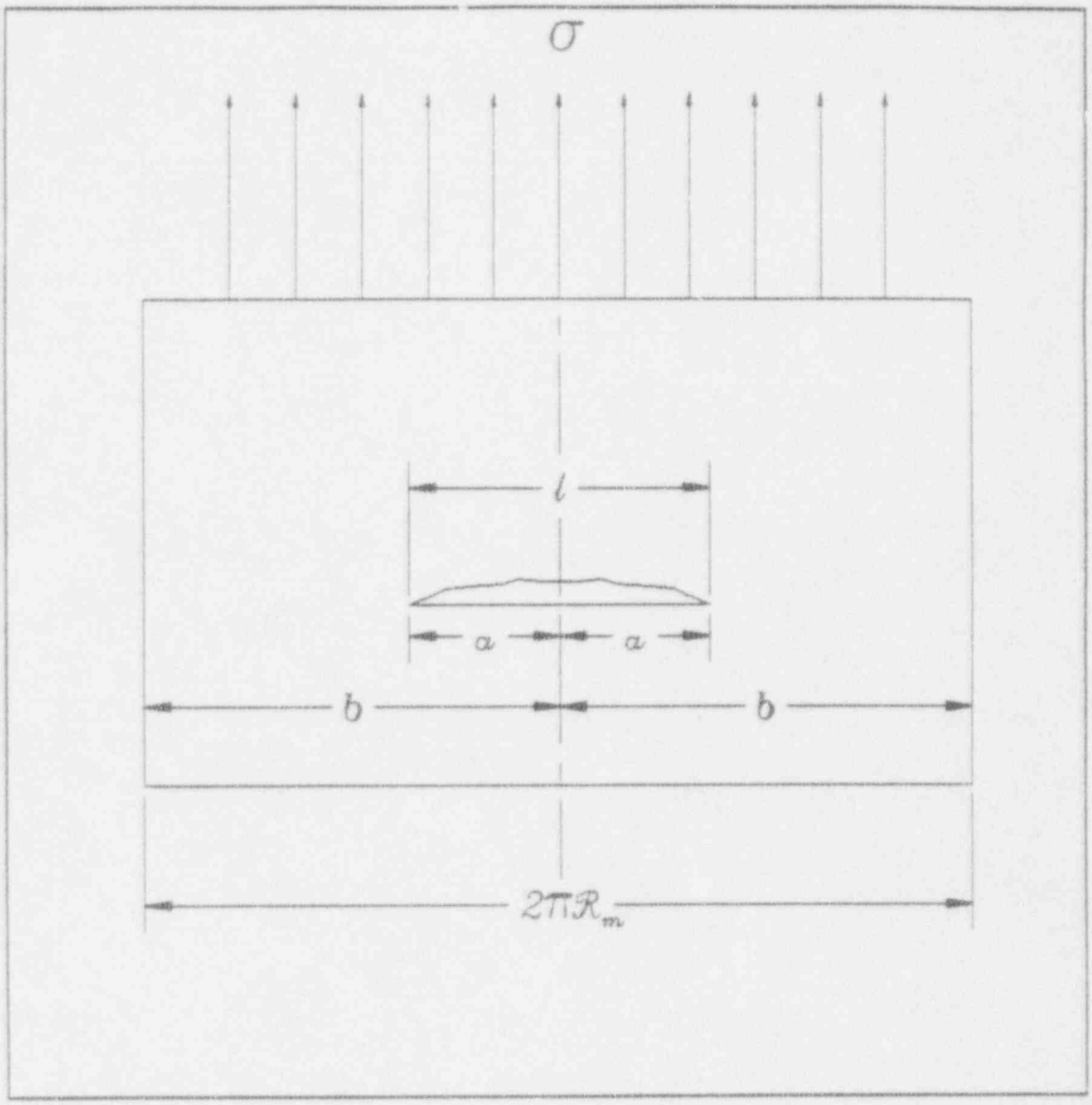


Figure 2.3.3
Infinite Plate Crack Propagation Model



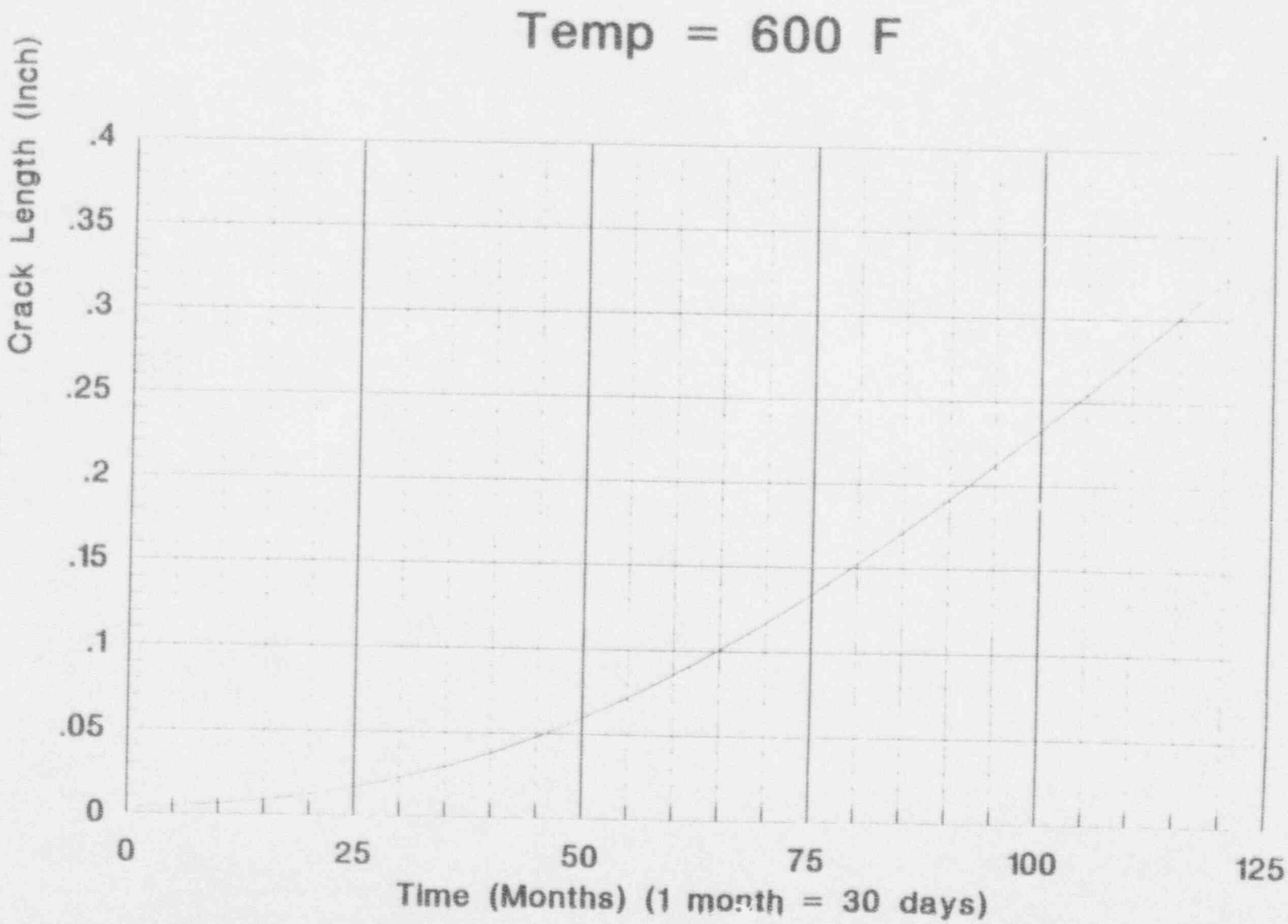


Figure 2.3.4
Thin-walled CEDM Through-wall Crack Growth Curve

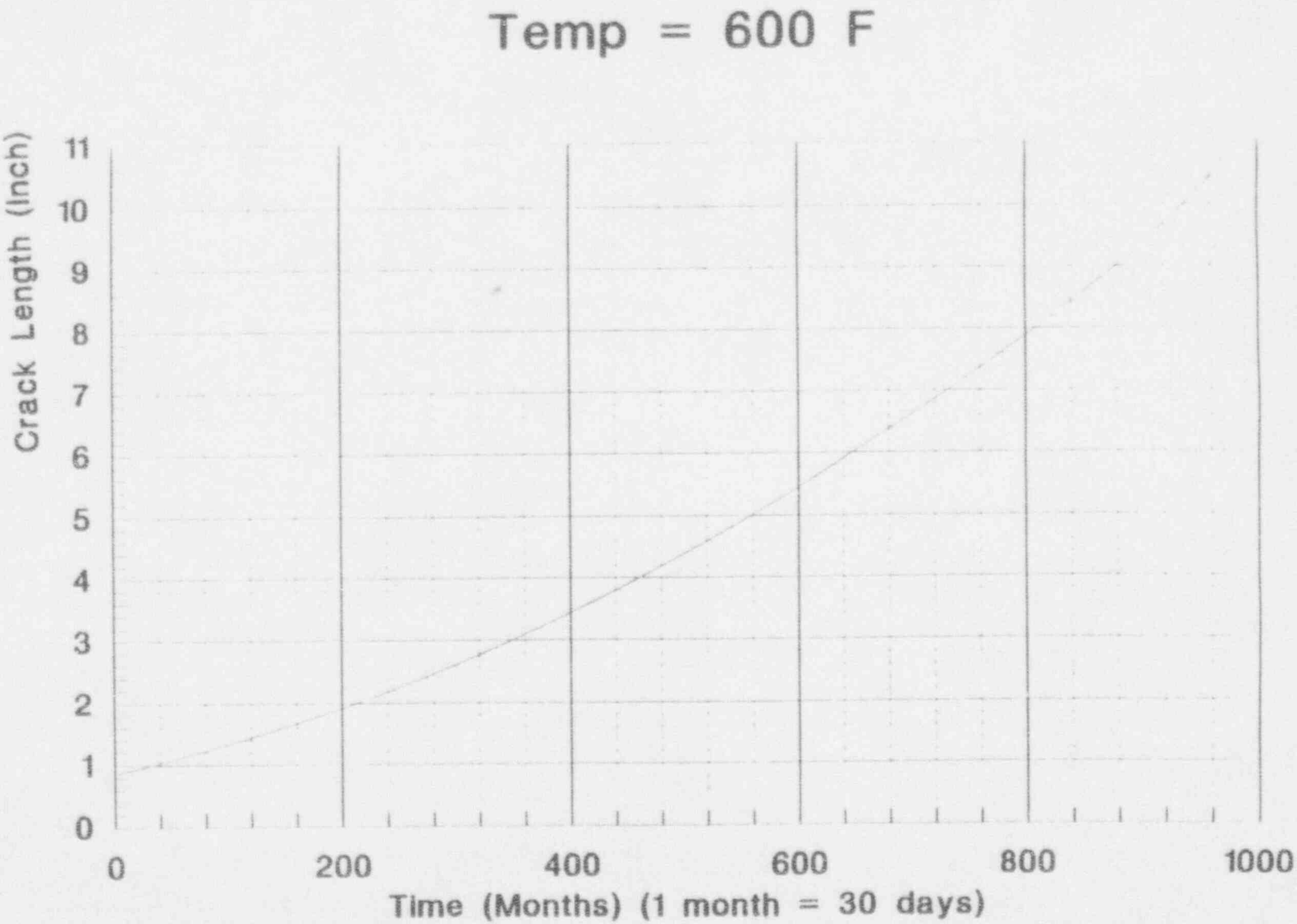


Figure 2.3.5
Thin-walled CEDM Around-wall Crack Growth Curve

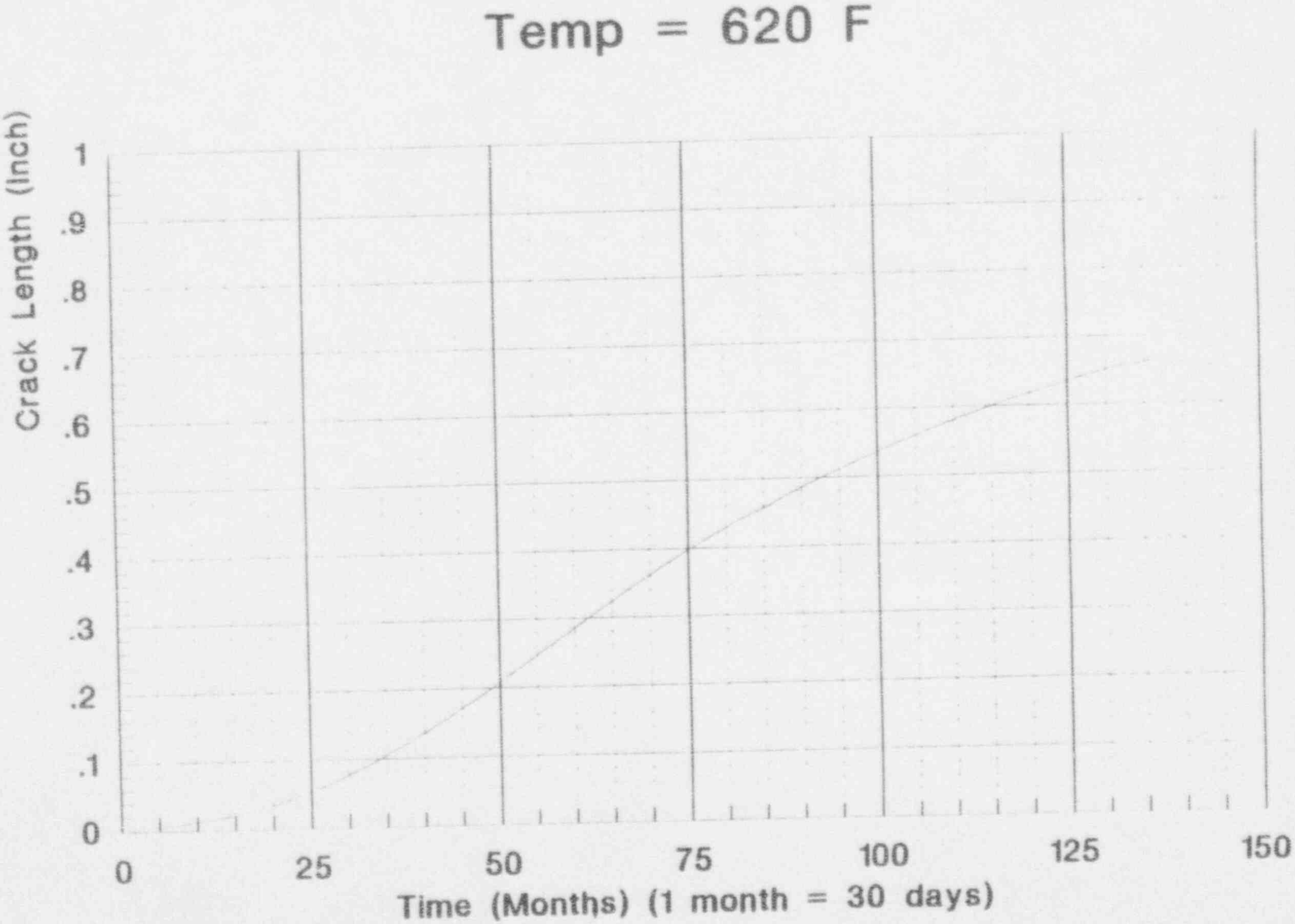


Figure 2.3.6
Nominal-walled CEDM Through-wall Crack Growth Curve

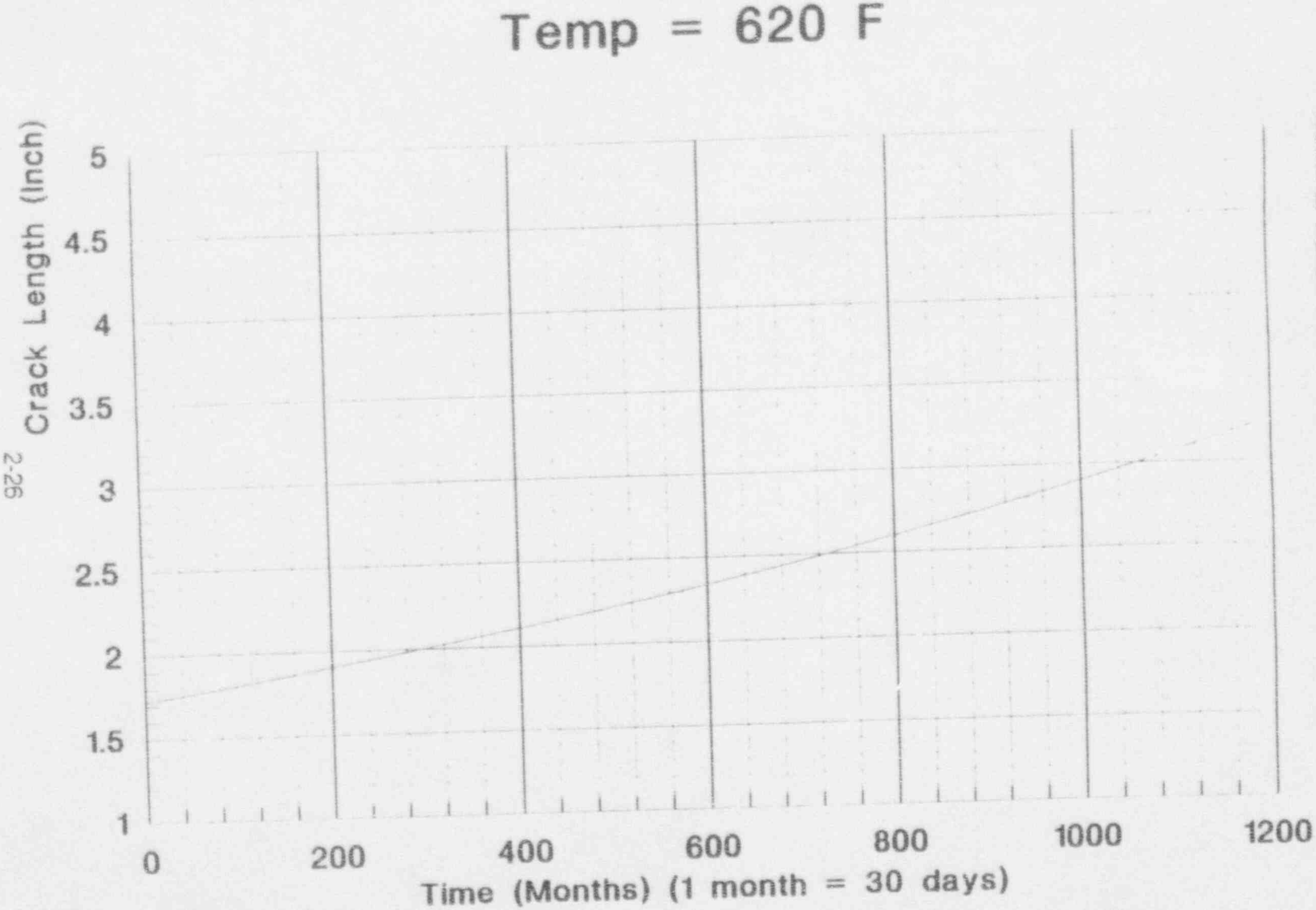


Figure 2.3.7
Nominal-walled CEDM Around-wall Crack Growth Curve

2.4 SAFETY SIGNIFICANCE DETERMINATION

This section reviews the potential adverse consequences from a postulated OD-initiated throughwall crack and its safety significance. Among the potential consequence that is specific to OD-initiated cracking is the possibility of catastrophic rupture. Field inspections continue to show that ID-initiated cracking is predominately axial in character. In Bugey-3, however, a circumferentially oriented crack was observed on the OD-surface. The presence of circumferential crack raises the concerns regarding catastrophic failure. The consequence of reactor coolant leakage, steam impingement, and boric acid wastage is amply discussed in CEN-607.

The potential for a catastrophic rupture resulting from OD-initiated PWSCC was evaluated to be unlikely due to the results of the crack propagation calculations. This calculation showed that a minimum of about 91 years is needed before a circumferential crack would grow sufficiently to compromise its structural integrity. In addition, the tight fit of the nozzle in the reactor vessel head penetration bore prevents significant opening of any crack. Catastrophic rupture is, therefore, not credible during the current licensed period. Thus, there is no safety significance associated with a circumferential crack on the OD of the CEDM.

The CEOG concludes that there is no safety significance associated with the potential for or consequence of OD-initiated PWSCC of the reactor vessel Alloy 600 penetration nozzles. This determination is based on the results of a stress analysis, crack propagation calculation, and an estimate of the minimum time need to reach the critical crack size.

If a throughwall crack is postulated, a review of the potential adverse affect of a PWSCC shows that CEOG plants continue to meet the requirements of General Design Criteria 14, 30, and 31 of Appendix A to 10 CFR 50. In particular, the reactor coolant pressure boundary continues to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Specifically, current surveillance guidelines provide sufficient means to detect and, to the extent practical, identify the location of the source of any reactor leakage.

All CEOG plants currently have adequate operational procedures and Technical Specification limits in place to respond quickly and appropriately to the leakage of less than 1 gpm from a throughwall crack.

2.5 UNREVIEWED SAFETY QUESTION EVALUATION

In 10 CFR 50.59, the NRC has provided criteria to determine whether changes to the facility involve an Unreviewed Safety Question. A change to the facility is deemed to involve an unreviewed safety question if:

(i) the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased;

or

(ii) a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created;

or

(iii) the margin of safety as defined in the basis for any technical specification is reduced.

The reactor vessel head penetration nozzles are primary pressure boundary components. They are designed and fabricated to the requirements of the applicable version of the ASME Boiler and Pressure Vessel Code, Section III. Each nozzle is welded to the reactor vessel head. The nozzles are made of Alloy 600.

Throughwall cracks have been found in similar European nozzles. Primary coolant leaked from one nozzle during a hydrostatic pressure test. No steam impingement damage or boric acid corrosion was evident in that case. The throughwall crack found was oriented axially and was near the weld. Other axial indications were also found; but, they did not extend throughwall. At the end of the axial throughwall crack, a circumferential indication was observed on the outside surface. The potential for similar OD cracking may exist in CEOG plants and, therefore, an Unreviewed Safety Question determination is performed.

The fact that reactor vessel head Alloy 600 nozzles do not create an unreviewed safety question is demonstrated by the following:

(i) **The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report will not be increased by the potential for or consequence of Alloy 600 reactor vessel head penetration nozzle cracking.**

The stress field analysis and crack growth calculations demonstrate that catastrophic failure of the CEDM nozzles is not expected in the current license period. The most likely source of reactor coolant leakage will be from an axial throughwall crack in a single, most-limiting, CEDM nozzle. According to Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANSI N18.2-1973 issued by American National Standards Institute, the probability of minor primary coolant leakage is categorized as Condition II, Incident of Moderate Frequency. One incident in this category is expected per plant during a calendar year. The occurrence of a throughwall crack in either a CEDM nozzle is categorized under this broad probability condition. That is, leakage past a throughwall crack would not result in the initiation of an incident of Condition III (Infrequent Incidents) or IV (Limiting Faults) without other independent incidents. Thus, the probability of occurrence of primary coolant leakage is not expected to increase.

The consequence of primary coolant leakage is also not expected to increase. Any minor primary coolant leakage would not prevent the orderly shutdown and cooldown of the reactor as currently prescribed by the CEOG plant Technical Specifications. It is expected that after appropriate corrective action, the plant can be returned to power operation. Leakage would continue to be measured using existing techniques.

With respect to accidents which involve breaching the pressure boundary, the FSARs of CEOG plants include both an evaluation of both the CEA Ejection Accident and Small Break LOCA. The CEA Ejection Accident is initiated as a result of a postulated mechanical failure of a CEDM housing or nozzle on the reactor vessel head in the form of a complete circumferential rupture. This mechanical failure is the initiating event that results in the complete and rapid expulsion of a control rod from the core. The consequence of the CEA ejection is a rapid positive reactivity insertion which combined with an adverse power distribution may result in localized fuel damage.

In analysis performed by ABB-CE, it is assumed that a CEA is ejected almost instantaneously from the core. In the unlikely event a CEDM nozzle should separate from the reactor vessel head, its potential vertical upward travel is limited by missile shield blocks placed over the reactor vessel head and drive mechanisms. Thus, the expected behavior of a CEA ejection accident is considered to be less severe than that evaluated in the safety analysis report. In all cases evaluated for CEOG plants, localized fuel damage is either averted or is kept by reactor protection system design and actions to be sufficiently small such that radiological dose limits are not exceeded.

A LOCA is defined as a breach of the reactor coolant system boundary which results in interruption of the normal mechanism for removing heat from the reactor core. The formal analysis of LOCA in CEOG plant confirms that the Emergency Core Cooling System (ECCS) design and operation provides adequate protection for the core for break-sizes up to the double-ended severance of the largest RCS pipe. This protection is provided even in the event of the most severe single active failure. In all cases, the most-limiting break location is analyzed and presented in the Safety Analysis Report of CEOG plants.

The catastrophic failure of a CEDM or ICI nozzle would cause a Small Break LOCA. A Small Break LOCA is defined as a LOCA whose break-size is less than that of the largest diameter pipe in the RCS. A breach in the RCS in the reactor vessel head is not a limiting location. Analytically, there is a high degree of assurance that the ECCS in CEOG plants will continue to provide adequate cooling of the core in the event of a CEDM or ICI rupture. Thus, for CEOG plants the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactor," will continue to be met.

Based on the above, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report will not be increased.

(ii) **The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report will not be created by the potential for or consequence of Alloy 600 reactor vessel head penetration nozzle cracking.**

In the worst case, an OD-initiated circumferential crack can be postulated to cause catastrophic failure of the CEDM nozzle. Although catastrophic failure is not credible during the current licensed period, the FSAR for all CEOG plants explicitly evaluates both a Control Element Assembly (CEA) Ejection Accident and a Small Break Loss of Coolant (LOCA) Accident. Either or both of these events could result from a postulated catastrophic failure of a CEDM nozzle. Therefore, there is no possibility of an accident or malfunction of a different type that already evaluated in the FSAR.

(iii) **The margin of safety as defined in the basis for any technical specification will not be reduced by the potential for or consequence of Alloy 600 reactor vessel head penetration nozzle cracking.**

A review of the Technical Specifications shows that the CEDM and ICI nozzles do not maintain the margin of safety as defined in the basis for any technical specifications other than contributing to the maintenance of the integrity of the primary pressure boundary.

The leakage limits in the Technical Specifications will remain the same and will, therefore, continue to provide the same degree of assurance that the chance of a crack will not progress to an unsafe condition without detection and proper evaluation. Thus, the margin of safety as defined in the basis for any technical specification will not be reduced.

Based on the above, the potential for or consequence of Alloy 600 reactor vessel head penetration nozzle cracking does not involve an unreviewed safety question and does not require a change to the Technical Specifications.

3.0 CONCLUSIONS

As described in Section 1.0, PWSCC were discovered in reactor vessel head Alloy 600 penetration nozzles at the French Bugey-3 PWR plant in 1991. One crack was above the weld and throughwall. The throughwall crack was detected when primary coolant leaked passed the crack during a hydrostatic pressure test of the reactor vessel. This raises the concern for OD-initiated cracking.

An extensive evaluation of the potential for and the consequence of PWSCC in CEOG plants was conducted and is described in Section 2.0. The evaluation supports the following conclusions:

- In Section 2.1, a stress field analysis shows that in the region above the weld the compressive zones do not extended uniformly around the OD of the nozzle. Thus, in the presence of tensile loading consideration must be given to the potential for PWSCC on the OD if an ID-initiated throughwall crack is postulated.

- In Section 2.2, the environmental chemistry assessment showed that one of two chemical environments could arise on the OD surface of a CEDM nozzle, depending upon whether a sealed or vented crevice exists. Neither environment, however, is expected to sustain OD-initiated PWSCC or caustic-induced SCC.

- In Section 2.3, crack propagation calculations indicate that it would take more than 91 years operation (well beyond the current license period) to propagate a crack before catastrophic failure would be possible.

- In Section 2.4, a determination of the safety significance of OD-initiated PWSCC indicated that because catastrophic rupture is not credible during the current licensed period, there is no safety significance associated with a circumferential crack on the OD of the CEDM. Previous CEOG report CEN-607, already affirmed that there is no safety significance to reactor coolant leakage, steam impingement damage or boric acid corrosion and wastage. CEOG plants currently have adequate operational procedures and Technical Specification limits in place to respond quickly and appropriately in the event that any detectable leakage.

- Finally, in Section 2.5, a 10 CFR 50.59 evaluation concluded that the potential for or consequence of Alloy 600 reactor vessel head penetration nozzle cracking does not involve an unreviewed safety question and does not require a change to the Technical Specifications.

Based on the evaluation performed and summarized in this report, the CEOG considers that there is no new safety significance associated with the potential for OD-initiated PWSCC. The safety implications of this phenomena in CEOG plants are fully addressed by current practices, precautions, and procedures in force. Specifically, catastrophic failure of the CEDM is not expected. In the event of leakage, all CEOG plants have adequate operational procedures and Technical Specification in place to respond quickly and appropriately. Accordingly, the CEOG considers that the consequence of an OD-initiated crack are primarily economic and closely related to issues of plant availability and the protection of capital equipment.

4.0 REFERENCES

[1] CEN-607, "Safety Evaluation of the Potential for and Consequence of Reactor Vessel Head Penetration Alloy 600 ID-Initiated Nozzle Cracking"

[2] Report of EPRI/EdF/Owners Group Meeting on CRDM Nozzle PWSCC Structure Integrity Issues, May 6, 7, 1993.

[3] M. Koike, M. Suda, H. Kawamura, and H. Hirano, "Effect of Boric Acid on IGA/SCC Propagation Rate of Alloy 600 in High Temperature Water". Presented at the EPRI Workshop on Secondary Side-Initiated IGA/SCC, Minneapolis, MN, October 14-15, 1993.

[4] H. Hirano, H. Takaku, and T. Kurosawa, Corrosion Science, 31, 557, (1990).

[5] Raju, I. S. and Newman, J. C., "Stress-Intensity Factors for Circumferential Surface Cracks in Pipes and Rods under Tension and Bending Loads", Fracture Mechanics: Seventeenth Volume, ASTM, STP 905, American Society for Testing and Materials, Philadelphia, 1986, pp. 789-805.

[6] U.S. Nuclear Regulatory Commission letter from W. T. Russell (USNRC) to W. Rasin (NUMARC) dated November 19, 1993.

[7] Smialowska, Z.S. et al., "Effects of pH and Stress Intensity on Crack growth Rate in Alloy 600 in Lithiated and Borated Water at High Temperatures," Proceedings, Fifth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Monterey, August 1991.

[8] Scott, P. M., "An Analysis of Primary Water Stress Corrosion Cracking in PWR Steam Generators," Proceedings, Specialists Meeting on Operating Experience With Steam Generators, Brussels, Belgium, September 1991.

CEOG COMBUSTION ENGINEERING OWNERS GROUP

Raymond Burski, Chairman
c/o Entergy Operations/Highway 18/Killona, LA 70066

December 29, 1993
CEOG-93-688

Nuclear Management and Resource Council
1776 Eye Street N.W., Suite 300
Washington, D.C. 20006-3706

Attention: Mr. Morris Schreim

Subject: CEOG Safety Evaluation for CEDM Cracking

Enclosure: "Safety Evaluation of the Potential for and Consequence of Reactor Vessel Head Penetration Alloy 600 OD Initiated Nozzle Cracking," CEN-614

Gentlemen:

This letter provides two copies of the enclosed safety evaluation. It is the CEOG's understanding that all PWR Owners Groups are submitting OD safety evaluations to NUMARC for NUMARC to forward to the Nuclear Regulatory Commission (NRC) as part of the coordinated industry initiative on Reactor Vessel Head Penetration Cracking.

The enclosed safety evaluation specifically addresses OD initiated cracking. The results of the safety evaluation demonstrate that CEOG plants may continue to operate for a significant time beyond plant licensed life before they must be concerned with postulated circumferential cracks. The CEOG concludes that the safety significance associated with the potential for or consequence of OD-initiated PWSCC of the reactor vessel Alloy 600 penetration nozzles is minimal.

The CEOG OD safety evaluation supports the industry position that PWSCC of reactor vessel head CEDM nozzle penetrations does not represent an immediate safety concern.

Should you have any questions or comments on this letter, please do not hesitate to contact me at (504) 739-6774.

Sincerely,

For Raymond Burski
Chairman
C-E Owners Group

RB/RDC:rn

Enclosure - as stated

cc: CEOG Executive Committee
C-E Owners Group
C-E Plant Managers
CEOG Alloy 600 Working Group
G. C. Bischoff, ABB
P. W. Richardson, ABB