Comparison of Piping Designed to ANSI B31.1 and ASME Section III, Class 1

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ABSTRACT

The evaluation of potential fatigue damage is a technical issue affecting the license renewal of nuclear power plants. The importance of this issue led the nuclear power industry, through the Nuclear Management and Resources Council (NUMARC), to develop a consistent technical position which asserted, in part, that the current licensing basis (CLB) for fatigue for Class 1 reactor coolant containing piping systems in early plants was adequate for the license renewal term, with clearly defined exceptions. These early plant piping systems were designed to the requirements of ANSI B31.1 for Power Piping and its predecessor codes. Class 1 piping systems in later plants were designed to the ASME Code, Section III, Class 1 rules. The NUMARC technical position on fatigue for license renewal identified the exceptions to CLB design adequacy as being associated with component locations having severe geometric (or material) and loading discontinuities, such as socket welds and slip-on flanges, and regions affected by step-change thermal transients. Typical examples of the latter are reactor coolant system nozzles for the decay heat removal system and charging system.

This report examines the validity of that asserted industry position by comparing the results of fatigue design evaluation methods for piping designed to the ANSI B31.1 Code to those of the ASME Code, Section III for Class 1 Piping. Although these Codes are fundamentally different, experience in operating plants has not shown that the former is inadequate. ASME Section III evaluation of two fatigue-sensitive reactor coolant piping systems, both originally designed to ANSI B31.1, are included. Both were evaluated using design-basis transient definitions consistent with modern nuclear plant design. These evaluations showed that the B31.1-designed systems had only very limited areas with high fatigue usage. In both systems, the locations of indicated high fatigue usage were those with geometric (or material) discontinuities that were also affected by severe step change thermal transients. The evaluations also showed that evaluation per the requirements of the current version of the ASME Code will produce significantly less indicated fatigue usage than the earlier Code

versions used for design of most nuclear plants in service in the United States today, with the reduced fatigue usage due to more realistic reclassification of through-wall thermal gradients.

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Section 1

INTRODUCTION

The evaluation of potential fatigue damage is a technical issue affecting the license renewal of nuclear power plants. The importance of the issue can be recognized by noting that six of the ten License Renewal Industry Reports (IRs) contain extensive fatigue evaluations for major plant systems, structures and components, while three of the other IRs contain at least some fatigue evaluations. This importance led the industry, through the Nuclear Management Resources Council (NUMARC) to formulate a technical position on fatigue evaluation for license renewal, in order to assure consistency throughout the IRs.

The essence of the NUMARC fatigue technical position was that the current licensing basis (CLB) for fatigue was adequate including both the current design basis (e.g., ANSI B31.1 [1,2], or ASME Section III, Class 1 [3]) and the current operating basis. The latter includes any inservice examination requirements (e.g., ASME Section XI [4]) and any licensing commitments related to fatigue (e.g., monitoring of operating transients). Any exceptions to the adequacy of the CLB were to be identified clearly in the IRs. The NUMARC fatigue technical position explicitly treated the question of Class 1 piping in older plants that were originally constructed to the rules of ANSI B31.1 or its predecessor standards, concluding that the CLB for such components is adequate except for specific locations and conditions. Two exceptions were identified. The first was regions with geometric and loading discontinuities (e.g., socket welds), the second was regions with step change thermal loadings (e.g., reactor coolant system nozzles for emergency core cooling system nozzles).

U.S. Nuclear Regulatory Commission (NRC) concerns about the fatigue issue led to their issuance of a draft technical position on fatigue for license renewal (BTP PLDR D-1) [5]. This draft BTP has since been withdrawn, and has been replaced by a generic regulatory evaluation of the fatigue issue for operating plants.

In order to provide further guidance to the industry regarding the fatigue design margins inherent in Class 1 piping system components constructed to B31.1 rules, the Electric Power Research Institute (EPRI) has initiated several efforts, in conjunction with the U. S. Department of Energy (DOE)/Sandia National Laboratories (SNL). These efforts are intended to describe the difference in design and construction of ANSI B31.1 and ASME Section III, Class 1 piping components, and to evaluate the inherent conservatism of the fatigue resistance of the former. The work reported herein is supportive of these joint efforts and provides an assessment of some typical ANSI B31.1 piping systems when subjected to ASME Section III, Class 1 analysis methods. The intent is to show where inherent conservatism exists, and to identify if there are certain situations where the B31.1 evaluation may not provide adequate assurance of fatigue resistance.

The approach taken in this report consists of: (1) examining existing Class 1 piping systems that were originally designed to ANSI B31.1 requirements; (2) assuring that these selected piping systems do, in fact, satisfy B31.1 design rules; (3) developing a set of modern Section III, Class 1 fatigue design-basis transients that would apply to these piping systems if subjected to current fatigue design requirements; and (4) determining the Section III, Class 1 fatigue usage factors for limiting locations in the piping systems. The selected piping systems were chosen to be representative of systems with locations of high fatigue usage and with a range of geometric features and loading conditions such that the NUMARC technical position on fatigue could be adequately examined. The thermal transients considered only those used in design. No thermal stratification caused by inadequate flow mixing was included since only a few Class 1 piping system have recently been identified as affected. Instead, thermal discontinuities along the length of the piping system are used to illustrate the effect of rapid changes in thermal loading.

To assist the reader in understanding the differences between the two Codes, Section 2 of this report summarizes the design and analysis requirements for each. Then, in Sections 3 and 4, ASME Section III, Class 1 piping analyses performed on two piping systems, originally designed to ANSI B31.1, are described. The analyses of both a Boiling Water Reactor (BWR) recirculation system and a Pressurized Water Reactor (PWR) charging line are described. Although actual existing B31.1 analyses were used, the plant names are not included in this report at the request of the utilities who allowed their plants to be evaluated.

The conclusions from the report are contained in Section 5.

Section 2
DISCUSSION OF CODE REQUIREMENTS

The need for a national code for pressure piping became evident in the early 1900's. The American Society of Mechanical Engineers initiated Project B31 in 1926. A first edition was published in 1935 as the American Tentative Code for Pressure Piping [2]. (See Forward to Reference 1 for the history of development of ANSI B31.1.)

As a result of continuing review over the years, the 1955 Edition of ANSI B31.1 [1] was a significant step forward in design of pressure piping systems. The work of Markl (and others) was incorporated and presented an approach for avoiding fatigue failures in power piping systems [6,7]. The developments included in that document form the basis for the current requirements for fatigue design of ASME Section III, Class 2 and Class 3 piping systems [8,9]. There have been no significant failures in the nuclear power industry that would indicate that the design rules presented in these codes are not sufficient [10].

In the following, the design requirements for ANSI B31.1 piping systems are described prior to introduction of ANSI B31.7 in 1969 [11]. Some additions to ANSI B31.1 in June 1973 are also discussed. The ANSI B31.1 requirements are then compared to the requirements for design of Class 1 piping in accordance with the current edition of ASME Section III. Where appropriate, reference is made to other editions of the Codes.

2.1 Original Requirements for Design of ANSI B31.1 Piping

The piping in the early U.S. nuclear plants was designed in accordance with the requirements of ASA B31.1-1955 [1]. Section 1 of ASA B31.1-1955 was written for Power Piping, and encompassed the "...minimum requirements for design, manufacture, test, and installation of power piping systems, as defined for steam generating plants, central heating plants, and industrial plants." Section 6 provided minimum standards for fabrication, but also included a Chapter 3 dealing with approved methods for providing for thermal expansion and flexibility in piping systems. Specific requirements that governed the design were as follows:

1. **Paragraph 122 - Thickness Pipe:** The equation, identical to that used in current codes, was described.

2. **Paragraph 607 - Allowable Stresses:** This paragraph, in a Chapter dealing with pipe hangers and other supporting elements, permitted the allowable stress to be increased by 20% for short time overloading conditions.

3. **Paragraph 622 - Stresses and Reactions:** This paragraph introduced the current methods for computing the allowable expansion stress (S_A) and stress range reduction factors. A formula was provided for computing the expansion stress:

$$S_E = \sqrt{S_b^2 + 4S_t^2}$$

where:

S_b	=	$i\,M_b/Z$ = resultant bending stress, psi
S_t	=	$M_t/2Z$ = torsional stress, psi
M_b	=	resultant bending moment, in-lb
M_t	=	torsional moment, in-lb
Z	=	section modulus of pipe, in^3
i	=	stress-intensification factor, as defined in tables provided in the code

The paragraph also stated that the effects of pressure, weight, and other sustained loadings shall not exceed S_h (the basic allowable stress), and that if these stresses are less than the allowable, then the difference can be added to the allowable thermal expansion stress range. Guidance was given for computing the longitudinal pressure stress based on the inside area, but there were no specific formuli provided for computing the axial stresses due to sustained and occasional moments.

4. **Paragraph 623 - Supports:** This paragraph required that "... design and spacing of supports shall be checked to assure that the sum of the longitudinal stresses due to weight, pressure and other sustained external loading does not exceed S_h." Again, no formula was provided for computing the stress.

The basis of these rules is described in Markl's paper on piping flexibility analysis [7]. From reviewing this information, it is clear that the intent was to assure that the overall thermal expansion stress ranges, considered to be the only significant cyclic loading, were accounted for by including the stress intensification factor in the thermal expansion moment evaluation.

The design requirements of USAS B31.1.0, the successor Code to B31.1-1955 [2], were essentially unchanged, with the following notable exceptions:

1. The allowance for variations from normal conditions was modified to allow an increase in allowable stress of up to 15 and 20%, for loadings occurring less than 10 percent and one percent of the operating time, respectively (para. 102.2.4).
2. The concept of equivalent thermal expansion ranges was introduced (para. 102.3.2).
3. Methods for evaluating moments at reduced outlet connections were provided (119.6.3(b)).
4. The number of allowed materials included in Appendix A of the Code was increased.

The applicable design requirements were essentially unchanged in ANSI B31.1-1973 [12].

2.2 Modifications to ANSI B31.1 in June 1973

In the June 1973 addenda to ANSI B31.1 [13], changes were made which modified the Power Piping Code to be much more like it appears in the early versions of ASME Section III, Class 2 and 3 and in the current version of ANSI B31.1. Specifically, the following major revisions occurred:

1. Equations were provided for computing the longitudinal stresses due to pressure and moment loadings (para. 104.8). For dead weight and occasional moments, the term 0.75i (but not less than 1.0) was added to account for the fact the primary loading (load controlled) stresses in some components were affected by stress intensification.

2. Revised stress intensification factors were added for butt welds, tapered transitions, reducers and branch connections.

As provided in the June 1973 addenda, the equations for piping design are as follows:

Longitudinal Stresses Due to Sustained Loads. The effects of pressure, weight, and other sustained mechanical loads must meet the requirements of Equation 11.

$$\frac{PD_o}{4t_n} + \frac{0.75iM_A}{Z} \leq 1.0S_h \qquad \text{(B31.1, Eq. 11)}$$

where:

P	=	internal design pressure, psig
D_o	=	outside diameter of pipe, in.
t_n	=	nominal wall thickness of component, in.
M_A	=	resultant moment loading on cross section due to weight and other sustained loads, in-lbs
Z	=	section modulus, in^3
i	=	stress intensification factor. (The product, 0.75i, shall never be taken less than 1.0).
S_h	=	basic material allowable stress at maximum temperature from allowable stress tables, psi.

Longitudinal Stresses Due to Occasional Loads. The effects of pressure, weight, other sustained, and occasional loads, including earthquake, must meet the requirements of Equation 12.

$$\frac{PD_o}{4t_n} + \frac{0.75i(M_A+M_B)}{Z} \leq kS_h \qquad \text{(B31.1, Eq. 12)}$$

where terms are the same as above except:

k = 1.15 for occasional loads acting less than 10% of operating period

k = 1.2 for occasional loads acting less than 1% of operating period

M_B = resultant moment loading on cross section due to occasional loads. If calculation of moments due to earthquake is required, use only one-half the earthquake moment range. Effects of anchor displacements due to earthquake may be excluded from Equation 12 if they are included in Equations 13 or 14.

Additive Stresses. The requirements of either Equation 13 or Equation 14 must be met:

Thermal Expansion Stresses:

$$\frac{iM_C}{Z} < S_A \qquad \text{(B31.1, Eq. 13)}$$

where the terms are the same as above except:

M_C = range of resultant moments due to thermal expansion. Also include moments effects of anchor displacement due to earthquake if anchor displacement effects were omitted from Equation 12.

S_A = the allowable stress range for expansion stresses.

where:

S_A = $f\,(1.25\ S_c + 0.25\ S_h)$

f = stress range reduction factor (function of number of thermal expansion cycles)

S_c = allowable stress at cold condition

Sustained Plus Thermal Expansion Stresses: The effects of pressure, weight, other sustained loads and thermal expansion must meet the requirements of Equation 14:

$$\frac{PD_o}{4t_n} + \frac{0.75iM_A}{Z} + \frac{iM_C}{Z} < (S_h + S_A) \qquad \text{(B31.1, Eq. 14)}$$

with terms as previously described.

In the above equations, the approach used prior to 1973 would be identical, except that the term 0.75i used in the moment terms for sustained and occasional loads (Equations 11, 12, and 14) was not present. The absence of this term for nuclear plants is not deemed to be significant, since ANSI B31.1 is also used for high temperature applications, where consideration of creep is required and stress limitation is an important factor in the design.

2.3 ASME Section III, Class 1 Requirements

The design requirements for Section III, Class 1 for piping components are based on the maximum shear stress theory (as compared to ANSI B31.1 which is based on maximum stress theory). The design is considered to be acceptable if the design passes a series of equations for the various loadings to which the component is exposed. The introduction to Reference 11 includes a discussion of the Class 1 piping design criteria and philosophy.

A primary stress limit is provided to show that the design is acceptable for load-controlled (primary) loadings and is similar to Equation 11 of B31.1. The primary stress intensity limit is satisfied if the requirements of Equation 9 (of Section III) are met:

$$B_1\frac{PD_o}{2t} + B_2\frac{D_o}{2I}M_i \leq 1.5S_m \qquad \text{(Section III, Cl. 1, Eq. 9)}$$

where:

B_1 , B_2	=	primary stress indices for the specific product under investigation
P	=	Design Pressure, psi
D_o	=	outside diameter of pipe, in.

t	=	nominal wall thickness of product, in.
I	=	moment of inertia, in^4
M_i	=	resultant moment due to a combination of Design Mechanical Loads, in-lb.
S_m	=	basic allowable design stress intensity value, psi.

For loading conditions classified as Service Level B (in the Design Specification), the above equation must also be met, except that the allowable stress may be increased from 1.5 S_m to 1.8 S_m. The magnitude of allowable increase is consistent with the 20% allowable increase in Equation 12 of ANSI B31.1.

The remainder of the equations for Service Levels A and B are provided to assure satisfactory cyclic behavior. To satisfy the range of primary plus secondary stresses (which will assure that shakedown occurs and that excessive distortion does not occur), Equation 10 must be met. The calculation of the stress range is based upon the effect of changes which occur in mechanical or thermal loadings which take place as the system goes from one load set, such as pressure, temperature, moment, and force loading, to any other load set which could also exist. Equation 10 must be satisfied for all pairs of load sets:

$$S_n = C_1 \frac{P_o D_o}{2t} + C_2 \frac{D_o}{2I} M_i$$

$$+ C_3 E_{ab} \left| \alpha_a T_a - \alpha_b T_b \right| \leq 3S_m \qquad \text{(Section III, Cl. 1, Eq. 10)}$$

where:

C_1, C_2, C_3	=	secondary stress indices for the specific component under investigation,
D_o, t, I, S_m	=	as defined for Equation 9,
P_o	=	range of service pressure, psi,
M_i	=	resultant range of moment which occurs when the system goes, from one service load set to another, in-lb.,

E_{ab}	=	average modulus of elasticity of the two sides of a material or structural discontinuties at room temperature, psi,
α_a, α_b	=	coefficient of thermal expansion on side a and side b of a structural or material discontinuity, in/in-°F.
T_a, T_b	=	range of average temperatures on side a and side b of a structural discontinuity, when the system goes from one service load set to another, °F.

The fatigue resistance of the component is assessed by evaluating the ranges of peak stress. For every pair of load sets, S_p values are calculated using Equation 11:

$$S_p = K_1 C_1 \frac{P_o D_o}{2t} + K_2 C_2 \frac{D_o}{2I} M_i + K_3 C_3 E_{ab} |\alpha_a T_a - \alpha_b T_b|$$

$$+ \frac{1}{2(1-\nu)} K_3 E\alpha |\Delta T_1| + \frac{1}{1-\nu} E\alpha |\Delta T_2| \qquad \text{(Section III, Cl. 1, Eq. 11)}$$

where:

K_1, K_2, K_3	=	local stress indices for the specific component under investigation,
$E\alpha$	=	modulus of elasticity (E) times the mean coefficient of thermal expansion (α), both at room temperature, psi °F,
ΔT_1	=	range of the temperature difference for each load set combination between the temperature of the outside surface T_o and the temperature of the inside surface T_i of the piping product assuming a moment generating equivalent linear temperature distribution, °F,
ΔT_2	=	range for that portion of the nonlinear thermal gradient through the wall thickness not included in ΔT_1, °F,

If Equation 10 cannot be satisfied for all pairs of load sets, the alternative analysis described below may still permit qualifying the component. Only those pairs of load sets which do not satisfy Equation 10 need to be considered.

(a) Equation 12 shall be met:

$$S_e = C_2 \frac{D_o}{2I} M_i^* \leq 3S_m \qquad \text{(Section III, Cl. 1, Eq. 12)}$$

where:

S_e	=	nominal value of expansion stress, psi
M_i^*	=	same as M_i in Equation 10, except that it includes only moments due to thermal expansion and thermal anchor movements, in-lb.

(b) When the limits of Equation 10 are exceeded and before the rules of Equation 13 can be utilized, the value of the range of ΔT_1 cannot exceed that calculated per NB-3653.7 as follows:

$$\Delta T_1 \; range \leq \frac{y' S_y}{0.7 E\alpha} C_4$$

where:

y'	=	3.33, 2.00, 1.20, and 0.80 for x = 0.3, 0.5, 0.7, and 0.8, respectively
x	=	$(PD_o/2t)(1/S_y)$
P	=	maximum pressure for the set of conditions under consideration, psi
C_4	=	1.1 for ferritic material
	=	1.3 for austenitic material
$E\alpha$	=	as defined for Equation 11, psi/°F

S_y = material yield strength value, psi, taken at average fluid temperature of the transient under consideration.

The primary plus secondary membrane plus bending stress intensity, excluding thermal bending and thermal expansion stresses, shall be $<3S_m$. This requirement is satisfied by meeting Equation 13:

$$C_1 \frac{P_o D_o}{2t} + C_2 \frac{D_o M_i}{2I} + C_3' E_{ab} \left| \alpha_a T_a - \alpha_b T_b \right| \leq 3S_m \qquad \text{(Section III, Cl. 1, Eq. 13)}$$

where:

M_i = moment as defined for Equation 9, in-lb, and all other terms as previously described,

C_3' = stress index (NB-3680).

(d) If these conditions are met, the value of S_{alt} shall be calculated by Equation 14:

$$S_{alt} = K_e \frac{S_p}{2} \qquad \text{(Section III, Cl. 1, Eq. 14)}$$

where:

S_{alt} = alternating stress intensity, psi,

S_p = peak stress intensity value calculated by Equation 11, (NB-3653.2), psi

K_e = 1.0 for $S_n \leq 3S_m$,

= $1.0 + [(1 - n)/n(m - 1)](S_n/3S_m - 1)$, for $3\,S_m < S_n < 3mS_m$

= $1/n$, for $S_n \geq 3mS_m$,

S_n = primary plus secondary stress intensity value calculated in Equation 10, NB-3653.1, psi,

m,n = material parameters provided in Table NB-3228.5(b)-1.

The alternating stress for all load sets is computed as one-half of the peak stress ranges calculated from Equation 11, or by the alternate approach of Equation 14 if Equation 10 is not met. The fatigue analysis is then performed using the applicable Code fatigue curve and the design number of cycles for each loading from the design specification.

It should be noted that for ASME Section III Code editions prior to the Summer 1979 Addenda, Equation 10 contained an additional term. In these earlier Code editions, the ΔT_1 term of the peak stress Equation 11 was also included in the secondary stress Equation 10:

$$S_n = C_1 \frac{P_0 D_0}{2t} + \frac{C_2 D_0}{2I} M_i + C_3 E_{ab} |\alpha_a T_a - \alpha_b T_b| + \frac{E\alpha |\Delta T_1|}{2(1-\nu)} \le 3S_m \quad \text{(Section III, Cl. 1, Eq. 10)}$$

Addition of this term frequently increased the stress, S_n, above $3S_m$. When this occurred, Equations 12 and 13 had to be met, and the fatigue analysis had to be conducted using a relatively high K_e factor, increasing the alternating stresses used in the fatigue analysis. The ASME Section III Committee on Piping Design justified that this was overconservative and modified the equation accordingly, starting with the Summer 1979 Addenda. However, most current Section III plants were designed to the earlier version of the Section III Code.

2.4 Allowable Stresses

The allowable stresses for B31.1 and ASME Classes 2 and 3 are basically those of ASME Section I. This requires that at any temperature below the creep range, the maximum allowable stress value shall be the lower of :

1/4 of the specified minimum tensile strength at room temperature;
1/4 of the tensile strength at temperature;

5/8 of the specified minimum expected 0.2% offset yield strength at temperature; except that for austenitic materials where some permanent strain distortion is acceptable, 90% of the yield strength at temperature may be used.

For austenitic materials, the higher allowables were not recommended where slight amounts of distortion could cause leakage or malfunction (e.g., at flanged connections). Thus, for piping, the higher allowables were normally used.

For Section III, Class 1 components, the allowable stress intensity must be the lower of:

1/3 of the specified minimum tensile strength at room temperature;
1/3 of the tensile strength at temperature;
2/3 of the specified minimum yield strength at room temperature;
2/3 of the yield strength at temperature (for ferritic materials), or 90% of the yield strength at temperature but not to exceed 2/3 of the specified minimum yield strength at room temperature (for austenitic materials).

2.5 Comparison of Design Requirements

Table 2-1 summarizes the key differences between the design requirements for piping designed to ANSI B31.1 and ASME Class 1 requirements. In general, a piping system designed to B31.1 requirements will have a thicker wall due to the lower allowable stresses, although for stainless steels, the difference may be small. It is clear that the "fatigue considerations" are not as rigorous for B31.1 design.

The Nuclear Regulatory Commission evaluated the design of older plants versus those being designed in the 1970's in the Systematic Evaluation Program (SEP). As part of this program a study was performed to assess the differences in the quality standards applied to design of reactor coolant pressure retaining components [14]. It was identified that there were a number of early Code Cases (N-1 through N-12) issued in 1960 to 1962 that provided additional guidelines for design of nuclear plant piping. Significant content of these Code Cases is as follows:

1. Code Case N-1 stated that nuclear piping (for which loss of fluid could result in a radiation hazard) may be designed to B31.1 (1955) supplemented by the requirements of case interpretations identified by the prefix "N".

2. Code Case N-2 required that valves used in nuclear power systems:

   a. be of materials recognized by ASA B31.1-1955 and conform to a recognized standard (e.g., ASA B16.5),
   b. meet physical and inspection requirements of Code Case N-10,
   c. have a positive sealing or some provision for stem and bonnet leak-off control, and
   d. screwed end valves (in which the thread provides the only seal) are not permitted.

3. Code Case N-4 permitted the temperature limit of 100°F for hydrostatic media to be exceeded.

4. Code Case N-7 permitted the use of nuclear piping made from austenitic stainless steels, provided that:

   a. materials conform to one of the following ASTM specifications: A376, A358, A312, and A430 for piping; ASTM-A403 for welded fittings; or ASTM-182 for forgings,
   b. full radiography of longitudinal and circumferential welds is performed; however, liquid penetrant methods are permitted when size or configuration precludes full radiography, or for services at or near atmospheric temperatures up to 212°F provided that piping is tested at 1.5 times the maximum allowable working pressure,
   c. allowable stress values are used as shown in Table 2-2, and
   d. reheat treating at 1950°F for one hour per inch of thickness for pipe sections subject to cold or hot formings followed by liquid penetrant testing of all accessible surfaces was performed.

3. Code Case N-9 allowed the use of centrifugally cast austenitic steel pipe for nuclear service provided that specified chemical and mechanical properties are satisfied: inside and outside surfaces shall, (1) be machine-finished to 250 micro-inch RMS or 225 micro-inch AA or finer; (2) be pressure tested at 1.5 times the rated pressure and fluid penetrant inspected; (3) be fully radiographed; (4) meet the requirements of ASTM E-71 for Class 2 quality casting; and (5) be reheat treated at 1950°F for hot formed sections. Stress allowables should be in accordance with Table 2-3.

   Note: These stress values were based on a casting quality factor of 1.00, and required a minimum specified tensile strength of 70 ksi.

4. Code Case N-10 permitted the use of cast austenitic steel butt welding fittings for nuclear service provided that ASTM Specifications A-351 and ASA B16.9 are augmented by the following requirements:

   a. specified chemistry and mechanical properties shall be satisfied,
   b. fittings shall be finished to 250 micro-inch RMS or 225 micro-inch AA or finer,
   c. fittings shall be tested at 1.5 times the rated pressure,
   d. fittings shall be inspected by the fluid penetrant method and be fully radiographed in satisfaction of the ASTM E-71 requirements for Class 2 quality castings,
   e. fittings shall be heat treated at 1950°F followed by rapid cooling in air or a liquid medium,
   f. stress allowables shall be in accordance with Table 2-4, provided that minimum specified tensile strength was 70 ksi.

5. Code Case N-11 indicated that <u>any</u> sound means of providing for thermal expansion may be used and the following requirements must be met:

   a. must meet requirements of Section 6, Chapter 3 of ASA B31.1-1955,
   b. material recognized by ASA B31.1-1955,
   c. if sliding or swivel type, have a positive seal or leak-off control,
   d. provide for thermal expansion due to rapid temperature fluctuations.

6. Code Case N-12 provided a procedure for qualifying new materials for use in nuclear piping systems. The following subjects were discussed: ASTM identification, alternate identification, creep and stress rupture data, physical properties, heat treatment, hardness measurements, impact strength and transition temperature, radiation and temperature effects, microstructure variations, availability, weldability, and test results.

Of special interest are the material properties used for stainless steel, since this is generally the type of material used in Class 1 piping system. Type 304 and Type 316, the most commonly used, were not included as allowed materials in the 1955 B31.1 Code. The values included in the later 1967 version of B31.1 were comparable to those from the Code Case. For all materials except the 316/316H, the allowable stresses at operating temperatures (500-650°F), are less than those in current Codes; those of 316/316H are comparable.

The Code Cases also point out that designers <u>were</u> considering additional requirements for overall quality of the installed piping systems beyond those included in B31.1 conventional power plant piping. These considerations were obviously the initial thoughts that formulated the design, fabrication, inspection and overall quality standards that eventually lead to development of the ANSI B31.7 [11] requirements for nuclear piping which later were included in ASME Code Section III requirements for Class 1 piping.

Table 2-1

Summary of Key Design Differences Between Codes

General Design:	B31.1:	ASME III, Class 1
Basic Allowable Stress	Lower	Higher
Allowable Stress Basis	Maximum Stress	Stress Intensity (& Stress Intensity Range)
Local Effects	Stress Intensification Factors	Stress Indices
Fatigue Basis:		
Consideration of Geometric Discontinuities	Limited Evaluations	Complete Range
General Thermal Expansion & Secondary Stresses	Stress Range Reduction Factors (which consider number of thermal expansion cycles)	$3S_m$ (but may be exceeded with additional evaluation of cyclic operations)
Bi-Metallic and Adjoining Thickness Difference Effects	Not Considered	Secondary Stress Intensity
Through-wall Transient Stresses	Not Considered	Peak Stress Intensity Secondary Stress Intensity
Maximum Thermal Expansion Stress	$1.25\ S_c + 1.25\ S_h$ ($\approx 1.6\ S_y$ Ferritic) ($\approx 2.2\ S_y$ Austenitic)	$3S_m$ ($\approx 2\ S_y$ Ferritic) (≈ 2-$2.2\ S_y$ Austenitic)
Anchor Movements	No Consideration (prior to 1973)	Included

Table 2-2

Code Case N-7 Allowable Stress Values (ksi) for Stainless Steel

Material	Temperature (°F)							
	≤ 100	200	300	400	500	600	650	700
304 304H	18.75	16.65	15.0	13.65	12.5	11.6	11.2	10.8
304L	17.5	15.3	13.1	11.0	9.7	9.0	8.7	8.5
316 316H	18.75	18.75	17.9	17.5	17.2	17.1	17.05	17.0
316L	17.5	16.25	14.5	12.0	11.0	10.15	9.8	9.45
321/321H 347/347H 348/348H	18.75	18.75	17.0	15.8	15.2	14.9	14.85	14.8
309	18.75	18.75	17.3	16.7	16.6	16.5	16.45	16.4
310	18.75	18.75	18.5	18.5	18.2	17.7	16.9	16.6

Table 2-3

Code Case N-9 Maximum Allowable Stress Values in Cast Stainless Steel Pipe (ksi)

Material	Temperature (°F)							
	≤ 100	200	300	400	500	600	650	700
CPFS	17.5	15.7	14.25	13.1	12.2	11.7	11.5	11.3
CPF8M	17.5	16.9	16.5	16.3	15.9	15.35	15.0	14.7
CPF8C	17.5	17.0	15.6	14.2	13.0	12.2	11.9	11.7

Table 2-4

Code Case N-10 Maximum Allowable Stress Values for Cast Fittings (ksi)

Material	Temperature (°F)							
	≤ 100	200	300	400	500	600	650	700
CF8	17.5	15.7	14.25	13.1	12.2	11.7	11.5	11.3
CF8M	17.5	16.9	16.5	16.3	15.9	15.35	15.0	14.7
CF8C	17.5	17.0	15.6	14.2	13.0	12.2	11.9	11.7
CH20	17.5	16.1	15.15	14.6	14.55	14.45	14.4	14.35
CK20	16.25	15.3	14.9	14.6	14.45	14.45	14.4	14.35

Section 3

EVALUATION OF A PWR CHARGING LINE

The charging system in a PWR is part of the Chemical and Volume Control System (CVCS). The charging system returns purified water from the CVCS to the reactor coolant loop cold leg. The returning water is heated to near-reactor conditions in a regenerative heat exchanger that is also in contact with the letdown flow which goes from the reactor to the CVCS.

Charging lines in PWRs are one of the most fatigue sensitive piping systems in these plants because of the rapid changes in temperature during periods of charging and letdown transient operating conditions. The PWR charging line selected for evaluation was constructed from 3-inch Schedule 160 stainless steel piping and fittings. Schematically the charging line can be divided into two zones based on the temperature transients, with the zones as shown in Figure 3-1.

- **Zone A** is the section of the charging line between the reactor coolant piping nozzle and the first check valve. The temperature in this zone typically remains at reactor coolant temperature when charging is stopped and, during charging, is controlled by the temperature of the charging line flow.
- **Zone B** is the remaining section of the charging line from the check valve to the outlet of the regenerative heat exchanger. This zone is affected by charging transients and will cool to ambient temperature over long periods with no flow.

During conditions with charging flow on, the charging system temperature changes with flow rate, with both zones seeing essentially the same temperature transients.

The charging nozzle, which connects the charging system to the reactor coolant cold leg, is another known fatigue-sensitive area. It experiences the Zone A transients. It was not considered in this evaluation since the original design was established as part of the reactor coolant piping stress analysis and design details were not available. It is known that the nozzle contains a thermal sleeve to better accommodate thermal transients.

In the sections that follow, results of a B31.1 Code evaluation are compared with an ASME Code Section III, Class 1 evaluation for selected critical locations in the charging line.

3.1 B31.1 Analysis

Design Requirements: As discussed in Section 2.2, the B31.1 Code for piping requires that Equations 11, 12, 13 and 14 be satisfied. Allowable stresses are shown in Table 3-1 for the charging line, constructed from Type 316 stainless steel. For Equation 12, the effects of earthquake loadings were included. The stress analysis of record was B31.1 with Summer 1973 Addenda.

Results: Results of the B31.1 analysis, are summarized in Figure 3-2. B31.1 stresses are shown in in this figure as a ratio of calculated stress to the allowable stress values tabulated in Table 3-1 ("stress ratios"). All the B31.1 stresses were less than 75% of the allowable values. Application of a stress range reduction factor less than one to the allowable thermal expansion stress was not required because the number of significant thermal expansion ranges was much less than 7000.

3.2 ASME Section III, Class 1 Analysis

Design Requirements: For a Class 1 analysis, satisfaction of the appropriate stress intensity Equations 9 through 14 is required as discussed in Section 2.3.

The discussions that follow will focus on fatigue usage at typical locations (weld, elbow, tee) in Zones A and B of the charging line. Because the cross-section of the charging line is the same for Zones A and B, different fatigue usages in the same zone are primarily due to different stress indices which are listed in Table 3-2. The higher stress indices shown in parentheses in Table 3-2 are from the 1974 Code; the other values are from the 1986 Code. Also shown in Table 3-2 are the B31.1 stress intensification factors (SIFs), for comparison.

Transient Thermal Analysis: Plots of temperature and charging flow versus time for various transient thermal events experienced by the charging line are contained in Appendix A, Figures A-1 through A-12. These are derived from the design of a relatively newer PWR plant with a Section III, Class 1 piping system design. For the transient thermal evaluation, each temperature cycle (cooldown or heatup) was assigned an event number 1 through 21 as shown in these figures. The transient thermal events experienced by Zones A and B differ only for those periods when there is no flow through the charging line (Figures A-1, A-3, A-4, A-5 and A-6). Results of the transient thermal analyses are summarized in Tables 3-3 and 3-4 for Zones A and B, respectively. The ΔT_1, ΔT_2 terms in these tables are for the 3.5-inch diameter, 0.437-inch thick straight pipe. The T_a-T_b term was calculated specifically for a location at the pipe-to-valve weld where a thickness of 0.66 inches was taken for the valve and was based on simple one dimensional analysis of the two thicknesses. It should be noted that the ΔT_1, ΔT_2, and T_a-T_b terms are time independent individual maximums which therefore conservatively predict the fatigue usage values computed, as described below. Load cases in Tables 3-3 and 3-4 are proceeded with "SSH+" or "SSH-" to indicate that these cyclic events occur while the reactor is operating at steady-state hot conditions (i.e., at approximately 560°F).

Fatigue Usage Results: Fatigue usage results computed for the Class 1 analysis are summarized in Table 3-5 for weld, elbow and tee locations in Zones A and B. Also listed in this table are pressure and moment loads for rated operating conditions. All fatigue usage values computed per the requirements of the 1986 Code were well below 0.2.

Fatigue usage values per the 1974 Code were greater than 1.0 for the valve-weld location on both the Zones A and B sides of the valve. These high usage values are due to the inclusion of the ΔT_1 term in Equation 10 (for S_n) as required by the 1974 Code

(See Section 2.3 for a description of the relevant Code equations.). To illustrate the difference in fatigue usage calculations between the 1986 and 1974 Codes, stresses and fatigue usage contributions for the most significant load set pairs are summarized in Tables 3-6 and 3-7. S_n values computed per the 1974 Code, which exceed $3S_m$, result in a high alternating stress, S_{alt}, because the multiplier K_e is much greater than 1.0.

The contributions to high usage are always associated with rapid temperature changes, especially for step-change transients. For example, load sets 11-30 and 11-17 are loading combinations that combine a step-change-down transient (Event 10A) with a step-change-up transient (Event 5B). Load set 6-9 combines a step-change-down (Event 2A) with a rapid-change-up transient (Event 4). These similar transients do not occur in Zone B, and the computed usage is much less. (See transient definitions in Appendix A.)

Revised Fatigue Usage Results using Increased B31.1 Moments: The moments existing in the charging line were less than the B31.1 allowable for this calculation. To demonstrate the effect of higher B31 moments in the Class 1 analysis (which could exist in other lines in other plants), revised fatigue calculations were performed for the Zone A valve-weld location as shown in Table 3-5. The code allowable limiting moment was chosen for this demonstration. Stresses due to occasional moments were increased to the B31.1 allowable stress limit of 19.92 ksi and thermal expansion stresses were increased to the allowable limit of 44.13 ksi. The revised maximum fatigue usage per the 1986 Code was 0.31, up by 0.21 but still well below the allowable limit of 1.0. This resulted because the most significant alternating stresses were those due to through wall gradients and differential thermal expansion stress terms.

3.3 Summary of PWR Charging Line Evaluation

- The PWR charging line is acceptable when designed to either the B31.1 Code or the present (1986) ASME Section III, Class 1 Code. This evaluation shows that no geometric or loading discontinuities exist that would call into question the CLB for fatigue for this system.

- Although the valve-weld location would appear to have a geometric or loading discontinuity of concern, current code requirements show that the fatigue usage factor is only 10% of the allowable. There is high computed usage only when the location is evaluated to the older 1974 code because of the inclusion of the ΔT_1 term in Equation 10.

- The bending moments in the evaluated system were considerably below the B31.1 allowable. However, even when bending moments up to the maximum allowed by B31.1 are used, the charging line is acceptable when evaluated using the 1986 Code Class 1 analysis requirements.

- The analysis demonstrated that high usage occurred only at regions with geometric discontinuities and was associated only with rapid-temperature change transients which suddenly changed the temperature boundary condition from reactor temperature to ambient temperature (or vice-versa).

- Although no geometric or loading discontinuities were found in this piping system that would compromise the B31.1 design fatigue resistance, a few critical locations--such as welds at locations between two regions with dissimilar transient thermal response--can be identified as the basis for a limited B31.1/ASME Section III comparative assessment.

Table 3-1

Allowable Stresses (ksi)

B31.1 Allowable Stresses (ksi)			
Eq. 11 S_h	Eq. 12 $1.2S_h$	Eq. 13 S_A	Eq. 14 S_A+S_h
16.600	19.920	27.525	44.125

Allowable stress derived from ANSI B31.1-1973.

Table 3-2

SIF Values and Class 1 Stress Indices
for Selected Locations

Loc.	B31.1 SIF	ASME Section III, Class 1 Indices								
		B_1	C_1	K_1	B_2	C_2	K_2	C_3	C_3'	K_3
Weld	1.0	0.5	1.0 (1.1)[2]	1.2	1.0	1.0	1.8	0.6 (1.0)	0.5	1.7
Elbow	0.997[1,3]	0.236 (1.0)	1.258	1.0	1.46 (1.64)	2.19	1.0	1.0	0.5	1.0
Tee	0.773[3]	0.5	1.5	4.0	1.153	1.545	1.0	1.0	0.5	1.0

Notes:

1. Stress intensification factor for a 1.5-D bend.
2. Indices in parentheses are per the 1974 Code; other values are per the 1986 Code.
3. SIF = 1.0 used in analysis.

Table 3-3

Load Cases Simulated and Transient Thermal Evaluation Results for Zone A (Hot Side)

Sequence	Load Case	T_{pipe}	ΔT_1	ΔT_2	T_a-T_b	# Cycles
1	Zero Load	70	0	0	0	200
2	Steady State Hot: SSH	560	0	0	0	100000
3	SSH + OBE	560	0	0	0	8
4	SSH - OBE	560	0	0	0	10
5	SSH + Event # 1	500	-17.31	5.206	11.4	60
6	SSH + Event # 2A	100	277.8	-97.97	-83.1	60
7	SSH + Event # 2B	200	-45.47	9.812	14.1	60
8	SSH + Event # 3	292.8	120.1	-27.57	-81.9	200
9	SSH + Event # 4	181.9	-116.8	25.57	74.9	200
10	SSH + Event # 5A	307.8	134.7	-32.91	-78.2	20
11	SSH + Event # 5B	70	-196.2	65.21	106.2	20
12	SSH + Event # 6A	100	277.8	-97.97	-82.3	20
13	SSH + Event # 6B	218.1	-46.34	10.98	19.3	20
14	SSH + Event # 7	500	-17.35	5.21	12.4	200
15	SSH + Event # 8A	494.4	48.25	-19.15	-13.4	200
16	SSH + Event # 9	500	-17.31	5.206	11.4	20
17	SSH + Event # 10A	100	278.8	-115.3	-83.5	18
18	SSH + Event # 10B	200	-44.96	8.6	14.3	20
19	SSH + Event # 11	510.4	-18.08	5.486	10.8	24000
20	SSH + Event # 12	549.4	18.55	-5.561	-9.0	24000
21	SSH + Event # 13	477.2	40.31	-12.26	-18.9	24000
22	SSH + Event # 14	422.1	27.58	8.21	15.7	24000
23	SSH + Event # 15	429.1	26.71	-5.608	-17.7	2000
24	SSH + Event # 16	435.1	-15.42	3.081	16.2	2000
25	SSH + Event # 17	545.9	21.89	-8.704	-10.8	2000
26	SSH + Event # 18	517.7	-9.306	1.777	6.3	24000
27	SSH + Event # 19	486.2	40.93	-11.19	-19.4	24000
28	SSH + Event # 20	404.6	11.51	-2.545	-9.7	24000
29	SSH + Event # 21	406.3	-47.52	12.52	24.2	24000
30	SSH + Event # 10A+OBE	100	278.8	-115.3	-83.5	2

Table 3-4

Load Cases Simulated and Transient Thermal Evaluation Results for Zone B (Cold Side)

Sequence	Load Case	T_{pipe}	ΔT_1	ΔT_2	T_a-T_b	# Cycles
1	Zero Load	70	0	0	0	200
2	Steady State Hot: SSH	560	0	0	0	100000
3	SSH + OBE	560	0	0	0	8
4	SSH - OBE	560	0	0	0	10
5	SSH + Event # 1	500	0	0	0	60
6	SSH + Event # 2A	100	0	0	-76.8	60
7	SSH + Event # 2B	200	-101.5	21.93	69.5	60
8	SSH + Event # 3	292.8	120.1	-27.57	-81.9	200
9	SSH + Event # 4	181.9	-116.8	25.57	74.9	200
10	SSH + Event # 5A	307.8	134.7	-32.91	-78.2	20
11	SSH + Event # 5B	70	0	0	0	20
12	SSH + Event # 6A	100	0	0	0	20
13	SSH + Event # 6B	218.1	-113.3	25.31	72.1	20
14	SSH + Event # 7	500	0	0	0	200
15	SSH + Event # 8A	494.4	12.18	-5.041	-3	200
16	SSH + Event # 9	500	0	0	0	20
17	SSH + Event # 10A	100	0	0	0	18
18	SSH + Event # 10B	200	-101.5	21.93	14.3	20
19	SSH + Event # 11	510.4	-18.08	5.486	10.8	24000
20	SSH + Event # 12	549.4	18.55	-5.561	-9.0	24000
21	SSH + Event # 13	477.2	40.31	-12.26	-18.9	24000
22	SSH + Event # 14	422.1	-27.58	8.21	15.7	24000
23	SSH + Event # 15	429.1	26.71	-5.608	-17.7	2000
24	SSH + Event # 16	435.1	-15.42	3.081	16.2	2000
25	SSH + Event # 17	545.9	21.89	-8.704	-10.8	2000
26	SSH + Event # 18	517.7	-9.306	1.777	6.3	24000
27	SSH + Event # 19	486.2	40.93	-11.19	-19.4	24000
28	SSH + Event # 20	404.6	11.51	-2.545	-9.7	24000
29	SSH + Event # 21	406.3	-47.52	12.52	24.2	24000
30	SSH + Event # 10A+OBE	100	278.8	-115.3	-83.5	2

Table 3-5

PWR Charging Line Evaluation (ASME III, Class 1)

Summary of Loads and Fatigue Usage

LOCATION	OPERATING PRESSURE (psi)	MOMENTS (FT-LBS)			FATIGUE USAGE	
		WEIGHT	OCC.	THERM.	1986 CODE	1974 CODE
ZONE A (HOT SIDE):						
VALVE-WELD	2235 2235 2235 2235	71 71 71 71	49 4016 (Note 1) 49 (Note 2) 4016 (Note 3)	3266 3266 (Note 1) 9820 (Note 2) 9820 (Note 3)	0.103 0.104 (Note 1) 0.234 (Note 2) 0.310 (Note 3)	3.624
WELD	2235	71	49	3266	0.041	0.531
ELBOW	2235	71	49	3266	0.009	0.289
ZONE B (COLD SIDE):						
VALVE-WELD	2235	1646	2191	3893	0.050	1.504
WELD	2235	1646	2191	3893	0.013	0.022
ELBOW	2235	1646	2191	3893	0.003	0.011
TEE	2235	1646	2191	3893	0.006	0.014

Notes:
1. Moments modified to satisfy limiting stress of 19.92 ksi per Eq. 12 of ANSI B31.1 Code.
2. Moments modified to satisfy limiting stress of 44.13 ksi per Eq. 14 of ANSI B31.1 Code.
3 Moments modified to satisfy limiting stresses of 19.92 ksi per Eq. 12 and 44.13 ksi per Eq. 14 of ANSI B31.1 Code.

Table 3-6

Detailed Fatigue Calculation (ASME III, Class 1)

PWR Charging Line: Zone A Pipe-Valve Weld - (1974 Code)

Load Set ID's	Eq. 10 [1] Stress, S_n (ksi)	Eq. 11 Stress, S_p (ksi)	K_e ($S_{alt} = K_e \times S_p/2$)	Eq. 14 Stress, S_{alt} (ksi)	Number of cycles	Allowable cycles	Fatigue Usage
11 30	127.31	278.06	3.33	463.44	2	25	0.08
11 17	127.11	277.69	3.33	462.82	18	25	0.71
6 9	107.42	224.96	3.33	374.93	60	41	1.46
9 12	107.22	224.63	3.33	374.39	20	41	0.48
9 10	83.19	161.74	3.01	243.40	20	120	0.17
8 9	81.13	156.37	2.88	224.99	100	148	0.67
27 29	27.89	55.74	1.00	27.87	23900	1.0E6	0.02
21 22	21.49	43.68	1.00	21.84	24000	2.4E6	0.01
						Total	3.62

Note:
1. Allowable stress, $3S_m$, is 51.9 ksi

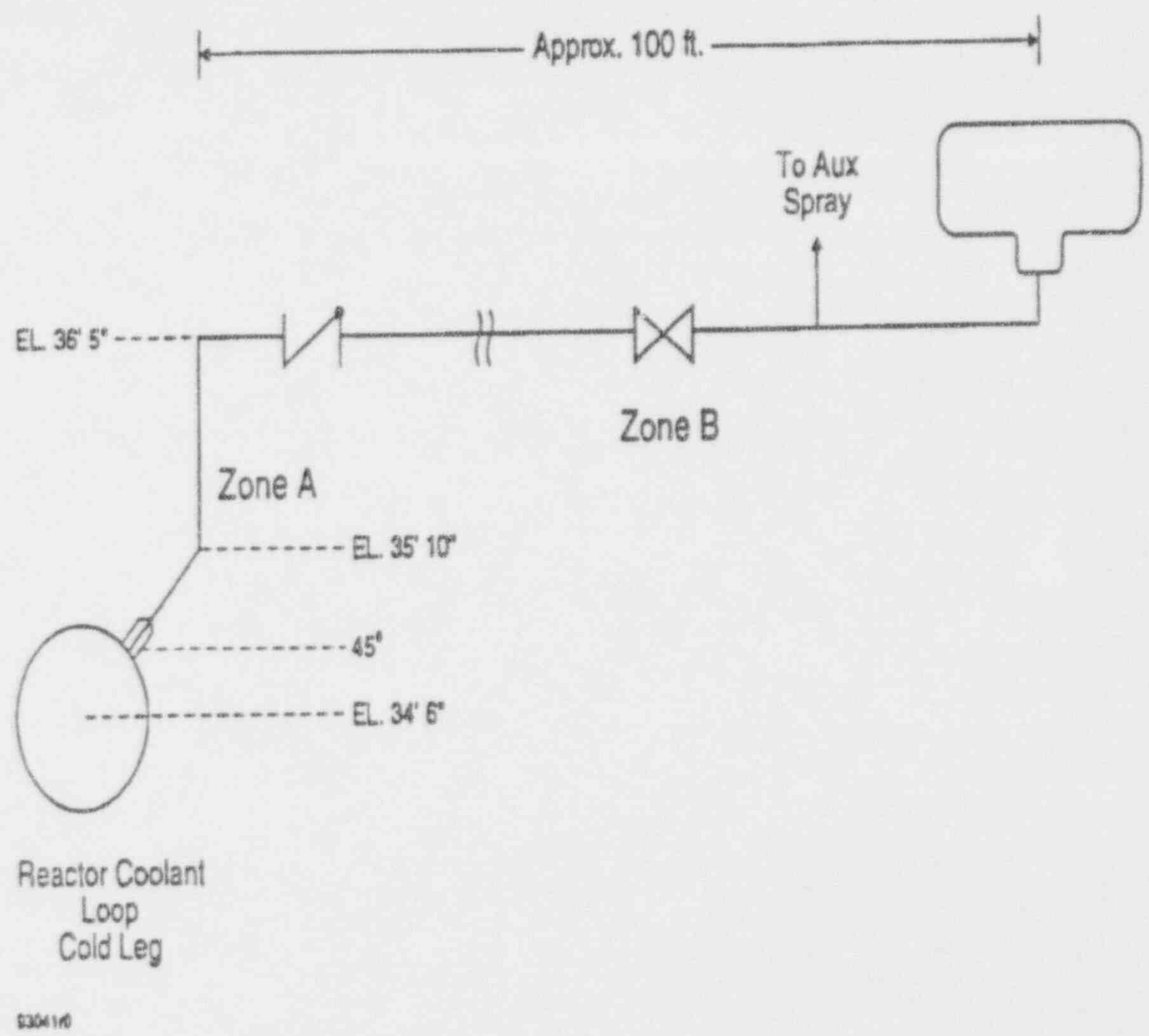
Table 3-7

Detailed Fatigue Calculation (ASME III, Class 1)

PWR Charging Line: Zone A Pipe-Valve Weld - (1986 Code)

Load Set ID's	Eq. 10 [1] Stress, S_n (ksi)	Eq. 11 Stress, S_p (ksi)	K_e ($S_{alt} = K_e \times S_p/2$)	Eq. 14 Stress, S_{alt} (ksi)	Number of cycles	Allowable cycles	Fatigue Usage
11 30	28.27	247.29	1.00	123.64	2	885	0.00
11 17	28.06	246.92	1.00	123.46	18	890	0.02
6 9	25.10	199.33	1.00	99.66	60	1825	0.03
9 12	24.98	199.13	1.00	99.57	20	1831	0.01
9 10	25.73	136.90	1.00	68.45	20	7625	0.00
8 9	25.80	130.94	1.00	65.47	100	9126	0.01
27 29	8.66	48.763	1.00	24.33	23900	1.6E6	0.01
21 22	6.62	38.07	1.00	19.03	24000	4.3E6	0.01
						Total	0.10

Note:
1. Allowable stress, $3S_m$, is 51.9 ksi



Notes:
1) Zone A affected by cold leg transients and refilling with cold leg water when flow stops.
2) Zone B affected by charging transients and cools to ambient over long periods with no flow.
3) When flow exists, transients in both lines are identical.

Figure 3-1. Schematic of the PWR Charging Lline SHowing Temperature Zones A and B

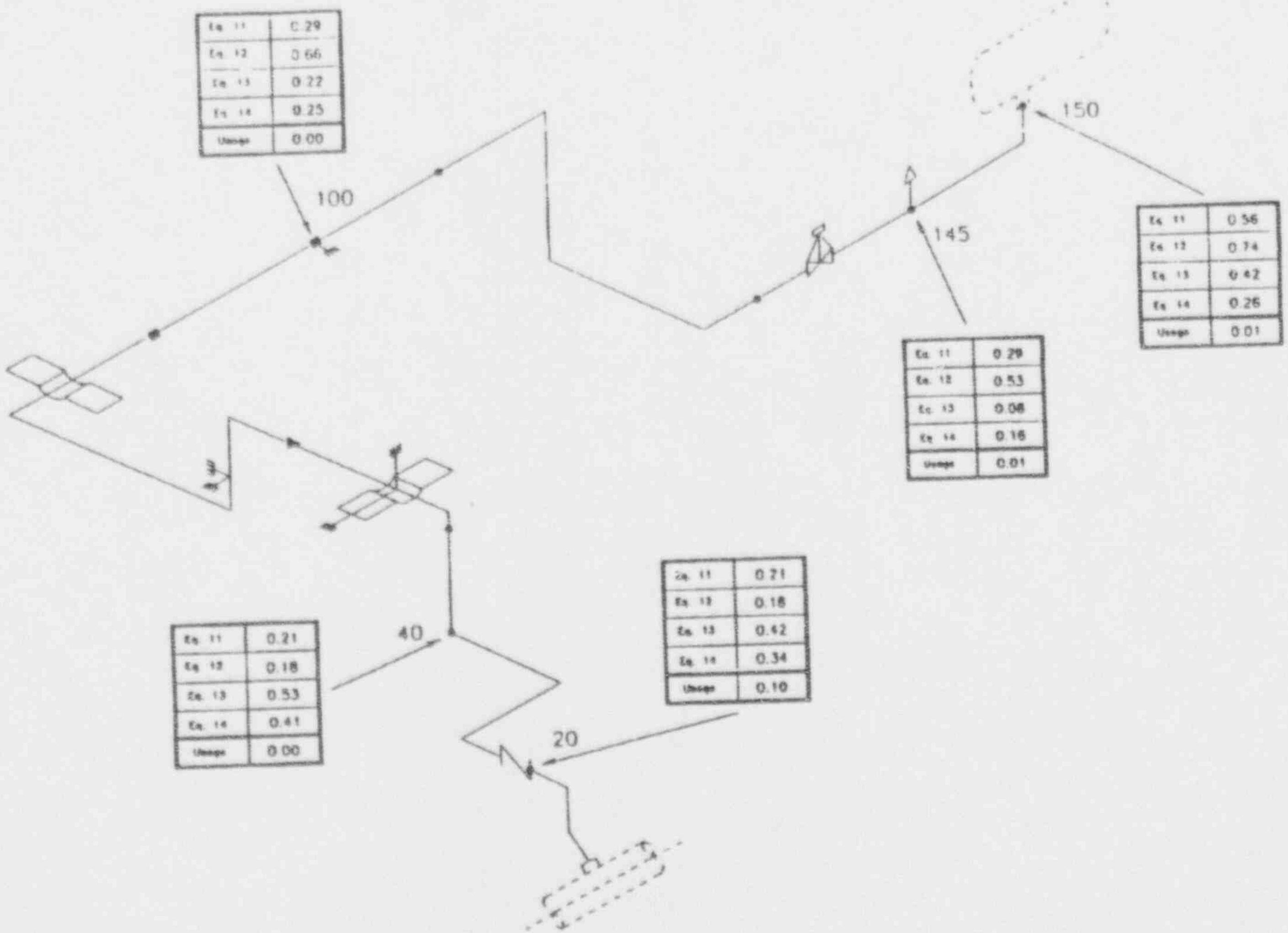


Figure 3-2. Isometric of the PWR Charging Line Showing B31.1 Stress Ratios and Class 1 Fatigue Usage Values at Select High Stress/High Usage Locations

Section 4

EVALUATION OF A BWR RECIRCULATION SYSTEM WITH ATTACHED LINES

The reactor recirculation system in a BWR is used to circulate excess water through the reactor core. During normal plant operation, it is exposed to reactor coolant temperature and pressure and due to the relatively slow rate of temperature changes required by the reactor vessel, it does not experience significant thermal stresses. However, during reactor shutdown, flow from the attached residual heat removal (RHR) system is initiated. This causes significant thermal stresses, especially at the piping near the RHR supply.

Isometrics of the BWR recirculation system along with the attached RHR supply and return lines selected for evaluation are shown in Figures 4-1 through 4-4. The recirculation piping is 28-inch Schedule 80 stainless steel. The attached RHR piping in 16-inch schedule 80 carbon steel, although there is a short section of 18-inch stainless steel piping adjacent to the recirculation piping. This system was originally designed to B31.1 and was more recently evaluated for compliance with the B31.1 Code (1977 Edition, Winter '78 Addenda). To assess the piping in accordance with ASME Section III, Class 1 requirements, analysis using the 1980 Edition of the Code (Summer 1982 Addenda) was conducted.

Results of the B31.1 and Class 1 analyses are summarized in the following sections along with a comparison of results for selected critical locations (component types) in these piping systems.

4.1 B31.1 Analysis

Design Requirements: As discussed in Section 2, the B31.1 Code for piping requires that Equations 11, 12, 13 and 14 be satisfied. Allowable stresses are shown in Table 4-1 for the recirculation system piping.

Results: Results of the B31.1 analysis are summarized in Figures 4-5 and 4-6 for selected

high stress locations in Loop A, which are typical of other locations in the recirculation piping system shown in Figures 4-1 through 4-4. The B31.1 stresses are shown in Figures 4-5 and 4-6 as a ratio of calculated stress to the allowable stress values ("stress ratios"). Also shown along with the B31.1 stress ratios are the fatigue usages computed for the Class 1 analysis described in the next
section.

For the Loop A recirculation lines shown in Figure 4-5, all B31.1 stresses were less than 70% of the allowable values. Stresses were typically high at the reactor pressure vessel (RPV) nozzle connections and at "Tee" branch connections.

Stress ratios shown in Figure 4-6 were high for the RHR return line from the recirculation line "Tee" (node 326) to the valve connection (nodes 408, 410). These stresses were less than 90% of the B31.1 Code allowable values shown above. B31.1 stress intensification factors for selected nodal locations are provided in Table 4-2 for comparison with Class 1 stress indices.

Calculated stresses at all locations in the recirculation piping system and the attached lines were well within the B31.1 allowables.

4.2 ASME Section III, Class 1 Analysis

Design Requirements: For a Class 1 analysis the stress intensity Equations 9 through 14 presented in Section 2.2 must be satisfied.

The discussions that follow will focus on a few locations in Loop A of the recirculation system where high fatigue usage values were computed for the Class 1 analysis. Loop B was similar. These locations and the associated Class 1 stress indices are summarized in Table 4-2.

Transient Thermal Load Cases: Load histograms for the recirculation piping are provided in Appendix B. The temperature/pressure/flow history for the shutdown that is a primary contributor to high fatigue usage values for the RHR return line is shown in Figure 4-7. During this event, a significant thermal shock results when cold water (50°F) from the RHR return line is suddenly injected into the recirculation line that is initially at approximately 375°F. For the design transient, it was assumed that there is an associated step-change-up after 15 seconds, when the RHR system has filled with hot reactor water. All other thermal transients are much less severe.

Results: Fatigue usage results computed for the Class 1 analysis are summarized in Figures 4-5 and 4-6 for selected locations in Loop A of the recirculation piping system along with B31.1 stress ratios for the purposes of comparison.

Low fatigue usage values (less than 0.2) were computed for all locations in the recirculation lines (Figure 4-5) except for instrument nozzle connections to the RPV inlet risers (nodes 16B, 247 are typical) where fatigue usages of 0.21 to 0.24 were reported.

Fatigue usages were highest in the section of the RHR return line upstream of the tee, as shown in Figure 4-6. The highest usage computed for the RHR return line was 1.56 located at the discharge side of the valve near the RHR return tee (node 408). Stresses and fatigue usages at node 408, for the most significant load set pairs are summarized in Table 4-3. The load set pair SD7/SD10 results in the most significant usage of 1.42. Transient thermal response temperatures (ΔT_1 and ΔT_2) and corresponding thermal stresses which contribute to the high fatigue usage at node 408 are summarized in Table 4-4.

This analysis brings up a point concerning the fatigue analysis of many Class 1 components. The idealized transient which has been evaluated is not expected to happen as analyzed. The analysis assumes an instantaneous opening of the RHR valve with admission of a maximum flow rate (with high heat transfer coefficient) at instantaneous minimum temperature (50°F). After a short period of cold shock, there is a step change to hot shock the component. No consideration was given to the fact that there would be considerable thermal heat capacity in the RHR system to modify the transient heatup at event SD10. No further evaluation was attempted in this scoping analysis to demonstrate that the usage could be reduced to less than unity. (A subsequent evaluation, performed employing a more detailed heat transfer analysis in the area of high fatigue usage, showed that the usage factors could be reduced to less than 1.0).

The high fatigue usage in the RHR return line near the valve connection is due to a combination of the following factors:

(i) High stress indices (Table 4-2)
(ii) Severe thermal transients simulated (ΔT_1 and ΔT_2 terms, Table 4-4)
(iii) Dissimilar temperature response on either side of the limiting location due to either geometry (node 408; valve-to-pipe weld) or material properties (node 406; carbon-to-stainless steel field weld).

4.3 Revised Class 1 Analysis with Limiting B31.1 Moments

Recognizing that stresses computed for the B31.1 analysis were well below B31.1 allowables, a simplified Class 1 analysis was performed with moments at the B31.1 limits to address the issue of how higher moments would change the fatigue usage values computed for the Class 1 analysis.

As shown in Table 4-5, B31.1 stresses due to thermal expansion moments (Eq. 13) at selected high usage locations were ratioed to the Code allowable value of 26.382 ksi and the resulting stress increase factors were applied to the Class 1 analysis to compute revised fatigue usage values. These results show that the increase in fatigue usage is minimal for an increase in up to 50% of the thermal expansion stress (nodes 401 and 404). The largest increase in fatigue usage by a factor of 4.58 was at node 247 corresponding to an artificial increase in expansion stress by a factor of 8.11.

These results illustrate that the higher system loads resulting in up to a 50% increase in computed B31.1 stresses will not significantly change the fatigue usage values computed for a Class 1 analysis. This result can be easily understood upon review of the Class 1 design requirements, discussed in Section 2.2, which show that fatigue usages are primarily a function of thermal shock stresses (i.e., the ΔT_1, ΔT_2, and T_a-T_b terms) and are less sensitive to typical changes in pressure and thermal expansion moments.

4.4 Summary of BWR Recirculation System Evaluation

- The BWR recirculation system satisfies the requirements of B31.1 with computed stresses well below the allowables. With the exception of the pipe-to-valve welds in the RHR return line, all other locations satisfy the fatigue requirements of the 1980 ASME Code Section III, for Class 1 components (Summer 1982 Addenda).

- For those locations with high fatigue usage, the most significant contributors were the combination of thermal loading discontinuities (step change transients) and geometric discontinuities (high stress indices), which is in agreement with the NUMARC technical position on fatigue.

- If code allowable B31.1 loads are used, all locations except for the valve-weld and a dissimilar metal weld (carbon/stainless steel) are acceptable per the 1980 ASME Section III Class 1 analysis. The recirculation riser location would have been acceptable if the moments had not been increased by such a high factor.

- The results show that the inherent fatigue resistance of piping components designed to B31.1 is compromised only by a combination of geometric and loading discontinuities, and that a few critical locations, such as welds and locations between two regions with dissimilar transient thermal response (i.e., metal welds or significant changes in cross section), are readily identifiable. These locations, when exposed to rapid thermal transients, will be the ones most adversely affected by metal fatigue.

Table 4-1

Allowable Stress (ksi)

Line	B31.1 Allowable Stress (ksi)			
	Eq. 11 S_h	**Eq. 12** $1.2*S_h$	**Eq. 13** S_A	**Eq. 14** S_A+S_h
Recirculation Loops* (Stainless Steel)	12.208	14.434	26.382	38.410
RHR Lines (Carbon Steel)	15.000	18.000	22.500	37.500

* RHR lines near recirculation loop are also stainless steel.

Table 4-2

B31.1 SIF Values and Class 1 Stress Indices for Selected Nodes

Loc. (loop A)	Node No.	B31.1 SIF	ASME Section III, Class 1 Indices								
			B_1	C_1	K_1	B_2	C_2	K_2	C_3	C_3'	K_3
RPV Inlet Riser	247	1.0	0.5	1.4	2.0	1.0	1.5	1.8	1.8	1.0	1.7
RHR Return Tee	326	1.7	0.5	1.5	4.4	2.1	3.6	1.1	1.0	0.5	1.1
RHR Return Elbow	401	2.7	0.5	1.0	1.6	1.0	1.5	2.0	0.7	0.6	1.9
RHR Return Reducer	404	2.0	0.5	1.0	1.7	1.0	1.0	2.0	0.6	0.5	1.9
RHR Return Field Weld	406	1.0	0.5	1.0	1.7	1.0	1.0	2.0	0.6	0.5	1.9
RHR Return Valve Weld	408	1.6	0.5	1.3	1.5	1.0	1.7	1.8	1.6	1.0	1.7

Table 4-3

ASME Section III, Class 1 Results for Node 408
RHR Return Line Pipe-Valve Weld

Thermal Event ID[1]	Eq. 9[2] Stress (ksi)	Eq. 10[3] Stress, S_n (ksi)	Eq. 11 Stress, S_p (ksi)	Eq. 12[3] Stress (ksi)	Eq. 13[3] Stress (ksi)	Eq. 14 Stress, S_{alt} (ksi)	No. of Cycles	Allow. Cycles	Usage
SD7 SD10	7.883	72.238	199.553	7.399	46.969	168.546	20	162	0.12
SD7 SD10	7.883	69.294	194.253	4.455	46.969	153.423	290	204	1.42
--[4] SD11	7.883	18.772	51.772	NA	NA	25.886	290	34784	0.01
All other events									.01
								Total	1.56

Notes:

1 Transient thermal event ID's for node 408 are summarized in Table 4-4.
2 Allowable stress, $1.5S_m$, is 17.908 ksi.
3 Allowable stress, $3.0S_m$, is 53.724 ksi.
4 Steady-state with reactor hot.

Table 4-4

Summary of ΔT's and Thermal Stresses for Node 408
ASME Section III, Class 1 Analysis

Thermal[1] Event	ΔT_1 °F	ΔT_2 °F	T_{avg} °F	ΔT_1 Stress ksi	ΔT_2 Stress ksi	T_a °F	T_b °F	T_a-T_b Stress ksi
SD7	80.0	20.0	488.0	9.440	4.720	546	430	19.163
SD11+	49.0	29.0	300.0	5.782	6.844	300	300	0.0
SD10-	197.0	79.0	92.5	23.246	18.644	50	135	14.042

Note:
[1] See Figure 4-7 for a description of thermal events.

Table 4-5

Revised Class 1 Fatigue Usages
Using Limiting B31.1 Thermal Expansion Stresses

Location (Loop A)	Node No.	B31.1 Eq.13 Stress (ksi)	Allowable S_a (ksi)	Factor S_a/Stress	Fatigue Usage	
					Original	Revised
RPV Inlet Riser	247	3.254	26.382	8.11	0.24	1.10
RHR Return Tee	326	14.246	26.382	1.85	0.35	0.70
RHR Return Elbow	401	21.183	26.382	1.25	0.55	0.57
RHR Return Reducer	404	17.171	26.382	1.50	0.34	0.36
RHR Return Field weld	406	7.203	26.382	2.83	0.74	1.71
RHR Return Valve Weld	408	10.103	26.382	2.61	1.56	2.76

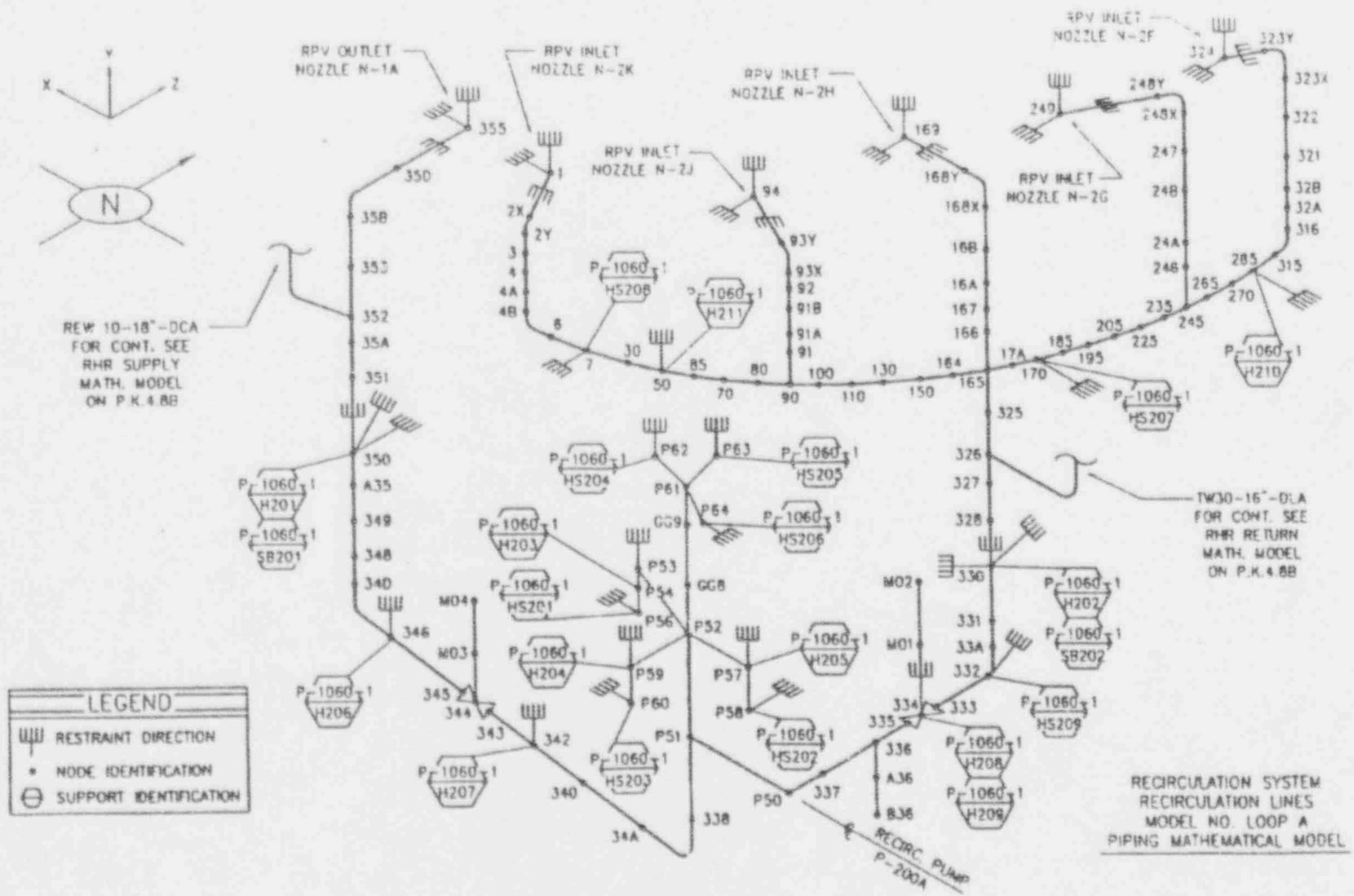


Figure 4-1. Isometric of the BWR Recirculation System Showing Node Numbers and Restraints: Loop A Recirculation Supply and Discharge Lines

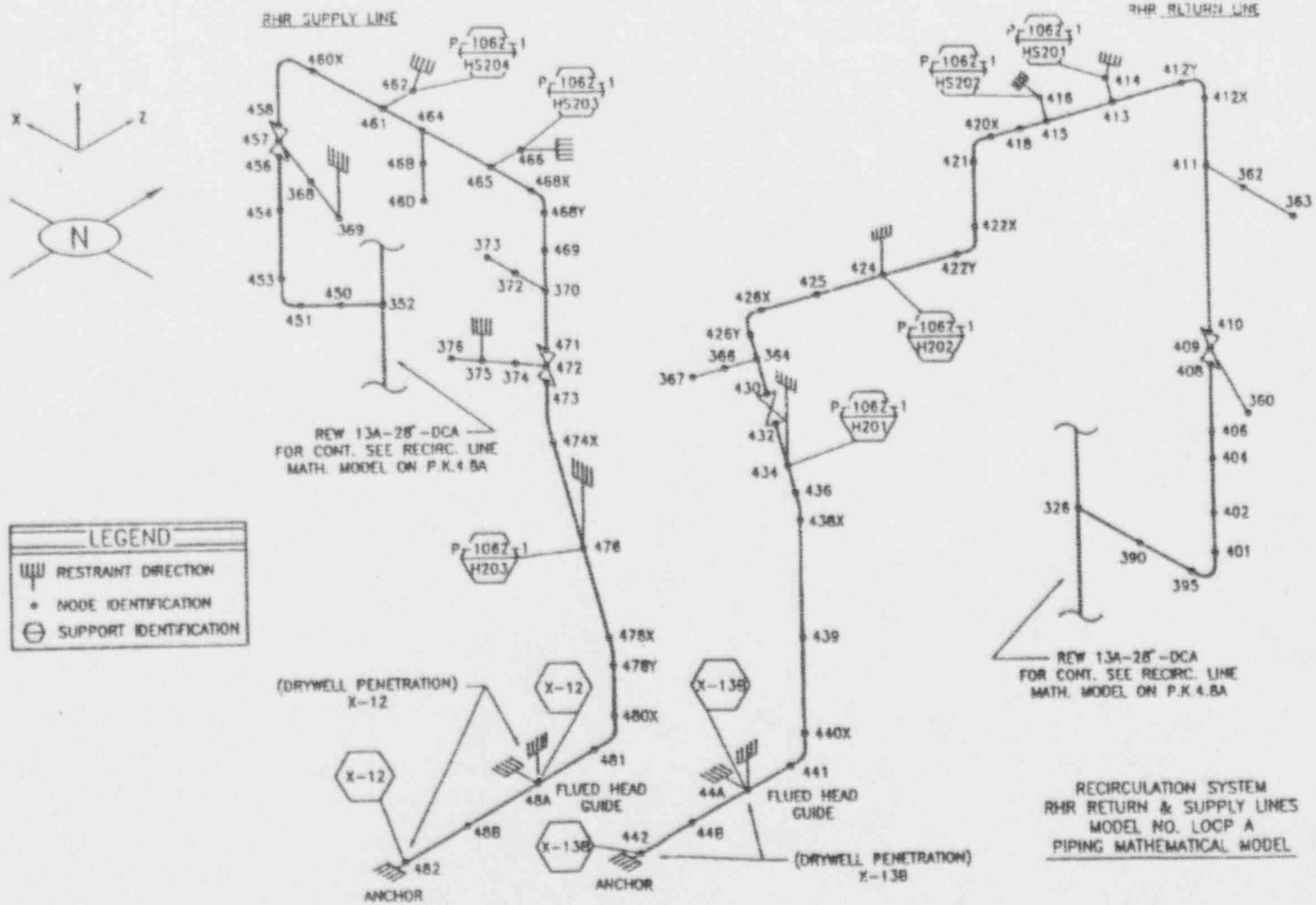


Figure 4-2. Isometric of the BWR Recirculation System Showing Node Numbers and Restraints: Loop A RHR Supply and Return Lines

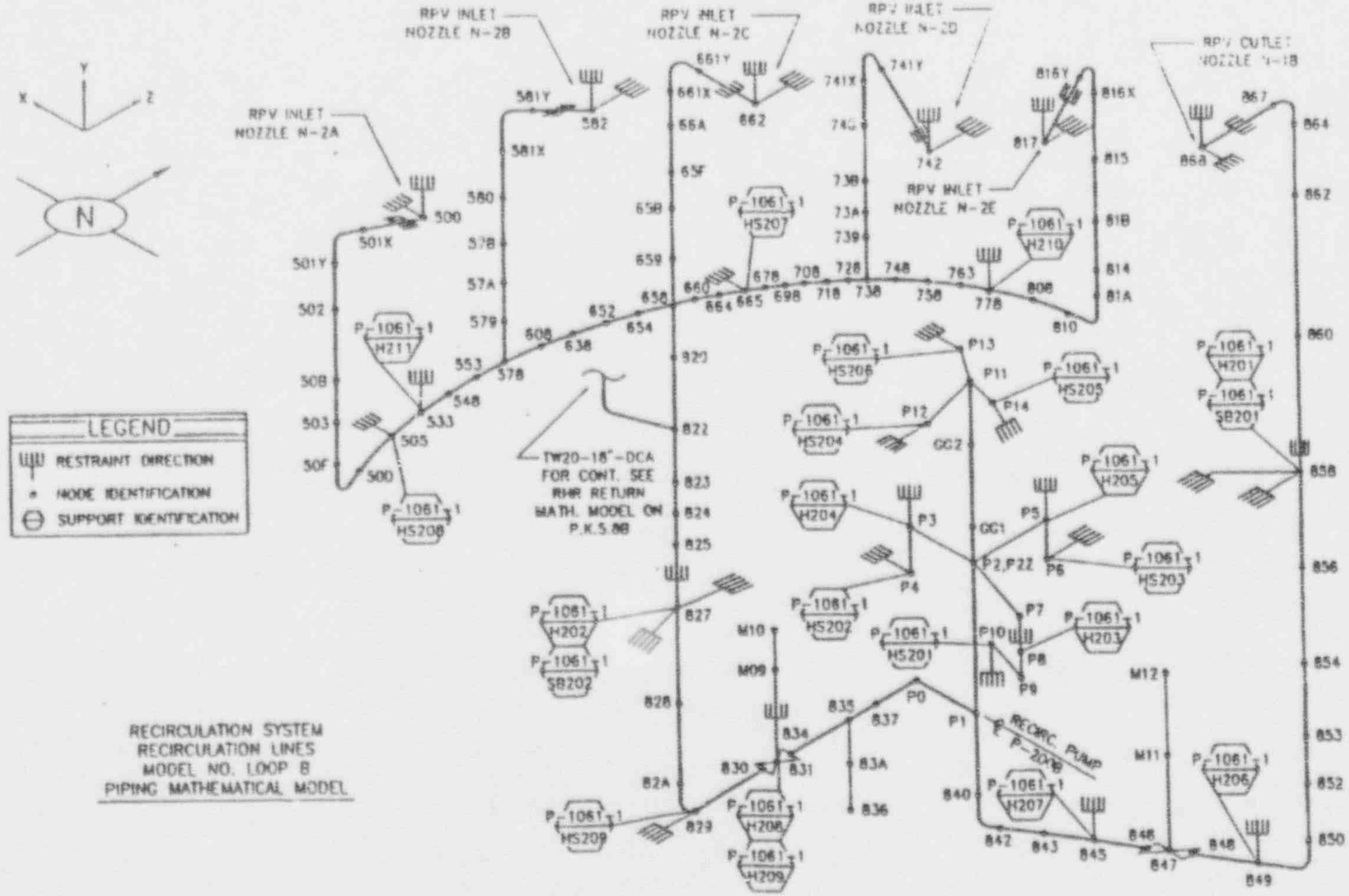


Figure 4-3. Isometric of the BWR Recirculation System Showing Node Numbers and Restraints: Loop B Recirculation Supply and Discharge Lines

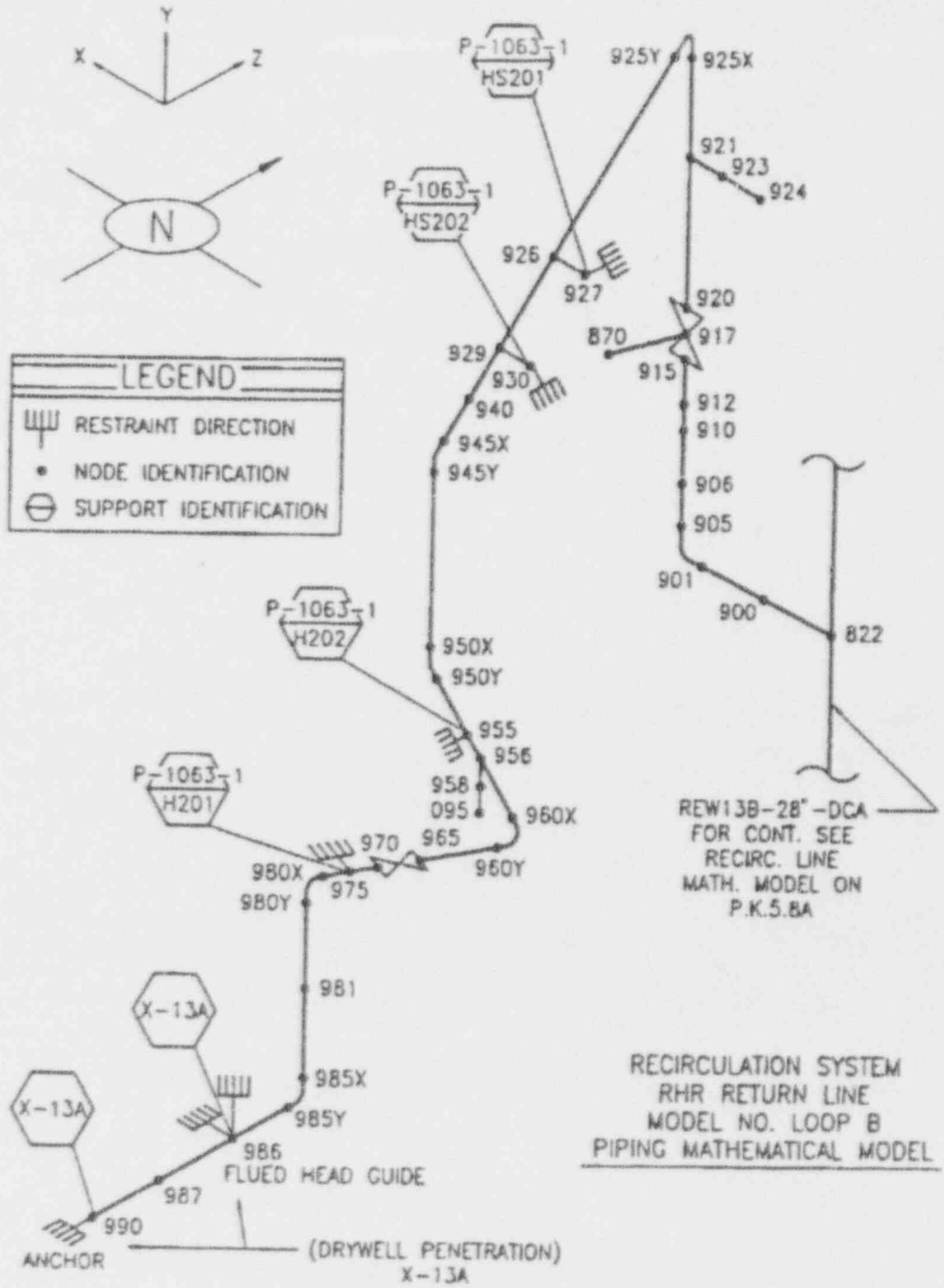


Figure 4-4. Isometric of the BWR Recirculation System Showing Node Numbers and Restraints: Loop B RHR Return Line

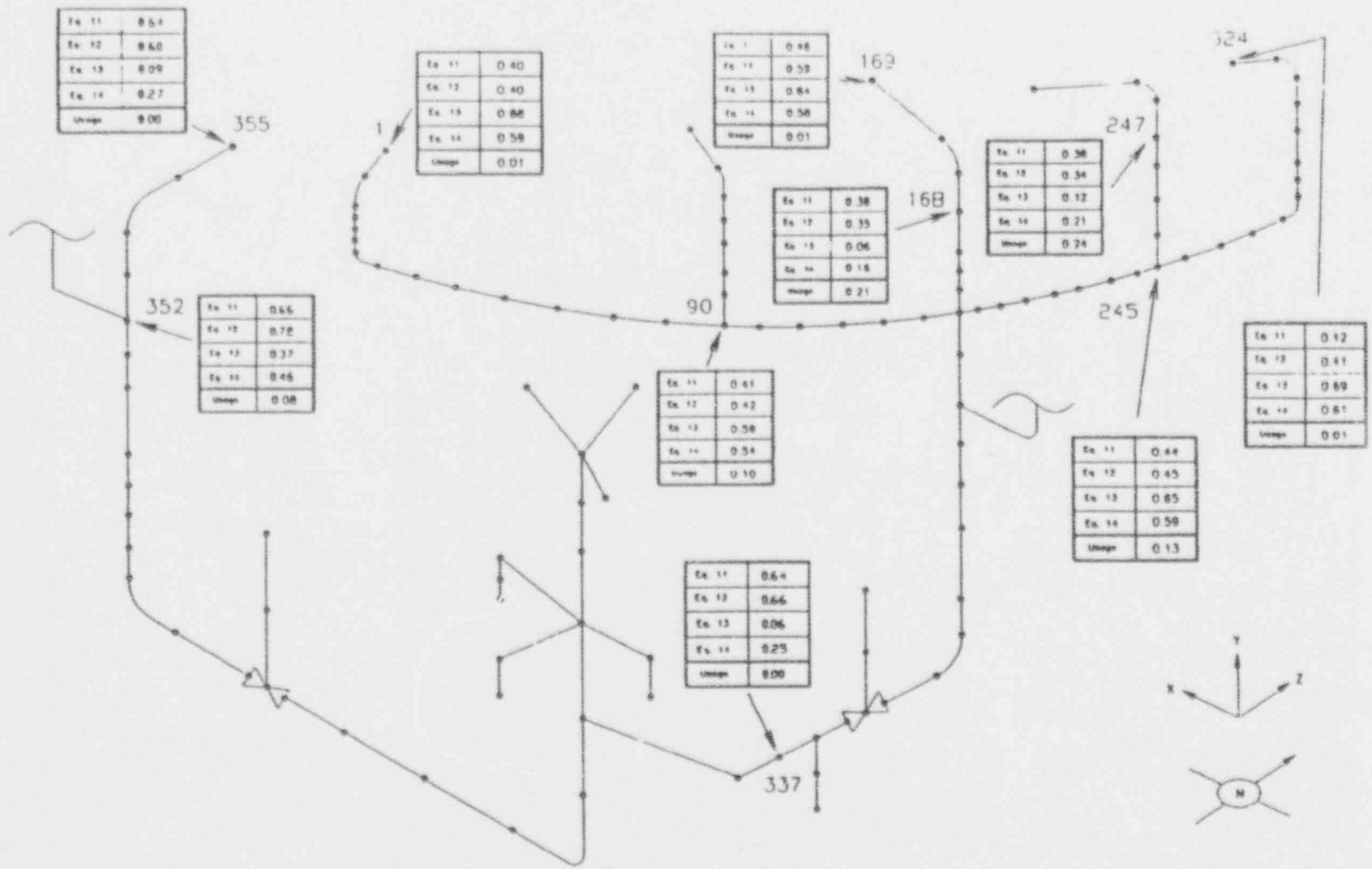


Figure 4-5. Isometric of the BWR Recirculation System Loop A Supply and Discharge Lines Showing B31.1 Stress Ratios and Class 1 Fatigue Usage Values at Select High Stress/High Usage Locations

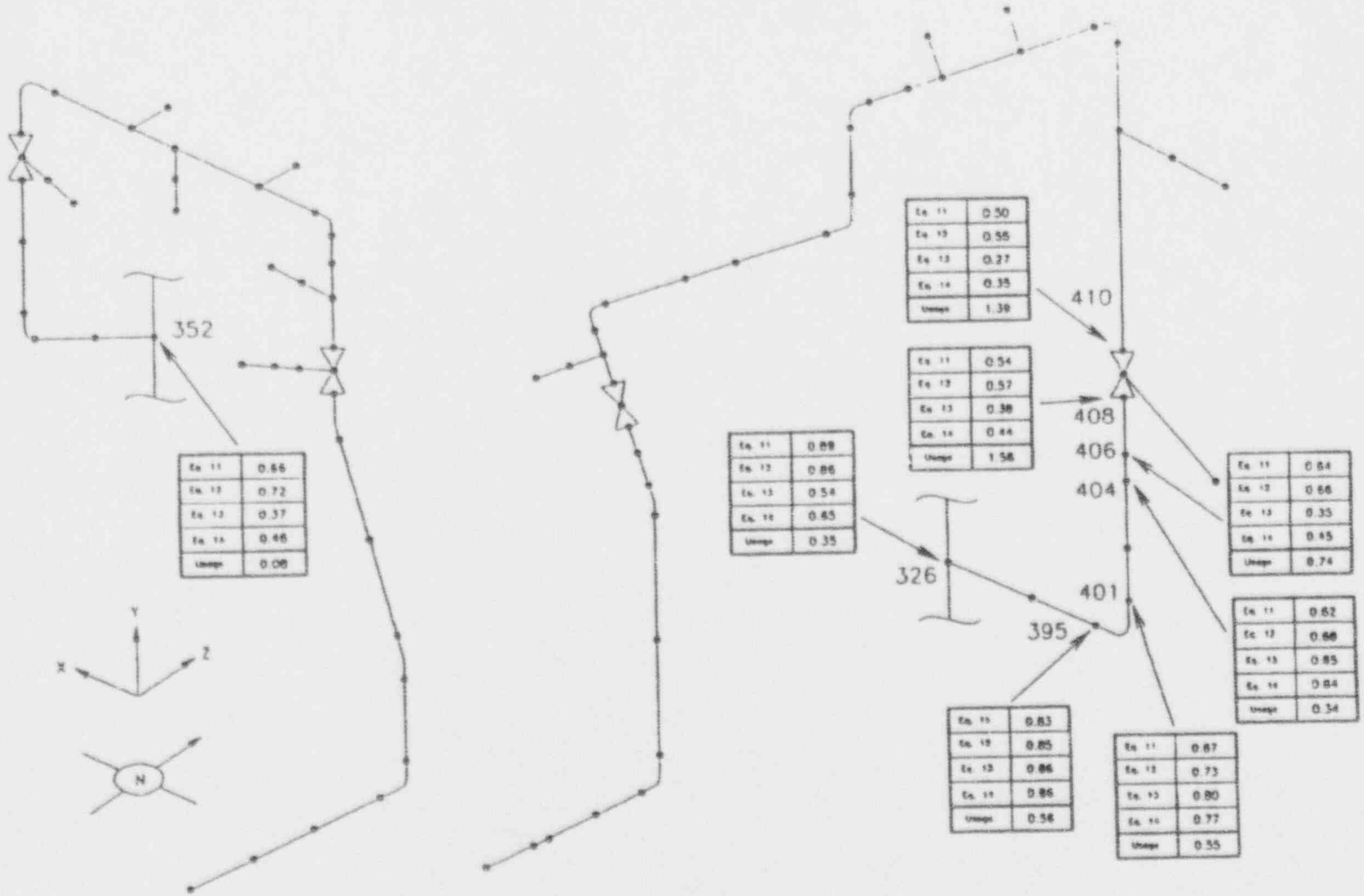
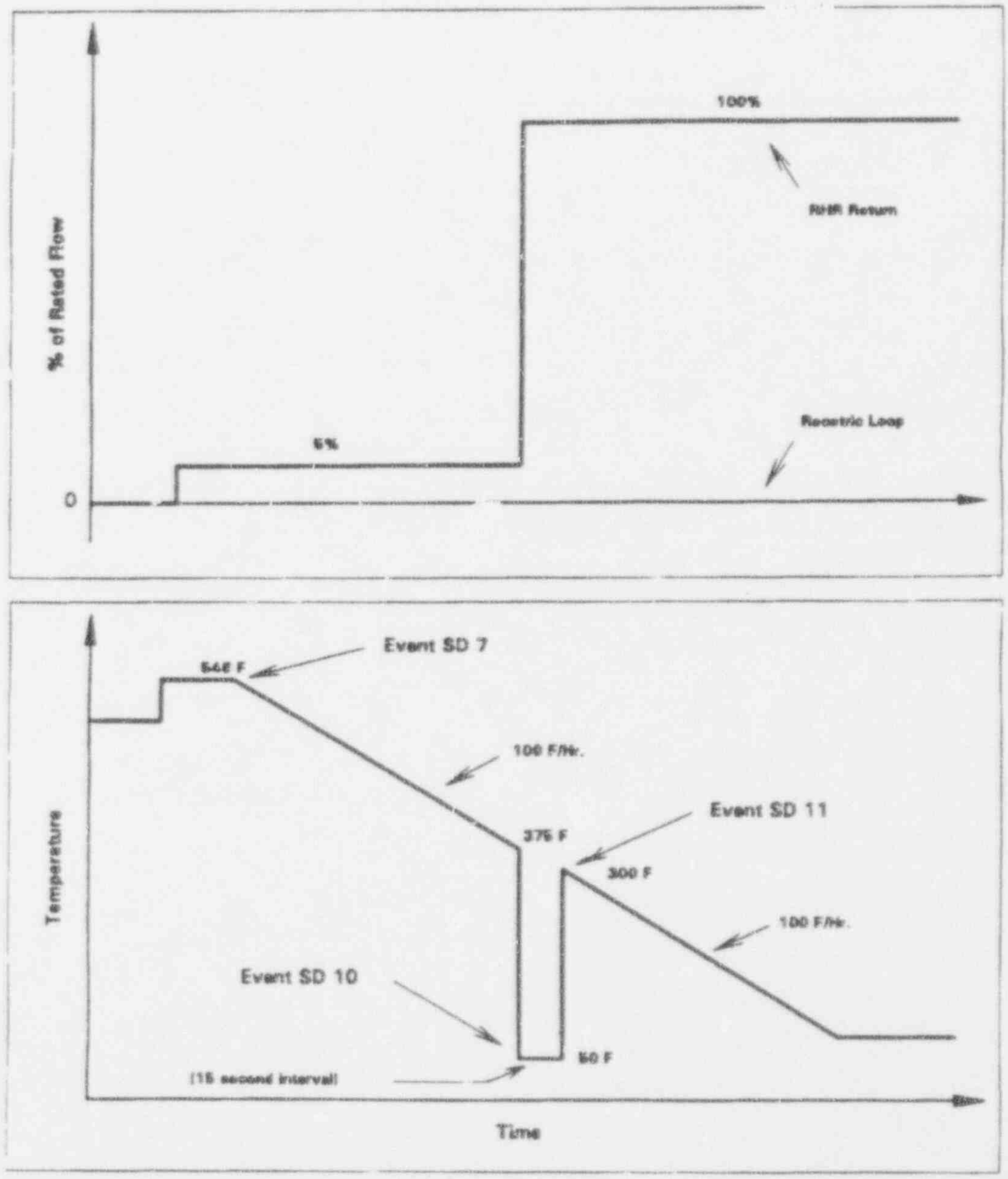


Figure 4-6. Isometric of the BWR Recirculation System Loop A RHR Supply and Return Lines Showing B31.1 Stress Ratios and Class 1 Fatigue Usage Values at Select High Stress/High Usage Locations



Note: Original plant design considered 120 of these events, associated with each plant shutdown. A plant re-evaluation was conducted and showed that up to 318 of these events could potentially occur so this number was included in the current evaluation.

Figure 4-7. Temperature, Pressure and Flow History for the Thermal Transient Event(s) that Result in High Fatigue Usage Values for the RHR Return Line (Nodes 395 to 408)

Section 5

CONCLUSIONS

In this report, the NUMARC technical position on fatigue was examined relative to the assertion that the fatigue CLB for Class 1 piping system in early nuclear plants is adequate for license renewal, with few and very clearly-defined exceptions. In order to examine this assertion, the fundamental differences between nuclear plant piping designed to ANSI B31.1 and ASME Section III, Class 1 have been explored. Differences in the Code stress evaluation methodology have been described in detail. Review of early ASME nuclear Code Cases shows that some additional considerations beyond the ANSI B31.1 requirements for power piping were applied in the design of reactor coolant piping systems for some early plants. However, these additional considerations related to material selection and initial inspection requirements are unrelated to fatigue design analysis methodology.

To assess the acceptability of ANSI B31.1 piping designs for reactor coolant system piping in older plants, ASME Class 1 piping analysis was conducted for two typical piping systems, one each for a PWR and a BWR plant. In both cases, a system (or portion thereof) that is normally identified as having high fatigue usage was chosen for evaluation. In both cases, the analysis per the requirements of ANSI B31.1 showed that the system was acceptable. When evaluated to the requirements of ASME Section III, Class 1, only very limited areas of the evaluated systems were found to have high usage.

In each case, the few potential locations of high fatigue usage are easily identified once one understands the controlling parameters. These are:

1. High usage will occur only at locations experiencing significant thermal transients. In both systems, the controlling transients included significant step changes in boundary temperature due to on/off flow conditions. The fatigue usage was always relatively low in portions of the system that did not experience the step-change transients. However, severe flow stratification and local thermal cycling effects might also lead to similarly high fatigue usage.

2. Structural or material discontinuities are also always present at locations of high fatigue usage. These locations are typically associated with high stress indices that result in multiplication of the stresses relative to those that occur in most of the adjacent piping and fittings.

3. At locations such as pipe-to-valve welds or other changes of thickness or at bimetallic joints, the relative heatup/cooldown rates and thermal expansion of the adjacent structures contribute significantly to computed fatigue usage. At these locations, the secondary stress range can be greater than $3S_m$, requiring that simplified elastic plastic analysis be conducted with its accompanying amplifying effects on the peak stress range and fatigue usage.

4. At material discontinuities, the high secondary stress ranges can occur even for slow transients. When combined with rapid transients, the effect is further amplified since simplified elastic plastic analysis may have to be conducted.

On the other hand, the majority of locations in piping systems are not affected by the effects mentioned above. As demonstrated by the analysis conducted in support of this report, most locations have low fatigue usage. Most piping systems do not experience severe thermal transients since the heatup and cooldown rates are determined by requirements for the relatively thick reactor pressure vessel. These 100°F heatup and cooldown transients <u>never</u> contribute significantly to the fatigue usage of piping systems.

The evaluation also showed that the fatigue usage for Class 1 piping systems designed prior to about 1980 (most plants in the US today) is very conservative compared to the fatigue usage computed using the current version of the ASME Code for Class 1 Components.

Thus, it is concluded that piping systems designed to the requirements of ANSI B31.1 are adequate for continued service in nuclear plants. In the absence of stress risers (high stress indices or material discontinuties) and severe thermal transients, there is no reason to expect fatigue usage to approach unity in these systems. However, a limited number of regions that experience severe thermal transients and contain structural or material discontinuities may indicate high fatigue usage when evaluated by conventional Class 1 piping methods. For these few easily-identifiable locations, more sophisticated analysis methods or considerations of actual (as compared to design) transients can probably be used to show that fatigue usage will not exceed that allowed by the ASME Section III, Class 1 requirements for piping systems.

These evaluations support the NUMARC technical position that the CLB for fatigue is adequate for piping constructed to the requirements of ANSI B31.1 and its predecessor standards, except for the few specific locations associated with geometric and loading discontinuities. Evaluation of these few potentially fatigue-sensitive locations can be used as a technical basis for justifying that the systems are acceptable for an extended license renewal term.

Section 6

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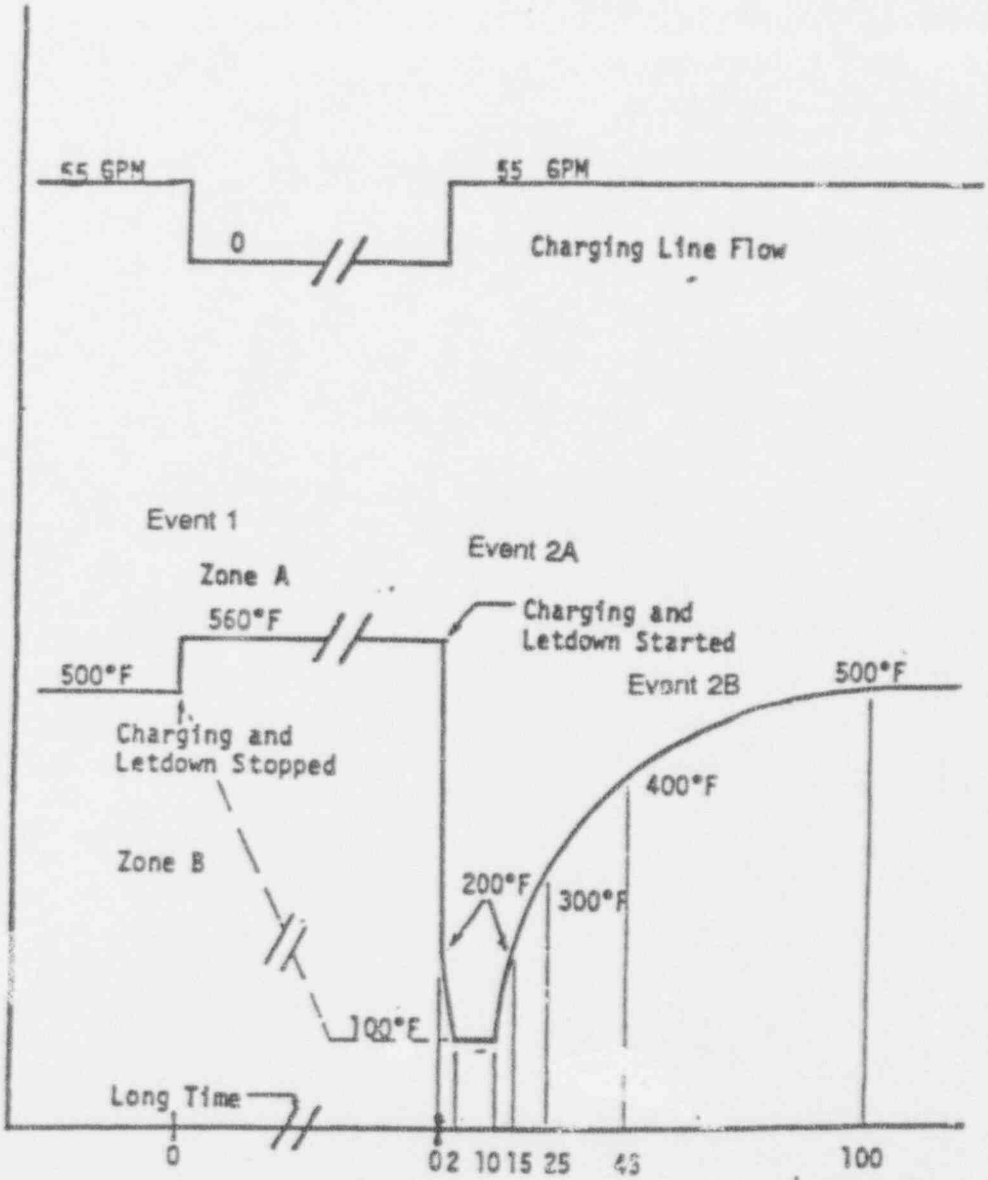
APPENDIX A

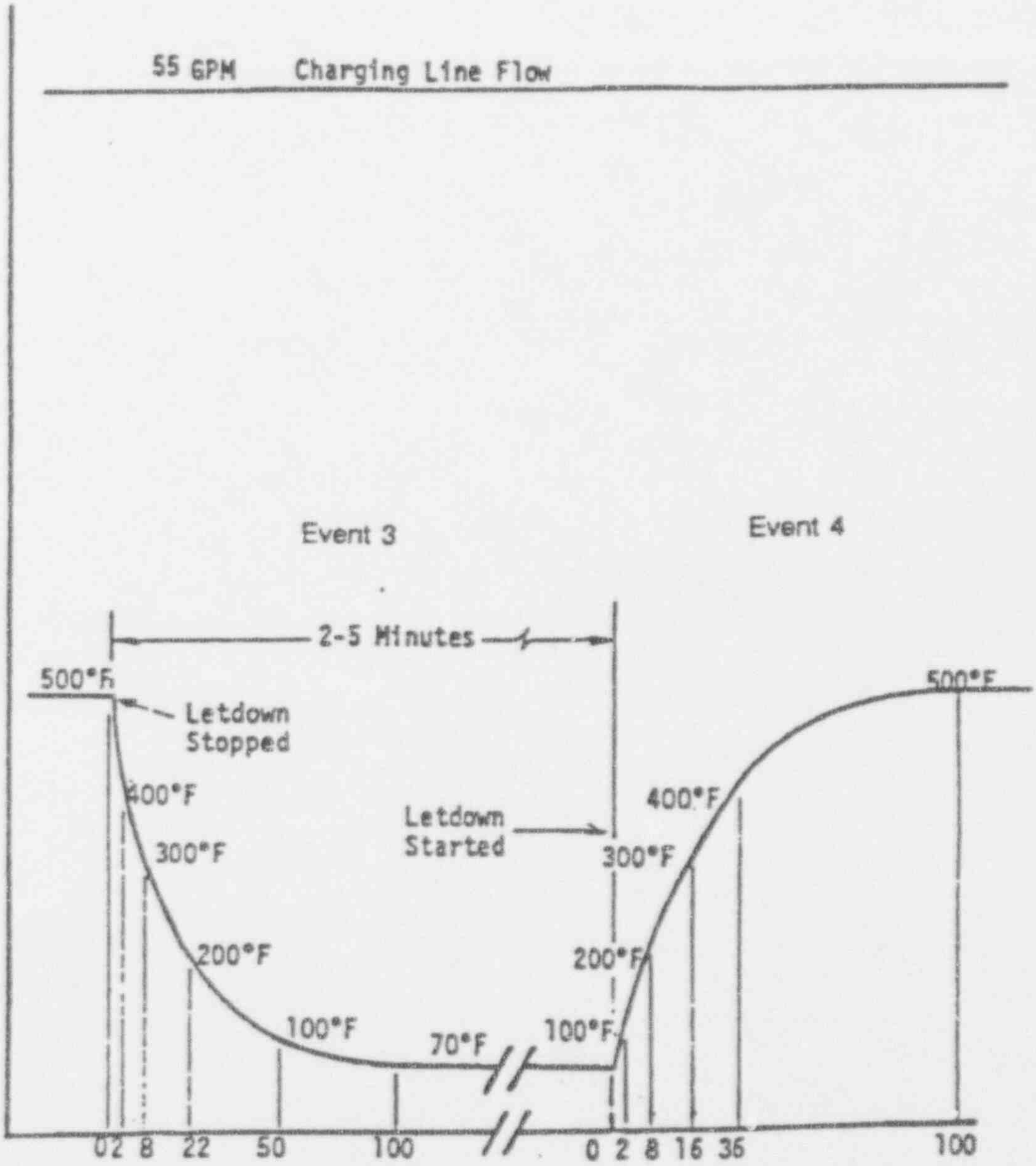
PWR CHARGING LINE
TEMPERATURE AND CHARGING FLOW VERSUS TIME
FOR VARIOUS TRANSIENT EVENTS



Time (Seconds)

Normal Charging and Letdown Shutoff
And Return To Service

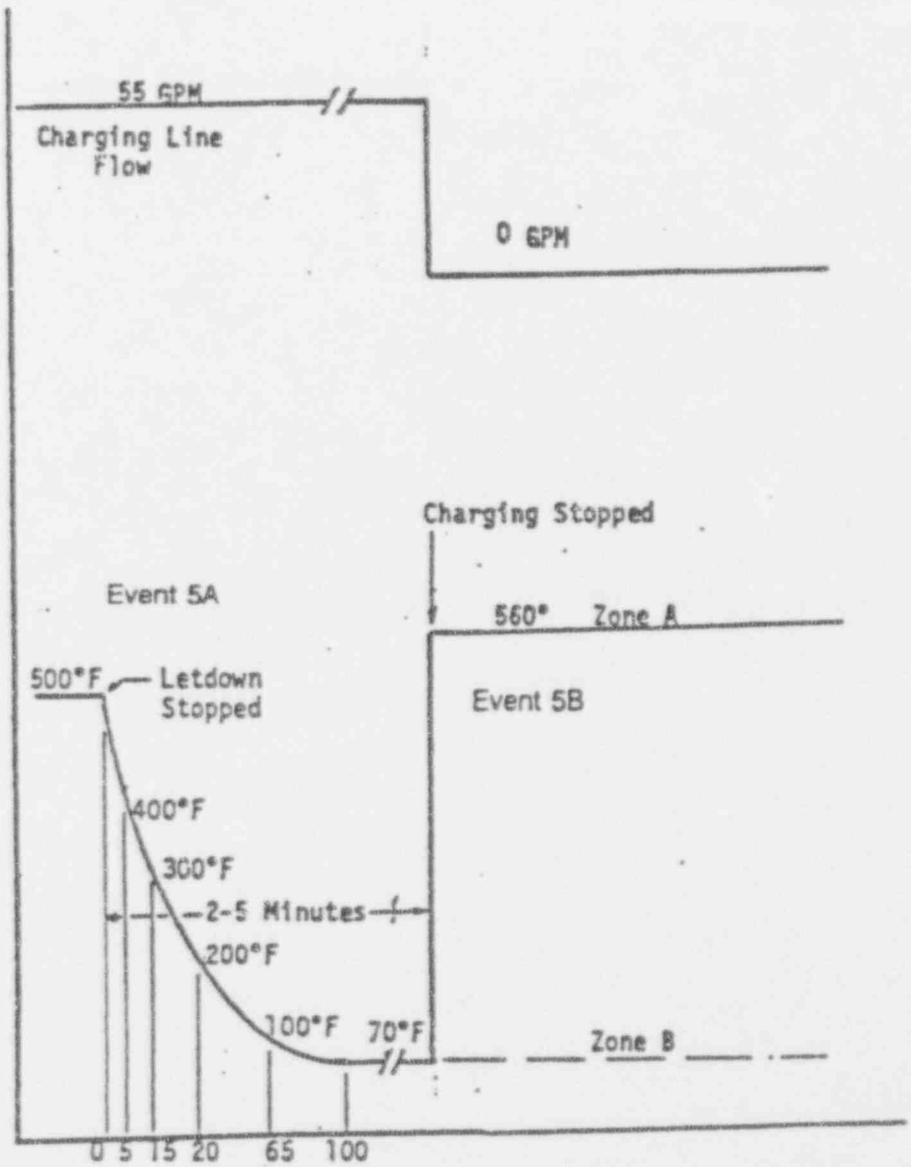
(60 Cycles)



Time (Seconds)

Letdown Trip With Prompt Return To Service

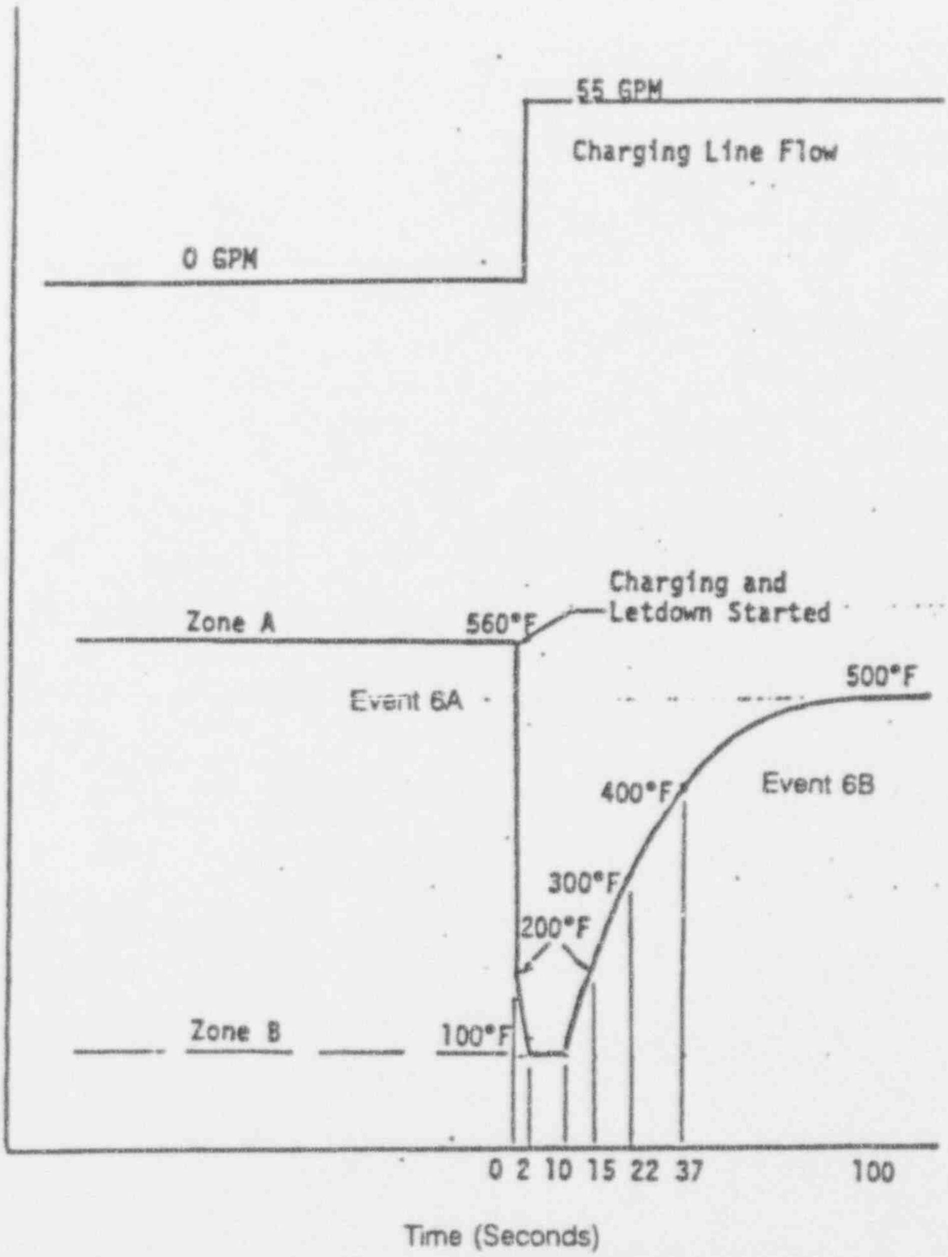
(200 Cycles)



Time (Seconds)

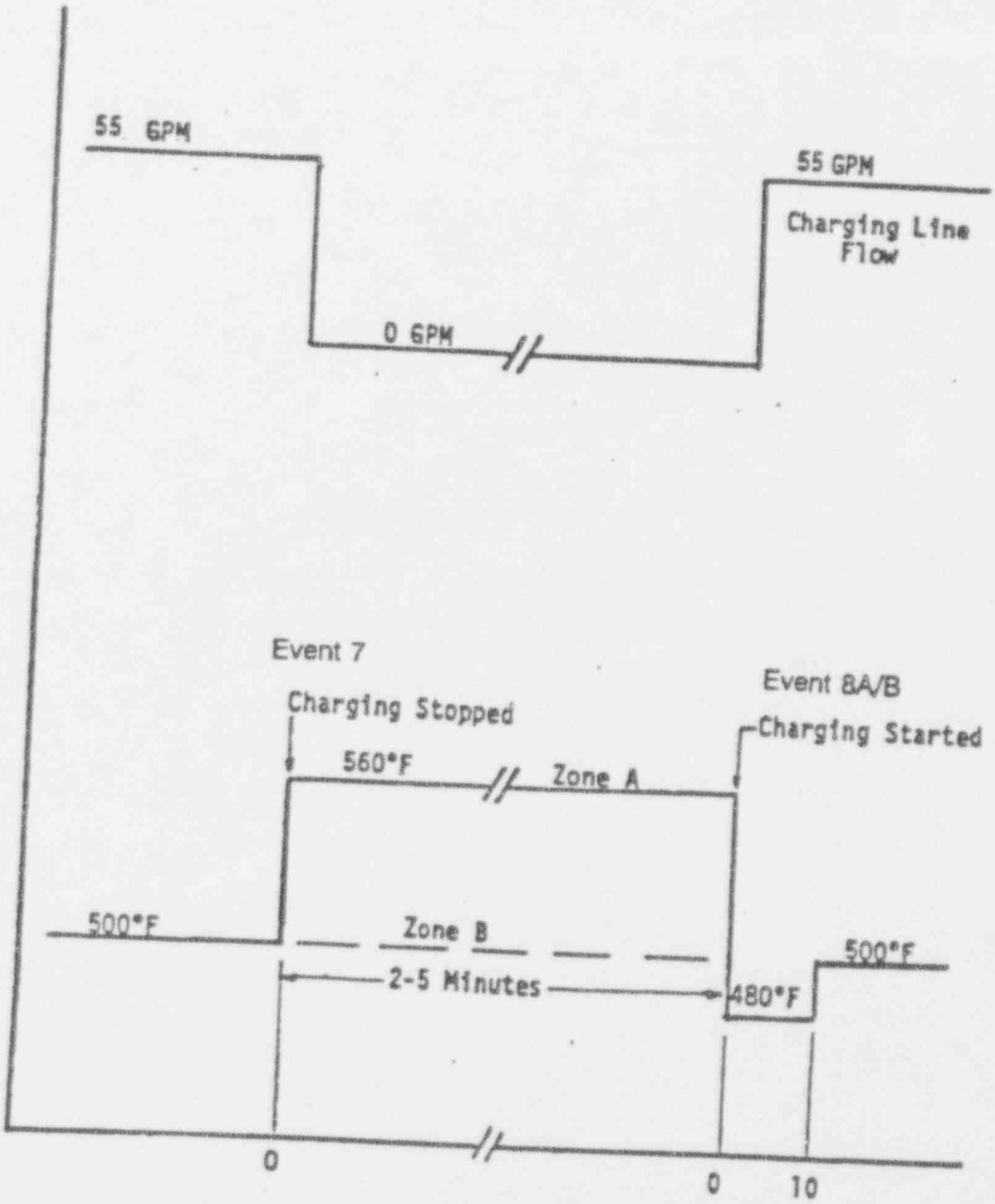
Letdown Trip With Delayed Return To Service
Letdown Trip Portion

(20 Cycles)



Letdown Trip With Delayed Return To Service
Return Portion

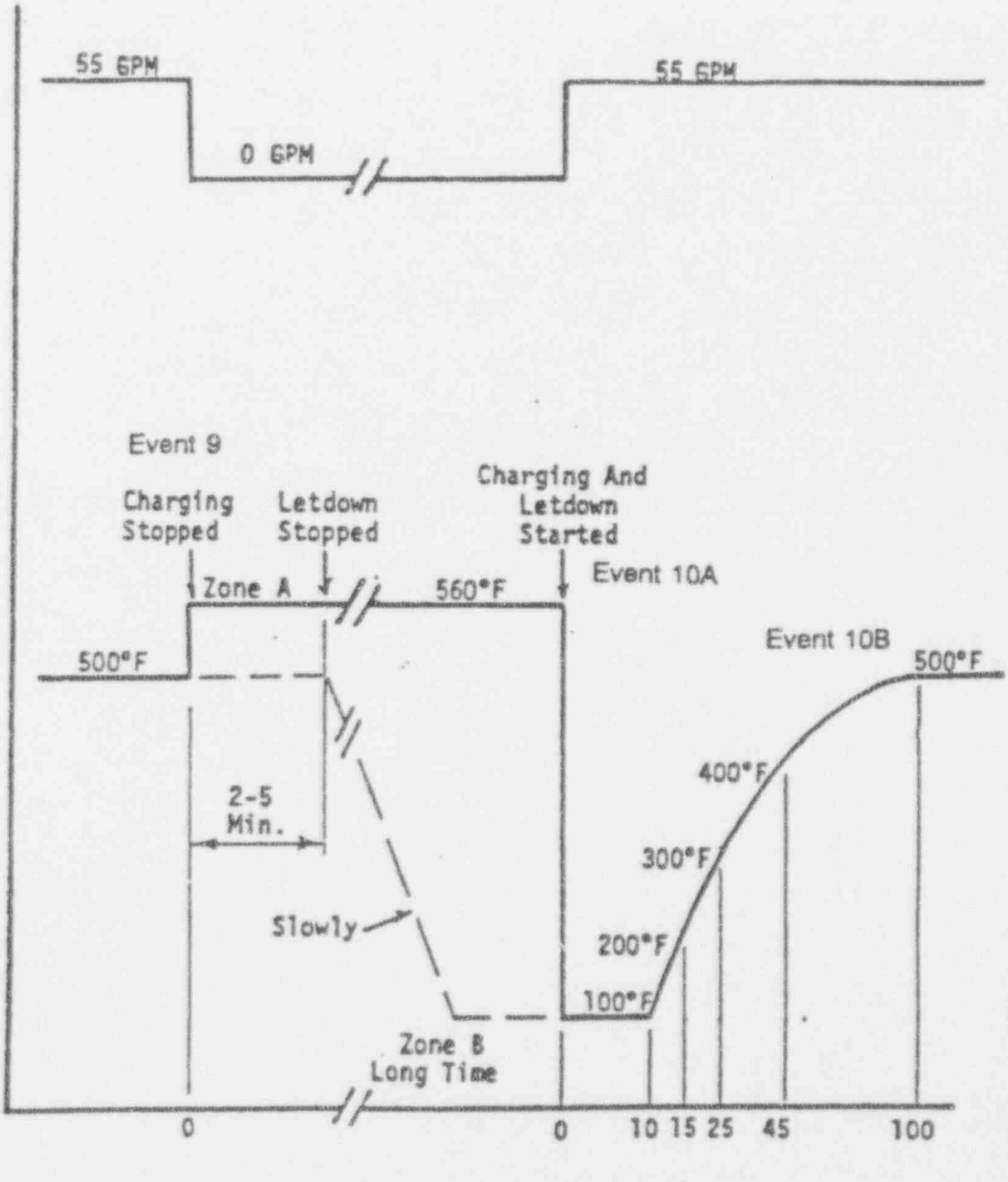
(20 Cycles)



Time (Seconds)

Charging Trip With Prompt Return To Service

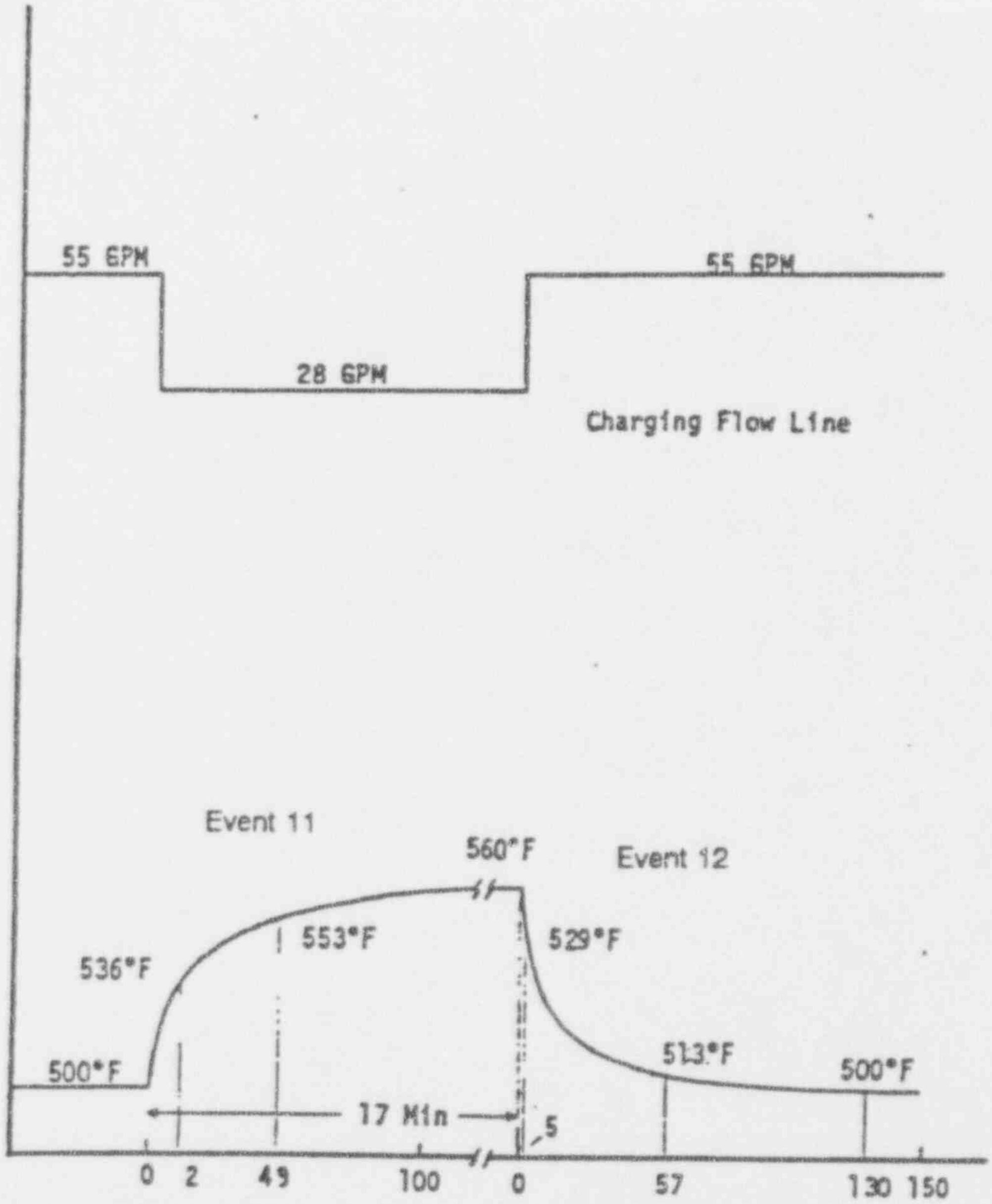
(200 Cycles)



Time (Seconds)

Charging Trip With Delayed Return To Service

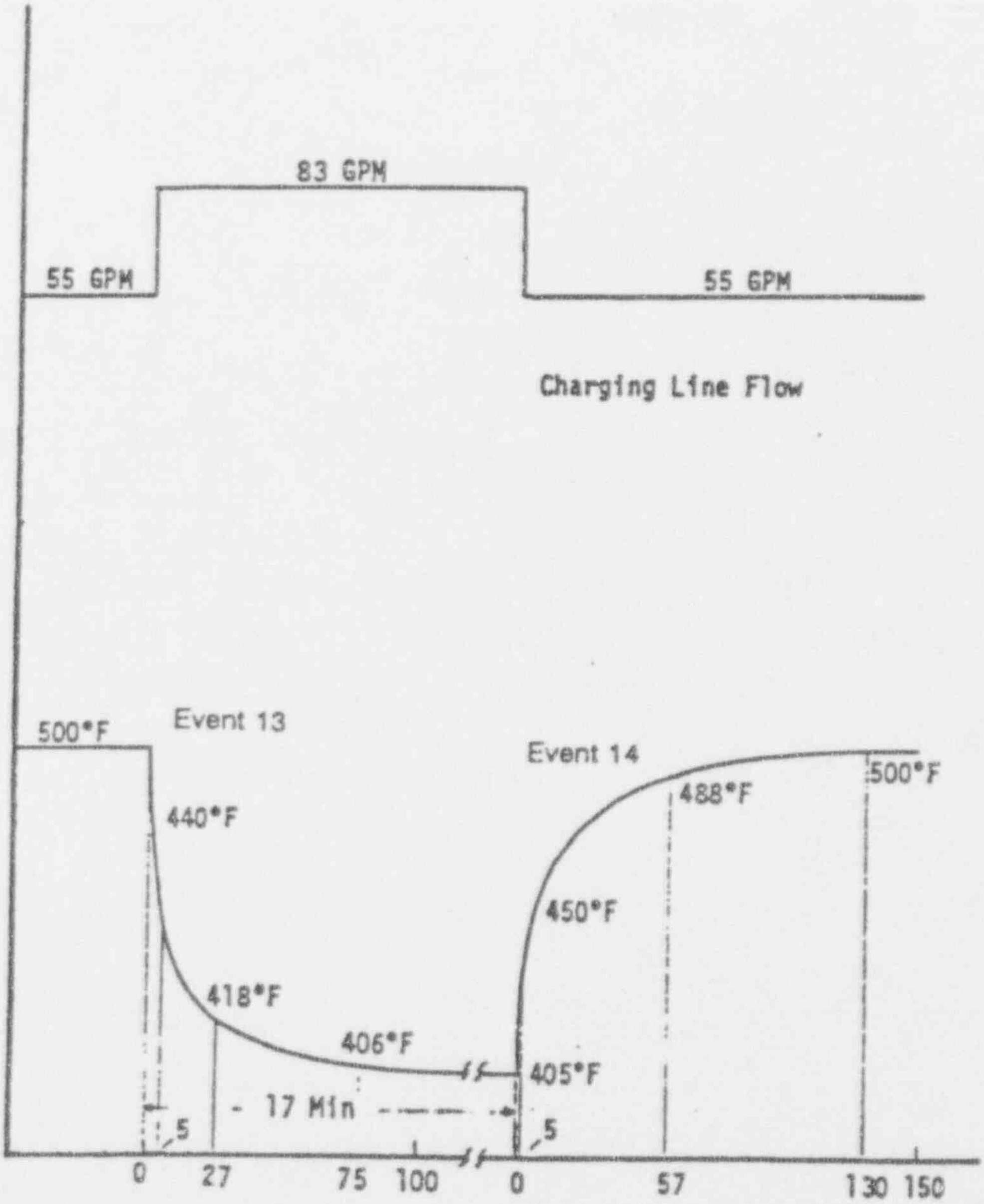
(20 Cycles)



Time (Seconds)

Charging Flow 50% Step Decrease And Return

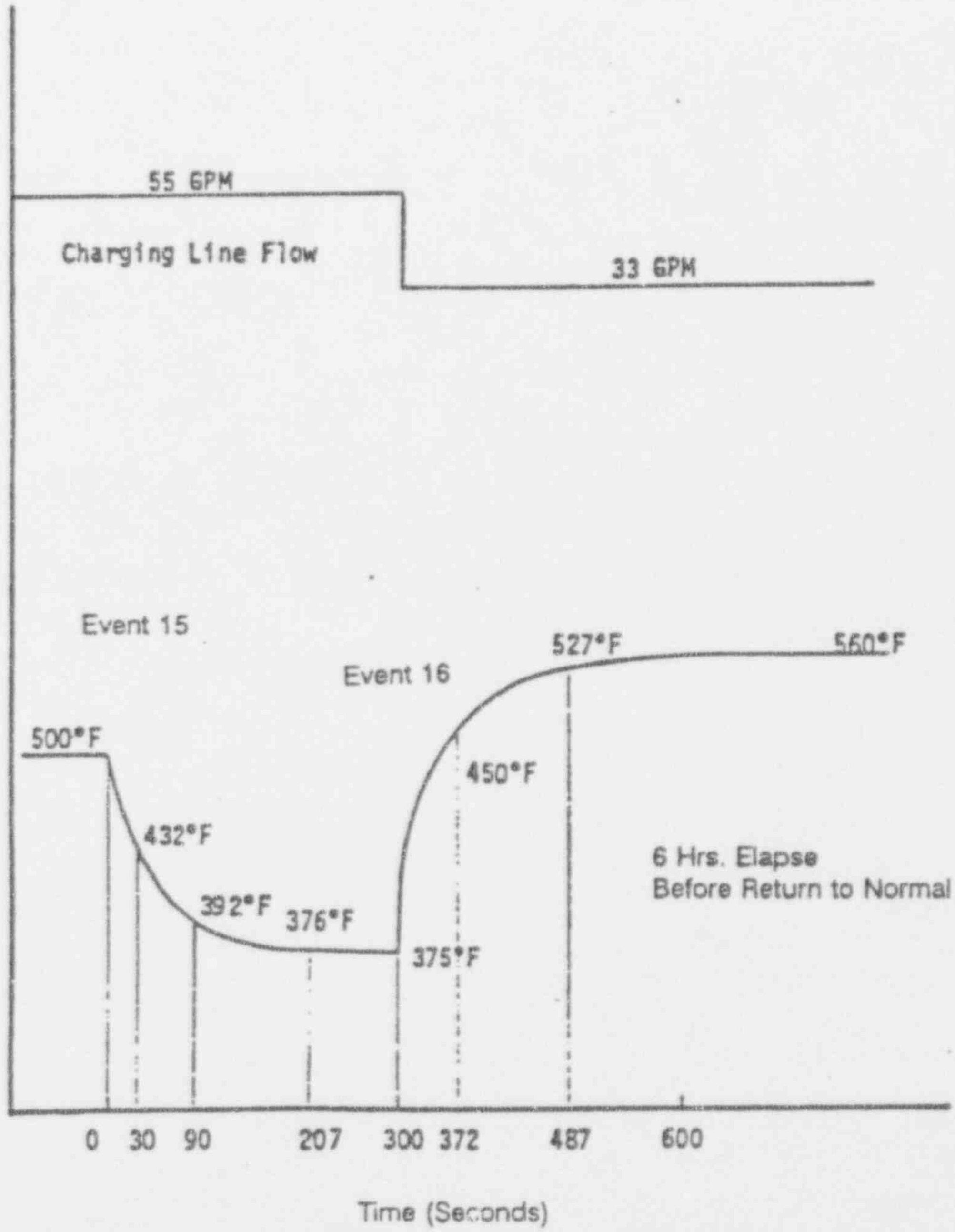
(24,000 Cycles)



Time (Seconds)

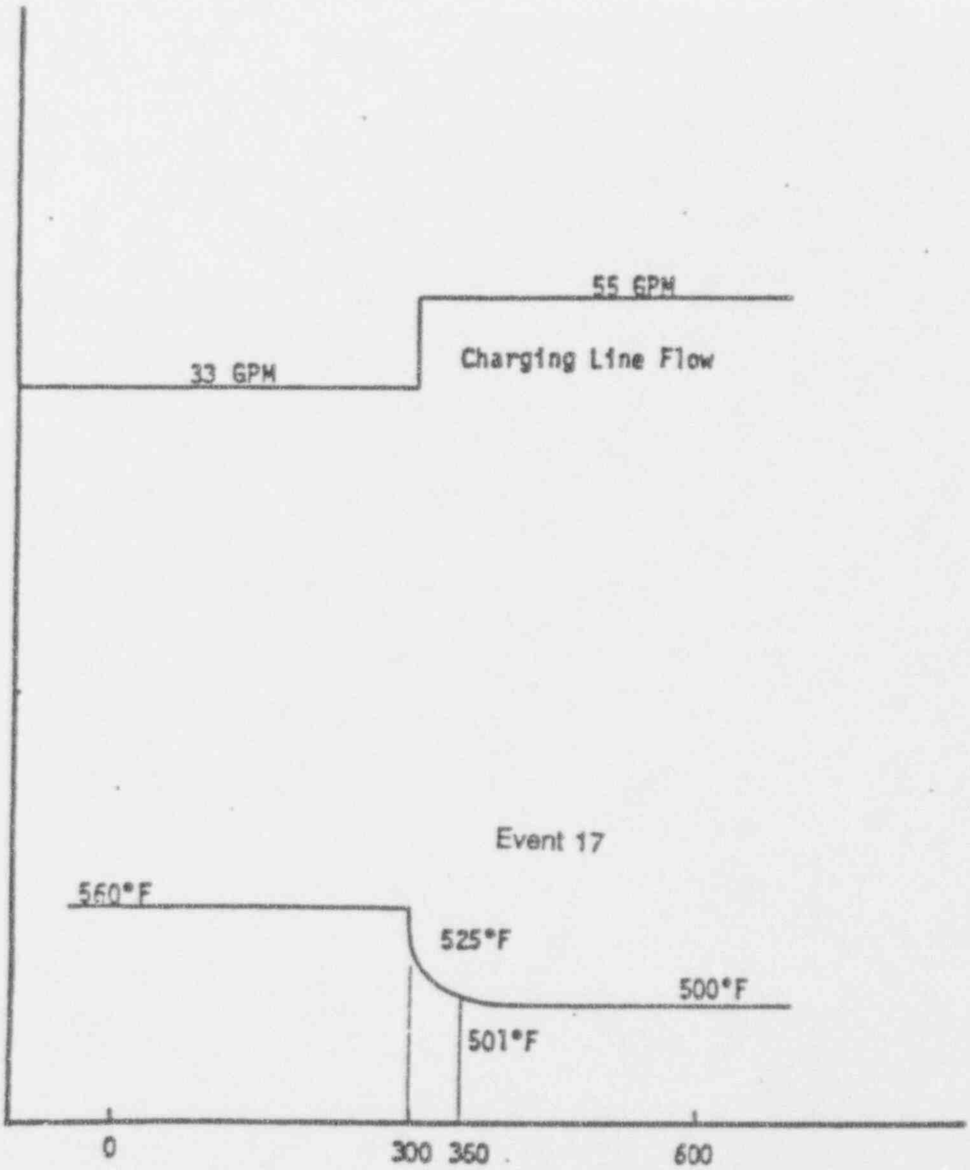
Charging Flow 50% Step Increase And Return

(24,000 Cycles)



Letdown Flow 40% Step Decrease

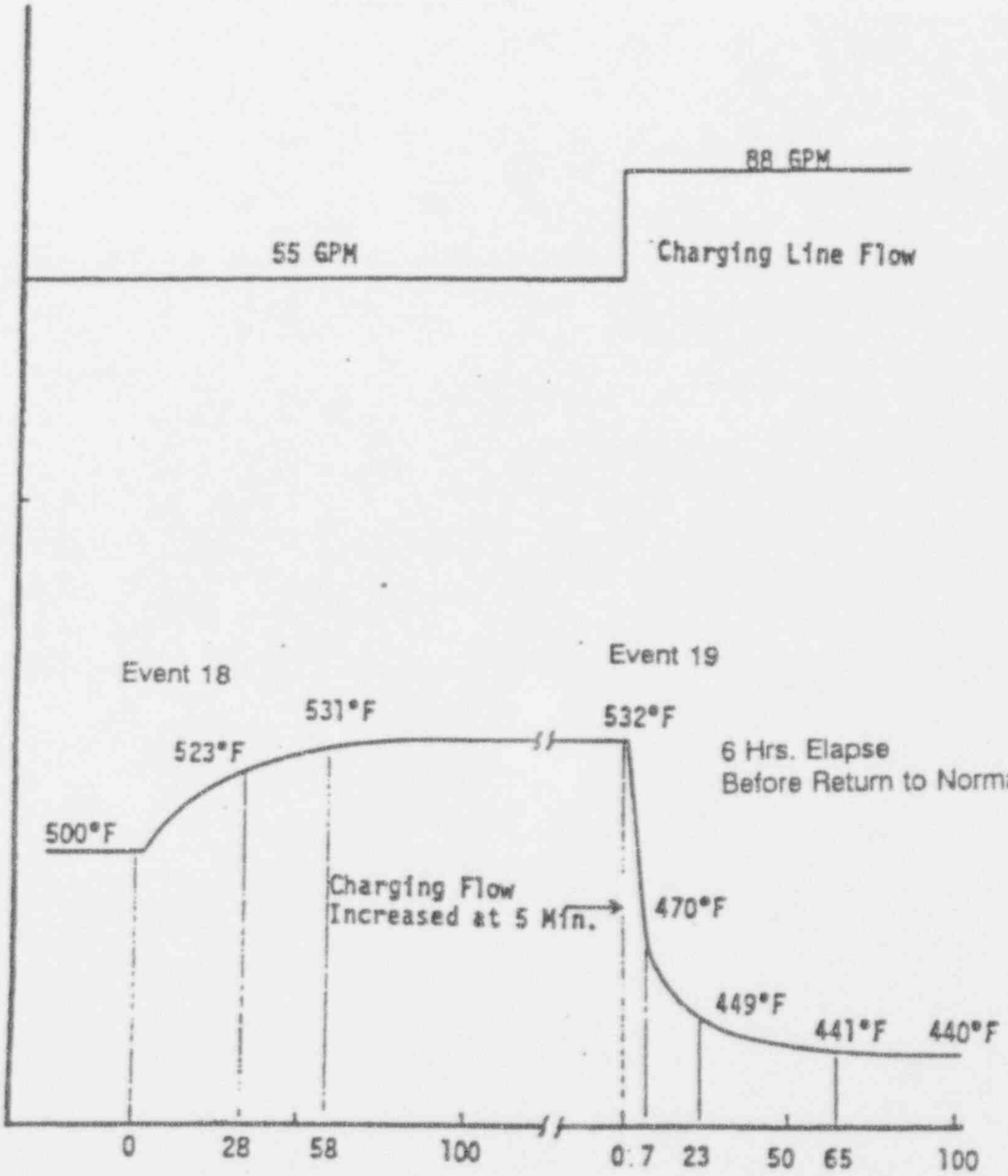
(2,000 Cycles)



Time (Seconds)

Letdown Flow 40% Step Decrease
Return to Normal

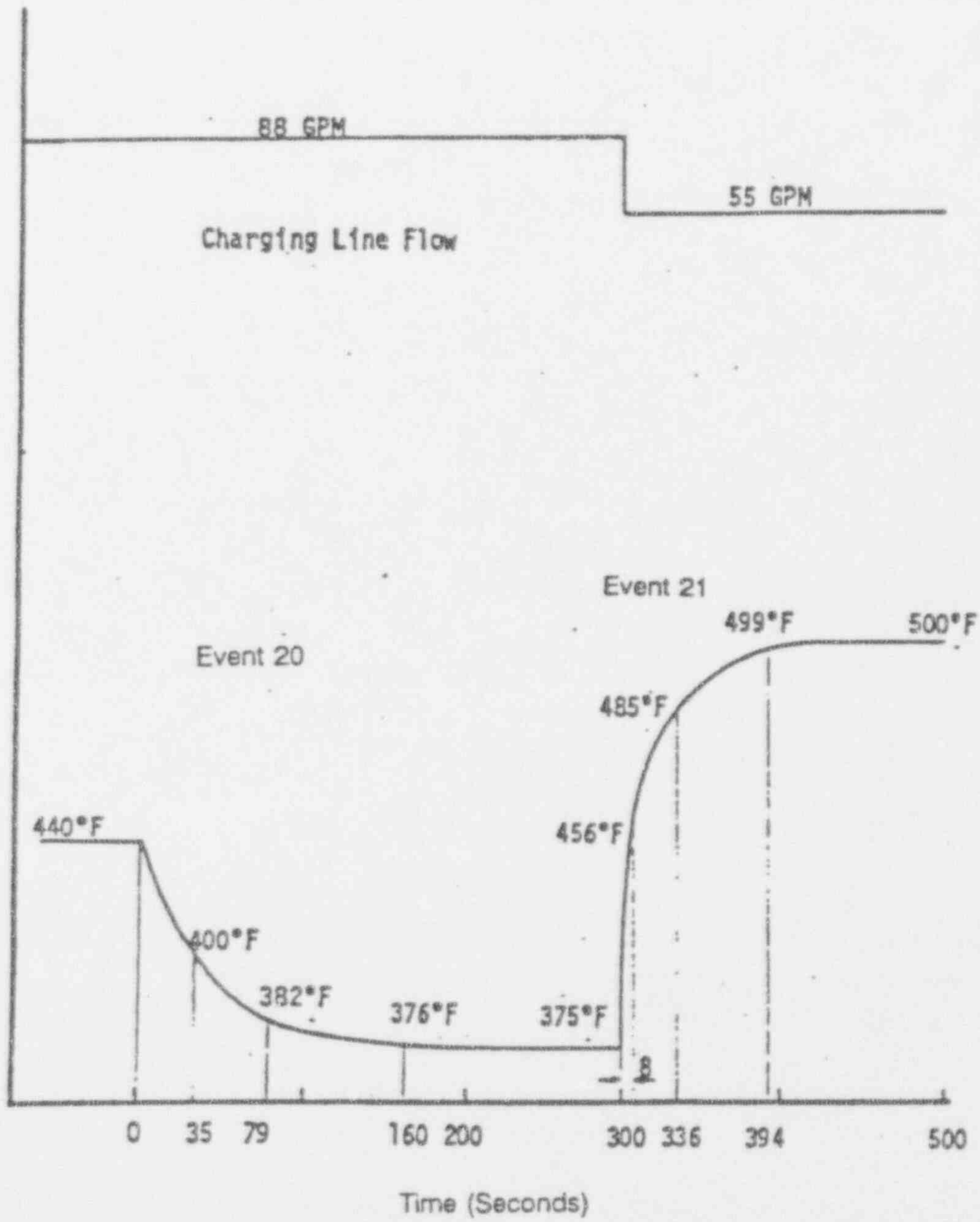
(2,000 Cycles)



Time (Seconds)

Letdown Flow 60% Step Decrease

(24,000 Cycles)



Letdown Flow 60% Step Increase
Return to Normal

(24,000 Cycles)

APPENDIX B

BWR RECIRCULATION SYSTEM
LOAD HISTOGRAMS FOR TRANSIENT THERMAL EVENTS

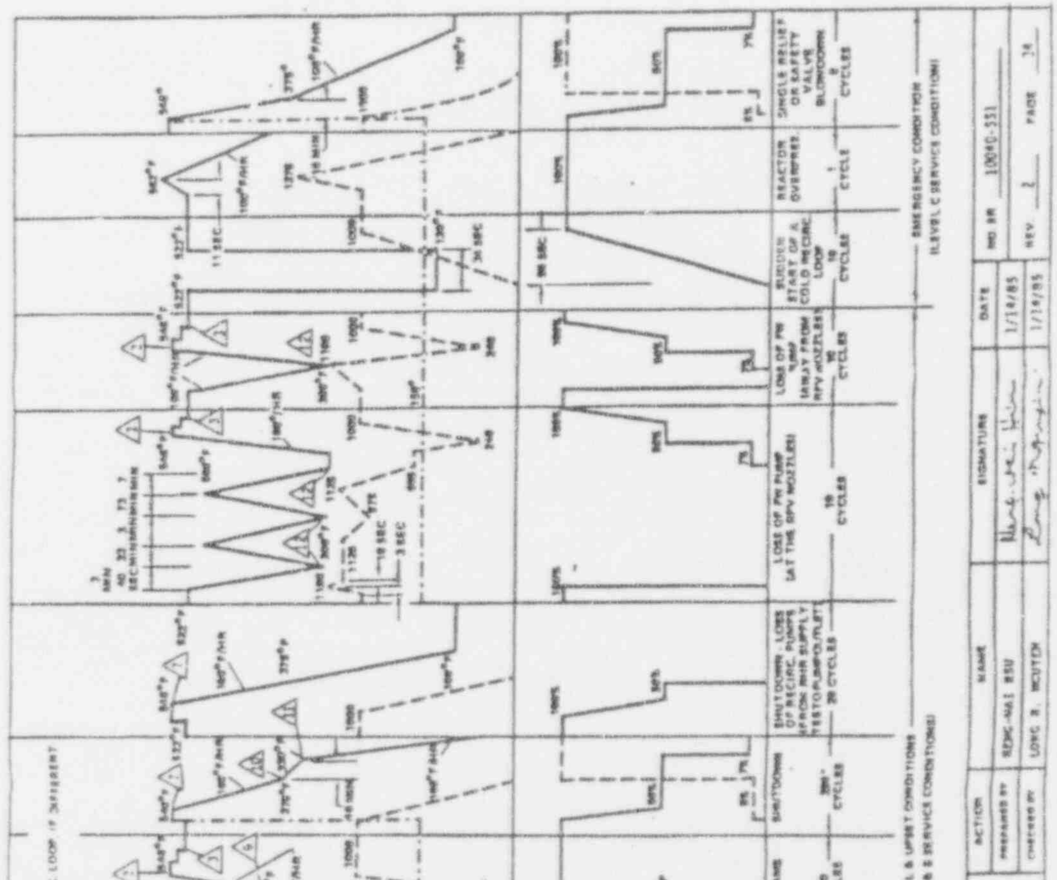


FIGURE B.1 LOAD HISTOGRAM FOR RECIRCULATION SUCTION AND RHR SUPPLY

ACTION	NAME	SIGNATURE	DATE
PREPARED BY	RENG-HAI RSU	[signature]	1/14/85
CHECKED BY	LONG B. NGUYEN	[signature]	1/14/85

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