TENNESSEE VALLEY AUTHORITY

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5N 157B Lookout Place

FEB 18 1988

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - FEEDWATER (FW) CHECK VALVE CLOSURE FOLLOWING PIPE RUPTURE - RESPONSE TO INSPECTION REPORT 87-27, OBSERVATION MEB-3

References: 1. NRC letter to TVA dated August 24, 1987, "Sequoyah Design Calculation Review: Report Nos. 50-327/87-27 and 50-328/87-27"

2. TVA letter to NRC dated May 16, 1979, "Sequoyah Nuclear Plant Units 1 and 2 - Potential Excessive Water Hammer Forces in the Main Feedwater System - NCR MEB 79-1 - Final Report"

3. TVA letter to NRC dated October 10, 1979, "Sequoyah Nuclear Plant Units 1 and 2 - Potential Excessive Water Hammer Forces in the Main Feedwater System - NCR MEB 79-1 - Revised Final Report"

4. NRC letter to TVA dated November 16, 1979, Inspection Report 50-327/79-63 and 50-328/79-31

The purpose of this submittal is to discuss the FW water hammer issue as it relates to SQN units 1 and 2. Initially addressed in the 1979-1980 timeframe and formally dispositioned, this item has again been raised and requires subsequent discussion. Most recently, NRC Inspection Report Nos. 50-327/87-27 and 50-328/87-27 (reference 1) concluded that the SQN water hammer issue (Observation No. MEB-3) "remains open pending (a) CEB's documented evaluation of the main feedwater system at Sequoyah Nuclear Plant with respect to the postulated water hammer forces and (b) TVA's justification for not issuing the feedwater water hammer analysis when it was identified by engineering as a licensing commitment" at other TVA facilities. The enclosure contains a response to the issues raised by NRC in 50-327/87-27 and 50-328/87-27 regarding Observation No. MEB-3.

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The original SQN water hammer analysis, performed in 1979, demonstrated the integrity of check valves and piping under the most severe FW line break postulated. Subsequent review by NRC in 1979 closed the issue for SQN. Recently, the effects of a postulated rupture of the main FW header were again reviewed by TVA. This review, based upon analysis and current industry status, has concluded that further actions are not warranted and that plant safety is not impaired.

If you have any questions, please telephone M. R. Harding at (615) 870-6422.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R. Gridley

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ENCLOSURE

On January 4, 1979, the Mechanical Engineering Branch (MEB) prepared nonconformance report (NCR) MEB-79-1 to indicate that TVA may not have properly considered main FW system water hammer at SQN and that other TVA plants might be involved. The corrective action, which the Civil Engineering Branch (CEB) detailed on the NCR, specified completion of an analysis of the main FW system for water hammer energy and an evaluation of the main FW check valve with respect to its ability to withstand the calculated closing energy assuming a postulated main FW line break upstream of the check valve.

On May 16, 1979, TVA provided NRC with a report on the disposition of NCR MEB-79-1 (reference 2), which indicated that MEB had evaluated the main FW check valves for the water hammer transient and that the main FW check valves would maintain their function and integrity following the most severe main FW line break postulated.

At a subsequent meeting with C. R. McFarland of NRC Office of Inspection and Enforcement (OIE), TVA was told that assurance of main FW piping integrity would be required in addition to the assurance of FW check valve integrity before closure of open 10 CFR 50.55(e) item (327/79-12-09; 328/79-07-09).

TVA submitted revision 1 to the main FW line break report (MEB 79-1), dated October 10, 1979 (reference 3). This revision analyzed the ability of the main FW check valves and associated piping to remain functional following the most severe postulated main FW line break. The results of these analyses concluded that the main FW line check valves and associated piping will maintain their integrity and remain functional under all expected accident conditions.

It is noted that the analytical methods used by TVA were consistent with the design bases for other plants of the SQN vintage. Similarly, the industry's design bases reflected the regulatory requirements of the 1979-1980 time period.

Subsequent NRC review of this issue during an inspection on October 30, 1979, to November 1, 1979, dispositioned this item as closed. The check valve integrity analysis and the hoop stress analysis of MEB-79-1, revision 1, were the bases for NRC (OIE) closure of the 10 CFR 50.55(e) items related to this matter. Documentation of closure is contained in NRC IE Inspection Report Nos. 50-327/79-63 and 50-328/79-31 (reference 4).

Analytical advances in the evaluation of water hammer transients for FW system piping did not occur until after 1980, the year SQN received its operating license. By 1983, accurate forcing functions were available for the anticipated FW system piping breaks. This data was generated by TVA for the apparent purpose of applying the current methodology to SQN despite the fact that SQN NCR MEB 79-1 had already been resolved with NRC.

Watts Bar Nuclear Plant (WBN), having not yet received its operating license, proceeded to reanalyze its main FW system for water hammer loads using forcing functions available in 1983. This analysis yielded snubber design loads greater than the original snubber seismic loads. SQN, on the other hand, determined that a similar detailed structural response analysis was not required. An engineering evaluation of the pipe whip steel downstream of the check valves indicated that there is as much steel in the available space as is practical to install, consistent with access for inspection and maintenance. At each restraint, the gap provided was judged to be the true minimum practical consistent with pipe shakedown, unanticipated thermal expansion, insulation, etc. In the unlikely event of this severe accident in the FW system, pipe movement downstream of the check valves would therefore be limited to the pipe support gap. Accordingly, TVA concluded that the FW piping at SQN was adequately designed and did not reanalyze this condition or fund continued evaluation efforts.

The dispositioning of this issue at SQN can be further justified by the following.

SAFETY ASSESSMENT OF FW CHECK VALVE CLOSURE TRANSIENT FOLLOWING PIPE RUPTURE

1. ANALYSIS

- Event required to cause check valve "slam" transient is instantaneous (i.e., 1 millisecond) double-ended guillotine FW pipe rupture in the main FW header. A simplified sketch of the main FW system is shown in attachment 1.

- Initiating causes of such a traumatic failure include a seismic event, material fatigue because of thermal cycling, material failure because of erosion/corrosion, and fluid transients in the FW system.

- Seismic Event - Investigations into the behavior of piping and pipe support systems subjected to substantially higher ground response than SQN site have demonstrated high survivability with regard to pressure boundary integrity and overall system reliability.

- Material Fatigue - Allowable stresses in piping analysis codes are based on 7000 cycles, a figure considered in itself to be conservative. Because of relatively short operating life (approximate 4 years) in comparison with overall plant life (40 years), Sequoyah has utilized only an estimated 700 cycles to date.

- Erosion/Corrosion - Not anticipated because of high water quality and low dissolved oxygen content. Also, system subcooling during operation avoids flashing of FW to steam, which can otherwise contribute to erosion in FW piping. Continued monitoring for pipe wall thinning in FW/condensate piping will be provided by the recently issued Surveillance Instruction (SI)-733. Design study of single-phase flow erosion/corrosion is in preparation stages for SQN systems considered susceptible to these physical/chemical phenomena.

- Fluid Transients - Water hammer events resulting in FW piping and support damage have occurred at other operating facilities; however, none have caused longitudinal splits, full separations, or guillotine ruptures in regions of the system pertinent to check valve closure analysis. Failures have occurred on the inlet side of FW pump with significantly less severe consequences because of the flow-limiting effect of the pump and its impeller.

SQN has performed mechanical improvement modifications to the unit 1 and unit 2 steam generators to mitigate the effects of water hammer events caused by abnormal steam generator conditions (e.g., draining of steam generator level below the elevation of the feed ring). Most notably, modifications have included plugging of original bottom flow holes and installation of "J-Tubes" on the feed rings. No water hammer was observed in a special test conducted by TVA and witnessed by NRC following this modification.

In the industry in general, water hammer events initiated in steam generators have resulted in damage that, for the most part, has been limited to pipe supports and has not involved complete separation in piping or significant release of fluid. Even in cases where main FW pressure integrity was lost (e.g., because of valve damage), the ability to achieve cold shutdown has never been compromised by a FW fluid transient and subsequent water hammer.

- SQN FW pipe rupture restraints will prevent uncontrolled pipe movements should a water hammer event occur that causes deformation of the conventional system pipe supports. The dampening effect of the rupture restraints will in turn minimize the translation of forces from the break location to the FW check valves, and thus will further lessen the potential for damage to the check valves.

- SQN water hammer analysis, originally performed in 1979, already demonstrated the integrity of check valves and piping under the most severe FW line break postulated. Subsequent review by NRC in 1979 closed the issue for SQN.

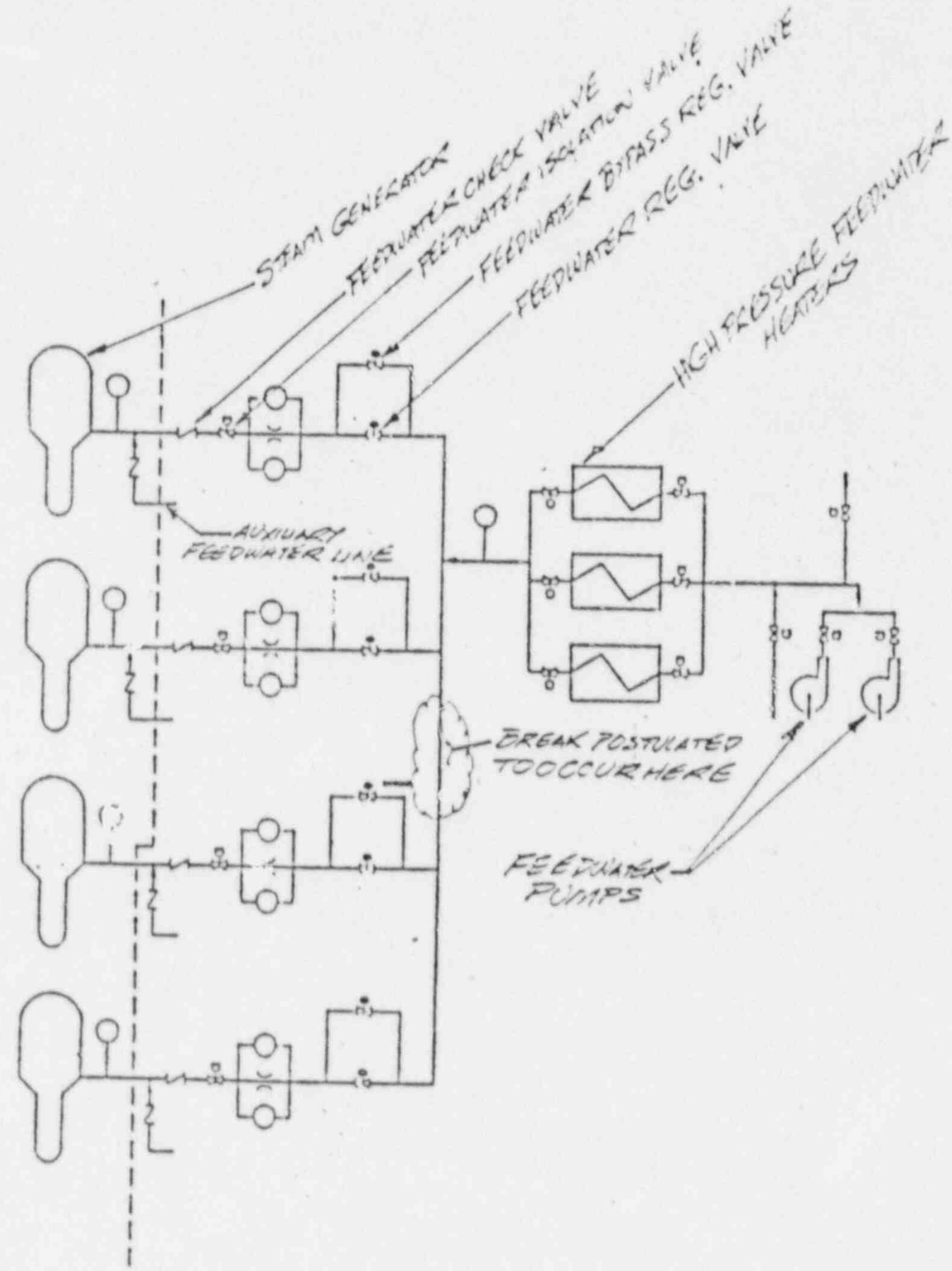
2. CURRENT INDUSTRY STATUS

- Generic issue (to postulate water hammer-induced instantaneous guillotine rupture and reanalyze/redesign all affected piping and pipe supports) has not been addressed by all nuclear utilities for similar vintage plants.

- Water hammer events have occurred but have not posed a significant safety issue in terms of functional capability of the FW system or safe shutdown of units involved. No instantaneous guillotine ruptures, of the type postulated in the check valve "slam" analysis, have occurred.

- Nuclear, fossil, and petrochemical industry experience indicates that actual pipe ruptures do not impair pressure boundary integrity remote from the break. Rather, plastic hinges have formed in localized piping and isolated supports have failed, but pressure boundary integrity remote from the break has remained intact. This experience provides strong indication that FW pipe rupture restraints and steam generator nozzles would be more than sufficient to mitigate the postulated pipe break.

Attachment 1



3.9.2 Safety Class B, C, and D Fluid Components[a]

3.9.2.1 Plant Conditions and Design Loading Combinations

Design pressures, temperatures, and other information that provide the basis for the design of safety-related systems and components are presented in the corresponding sections that describe the system functional requirements. Codes that govern the analysis of vessels, tanks, valves, pumps, and piping are defined in Table 3.9.2-1. Environmental equipment such as ventilation and air treatment components and other equipment which is safety related but which have no applicable code for design are defined in Subsection 3.9.3.

3.9.2.2 Design Loading Combinations

The design loading combinations and the allowable stress intensity levels considered in the component or system design for TVA safety classes B, C, and D are shown in Table 3.9.2-2. Design loading combinations are categorized with respect to normal, upset, and faulted conditions. The categories are defined as follows:

1. Normal Conditions   Any condition in the course of system startup, operation in the design power range, and system shutdown, in the absence of upset, faulted, and test conditions.

2. Upset Conditions   Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients* which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, transients due to loss of load or power, and any system upset not

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[a]Includes Class A piping (Reactor Coolant Loop Branch Lines) analyzed by TVA.

resulting in a forced outage. The upset condition includes the effects of a one-half safe shutdown earthquake for which the system must remain operational or must regain its operational status.

3. Faulted Conditions Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that considerations of public health and safety are involved. The faulted condition includes the effects of the safe shutdown earthquake and the dynamic effects of postulated pipe rupture considered as separate events (see Table 3.9.2-2).

A summary of the loading conditions and stress limits for safety class B, C, and D fluid components appears in Table 3.9.2-3.

3.9.2.3 Inelastic Deformation

The proposed stress limits for the faulted condition for groups B, C, and D, components are within the code allowable for primary loads. Consequently, functional and structural integrity are assured for the faulted condition.

3.9.2.4 Design and Installation Criteria, Pressure Relieving Devices

The design and installation of pressure relieving devices are consistent with the requirements established by Regulatory Guide 1.67, 'Installation of Overpressure Protective Devices.'

Each main steam line is provided with one (1) power operated atmospheric relief valve and five (5) safety valves sized in accordance with ASME, B&PV, Section III, 1968 edition.

The safety valves are set for progressive relief in intermediate steps of pressure within the allowed range (105 percent of the design pressure) of pressure settings to prevent more than one valve actuating simultaneously. The valve pressure settings at which the individual valves open are shown in the table below in the column identified as "Set Pressure." The valves are designed to reset at the pressure levels identified in the column "Blowdown Pressure." Credit for the pressure drop (21 psi) between the steam generator outlet and the valve inlet was taken when establishing the "Set Pressure" valves.

TVA Class B, C, D, and Non-Nuclear Safety Piping

TVA has evaluated the necessity of performing a complete analysis on all piping systems and identified the limits of the analysis using the following guidelines:

1. Analyze all TVA Class B and C lines 6-inch diameter and larger.

2. Analyze all piping in Category I structures larger than 1 inch diameter that has an operating temperature of 200°F or greater and an operating pressure of 275 psig or greater, or any piping where thermal movements might create support conditions outside the scope of the alternate criteria.

3. Analyze all piping whose loads could influence the operation of equipment essential to safe shutdown.[3]

4. Analyze all branch lines, outside the scope of these requirements, whose failure would interrupt normal function of an essential system. This analysis is to extend a sufficient distance from the essential system to ensure its safety and to a convenient termination point.

5. All piping rigidly attached to the steel containment vessel specifically piping which passes through the containment structure rigid penetrations or is supported off the containment structure, or both. 3

6. All systems requiring seismic qualification, but not requiring complete analysis as outlined above, will be evaluated according to the procedures outlined in Paragraph 3.9.2.6.

The following systems are within the scope outlined above, and are being completely analyzed for thermal, seismic, seismic anchor movement, and deadweight conditions:

1. Main steam system

2. Main steam blowdown system

3. Feedwater system

4. Auxiliary feedwter system

5. Chemical and volume control system

6. Safety injection system

7. Containment spray system 3

[3]Some Class G, K, and M piping is analyzed in this category.

8. Residual heat removal system

9. Component cooling system

10. Essential raw cooling water

11. Auxiliary boiler piping

12. Upper head injection piping

13. Parts of other systems which require rigorous analysis.

3.9.2.5.2 Analytical Methods

Loading Conditions and Stress Limits

not 3.9.2-2

The design loading combinations and the allowable stress limits considered in the design of TVA piping systems within the scope of Subparagraph 3.9.2.5.1 are shown in Table 3.9.2-5. Design loading combinations are categorized with respect to normal, upset, and faulted conditions. Piping components have been designed to allowable stress intensity levels given by the ANSI B31.1-1967 Power Piping Code.

While the referenced code did not define stress levels for the loading combinations considered in Table 3.9.2-5, the allowable stress intensity levels are in agreement with subsection NC3000 of the ASME, Section III, Winter 1972 Addenda. The referenced subsection is considered to be equivalent to ANSI B31.1 with appropriate consideration to the modifications where they exist.

Analyses

1. Stress evaluations due to loadings such as deadweight, thermal expansion, and anchor movements are performed using static analysis techniques, while stress evaluations due to earthquake loadings are performed using dynamic analysis techniques. The computer programming for application of both techniques is described in Subparagraph 3.9.2.5.3.

2. Loads on equipment nozzles are combined and evaluated against allowables as follows:

$$F_{DL} + F_{ST} + F_{1/2SSE} \leq \text{Allowable}$$

3. Seismic valve accelerations are generaly maintained below 2 g vertical, and 3 g horizontal. Cases exist such that valve accelerations can exceed these standard limits. Such cases are evaluated and approved individually; this process is controlled by the Rigorous Piping Analysis Handbook 4

Table 3.9.2-2

DESIGN LOADING COMBINATIONS

FOR GROUP CLASSES B, C, AND D COMPONENTS AND SUPPORTS

Loading Cases	Operating Condition
a. Pressure + deadweight + thermal	Normal
b. Pressure + deadweight + thermal + 1/2 SSE	Upset
c. Pressure + deadweight + SSE or Pressure + deadweight + pipe rupture loads	Faulted

Table 3.9.2-5

CATEGORY I PIPING SYSTEMS

LOADING CONDITIONS AND STRESS LIMITS

Plant Condition	Loading Combinations	Stress Limits	NC-3652 Equations
Normal	$S_{LP}+S_{DL}$	S_h	8
	S_{ST}	S_A	10
	$S_{LP}+S_{DL}+S_{ST}$	S_A+S_h	11
Upset	$S_{LP}+S_{DL}+S_{1/2SSE}+S_{EXT}$	$1.2\ S_h$	9
Faulted	$S_{LP}+S_{DL}+S_{SSE}$	$2.4\ S_h$	9
	$S_{LP}+S_{DL}+S_{SSE}+S_{DBA}+S_j$	$2.4\ S_h$	9

where

S_{LP} = Longitudinal pressure stress.

S_{DL} = Longitudinal bending stress due to dead load.

$S_{1/2SSE}$ = Maximum bending stress due to inertial loadings of the 1/2SSE.

S_{SSE} = Maximum bending stress due to inertial loadings of the SSE.

S_{EXT} = Longitudinal bending stress due to external loading.

S_{ST} = Secondary stress due to thermal expansion and anchor movement stress associated with the 1/2SSE.

S_{DBA} = Stress due to design basis accident.

S_h = Basic material allowable stress at hot temperature.

S_A = Allowable stress range for expansion stress.

S_j = Stress due to jet impingement loads.

Generally, only four of the state points presented in Table 15.4.2-1 are subjected to detailed nuclear and thermal-hydraulic analysis. For Case B, the point with the highest power level is analyzed, since past experience has indicated this point is the one which will probably have the lowest DNBR. In addition, either the preceding or succeeding point (depending on the conditions) is analyzed.

For Case D (inside break with loss of offsite power), the point most likely to have the lowest DNBR is the point with the highest power/flow ratio. Usually, this point is the one with the highest power. As with Case B, either the preceding or succeeding point is also analyzed. Should any of the points analyzed result in DNBR's near 1.30, additional points may be analyzed to insure that the point with the minimum DNBR condition has been analyzed.

The points analyzed for this application had DNBR's greater than 1.30. Thus, it is concluded that the minimum DNBR for a steam break is greater than 1.30.

The maximum linear heat rate for the most limiting steambreak case presented in the FSAR was less than 10 kW/ft, which is less than the linear heat rate which results in fuel melting. There is no known failure mechanism associated with this peak linear heat rate.

15.4.2.2 Major Rupture of a Main Feedwater Pipe

15.4.2.2.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of auxiliary feedwater to the affected steam generator. (A break upstream of the feedline check valve would affect the Nuclear Steam Supply System only as a loss of feedwater. This case is covered by the evaluation in Subsection 15.2.8.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy discharge through the break), or a RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluted in Paragraph 15.4.2.1, "Major Rupture of a Main Steam Pipe." Therefore, only the RCS heatup effects are evaluated for a feedline rupture.

within expected ranges. The parameters of importance for these transients include reactor coolant system pressure, steam generator pressure, fluid temperatures, fuel and clad temepratures, break discharge flow rate, steamline and feedwater flow rates, safety and relief valve flow rates, pressurizer and steam generator water levels, mass and energy transfer within the containment (for breaks inside containment), reactor power, total core reactivity, hot and average channel heat flux, and minimum departure from nucleate boiling ratio (DNBR).

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and ICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe conditions.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model. RSB also reviews the values of all the parameters used in the analytical model. CPB reviews the initial conditions of the core and all nuclear design parameters. This includes power level, power distribution, Doppler coefficients, moderator temperature coefficients, void coefficients, reactor kinetics, DNB correlations, and control rod worth.

A secondary review is performed by the Accident Evaluation Branch (AEB) and the results are used by ASB to complete the overall evaluation of the break analysis. The AEB evaluates the fission product release assumptions used in determining any offsite releases and verifies that the radiological consequences resulting from a feedwater pipe break are within acceptable limits as part of its primary review responsibility for SRP Section 15.6.5. The result of AEB's analysis is transmitted to RSB for use in the SER writeup.

In addition, the RSB will coordinate other branches' evaluations that interface with the overall review of feedwater system pipe breaks as follows: The Auxiliary Systems Branch (ASB) reviews the auxiliary feedwater system to verify that it can function following a feedwater line break, given a single active component failure and with either onsite or offsite power as part of its primary review responsibility for SRP Section 10.4.9. RSB reviews the auxiliary feedwater system to confirm that the flow provided is acceptable for controlling the transient following a feedwater line break. The Mechanical Engineering Branch (MEB) evaluates potential water-hammer effects on safety valve integrity as part of its primary review responsibility for SRP Section 3.9 series. The Containment Systems Branch (CSB) reviews the methodology which evaluates the response of the containment to breaks of feedwater lines with regard to the effects of pressure and temperature on the containment functional capabilities as part of its primary review responsibility for SRP Section 6.2.1. The ICSB reviewer concentrates on the instrumentation and control aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. ICSB also evaluates potential bypass modes and the possibility of manual control by the operator as part of its primary reactor responsibility for SRP Sections 7.1 through 7.7.

SUBJECT: Feedwater System Fluid Forcing Function White Paper for Energy Balance Analysis

ATTACHMENT 2

METHOLDOLOGY FOR ENERGY BALANCE ANALYSIS FOR FEED WATER CHECK VALVE CLOSURE TRANSIENT

Description of Method

In order to assess the structural adequacy of the FW piping from the steam generator (SG) nozzle to the first anchor outside containment, an elastic time history evaluation of the major pipe movement due to the FW transient following postulated pipe rupture will be performed. The fluid transient forcing function force-time history developed (Ref 3) will be applied to critical piping segments, and a limiting plastic strain of the piping will be considered.

This energy balance analysis will be performed using the Stone & Webster proprietary program DINASAW (ME-147). This program is an extension of a series of computer programs developed at Massachusetts Institute of Technology for analyzing planar structural response problems which may involve impacts between various structures (Ref. 1). DINASAW may be used to predict the nolinear, dynamic behavior of plane frames (pipes, rings or beams) including large displacements, plasticity and impacts. Arbitrary force-time relations may be applied at any location. The scope of DINASAW is to predict the transient response of plane-frame structures such as straight or curved pipes or beams.

The analytical method used in DINASAW is an energy consistent finite-element technique, based on the Principle of Virtual Work, which is employed to generate on an incremental basis the structure's internal reactions directly. A lumped mass technique is used to generate the nodal masses and central-difference finite-difference operator is employed to integrate the equations of motion in time. The Collision-Induced Crush-Force and Collision-Induced Velocity Methods (CICFM and CIVM, respectively) are available for analyzing impacts.

FW PIPING MODEL

Each of the FW piping loops differs slightly (see Fig 1), however each of them has approximately six pipe segments between the SG nozzle and the first anchor outside containment. Segment 1101 in Firgure 2 contains this anchor for Loop 1 to SG 1 at Sequoyah Unit 2. The piping from the first anchor to the SG is approximated and modelled for input to the DINASAW program. Force-time histories developed for the check valve closure following postulated pipe rupture in the FW header are applied to critical pipe segments.

Figure 3 shows the force-time plot for segment 1502, which appears to be the most critical segment for this SG loop. Other pipe segment forces will be evaluated also.

The output of the DINASAW program will include plastic strain levels in the piping segments as well as a determination of the energy balance in the system resulting from the input forcing functions.

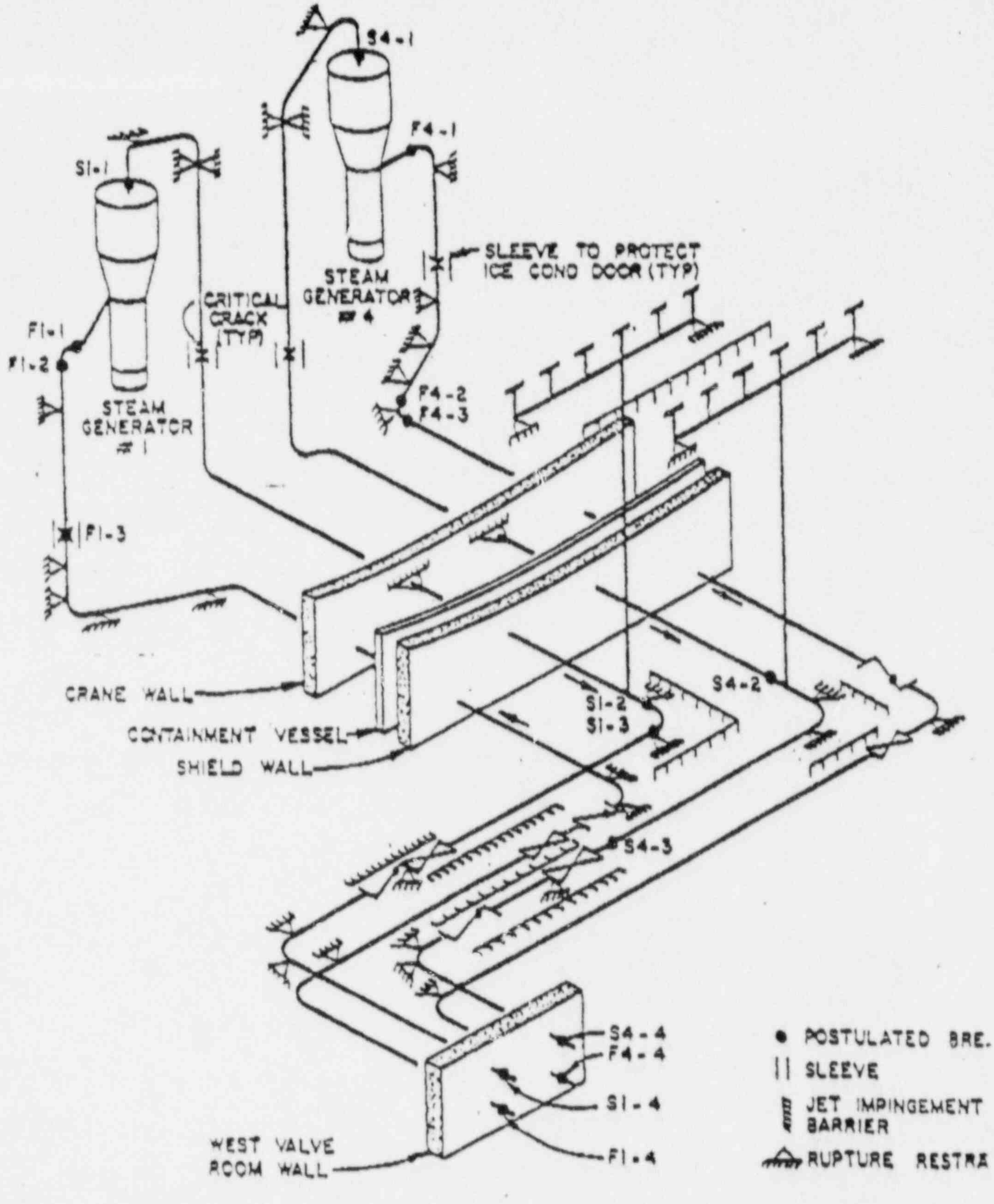
It is assumed that one or more of the pipe whip restraints will have to be shimmed to act as rigid pipe supports. The allowable plastic strain limits will be determined in a separate criteria document, but it is expected that the piping itself will be able to absorb the input forcing functions without rupturing or exceeding the ASME III Appendix F plastic strain limits.

REFERENCES

1. Collings, T. P. and Witmer, E. A., Application of the Collision-Impacted Velocity Method for Analyzing the Responses of Containment and Deflector Structures to Engine Rotor Fragment Impact. ASRL TR 154-8, Aeroelastic and Structures Research Laboratory, Massachusetts Institute of Technology, October 1973.

2. Terry, R. A., Dynamic Inelastic Nonlinear Analysis (DINASAW), User's Manual ME-147, Stone & Webster Engineering Corp., June 1982.

3. Stone & Webster calculation 17341.11-NP(B)-F01, Rev. 1 dated November 6, 1987.



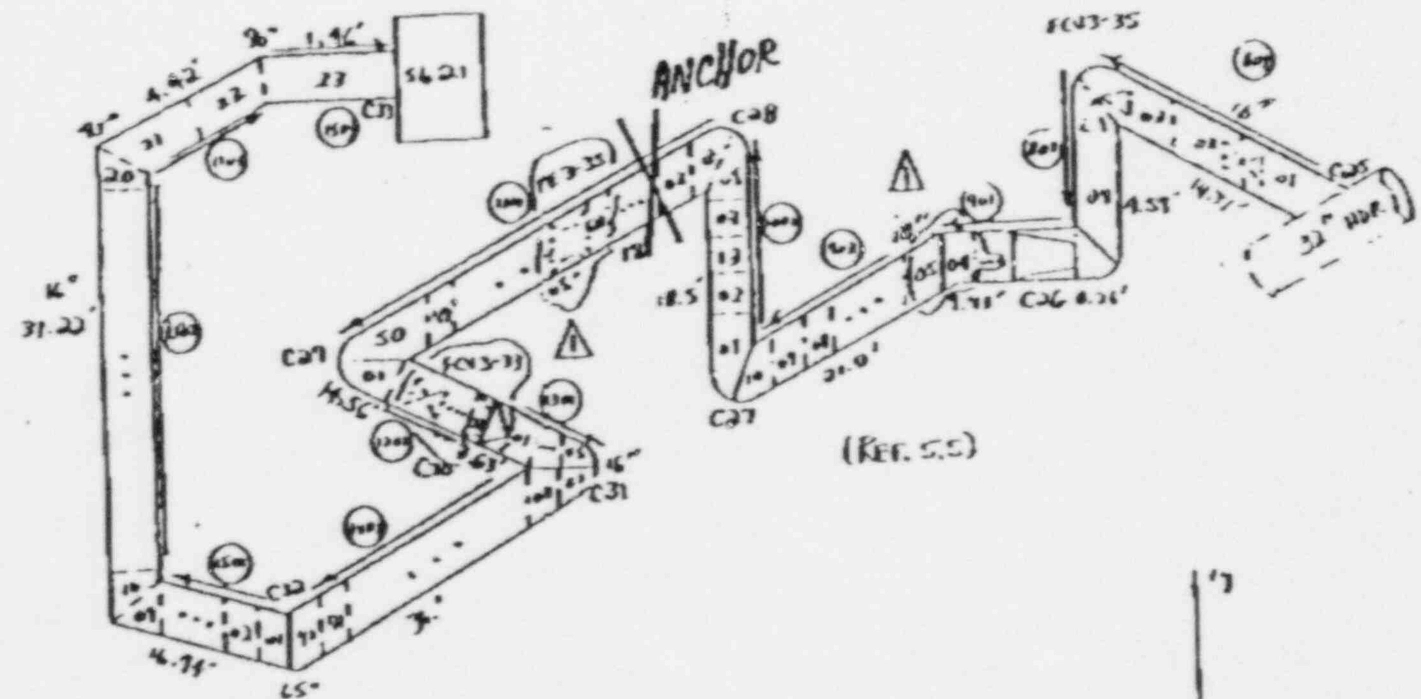
STEAM GENERATORS 1 AND 4 POSTULATED BREAK LOCATIONS AND FIXES

Revised by Amendment 1

FIGURE 1

TVA SQ2 FW LOOP1 -

REPIPE MODEL

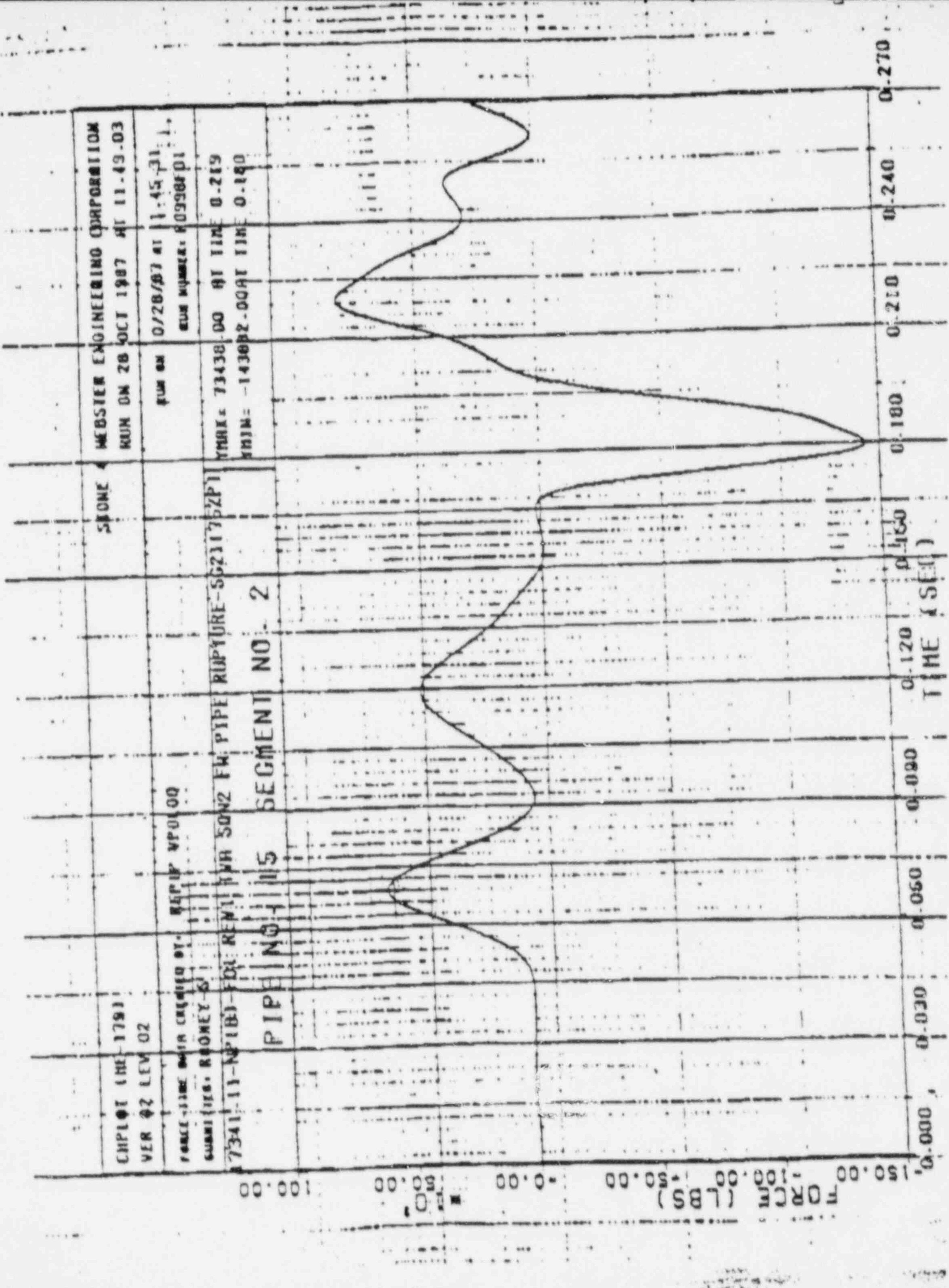


XX: REPIPE PIPE NO., YY: SEGMENT NO.

CZZ ZZ: RELAP COMPONENT NO.

POSITIVE FORCING FUNCTION DIRECTION

FIGURE 2


CHPLOT (NE-1791)
VER 02 LEV 02
FORCE-TIME DATA CREATED BY: REPIP VP0100
QUANTITIES: ROONEY-S
77341-11-NP(B)-FDL REV1 RVR SGN2 FW PIPE RUPTURE-SG2117SZP1
STONE & WEBSTER ENGINEERING CORPORATION
RUN ON 28 OCT 1987 AT 11.49.03
RUN ON 10/28/87 AT 11:45:31
RUN NUMBER: R0998FD1
YMAX= 73438.00 AT TIME 0.219
YMIN= -143082.00AT TIME 0.180
PIPING NO. 105 SEGMENT NO. 2
0.000
0.030
0.060
0.090
0.120
0.150
0.180
0.210
0.240
0.270
TIME (SEC)
100.00
50.00
0.00
-50.00
-100.00
-150.00
FORCE (LBS)

SEQUOYAH NUCLEAR POWER PLANT - UNIT 2
TVA CONTRACT TV-72102A

ACCEPTANCE CRITERIA FOR ENERGY BALANCE ANALYSIS FOR FEED WATER CHECK VALVE CLOSURE TRANSIENT

November 23, 1987

Prepared by: Stone & Webster Engineering Corporation

SEQUOYAH NUCLEAR POWER PLANT - UNIT 2

INTERIM ACCEPTANCE CRITERIA FOR ENERGY BALANCE ANALYSIS FOR FEED WATER CHECK VALVE CLOSURE TRANSIENT

1.0 INTRODUCTION

In order to evaluate the affect of a pipe rupture upstream of the feed water (FW) check valves elasto plastic time-history analysis will be performed on FW piping between the steam generators and containment isolation valves. The result of these analyses will be compared with the interim criteria herein in order to establish the structural adequacy of the FW piping inside containment.

1.1 PURPOSE

The purpose of this document is to provide interim acceptance criteria for the FW piping which must be satisfied prior to Unit 2 restart.

1.2 SCOPE (APPLICABILITY)

The criteria are to be applied to the results of a non linear finite element analysis, the methodology for which is summarized in Section 1.3.

1.3 METHODOLOGY

The sequence of this elasto-plastic evaluation is as follows:

- Perform time history analysis due to FW check valve closure followi[ng] postulated pipe rupture resulting in force-time histories on piping segments (Ref. 10).

- Evaluation of non linear response using segment force-time histories applied to analytic model using values for materials developed in accordance with criteria in Section 2.0.

The results of the analyses (Piping strain) are compared to acceptance criteria based on limits identified in Appendix F of ASME Section III as specifically augmented by Section 5.0 of this document.

The steam generator nozzle loads will be provided to Westinghouse for their acceptance.

2.0 ACCEPTANCE CRITERIA

The acceptance criteria of Appendix F of the ASME Code, Section III are utilized for the allowable stresses which result from a plastic analysis (Attachment 7.6).

These stress limits of 0.7 S_u and 0.9 S_u are converted to acceptable strains by using the power law for the exponential model for the FW piping material - SA 333, Grade 1 carbon steel - as developed in this document.

Westinghouse will have to indicate their acceptance of the steam generator nozzle loads developed by this plastic analysis.

3.0 MECHANICAL PROPERTIES OF STEELS

3.1 Parameters of Interest

The mechanical properties of carbon and stainless steels commonly used for piping and pipe rupture restraints are a basic input to the plastic analyses required for pipe rupture. The basic parameters are those which define the stress-strain curve (Attachment 7.1):

1. Yield strength
2. Young's modulus
3. Ultimate strength
4. Uniform ultimate strain

The effects of temperature and strain rate on those parameters must also be considered.

3.2 Data Sources

The basic data references are the ASME Boiler and Pressure Vessel Code (References 1 and 2) and ASTM Standards (Reference 3). These standards contain minimum specifications for strength and ductility (elongation). The temperature variation of the minimum yield strength, of the ultimate strength, and of Young's modulus are also tabulated in Reference 1. The uniform ultimate strain is only required for inelastic members and, for the primary energy-absorbing components, is given in the fabrication specification.

3.3 Probable Range of Values

Minimum strengths may be greatly exceeded by actual materials. For example, it is possible for the actual static yield strength to be double the Code specified minimum yield. The range in ultimate strengths is considerably less. SINCE PIPE RUPTURE ALLOWABLES ARE BASED ON STRAIN (REFERENCE 4), USE MINIMUM STRENGTH VALUES FOR CALCULATIONS BECAUSE THIS WILL PRODUCE MAXIMUM STRAIN.

3.4 Strain Rate Effects

Strain rate influences mechanical properties. For steels, high strain rates increase the yield strength but decrease the ductility. This effect is less significant at elevated temperatures. In the absence of specific data for a given material, it is most realistic to assume that the strain rate influences only the yield strength at temperatures less than 400°F. Thus, no correction should be used, unless otherwise justified, to account for strain rate in hot pipes (~550°F).

4.0 DERIVATION OF PARAMETERS

The following steps are to be followed to derive the material parameters used for pipe rupture analysis. For commonly encountered materials, these data are provided in Attachment 7.2.

4.1 Temperature, T

Since all material properties are temperature dependent, the first step is to estimate the component temperature.

For the feed water piping use an approximate normal operating temperature of 430°F.

4.2 Young's Modulus, E

The modulus of elasticity (E) at temperature is provided in Table I-6.0 of ASME Section III, Appendixes (Reference 1). To use this table, it may be necessary to refer to References 2 or 3 to first determine the chemical composition of the steel.

4.3 Yield Strength, Sy

Minimum values of the yield strength at room temperature may be found in ASME or ASTM (References 1, 2, and 3). For elevated temperatures, find the minimum yield strength from Tables I-2.1 and 2.2 of Reference 1.

4.4 Ultimate Strength, S_u

Minimum values of the ultimate strength at room temperature may be found in ASME or ASTM (References 1, 2, and 3). Minimum values of the ultimate strength at elevated temperatures are tabulated in Tables I-3.1 and 3.2 of the Appendices to ASME Section III (Reference 1).

4.5 Uniform Ultimate Strain, ε_{uu}

The uniform ultimate strain occurs at the ultimate strength which is reached the instant a tensile deformation instability called necking begins. The strain is uniformly distributed along the tensile member prior to necking.

For energy-absorbing restraint components, take the minimum value in the fabrication specification. For other materials which may exceed the elastic limit, such as the pipe, minimum values should be determinnd from representative test data such as contained in the Nuclear Systems Materials Handbook (Ref. 8). Due to the sparcity of available data, it is appropriate to determine ε_{uu} from total elongation using a ratio of minimum uniform strain to minimum total elongation for a similar material. For some analyses, such as for rings (source shields), the lower transverse elongation may be required. At elevated temperatures, the uniform ultimate strain may be modified in the same proportion as the elongation (Attachment 7.4).

4.6 Allowable Strain, ε_A

The allowable strain for tensile energy absorbers is one-half the uniform ultimate strain obtained in Section 3.5 (Reference 4).

4.7 Allowable Stress, S_A

Use a power law to determine the stress associated with the allowable strain (Attachment 7.3)

$$S = S_o \varepsilon^n$$

Where:

S = Stress

ε = Strain

$$n = \text{Log}\left(\frac{S_u}{S_y}\right) \Big/ \log\left(\frac{\varepsilon_{uu}}{\varepsilon_y}\right)$$

$$S_o = \frac{S_y}{\varepsilon_y^n} = \frac{S_y}{(0.002 + S_y/E)^n}$$

The values of Sy, S_u, and ε_{uu} were derived in Sections 4.3, 4.4 and 4.5. The value of ε_y is the normal 0.2 percent offset used for yield strength measurements. Calculate the stress (S_A) corresponding to the allowable strain (ε_A)

$$S_A = S_o \varepsilon_A^n$$

5.0 SPECIFIC PARAMETERS AND ALLOWABLE STRESSES AND STRAINS

In order to determine properties for SA 333, Grade 1 carbon steel relative to the SA 106, Grade B that is contained in Reference 11, we must first compare the chemical and tensile properties for the two materials. The following page contains excerpts from Section II of the ASME Code for the related materials. It can be quickly observed that the two materials have similar chemical properties except for a small amount (0.10%) of silicon in SA 106B and none required in SA 333-1. The next page shows the similar tensile properties, but the differences should be noted. SA 106B has an ultimate strength of 60,000 psi while SA 333-1 has a value of 55,000 psi. However, SA 333-1 has a greater elongation in 2 inches (35% versus 30% for SA 106B).

Three mechanical sublayers are used in the strain hardening model of the SA 333-1 carbon steel pipe material used in the computer model in the energy balance analysis. The yield and ultimate strength properties are obtained from the ASME Code as noted previously. Tabulated values of properties for SA 106B from Reference 11 are presented in Attachment 7.21 The following table lists the parameters for SA 106B and the needed values for SA 333-1 at $430^{\circ}F$. Attachment 7.5 shows these parameters on a sketch of the exponential and the biliniear strain hardening model. The curve is not drawn to scale as at S_y the 0.002 offset is quite exaggerated. The equation of $S = S_o \epsilon^n = 86177\, \epsilon^{0.210}$ was developed. Allowable values for P_m and $(P_L + P_b)$ stress intensities and strains are developed as follows:

$$P_m \le 0.7\ S_u = 38500\ \text{psi} \qquad \epsilon_{P_m} \le 2.16\%$$

$$(P_L + P_b) \le 0.9\ S_u = 49500\ \text{psi} \qquad \epsilon_{(P_L + P_b)} \le 7.135\%$$

COMPARE THE TENSILE REQUIREMENTS FOR SA 106 GRADE B WITH SA 333 GRADE 1

SECTION II – MATERIAL SPECIFICATIONS

1983 Edition

W85

TABLE 2 Tensile Requirements

SA106

	Grade A		Grade B		Grade C	
Tensile strength, min, psi (MPa)	48 000 (330)		60 000 (415)		70 000 (485)	
Yield strength, min, psi (MPa)	30 000 (205)		35 000 (240)		40 000 (275)	
	Longitudinal	Transverse	Longitudinal	Transverse	Longitudinal	Transverse
Elongation in 2 in. or 50 mm, min, %: Basic minimum elongation for walls 5/16 in. (7.9 mm) and over in thickness, strip tests, and for all small sizes tested in full section	35	25	30	16.5	30	16.5
When standard round 2-in. or 50-mm gage length test specimen is used	28	20	22	12	20	12
For strip tests, a deduction for each 1/32-in. (0.8-mm) decrease in wall thickness below 5/16 in. (7.9 mm) from the basic minimum elongation of the following percentage points	1.75[A]	1.25[A]	1.50[A]	1.00[A]	1.50[A]	1.00[A]

TABLE 3 Tensile Requirements

SA333

	Grade 1		Grade 3		Grade 4		Grade 6	
	psi	MPa	psi	MPa	psi	MPa	psi	MPa
Tensile strength, min	55 000	379	65 000	448	60 000	414	60 000	414
Yield strength, min	30 000	207	35 000	241	35 000	241	35 000	241
	Longitudinal	Transverse	Longitudinal	Transverse	Longitudinal	Transverse	Longitudinal	Transverse
Elongation in 2 in. or 50 mm, min, %: Basic minimum elongation for walls 5/16 in. (7.94 mm) and over in thickness, strip tests, and for all small sizes tested in full section	35	25	30	20	30	16.5	30	16.5
When standard round 2-in. or 50-mm gage length test specimen is used	28	20	22	14	22	12	22	12
For strip tests, a deduction for each 1/32 in. (0.80 mm) decrease in wall thickness below 5/16 in. (7.94 mm) from the basic minimum elongation of the following percentage	1.75[A]	1.25[A]	1.50[A]	1.00[A]	1.50[A]	1.00[A]	1.50[A]	1.00[A]

IT MAY BE OBSERVED THAT SA106 GR B HAS A S_u = 60,000, WHILE SA 333 GR 1 HAS S_u = 55,000 PSI. THE ELONGATION IN 2" IS 30% AND 35%, RESPECTIVELY, SO THE UNIFORM ULTIMATE STRAIN, ϵ_{uu}, MAY BE RATIOED: $\epsilon_{uu(SA106B)}\left(\frac{35}{30}\right) = \epsilon_{uu(SA333)}$

$\therefore \epsilon_{uu} = 10.1\left(\frac{35}{30}\right) = 11.8\%$ @ 430F FOR SA 333 GR 1.

COMPARE CHEMICAL REQUIREMENTS FOR SA 106 GR. B WITH SA133 GRADE 1

WB5

SA 106

TABLE 1 Chemical Requirements

	Composition, %		
	Grade A	Grade B	Grade C
Carbon, max[A]	0.25	0.30	0.35
Manganese	0.27–0.93	0.29–1.06	0.29–1.06
Phosphorus, max	0.048	0.048	0.048
Sulfur, max	0.058	0.058	0.058
Silicon, min	0.10	0.10	0.10

[A] For each reduction of 0.01 % below the specified carbon maximum, an increase of 0.06 % manganese above the specified maximum will be permitted up to a maximum of 1.35 %.

SA 333

TABLE 2 Chemical Requirements

Element	Composition, %						
	Grade 1[A]	Grade 3	Grade 4	Grade 6[A]	Grade 7	Grade 8	Grade 9
Carbon, max	0.30	0.19	0.12	0.30	0.19	0.13	0.20
Manganese	0.40–1.06	0.31–0.64	0.50–1.05	0.29–1.06	0.90 max	0.90 max	0.40–1.06
Phosphorus, max	0.05	0.05	0.04	0.048	0.04	0.045	0.045
Sulfur, max	0.06	0.05	0.04	0.058	0.05	0.045	0.050
Silicon	...	0.18–0.37	0.08–0.37	0.10 min	0.13–0.32	0.13–0.32	...
Nickel	...	3.18–3.82	0.47–0.98	...	2.03–2.57	8.40–9.60	1.60–2.24
Chromium	...	...	0.44–1.01	...	...	...	...
Copper	...	...	0.40–0.75	...	...	...	0.75–1.25
Aluminum	...	...	0.04–0.30	...	...	...	...

[A] For each reduction of 0.01 % carbon below 0.30 %, an increase of 0.05 % manganese above 1.06 % would be permitted to a maximum of 1.35 % manganese.

THE CONCLUSION IS THAT THESE TWO MATERIALS HAVE VERY SIMILAR CHEMICAL PROPERTIES.

Tabulated Parameters for Steels (Ref. 1)

Material	Temperature (F)	E (10^6 psi)	S_y (ksi)	S_u (ksi)	ε_{uy} (%)	ε_y (%)	S_o (ksi)	n -	ε_A* (%)	S_A* (ksi)
Carbon Steel										
SA106B	Rm Temp	29.5	35.0	60.0	14.0	.319	79.2	.142	7.0	55.2
	150	29.1	33.5	60.0	13.0	.315	82.8	.157	6.5	54.7
	430	27.6	29.5	60.0	10.1	.307	95.5	.203	5.0	52.1
	550	27.0	27.1	60.0	10.0	.300	101.3	.227	5.0	51.3
CARBON STEEL SA 333 GRADE 1	RM TEMP	29.5	30.0	55.0	16.3	.302	–			
	400	27.7	25.7	55.0	–	–	–			
	430	27.6	25.3	55.0	11.8	.292	86.2	.210	5.9*	47.6*
	500	27.3	24.5	55.0	–	–	–			

* VALUES ARE FOR ENERGY-ABSORBING RESTRAINT COMPONENTS. ALLOWABLE STRAIN, ε_A, AND ALLOWABLE STRESS, S_A, FOR THE PIPING WILL BE CALCULATED SEPARATELY.

CALCULATE Allowable Stress, S_A, AND ALLOWABLE STRAIN, ϵ_A

Use a power law to determine the stress associated with the allowable strain.

$$S = S_o \varepsilon^n$$

Where:

S = Stress

ε = Strain

$$n = \text{Log}\left(\frac{S_u}{S_y}\right) \Big/ \log\left(\frac{\varepsilon_{uu}}{\varepsilon_y}\right)$$

$$S_o = \frac{S_y}{\varepsilon_y^n} = \frac{S_y}{(0.002 + S_y/E)^n}$$

The values of Sy, S_u, and ε_{uu} were derived in Sections 3.3, 3.4, and 3.5. The value of ε_y is the normal 0.2 percent offset used for yield strength measurements. Calculate the stress (S_A) corresponding to the allowable strain (ε_A)

$$S_A = S_o \varepsilon_A^n$$

$$n = \text{LOG}\left(\frac{55000}{25300}\right) \Big/ \text{LOG}\left(\frac{0.118}{0.00292}\right) = 0.210$$

$$S_o = \frac{25300}{(0.00292)^{0.210}} = 86177$$

FOR GENERAL PRIMARY MEMBRANE STRESS INTENSITY, P_m:

$$P_m \leq 0.7 S_u = 38500 \text{ PSI}$$

$$\epsilon_A = \left(\frac{38500}{86177}\right)^{(1/.210)} = 0.0216 = 2.16\% \text{ FOR } P_m$$

FOR MAXIMUM PRIMARY STRESS INTENSITY AT ANY LOCATION $(P_L + P_B)$

$$(P_L + P_b) \leq 0.9 S_u = 49500 \text{ PSI}$$

$$\epsilon_A = \left(\frac{49500}{86177}\right)^{(1/.210)} = 0.07135 = 7.135\% \text{ FOR } (P_L + P_B)$$

6.0 REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III, Appendices

2. ASME, Boiler and Pressure Vessel Code, Section II

3. Annual Book of ASTM Standards, Parts 1 and 4

4. NUREG-0800, Standard Review Plan 3.6.2, Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

5. ANSI/ANS 58.2-1980, Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture

6. EMTR-2, A Guide to Extrapolation and Application of Experimental Pipe Crush Data

7. Calculation 599.47.01-NM(B)-51-DZ, Derivation of Parameters for EMTR-400.

8. Nuclear Systems Materials Handbook, Hanford Engineering Development Laboratory, U.S. Department of Energy.

9. Kendall, D. P., The Effect of Strain Rate and Temperature on Yielding in Steels, Journal of Basic Engineering, p. 207, March 1972

10. Stone & Webster calculation 17341.11-NP(B)-F01, Rev. 1 dated November 6, 1987.

11. SWEC EMD Technical Re ort EMTR-400-B, "Material Properties for Pipe Rupture Analysis," July 5, 1985.

7.0 ATTACHMENTS

7.1 Typical Uniaxial Stress - Strain Curve

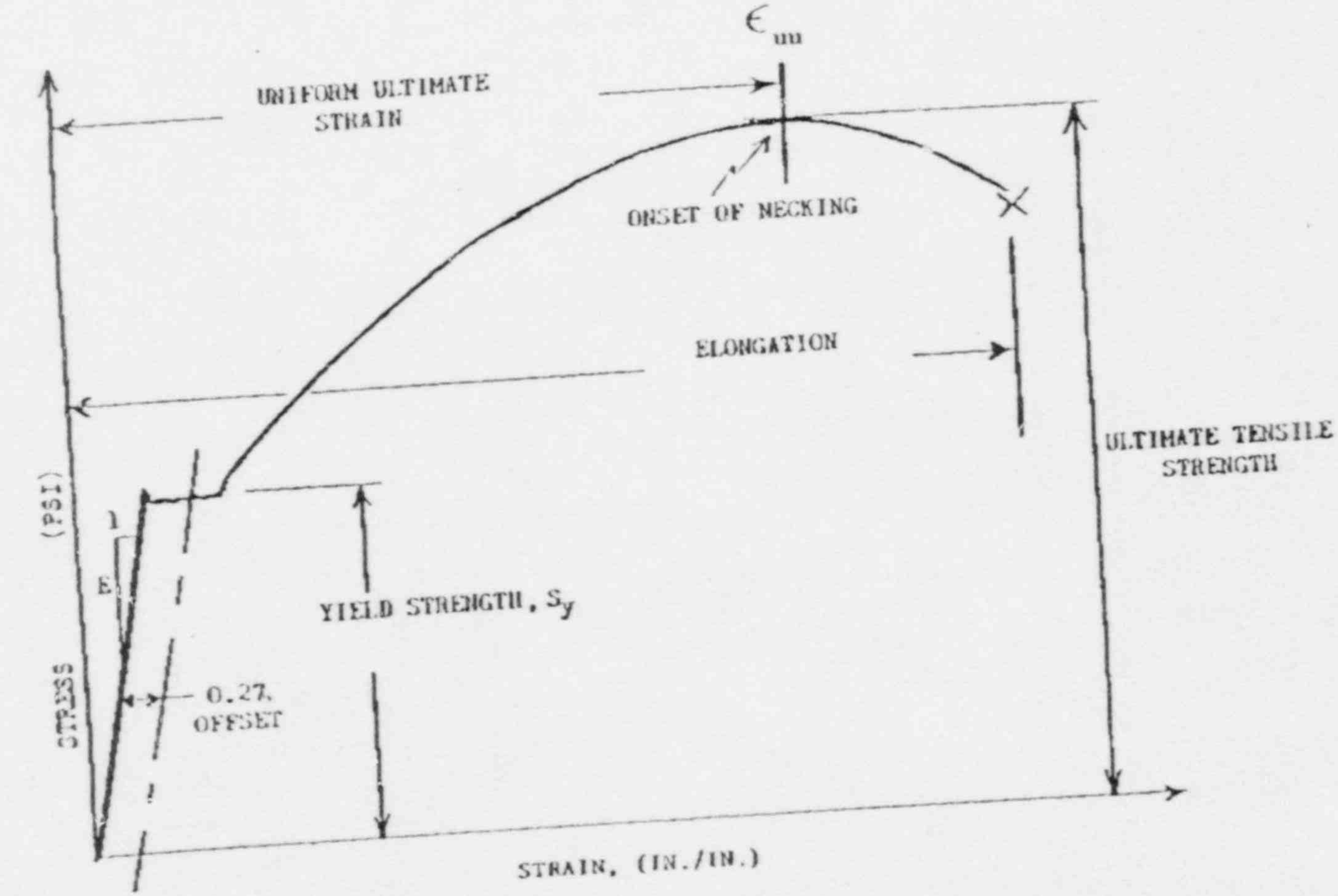
7.2 Tabulated Parameters for Steels

7.3 Exponential and Bilinear Strain Hardening Models

7.4 Elongation at Temperature

7.5 Exponential and Bilinear Strain Hardening Models for SA 333, Grade 1 Carbon Steel at 430°F

7.6 Excerpts from Appendix F of the ASME Code, Section III



ATTACHMENT 7.1

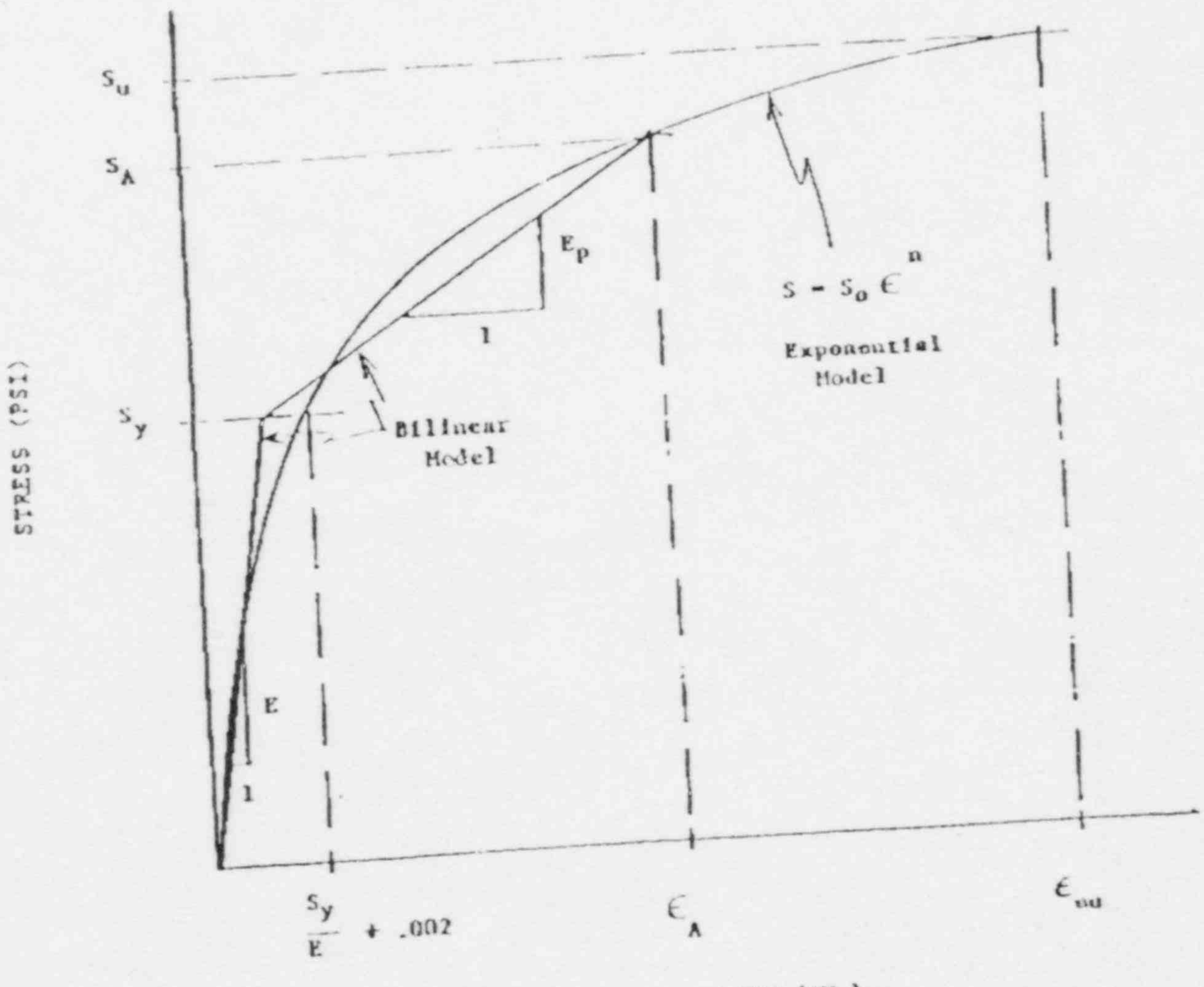
TYPICAL UNIAXIAL STRESS - STRAIN CURVE

Attachment 7.2

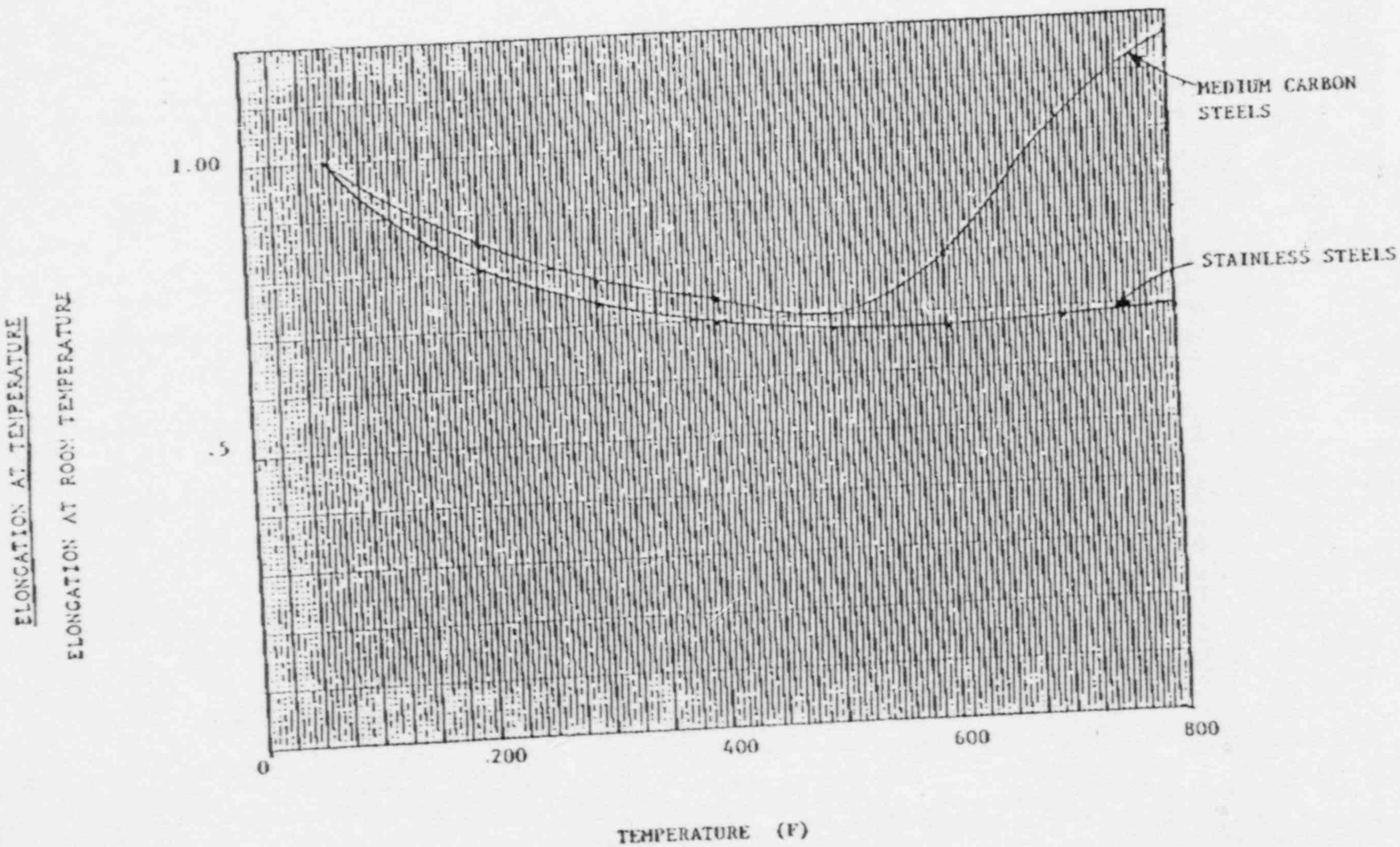
Tabulated Parameters for Steels (Ref. 7)

Material	Temperature (F)	E (10^6 psi)	S_y (ksi)	S_u (ksi)	ε_{uy} (%)	ε_y (%)	S_o (ksi)	n -	ε_A (%)	S_A (ksi)	E_P (10^6 psi)
Carbon Steel											
SA106B	Rm Temp	29.5	35.0	60.0	14.0	.319	79.2	.142	7.0	55.2*	.239*
	150	29.1	33.5	60.0	13.0	.315	82.8	.157	6.5	54.7*	.275*
	430	27.6	29.5	60.0	10.1	.307	95.5	.203	5.0	52.1	.448
	550	27.0	27.1	60.0	10.0	.300	101.3	.227	5.0	51.3	.484
SA155KC60	Rm Temp	29.5	32.0	60.0	12.0	.309	86.4	.172	6.0	54.2*	.316*
SA155KCF60	150	29.1	30.6	60.0	11.2	.305	90.4	.187	5.6	53.6*	.357*
SA515-60	430	27.6	27.0	60.0	8.6	.298	107.2	.237	4.3	50.9	.553
SA516-60	550	27.0	24.7	60.0	8.6	.292	114.2	.262	4.3	50.0	.590
Stainless Steel											
SA376-304	Rm Temp	28.3	32.0	75.0	30.3	.306	94.9	.199	15.1	66.2*	.219*
	150	27.9	27.5	73.0	27.8	.299	96.0	.215	13.9	63.8*	.241*
	430	26.3	20.3	64.1	20.6	.277	97.8	.267	10.3	53.3	.320
	550	25.5	18.8	63.5	20.0	.274	100.5	.284	10.0	52.3	.335
SA376-316	Rm Temp	28.3	30.0	75.0	29.0	.306	96.0	.201	14.5	66.2*	.229*
	150	27.9	27.9	75.0	24.1	.300	103.1	.225	12.0	65.0*	.285*
	430	26.3	20.9	71.8	21.8	.280	110.4	.283	10.9	59.0	.350
	550	25.5	19.3	71.8	22.1	.276	113.1	.300	11.0	58.4	.354
A240-304	Rm Temp	28.3	30.0	75.0	45.0	.306	87.0	.184	22.5	67.0*	.153*
A479-304	150	27.9	27.5	73.0	41.4	.299	86.9	.198	20.7	64.2*	.167*

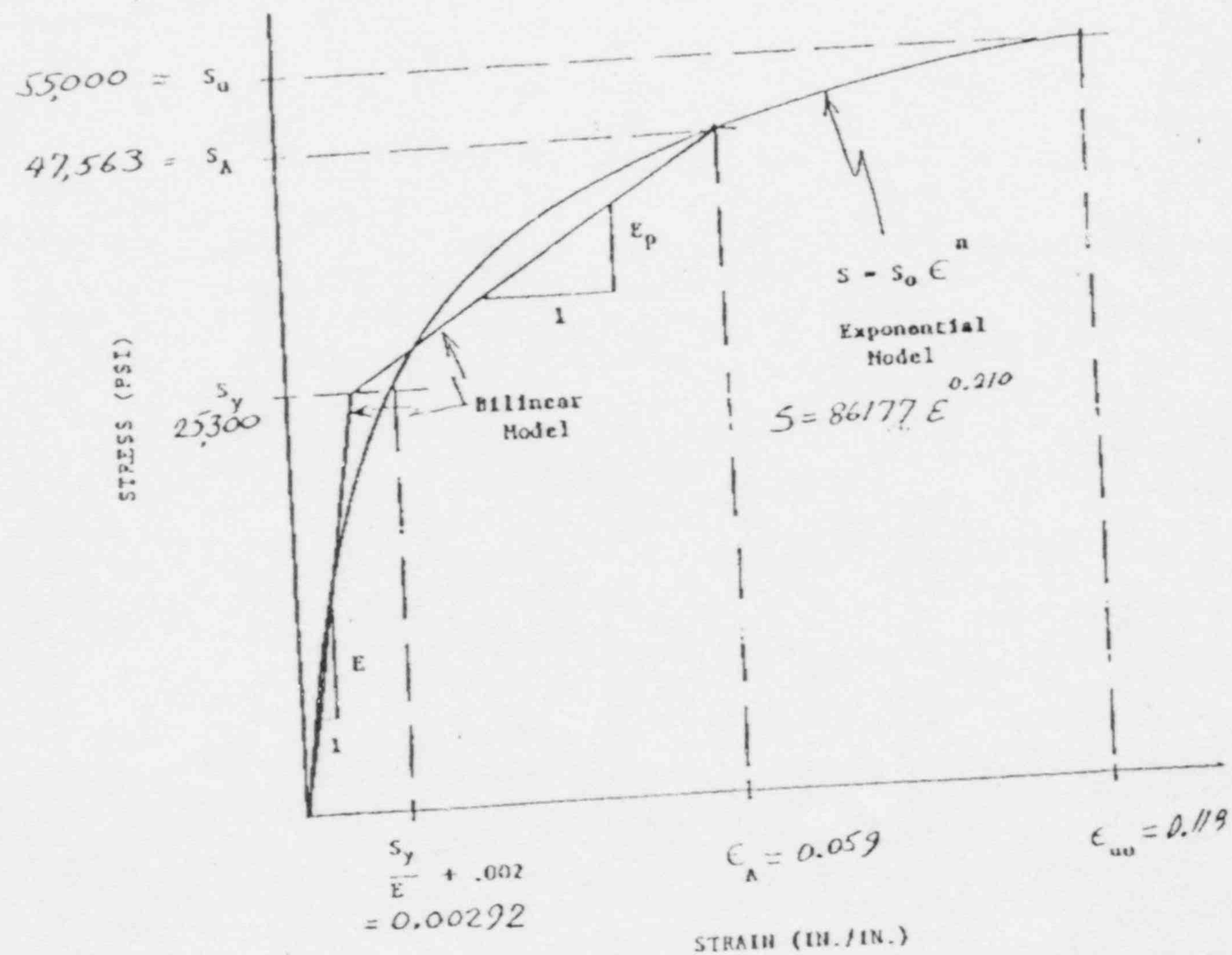
*Strain rate effect included.



ATTACHMENT 7.3 EXPONENTIAL AND BILINEAR STRAIN HARDENING MODELS



ATTACHMENT 7.4 (REF. 7)



7.5 EXPONENTIAL AND BILINEAR STRAIN HARDENING MODELS
CARBON STEEL AT 430°F

ATTACHMENT 7.6 EXCERPTS FROM APPENDIX F OF THE ASME CODE, SECTION III

F-1340 ACCEPTANCE CRITERIA USING PLASTIC SYSTEM ANALYSIS

The acceptance criteria in this section may be applied provided the system analysis considers effects of material nonlinear behavior. The criteria are subject to the restrictions on methods of evaluation stated in F-1322. (attached page)

F-1341 Criteria for Components

Acceptability of components may be demonstrated using any one of the following methods:

(a) elastic analysis
(b) plastic analysis
(c) collapse load analysis
(d) plastic instability analysis
(e) interaction method

F-1341.2 Plastic Analysis. Where the component is evaluated on a plastic basis the following primary stress limits shall be applied.

(a) The general primary membrane stress intensity P_m shall not exceed $0.7S_u$ for materials included in Table I-1.1 and the greater of $0.7S_u$ and $S_y + \frac{1}{3}(S_u - S_y)$ for materials included in Table I-1.2.

(b) The maximum primary stress intensity at any location shall not exceed $0.90S_u$.

(c) The average primary shear across a section loaded in pure shear shall not exceed $0.42S_u$.

F-1322 Methods and Requirements for Analysis

The following requirements shall be satisfied in the evaluation of components or component supports under the loads or load combinations for which Level D Service Limits are specified.

F-1322.1 Analysis Combinations

(a) System analysis may be performed by elastic analysis methods as defined in F-1321.3 or by plastic analysis methods as defined in F-1321.4(b). If elastic system analysis is used, the components and component supports shall be designed to meet the acceptance criteria in F-1330. If plastic system analysis is used, the components and component supports may be designed based on the acceptance criteria in F-1340.

(b) If elastic system analysis is used, the components and component supports may be designed based alternatively on the acceptance criteria of F-1340 provided a reevaluation of the system analysis is performed to determine that it has not been significantly invalidated due to load and stress redistribution and changes in geometry. Some of the conditions under which this analysis combination may be acceptable are:

(1) the plastic deformation is highly localized; or

(2) the changes in geometry are not significant.

This combination may also be considered valid if bounding solutions are established which conservatively account for redistribution of loads and stresses due to plasticity.

(c) If all loads on a component or component support are determined independently from system behavior (e.g., specified pressures), then the component or component support may be designed based on acceptance criteria in either F-1330 or F-1340.

(d) The Design Specification for the components and component supports shall indicate what type of system analysis (if any) has been used to derive the specified loads.

F-1322.2 Dynamic Effects. Postulated events for which Level D Service Limits are specified are generally dynamic in nature. The determination of loads for components and component supports shall account for dynamic amplification of structural response, both in the component and in the system.

F-1322.3 Material Behavior

(a) The mechanical and physical properties shall be taken from Appendix I at the actual temperature of the material. The allowable stresses shall be based on material properties given in Appendix I at temperature. If S_y values at temperature are not tabulated in Appendix I,[1], the value used shall be included and justified in the Design Report.

(b) The stress and strain allowables given in this Appendix are based on an engineering stress-strain curve. If another type of stress-strain curve (e.g., true stress-strain or Kirchoff stress-strain) is used, the results from the analysis or the allowables given in this Appendix shall be appropriately transformed.

(c) When performing plastic analysis, the stress-strain curve used shall be included and justified in the Design Report. It is permissible to adjust the stress-strain curve to include strain rate effects resulting from dynamic behavior. However, the allowables shall be selected in accordance with (a) above.

(d) The yield criteria and associated flow rule used in the inelastic analysis may be either those associated with the maxium shear stress theory (Tresca) or the strain energy distortion theory (Von Mises).

F-1322.4 Geometric Nonlinearities. Geometric nonlinearities may be produced by relatively large deformations and/or rotations and by gaps between parts of the structure. Analyses performed for derivation of loads and for evaluation of acceptability of components and component supports shall consider geometric nonlinearities if appropriate.

F-1322.5 Strain and/or Deformation Limits. In addition to the limits given in this Appendix, the strain or deformation limits (if any) provided in the Design Specification shall be satisfied.

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CEB	W9 D227A C-K	Panafax UF-600	615-632-0792	615-632-3675
EEB	W8 B122 C-K	Panafax PX-100	615-632-6893	615-632-2673
MEB	W7 A60 C-K	Panafax PX-100	615-632-6836	615-632-3334
NEB	W10 A3 C-K	Panafax PX-100	615-632-6794	615-632-2943
DNE DO	W12 A15 C-K	Panafax PX-100	615-632-6102	615-632-2992
SQEP	NUC Bldg A	Panafax PX-100	615-874-5218	615-874-5217
SQEP	NUC Bldg.C		615-874-5443	615-874-5468
SQEP	Staff Bldg E	Panafax UF-400AD	615-874-5351	615-874-5354
SQEP	Civil ~~Mech~~ Bldg G	Panafax UF-400AD	615-874-5443	615-874-5468
SQEP	Mech ~~Civil~~ Bldg H	Panafax UF-400AD	615-874-5517	615-874-5516
SQEP	Elec Bldg P	Panafax UF-400AD	615-874-5024	615-874-5023
WBEP	P-112 SB-K	Panafax PX-100	615-632-8285	615-632-8287
WBEP	C102, IOB WBN	Panafax PX-100	615-365-3695, 3231	615-365-1598
WBEP	DNE, SSFB WBN	Panafax PX-100	615-365-9231	615-632-7100

TECHNICAL ASSESSMENT OF MAIN FEEDWATER PIPING

In order to provide confidence in Feedwater Piping Integrity during and after the postulated event in the Turbine Building, TVA has initiated activities on one loop (loop 1) of the Main Feedwater System to verify its integrity.

Forcing functions have been developed and utilized as input to a three dimensional inelastic finite element analysis. The finite element analyses have been run using a variety of support/rupture restraint configurations. The variety of support/rupture restraint configurations ranged from no supports and gapped rupture restraints to modeling of support load deflection curves for ductile supports and gapped rupture restraints.

The finite element analysis results were evaluated to an acceptance standard based on Appendix F of ASME Section III. This effort indicates that piping system integrity will be maintained during and after the postulated event, although some pipe supports may fail or deform.

KSS:BSR

From Karl Seidle

DNE1 - 1744k