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# U.S. ELEVATED TEMPERATURE STRUCTURAL DESIGN STANDARDS: CURRENT STATUS AND FUTURE DIRECTIONS



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## ABSTRACT

The technical content and scope of coverage of nuclear structural design standards reflects their role in the procurement and licensing of nuclear power plants. In the United States licensing of nuclear power plants requires that the owner of the plant demonstrate that the health and safety of the public is not and will not be endangered by the operation of the plant. That demonstration is a matter of public record and is subject to review and criticism, in an advisary hearing, by state and federal licensing authorities and any member of the public. National concensus structural design standards have been one of the responses to this form of power plant licensing since they effectively remove structural design rules from the arena of conflict. The resulting national standards tend to be generally applicable to all plant types and to relatively diverse operating conditions and material types. Code Case 1592 which is the elevated temperature nuclear design criteria is an example of such a national standard. Its development was the spontaneous outgrowth of the U.S. LMFBR program which demanded the best possible assurance of integrity. Being written within the framework of the ASME Boiler Code it was developed as a general standard, not just a special case for the FFTF or the CRBR Project. Now that its contents are becoming known and accepted there is a desire to apply them to other areas (core support structures, component supports, containment vessels, and to non-radioactive systems). There also are efforts being made to refine the definition of the interface between low temperature rules and creep-temperature rules.

In this paper the development of Code Case 1592 is traced. The current and future technical content of the elevated temperature design standards for nuclear service are discussed. The relationship of Code Case 1592 to other ASME Standards and to certain U.S. industrial, governmental and regulatory standards is examined.

# U.S. ELEVATED TEMPERATURE STRUCTURAL DESIGN STANDARDS: CURRENT STATUS AND FUTURE DIRECTIONS

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## Role

National structural design standards for pressure vessels are a response to state and federal requirements that pressure containing components must be safe. In 1900, this country was experiencing an average of one catastrophic pressure vessel failure a day with a fatality rate of nearly one death per day. The public clearly was endangered by unsafe design (and fabrication and inspection) practices. As a result, several states individually imposed design restrictions in an attempt to protect the public. In 1911, the American Society of Mechanical Engineers started development of a set of pressure vessel design rules which would assure the safety of public, which could be adopted by all states, and which would not unnecessarily restrict the design freedom of the manufacturers. The result of those initial efforts was a design code which we know today as the ASME Boiler and Pressure Vessel Code.

Licensing is the key in the approval chain for a new nuclear power plant. Licensing requires that the need for the plant and the safety of the plant be demonstrated. This typically involves interactions between the evaluators, the energy suppliers, the energy users, and the critics (Figure 1). The safety of the plant can be subdivided into a) the response under specified conditions for which the plant is designed and b) the response of the plant under simultaneous multiple abnormal conditions. The latter case could involve evaluating the effect of a loss of all plant cooling combined with a loss of all plant protective system action combined with a loss of all operator intervention. It is beyond the scope of the ASME Boiler Code. However, the first case, the specific conditions for which the plant is designed, is the focus of the ASME rules. The normal and single abnormal events are the loading conditions which the ASME Code considers in its design evaluation process. Thus, the adequacy of the ASME Code rules for nuclear components is an important contributor to the safety of the proposed plant. Since the ASME Boiler Code is an important aspect of nuclear power plant licensing, its validity would be a legitimate subject for criticism in the licensing process.

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At the present time the structural integrity of nuclear components is not a viable issue [1] for those loadings which have been evaluated using Section III of the ASME Boiler Code [2, 3]. The rules for nuclear equipment have been extremely effective in preventing catastrophic structural failures. The physical troubles which have existed in the nuclear plants are due primarily to the occurrence of unexpected loadings such as flow induced vibration of reactor or heat exchanger internals, check valve slams, and unfortunate coolant chemistry control. Functional troubles continue to occur, especially in valves, but the ASME Code generally does not address function. Two recent comprehensive studies [4, 5] of reactor safety have concluded that current plants (which all use ASME Code designed components) pose less of a threat to public welfare than do meteors. Technically this success is due to the Code's conservatism, recognition of all relevent failure modes, scope (covering materials, fabrication, inspection, testing, and in-service inspection as well as design), and the truly remarkable advances in the area of computer aided evaluation (finite element analysis).

United States regulatory authorities accept the ASME Code rules as an adequate demonstration of structural integrity not only for their technical merit, but also because the Boiler Code represents a national concensus among the technical community. The Code rules are prepared in open meetings at which industry, regulatory, academic and public representatives work together to reach mutually acceptable rules, which are then made available to everyone for use. It may be significant that the intervenors who shout so loudly to the press about nuclear safety do not choose to participate in the Code making procedure where their views would be subjected to comparison with established scientific principles by some of the most competent people in the country. The federal Nuclear Regulatory Commission now is an active participant in the development of the ASME Boiler Code

The ASME Boiler Code will fulfill its role if it keeps the structural adequacy of nuclear components from becoming an active licensing issue. That is, if it can settle the disagreements based on the technical merits, not in an advisory confrontation.

## Form

Section III of the ASME Boiler and Pressure Vessel Code contains the design rules for nuclear structures. As shown in Figure 2, Section III provides seperate rules for metal structures and for concrete structures. Today we are concerned with the rules for metal structures.

There are, as shown here, seperate rules for each of three quality classes plus seperate rules for metal containment structures, for component supports, and for core support structures. The rules of Section III are phrased in as general terms as possible. This avoids, in either a negative or positive sense, special treatment of the products of one company. The Section III requirements apply both to the pressurized water reactor as well as to the boiling water reactor. The rules strive for generality rather than being writtin specifically for, let us say, a particular design of heat exchangers. It would be much easier to write and agree upon rules for a particular plant, but we feel it is to our eventual advantage to phrase the rules more generally.

Section III generally provides rules which assure specific maximum limits on damage when the design rules are satisfied. Figure 3 illustrates that the design rules for what are termed Normal and Upset Events assure that negligible damage to the structural integrity has occurred. The rules for Emergency Conditions produce limited structural damage. The Faulted Condition rules allow the threshold of structural failure to be approached. These several damage levels are provided for the use of the Owner and Manufacturer to demonstrate the desired structural response to specific loads.

Section III established, in their rules for low temperature service, an approach to design which has been continued at elevated temperatures (Figure 4). The designer determines (or upper bounds) the response of his component to the specified loads of service. The structural response is then evaluated using ASME Code procedures and limits to determine whether the resulting structural damage is acceptably small. The general approach of determining the structural response is called design by analysis. The evaluation procedure is failure mode oriented because a series of evaluations are performed where each stage of the evaluation protects against a specific failure mode.

The design rules for low temperatures protect against tensile instability, gross yielding, plastic ratcheting, fatigue, non-ductile rupture, and buckling. You are certainly acquainted with design rules for these failure modes. Most of these rules are used at elevated temperatures too, because these failure modes are still viable at elevated temperatures.

Code Case 1592

The elevated temperature structural design rules for Class 1 nuclear components are contained in Code Case 1592. The use of the Code Case form (rather than imbedding the rules in a Subsection of Section III) means that the ASME Code feels that the rules are interim in nature and are not yet appropriate for permanent adoption.

Code Case 1592 contains design rules (Figure 5) which are intended to protect against structural failure by excessive creep deformation, stress-rupture, creep enhanced ratcheting, creep-fatigue, and creep buckling as well as all of the low temperature failure modes.

Load Controlled Quantities

In the Code Case several new terms are defined. Load controlled quantities are defined as the sum of the primary stresses and those deflection controlled stresses which have significant elastic follow-up.

To prevent gross creep deformation and stress-rupture (ignoring stress concentrations) the membrane stress intensity (elastically calculated) is restricted to (Figure 6):

> 2/3rds of the minimum stress to cause rupture in the loading duration,
>
> 80% of the minimum stress to cause the onset of tertiary creep in the loading duration, and
>
> 100% of the average stress to cause 1.0% (elastic + plastic + thermal creep) strain in the loading duration.

This stress limit is denoted by the symbol, $S_t$. When two or more temperature/loading duration combinations exist, the limit becomes (Figure 7):

$$\sum_{i=1}^{p} \left(\frac{t_i}{t_{im}}\right)_i \leq 1.0$$

where $t_i$ is the total time spent at the specific stress/temperature combination and $t_{im}$ is the allowable time obtained by entering a plot of $S_t$ vs. $t_{im}$ at the membrane stress and current temperature level to determine $t_{im}$.

The elastically calculated membrane + linearized bending stress due to load controlled quantities is limited to (Figure 8):

$$(P_L + P_b) \leq [1+0.5(K-1)]\ S_t$$

where K is the elastic-perfectly-plastic shape factor for the section being considered. The value, 0.5, is a judgement of the degree to which the actual surface stress in combined membrane + bending will be reduced by the effects of plasticity and primary creep. The 0.5 value has been shown to be appropriate for solution treated types 304 and 316 austenitic stainless steels by Dr. Corum of ORNL and has been reported in Reference 6. The 0.5 value may not be appropriate for some other materials.

These limits on the elastically calculated membrane and membrane + linearized bending stresses due to load-controlled quantities are believed to be sufficient to protect against stress-rupture, gross thermal creep, and onset of tertiary creep (Figure 9). The thermal creep equations used to derive the thermal creep strains for the $S_t$ limit for 304 and 316 have a double exponential form and were derived by Dr. Blackburn [7, 8].

The stress-rupture tables are believed to have been based on the evaluations of Dr. Smith [9,10]. The "minimum" value used has been defined as the mean curve minus 1.65 standard deviations.

The tertiary creep limit is based on the work of Leyda and Rowe [11]. The time to the onset of tertiary creep being a temperature dependent function multiplied by the time to rupture.

For the materials now in Code Case 1592 (types 304 and 316 austenitic stainless steel, 2-1/4 Cr - 1.0 Mo., and Alloy 800H) the $S_t$ value is nearly always controlled by the stress rupture term. For materials with non-classical thermal creep curves the tertiary creep term is severely limiting. For this reason Alloy 800, Grade 1, is not in the Code Case. However, the first departure from linear creep for such materials now appears to be the result of factors other than the onset of failure. The Code is now reconsidering the tertiary creep limit. Eventually, the designers may have to adopt creep equations with an accelerating rate term in such cases.

## Deformation Controlled Quantities

Thermal loadings with little elastic follow-up and imposed deformations produce stresses which are termed deformation controlled quantities. These loads are capable of producing ratcheting and fatigue failures but they are not capable of producing stress rupture or gross distortion in a single application (Figure 10).

## Functional Deformation Limits

The ASME Code normally does not address function. But in elevated temperature applications it is possible to achieve distortions that make a component non-functional without violating the structural integrity limits. Distortion of a valve seat, for example, may be well within integrity limits but may cause loss of function due to leaking. The Owner is required to identify appropriate deformation limits to avoid impairment of function. The Manufacturer is also required to identify any additional functional deformation limits that may be required. A design is functionally acceptable if the functional deformation limits are not violated either by:

1) presuming the existence of a 1% strain in a manner consistent with the load, or service, or
2) upper bounding the inelastic strain in the component for the design lifetime.

## Structural Deformation Limits

The philosophy of the low temperature design code, Section III, is to assure "shakedown" to elastic behavior so there is not a continual accumulation of inelastic strain during the design lifetime. At elevated temperatures where creep occurs, it is generally impossible to avoid strain accumulation. However, it is necessary to limit strain accumulation to avoid excessive structural distortion and fracture. Inelastic strains can be accumulated by 1) a severe load that causes large inelastic (plastic) strains, 2) creep response to maintained loads, or 3) a small increment of plastic and/or creep strain on each of many load cycles. The primary stress limits generally preclude accumulation by the first two mechanisms. However, the third mechanism, called ratcheting, results from a combination of primary and secondary stresses. The net strain rate due to the interaction of primary and secondary stresses is higher than the sum of the rates for each of the individual stress categories. Ratcheting is the progressive inelastic deformation that can occur in a component subjected to cyclic variations of mechanical or thermal secondary stress in the presence of sustained primary stress. Ratcheting could occur, for example, in a pressurized vessel subjected to repeated rapid temperature changes initiated on one side of the vessel wall.

The prediction of incremental growth from cycle to cycle for typical component geometries and realistic operating histories generally requires an inelastic analysis. However, recognizing that inelastic analysis is expensive and time consuming, Code Case 1592 provides screening rules (tests) that are based on elastic analysis results and that identify whether or not ratcheting is a problem to be further evaluated. The elastic rules provide limits on the sum of the maximum value of the primary stress intensity, $(P_L+P_b/K_t)_{max}$, and the maximum range of secondary stress intensity, $(Q_R)_{max}$, that are intended to avoid both plastic and creep ratcheting. Another rule, applicable to a limited range of geometries, limits the accumulated inelastic strains to 1% for base metal and 1/2% for weld metal. The inelastic strains are bounded using an elastic analysis and an isochronous stress-strain representation of the material behavior. Since these rules and procedures ignore many of the complexities of the ratcheting phenomenon and the possible interactions involved, they are necessarily very conservative. Nonetheless, they allow the designer to quickly assess whether ratcheting is a problem and in many cases avoid more extensive analysis.

One of the ratcheting screening rules is based on the observation that shakedown can occur if just one end of the load cycle is at non-creep temperatures. At that end of the cycle, a residual stress can be maintained. Thus, a non-zero stress range can be accommodated without anywhere exceeding yield. In such cases, shakedown can be achieved.

In many cases, 1-D inelastic analysis will be sufficient to determine the accumulated inelastic strains. The PLACRAE [12] and CHERN [13] computer programs can perform 1-D inelastic analyses very efficiently.

When the elastic screening rules cannot be met, it is necessary to perform a detailed inelastic analysis. The calculated maximum accumulated positive principal inelastic (plastic plus creep) strains must meet each of the three limits:

1) membrane (strain averaged through the thickness) $\leq$ 1%

2) membrane + bending (surface strain, linearized distribution) $\leq$ 2%

3) local (maximum strain anywhere) $\leq$ 5%

Base metal properties are used for welds in the analysis, but the strain limits for welds are half of the above values. This has the effect of encouraging placement welds in low stress regions. The imposition of strain limits in Code Case 1592 is a departure from previous Code philosophy which utilized only stress limits, and provides an extra measure of assurance against time-dependent modes of failure.

The strain limits are also being re-examined by the Code to determine if they duplicate other limits or if they can be more directly linked to specific failure modes.

## Creep-Fatigue Limits

At low temperatures a plot of plastic strain range vs. the number of cycles to failure provides a basis for setting limits on the stress range to avoid failure by fatigue. At elevated temperatures there is also damage due to creep or relaxation during portions of the load cycles that reduces the fatigue life. Code Case 1592 accounts for the interaction between creep and fatigue by a linear summation of cumulative creep and fatigue damage. Fatigue damage is accounted for using Miner's cumulative damage criterion, and creep damage is accounted for on a time fraction basis. The design limit is given by:

$$\sum_i \left(\frac{n}{N}\right)_i + \sum_j \left(\frac{t}{T}\right)_j \leq D \qquad (1)$$

where the cumulative damage limit D is 1.0 for alloy 800H and a function of the cycle and time fraction sum for type 304 and 316 stainless steels; n is the number of applied cycles at load condition, i; N is the number of design allowable cycles at load condition, i; t is the time duration of load condition, j; and T is the allowable time, based on stress-to-rupture, for the stress at load condition, j.

Seperate evaluation procedures are given for use with elastic and inelastic analysis. For inelastic analysis, fatigue damage is assessed using the peak (local) strain range to determine the allowable number of cycles, N, from a fatigue curve constructed in a manner similar to the low temperature curve but for elevated temperatures. Strain rates are greater than $10^{-3}$ per second and no hold time effects are included. The creep damage term t/T may be replaced by $\int (dt/\tau)$.

A number of techniques are used in the Code Case to assess and bound the creep-fatigue damage using only an elastic analysis. The major features include use of 1) adjusted fatigue curves to account for creep damage due to peak stress relaxation during hold-times and slow strain rates, 2) a special equation to determine the maximum strain range in the presence of inelastic deformation for use with the fatigue curve, 3) a procedure for evaluating creep damage due to primary and secondary stresses, and 4) a cumulative damage limit D of 1.0.

As for the structural deformation limits, the design limits to be used with an elastic analysis are very conservative but provide a relatively quick method for a designer to determine whether additional inelastic analysis is required.

## Buckling Limits

The low temperature buckling limits are provided in the form of charts for allowable loads for spheres and cylinders under external pressure, and cylinders under axial loads. Design limits are not provided for other geometries and loading conditions, nor for conditions under which creep buckling can occur. Code Case 1592 provides minimum design factors for use with calculated buckling loads for any geometry and loading condition where instability due to compressive loads or strains may be a possible failure mode.

Two classes of design limits are provided, one for time-independent (instantaneous) buckling and one for time-dependent (creep) buckling. The time-dependent limit (Figure 11) is similar to the low temperature code except is applicable to general geometries and loading conditions. A load factor of 3.0, for example, means that the designer must assure that the calculated buckling load is more than three times the specified load, taking into account the effects of initial imperfections. The time-dependent limit (Figure 12) requires a factor on the load applied continuously through lifetime. That is, a load factor of 1.5 means that the designer must assure that buckling will not occur during the design lifetime for a load 1.5 times the load applied continuously through life, taking into account the effects of creep enhanced initial imperfections. For a creep exponent of 5 in the power law for steady creep, this stress factor corresponds to a life factor of about 10. In essence, the time-independent limits protect against instantaneous buckling at any point in life, and the time-dependent limits orotect against long-term, creep buckling.

Another new feature of the buckling limits is that distinction is made between load-controlled buckling characterized by continued application of the load during buckling, and strain-controlled buckling where the load is relieved by buckling. Strain limits for strain-controlled buckling are lower than load limits for load-controlled buckling because of the self-limiting nature of deformation in the post-buckled state and the relative insensitivity to initial imperfections. Time-dependent limits are not required for strian-controlled buckling because strain-controlled loads are reduced concurrently with resistance of the structure to buckling when creep is significant.

## Use of Low Temperature Rules

It has been recognized that calculations required to demonstrate satisfaction of the creep-fatigue ratcheting, and buckling limits are difficult at elevated temperatures. Further, it is recognized that those effects we call time-dependent are really stress-time- and temperature dependent. A procedure has been developed by the ASME Code to test the specified time-temperature combination to determine whether creep strain is significant or whether low temperature design procedures would be relevent. The procedure uses two criteria:

a) Is the thermal creep strain less than 0.2% for a stress level of $S_y^{ave}$ and a time duration equal to the total time in the design life at creep temperatures?

b) Is the stress rupture damage less than 0.1 at a stress level of 1.2 $S_y^{ave}$ for a time period equal to the total time specified in the design life at creep temperatures.

In those cases where both criteria are satisfied, creep strain is insignificant and the low temperature rules of Section III may be used. The locus of time-temperature points which satisfy both rules are shown in Figure 13. This option allows many applications to use low temperature procedures even though metal temperatures exceed Section III limits for moderate periods of time.

This concept for an upward extension of low temperature rules is also being considered by the Code for use with Class 2 and 3 components, and core support structures.

Interaction with Other Code Requirements

Code Case 1592 provides rules for the design of elevated temperature Class 1 nuclear components (Figure 14). It also modifies the material rules of Article NB-2000 of Section III.

The rules for fabrication and examination are given by Articles NB-4000 and NB-5000 as modified by Code Cases 1593 and 1594.

The rules for testing and protecting against overpressure are given by Code Cases 1595 and 1596.

An elevated temperature component must have material, fabrication, examination, testing, and overpressure protection rules that reflect the service requirements, not just appropriate design rules.

Interaction with RDT Standards

The Energy Research and Development Administration (ERDA) of the U.S. federal government has funded the development of standards to supplement existing industrial standards where they deem it necessary. For example, Code Case 1592 provides rules for the design of elevated temperature nuclear Class 1 components.

RDT Standard F9-4 [Reference 14] supplements the design rules of Code Case 1592 (Figure 15). It has been applied as a mandatory standard for the Clinch River Breeder Reactor. It tightens a few design rules. Where Code Case 1592 limits

$$(P_L+P_b) \leq KS_t$$

RDT Standard F9-4 adds the requirement that:

$$(P_L+\frac{P_b}{K}) \leq S_t$$

RDT Standard F9-4 requires the preparation and systematic updating of a Structural Evaluation Plan (SEP) to permit all parties to see what analysis (form, type, extent) will be available at each decision point in the construction sequence. Of special note is the fact that such a plan inevitably identifies the manner and extent to which each analysis tool will be used and thus identifies the required extent of verification and qualification.

RDT Standard F9-4 requires the use of <u>average</u> response properties (stress-strain, thermal creep) in inelastic analyses being performed to demonstrate freedom from ratcheting and where creep-fatigue damage is to be calculated. We believe that a *uniform approach* to the selection of properties for inelastic analysis is currently appropriate. When we know precisely how variations in all response and failure properties are inter-related, better (and perhaps more flexible) guidance can be provided.

RDT Standard F9-5 (Figure 16) contains information designed to aid the structural analyst. A thorough discussion of the intent and suggested content of a Structural Evaluation Plan is provided. The terms, verification, and qualification are defined. The ways in which analysis methods can be verified are discussed. Sources of qualification data are provided. A typical verification/qualification plan is provided.

The general concepts and the specific mathematical representations suggested for implementing time independent plasticity and time dependent thermal creep are provided. These recommendations were provided by ORNL [See Reference 15].

RDT Standard F9-4 contains detailed guidance on how certain of the Code Case rules can be satisfied. Code Case 1592 permits alternate rules to be used for ratcheting and creep fatigue when the Owner adopts the rules in his specification. RDT Standard F9-5 contains several important rules which, when adopted by the Owner, become alternates to the Code Case 1592 rules. The rules allowing the use of low temperature evaluation methods when metal temperatures and time durations are suitably limited, were implemented in RDT Standard F9-5 before they were adopted by the Code.

Finally, appendicies in RDT Standard F9-4 provide a capsule summary of the response characteristics of type 304 and 316 austenitic stainless steel [Reference 15] at elevated temperatures.

There are a number of other RDT Standards. In most cases, the additional restrictions which they impose are relatively few in number. Their restrictions are almost always well founded, technically.

## Regulatory Guides

The U.S. Nuclear Regulatory Commission has issued guides which NRC will use in evaluating compliance with their regulations. For example, Regulatory Guide 1.87 [Reference 16] accepts Code Case 1592 as an adequate means for demonstrating structural integrity of elevated temperature Class 1 components. A limited number of items require further justification by the Owner for his application. For example, the selection of response properties (maximum, average, or minimum) for inelastic analysis and for buckling analysis should be justified on a case basis. The net effect of the Regulatory Guide is most significant because it pre-emptorally accepts Code Case 1592 as a valid method of demonstrating structural integrity of high temperature components.

There are a number of other Regulatory Guides which apply to other aspects of nuclear plants.

## Practical Application of Code Case 1592

A typical LMFBR application will involve a design specification which specifies the steady state pressure, some small pressure fluctuations, a couple of earthquakes, and ten to twenty different thermal transient events. The pressure and earthquake loads rapidly define the minimum allowable wall thickness. Ratcheting and creep-fatigue may prove to be difficult to handle since they often produce conflicting thickness requirements.

The first step in evaluating the thermal transients usually is to run a one-dimensional transient thermal analysis of a shell thickness that is perhaps twice as thick as required to satisfy the Code Case's primary-stress limits. Elastic analysis of the thermal responses can be evaluated using the maximum surface stress and the bending stress from a linear through-the-wall stress distribution which has the same net moment and force as the actual elastic distribution.

Figure 18 contains the elastic stresses from a one-dimensional transient thermal/elastic analysis for each of nine specified thermal transients. A one-dimensional analysis simulates the response of most shell structures subjected to internal thermal transients. These stress results allow the analyst to compare the nature and severity of the several thermal transients on a relative elastic basis. Here we see that we can seperate all nine transients into just three groups. The severity of each group of transients is represented by the most severe transient of the group. The number of occurrences of the grouped event is the total of the number of occurrences of each member of the group. With only three transients, it is feasible to perform 2-D transient thermal/elastic analyses of the actual component for use in ratcheting and creep-fatigue evaluations.

If the design does satisfy the ratcheting or creep-fatigue limits of the Code Case using elastically calculated stresses, then inelastic analysis will not be needed. If the Code Case limits are not satisfied using elastically calculated stresses, then one-dimensional inelastic analysis is usually employed (Figure 19). If a one-dimensional model is applicable, then 1-D inelastic analysis is used. At least one of the 1-D programs [13] will perform more than one load cycle. That is, if the user asks for seven load cycles, the program will perform the specified load cycle seven times successively. One can afford inelastic analysis of a large number of load cycles because these programs are so quick. The program being discussed also has limited (post processor) capabilities for evaluating the results in terms of the ASME Code rules.

If the design still does not satisfy the Code limits, redesign usually is considered. Multidimensional inelastic analysis is so expensive that it has not proved to be practical as a routine design tool. In many cases, the inability to demonstrate that the design meets the ASME Code with elastic or 1-D inelastic analysis also means that the design poorly accommodates the loads of service. Thus, redesign is appropriate.

Multidimensional inelastic analysis serves as a design tool primarily to demonstrate a specific point which validates an elastic or simplified inelastic approach. If inelastic analysis can be used to demonstrate that the structural response predicted by less difficult analysis techniques is correct or conservative for a key load, then the simplified methods can be used for the spectrum of loads. At the present time this seems to be the appropriate design use for inelastic analysis.

## Summary

The elevated temperature structural design rules of the ASME Boiler and Pressure Code for nuclear components have been openly derived, are well accepted, and have been tested by application. We know that they can be used and that inelastic designs can meet their requirements. The service performance of the FFTF and CRBRP systems will tell us whether the current rules are completely adequate. We know of a number of areas where our rules can be improved. The basis of the ASME Code rules has been placed in the open literature [17-22]. Unfortunately, the ASME Code has not considered the design criteria used by other countries because they are not in the public domain. The Boiler Code members are extremely interested in your rules, thoughts, and suggestions. Your direct or indirect participation is invited.

## Acknowledgements

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20. Pressure Vessels and Piping: Analysis and Computers, 1974 Pressure Ves;el and Piping Conference, the American Society of Mechanical Engineers, New York City.

21. Fatigue At Elevated Temperatures, ASTM STP 520, The American Society for Testing and Materials, Philadelphia, Pa.

22. "Criteria for Design of Elevated Temperature Class 1 Components in Section III of the ASME B&PV Code", The American Society of Mechanical Engineers, New York City, to be issued.

Evaluator

NRC

State

Local

Energy Supplier

Utilities

ERDA

NSSS

Component Manufacturer

Energy User

Public

Industry

Conservationist

Critic

Local Resident

No Growth Advocate

Alternate Power Advocate

Anti-Nuclear Groups

EPA

FIGURE 1

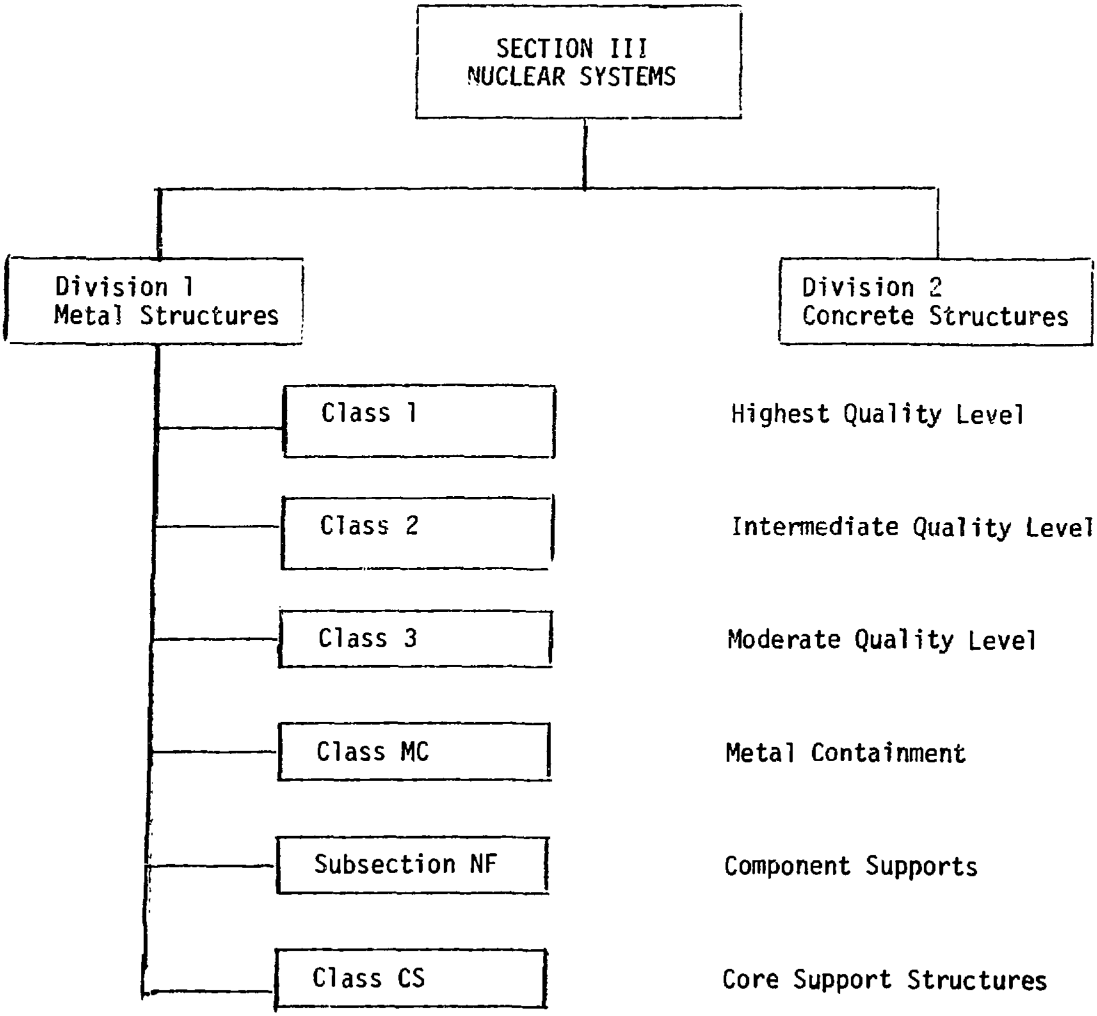

SECTION III
NUCLEAR SYSTEMS
Division 1
Metal Structures
Division 2
Concrete Structures
Class 1
Highest Quality Level
Class 2
Intermediate Quality Level
Class 3
Moderate Quality Level
Class MC
Metal Containment
Subsection NF
Component Supports
Class CS
Core Support Structures

FIGURE 2

| Operating Condition | Maximum Damage Level |
|---|---|
| Normal | Negligible |
| Upset | Negligible |
| Emergency | Limited |
| Faulted | Approaches the Threshold of Failure |

FIGURE 3

DESIGN BY ANALYSIS

and

FAILURE MODE ORIENTATION

- Tensile Instability
- Gross Yielding
- Plastic Ratcheting
- Fatigue
- Non-Ductile Fracture
- Buckling

FIGURE 4

HIGH TEMPERATURE FAILURE MODES

- Excessive Creep Deformation
- Stress Rupture
- Creep Enhanced Ratcheting
- Creep-Fatigue Failure
- Creep Buckling

FIGURE 5

$$\underline{P_m \leq S_t}$$

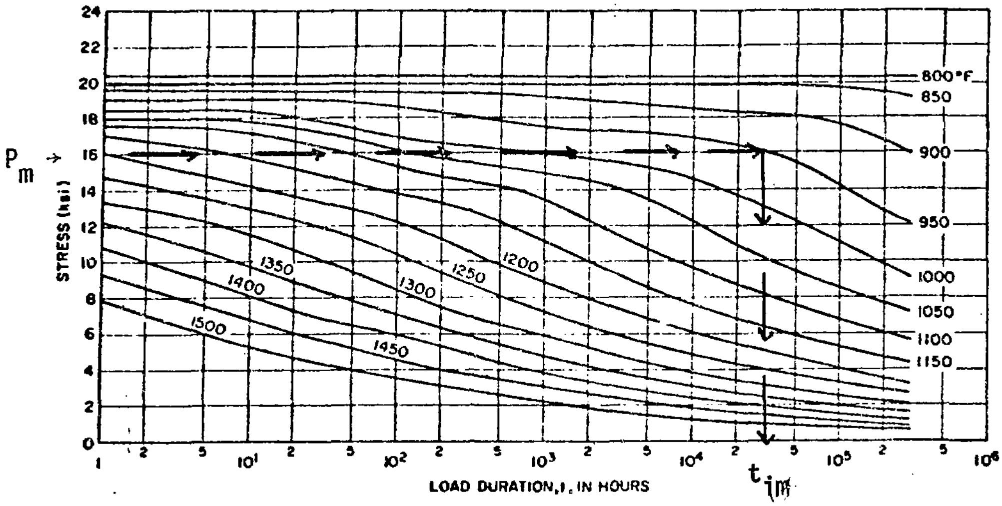
$S_t$ is the lower of:

$2/3\ S_R^{Minimum}$ in $t_i$ hours, and

$.80\ S_{Tertiary}^{Minimum}$ in $t_i$ hours, and

$1.00\ S_{1\%}^{Average}$ in $t_i$ hours.

FIGURE 6




$$\sum_i \left(\frac{t_i}{t_{im}}\right) \leq 1.0$$

FIGURE 7

$$(P_L + P_b) \leq [1+0.5(K-1)]\, S_t$$

or

$$\sum_i \left(\frac{t_i}{t_{ib}}\right) \leq 1.0$$

or

where $t_{ib}$ is determined at a stress of

$$(P_L + P_b) \div [1+0.5(K-1)]$$

FIGURE 8

$$\varepsilon_c = \varepsilon_t(1-e^{-rt}) + \varepsilon_x(1-e^{-st}) + \dot{\varepsilon}_m t$$

$$S_R^{Min} \equiv S_R^{Ave} - 1.65\ (\text{Standard Deviation})$$

$$t_{tertiary} \equiv f\ (\text{Temperature}) \times t_{Rupture}$$

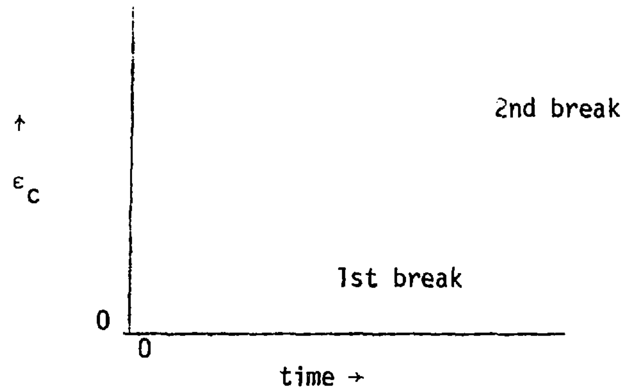



FIGURE 9

FAILURE MODES FOR
DEFORMATION CONTROLLED QUANTITIES

Functional Deformation Limits

Structural Deformation Limits

1% Membrane

2% M+B

5% Peak

Creep-Fatigue

$$\sum_i \left(\frac{t_i}{t_{id}}\right)_i + \sum_j (n/N_d)_j \leq D$$

FIGURE 10

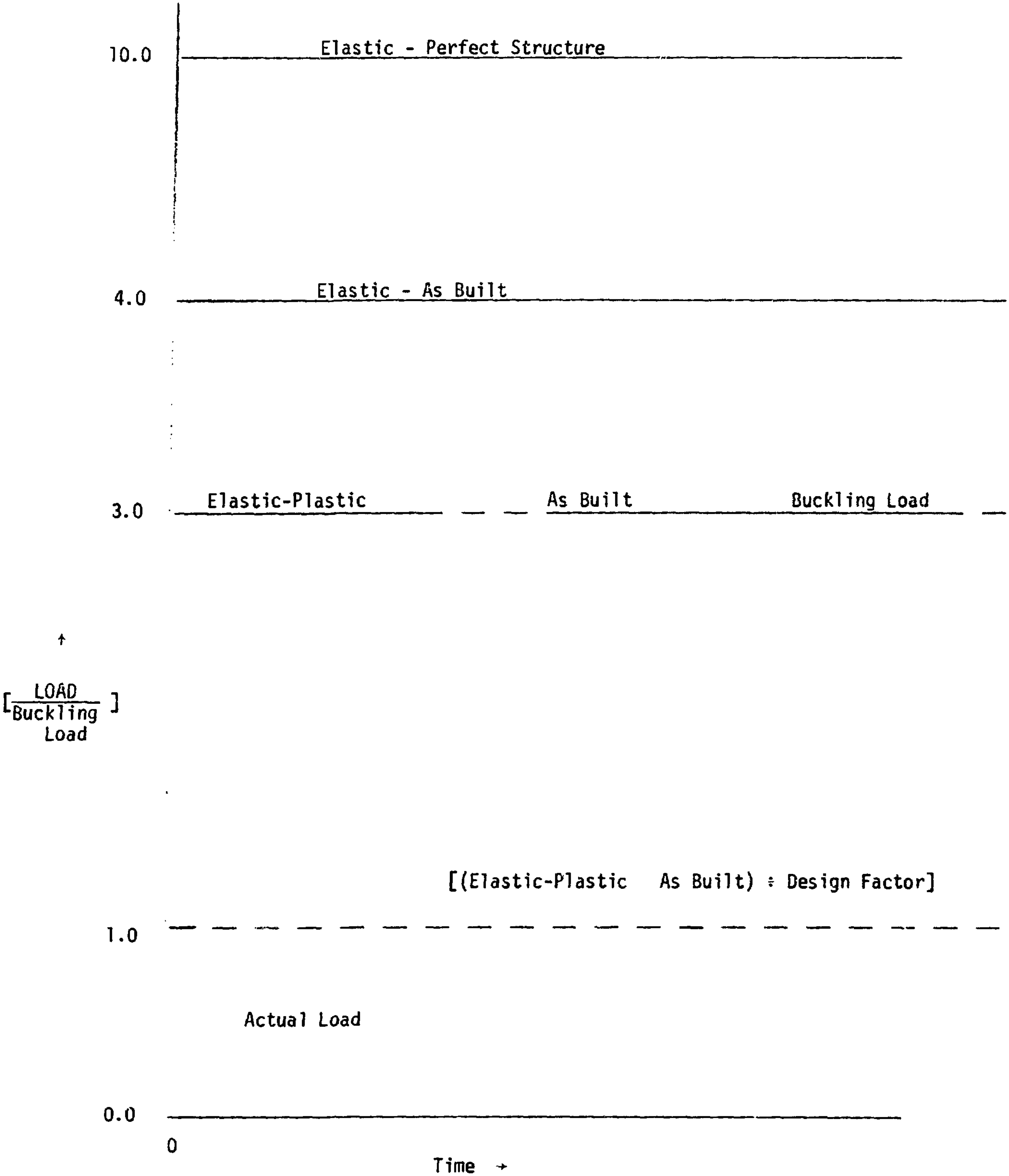

10.0
Elastic - Perfect Structure
4.0
Elastic - As Built
3.0
Elastic-Plastic
As Built
Buckling Load
[LOAD / Buckling Load]
[(Elastic-Plastic As Built) ÷ Design Factor]
1.0
Actual Load
0.0
0
Time

FIGURE 11

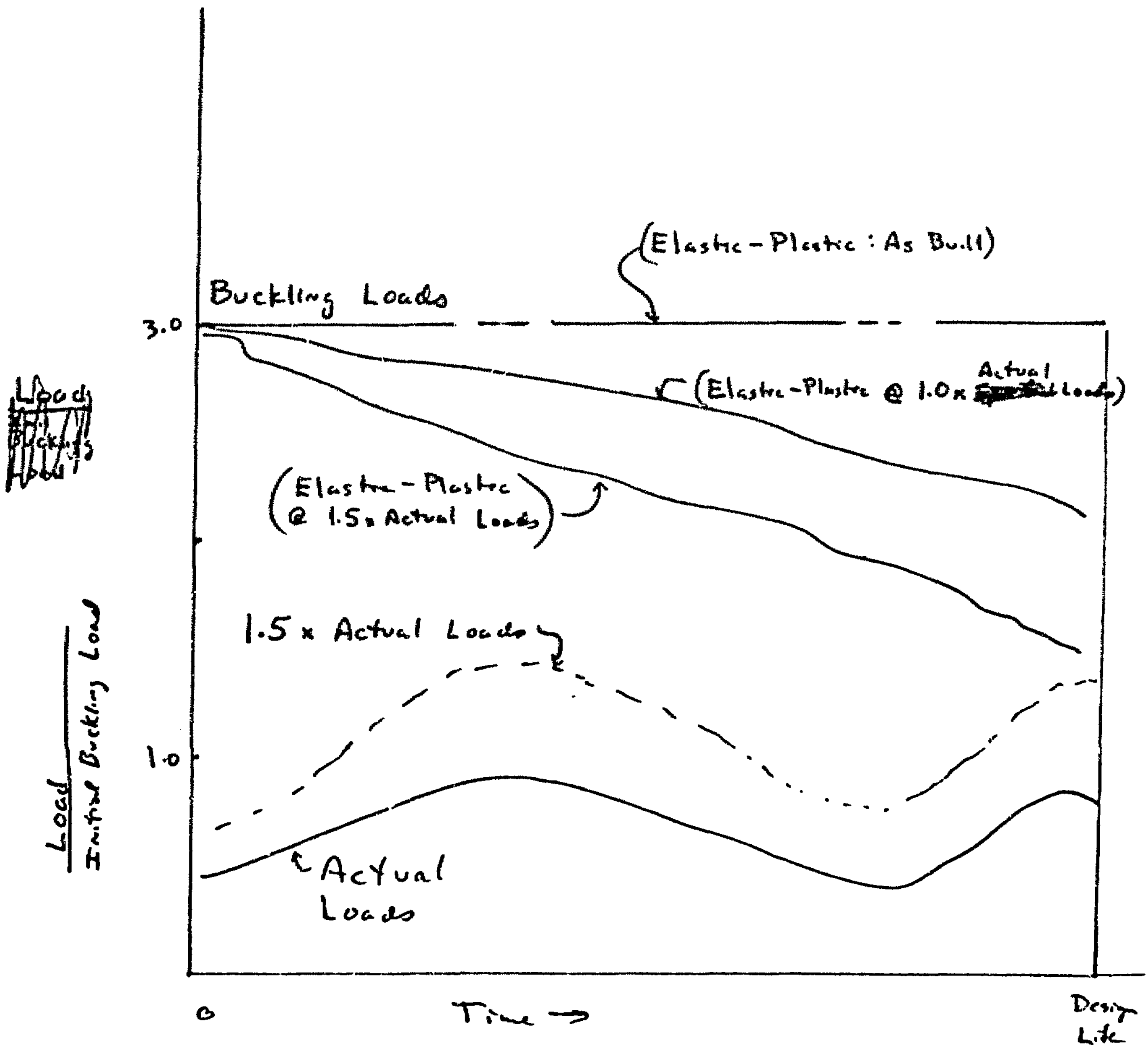

(Elastic - Plastic : As Built)
Buckling Loads
3.0
(Elastic-Plastic @ 1.0x Actual Loads)
(Elastic - Plastic @ 1.5x Actual Loads)
1.5 x Actual Loads
1.0
Actual Loads
Load
Initial Buckling Load
0
Time →
Design Life

FIGURE 12

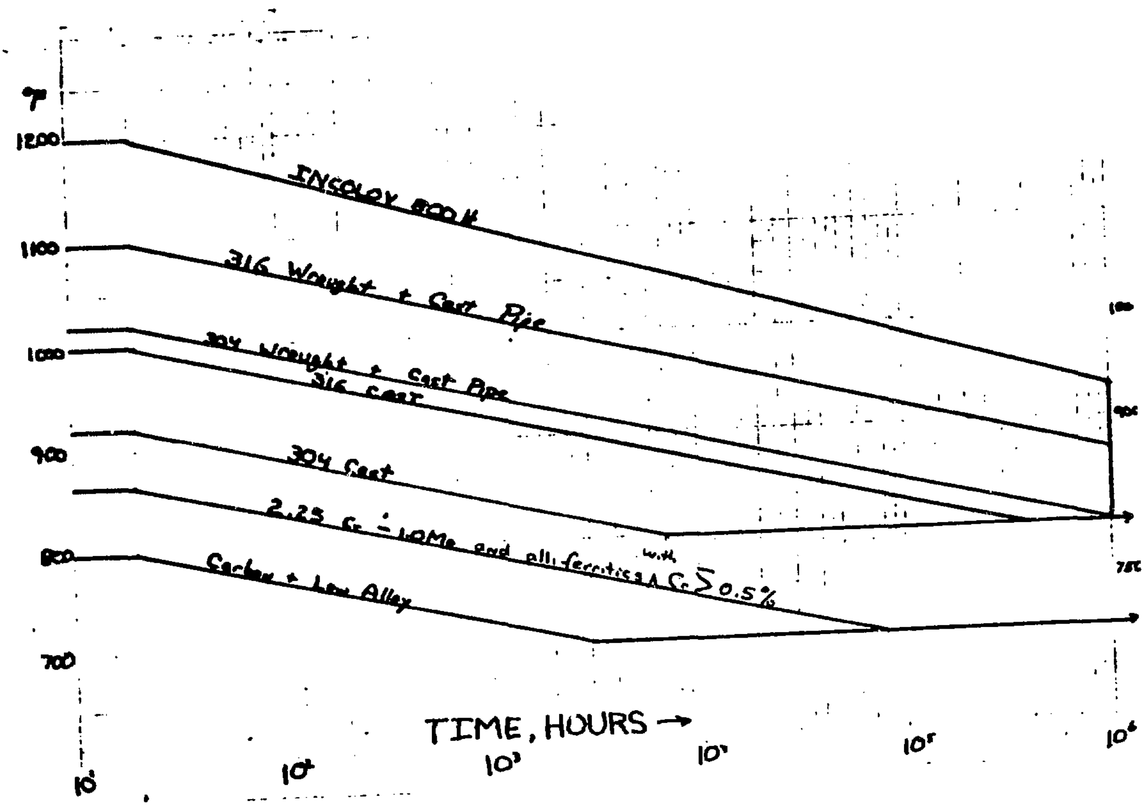

°F
1200
1100
1000
900
800
700
INCOLOY 800 H
316 Wrought + Cast Pipe
304 Wrought + Cast Pipe
316 Cast
304 Cast
2.25 Cr - 1.0Mo and all ferritics with Cr ≥ 0.5%
Carbon + Low Alloy
TIME, HOURS →
10
$10^2$
$10^3$
$10^4$
$10^5$
$10^6$

FIGURE 13

## ELEVATED TEMPERATURE CONSTRUCTION RULES

| | |
|---|---|
| Materials | NB-2000 + Code Case 1592 |
| Design | Code Case 1592 |
| Fabrication | NB-4000 + Code Case 1593 |
| Examination | NB-5000 + Code Case 1594 |
| Testing | Code Case 1595 |
| Overpressure Protection | Code Case 1596 |

FIGURE 14

RDT STANDARD F9-4

Supplements Code Case 1592

Defines Terms

Requires Structural Evaluation Program

Specifies Use of "Average" properties

Identifies Environmental Factors

Invokes Other Ratcheting, Creep-Fatigue Rules

FIGURE 15

## RDT STANDARD F9-5

Describes Structural Evaluation Program

Guides Verification and Qualification

Describes How to Use Code Case 1592

Provides Alternate Rules for

Ratcheting

Creep-Fatigue

Describes Plasticity and Creep Implementation

Illustrates Behavior of 304 SS

FIGURE 16

Revision 1
June 1975

U.S. NUCLEAR REGULATORY COMMISSION

# REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 1.87

## GUIDANCE FOR CONSTRUCTION[1] OF CLASS 1 COMPONENTS IN ELEVATED-TEMPERATURE REACTORS (SUPPLEMENT TO ASME SECTION III CODE CASES 1592, 1593, 1594, 1595, AND 1596)

### A. INTRODUCTION

Section 50.55a, "Codes and Standards," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires, in part, that structures, systems, and components be designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed. General Design Criterion 1, "Quality Standards and Records," of Appendix A,[2] "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 permits use of recognized codes and standards, provided they are identified and evaluated to determine applicability, adequacy, and sufficiency and are supplemented or modified as necessary to ensure a quality product in keeping with the required safety function. This guide describes interim licensing guidelines to aid applicants in implementing these requirements with respect to ASME Class 1 components operating at elevated temperatures. This guide applies to high-temperature gas-cooled reactors (HTGRs), liquid-metal fast-breeder reactors (LMFBRs), and gas-cooled fast-breeder reactors (GCFBRs).

[1] As defined in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PV Code), construction is an all-inclusive term comprising materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components.

[2] Appendix A to 10 CFR Part 50 is directly applicable to water-cooled nuclear power plants; however, as indicated in that appendix, the General Design Criteria are considered applicable to other types of nuclear power units and are intended to provide guidance in establishing principal criteria for such other units.

### B. DISCUSSION

The rules for construction of nuclear components given in Section III of the ASME *Boiler and Pressure Vessel Code*, including Class 1 nuclear components that are covered in Subsection NB of Section III, apply to components at temperatures where creep effects are insignificant. Material behavior considerations are limited to either elastic or elastic-plastic response, which, in effect, provides protection against only time-independent failure modes such as ductile rupture, gross distortion, and fatigue.

The service temperatures and load conditions for HTGRs, LMFBRs, and GCFBRs are such that time-dependent phenomena such as creep and relaxation are important. Subsection NB of Section III does not provide adequate guidance for construction of components subject to elevated-temperature service. Therefore, as an interim step, the ASME has developed five Code Cases (1592, 1593, 1594, 1595, and 1596) to provide guidance in this area. Code Cases 1593, 1594, 1595, and 1596 were approved on November 5, 1973, as interpretations of the ASME Boiler and Pressure Vessel Code. Code Case 1592 was approved on April 29, 1974.

These Code Cases cover design, fabrication and installation, examination, testing, and protection against overpressure. They reflect both time-independent and time-dependent material properties and structural behavior (elastic and inelastic) by considering the following modes of failure:

1. Ductile rupture from short-term loadings;

2. Creep rupture from long-term loadings;



FIGURE 17

| Event | Elastic Stress Intensity | | | Category |
|---|---|---|---|---|
| | Membrane | Linearized Bending | Peak | |
| U1 | 30 | 6,630 | 8,900 | b |
| U2 | 126 | 27,060 | 37,400 | a |
| U3 | 28 | 6,080 | 8,400 | b |
| U4 | 31 | 6,710 | 9,150 | b |
| U6 | 27 | 5,820 | 8,450 | b |
| U10 | -7 | -1,410 | -1,900 | c |
| | 31 | 6,560 | 8,800 | b |
| E1 | 10 | 2,080 | 2,800 | b |
| E2 | 28 | 5,920 | 8,200 | b |
| E10 | 90 | 19,300 | 26,500 | a |

FIGURE 18

ELASTIC ANALYSIS

1-D INELASTIC ANALYSIS

REDESIGN

MULTI-DIMENSIONAL INELASTIC ANALYSIS

FIGURE 19