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LESSONS LEARNED THROUGH THE CRBRP LICENSING PROCESS: HIGH-TEMPERATURE STRUCTURAL DESIGN

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ABSTRACT

This report contains a "road map" of that part of the licensing process for the Clinch River Breeder Reactor Plant (CRBRP) related to high-temperature structural design. The CRBRP licensing experience pertaining to high-temperature structural design, while not carried through to completion, has significant implications for the licensing of future liquid metal reactors (LMRs). The licensing experience, from development of the CRBRP structural design basis to "final" resolution of all regulatory concerns about structural adequacy, is reviewed in this report and significant lessons learned are noted. It is concluded that the structural design issues raised by regulatory authorities must be meaningfully addressed before any future LMRs can be licensed for operation.

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1.0 INTRODUCTION

Structural design of the Clinch River Breeder Reactor Plant (CRBRP) required consideration of thermal, chemical, and mechanical environments quite different from those associated with conventional light water nuclear reactors. Assurance of structural adequacy appropriate for nuclear service under these conditions required the development of new design methodologies and structural criteria. Development efforts were therefore undertaken by the CRBR Project and participants in the U.S. Department of Energy (DOE) Liquid Metal Reactor (LMR) Base Technology program. These efforts generated a substantial structural design basis for elevated-temperature nuclear components.

Although the structural design basis for the CRBRP was extensively developed when the licensing process began in earnest, there were some areas where development was not yet complete. Initial reviews by the Nuclear Regulatory Commission (NRC) and the Advisory Committee on Reactor Safeguards (ACRS) revealed a number of these areas. However, the extent of the weakness on any given issue and its implications with regard to the health and safety of the general public were not (and perhaps could not be) conclusively defined. Disagreements between the Project and the regulatory agencies were thus resolved by negotiations between the principals. As the issues themselves were as much or more matters of judgement than demonstrable fact, the negotiations were conducted in a rather wide arena. Negotiated resolution of structural design issues became a dominant Project licensing activity.

The CRBRP licensing process was not carried through to completion, and severa of the structural design issues were not resolved beyond the stage of a Project commitment to perform additional work in areas of concern. The activities performed and the structural issues raised during the CRBRP licensing proceedings nevertheless possess significant implications with respect to the licensability of future Liquid Metal Reactors (LMRs). It is certain that the regulatory licensing process for such future plants will be based on the CRBRP precedent. As the structural design issues identified in the CRBRP licensing process are now matters of public record which have yet to be resolved, it is unlikely that a future LMR will be licensed for operation until these concerns have been addressed in a meaningful way.

The lessons concerning demonstration of structural integrity for elevated-temperature components learned from the CRBRP licensing experience are documented in this report. The evolution of the structural design basis for the CRBRP is briefly described, the corresponding licensing positions of the Project and the regulatory agencies are outlined, and the gradual shifts in these positions are traced. The structural design issues are defined and their resolution is discussed. Although the CRBRP licensing process may not have been typical, particularly in the area of structural design, it is clear that the identified concerns must be addressed before any future LMR can be licensed for operation.

2.0 EVOLUTION OF HIGH-TEMPERATURE STRUCTURAL DESIGN CRITERIA

Structural design of the nuclear steam supply system portion of commercial nuclear power plants in the United States is based on the criteria of Section III of the ASME Boiler and Pressure Vessel Code (referred to hereinafter as the Code). Design guidance and criteria corresponding to that given in Section III are required for current LMR designs such as the CRBRP, where metal operating temperatures exceed the maximum values permitted by Section III. Applicable structural design guidance and criteria for the elevated-temperature portions of Class 1 pressure boundaries of the CRBRP are given in Code Case N-47, the history and development of which are described elsewhere [1].

The intent of Code Case N-47 is to provide the same level of confidence and assured protection against structural failure as is given by the provisions of Section III. This intent is met by the structural criteria of Code Case N-47, each of which provides protection against the occurrence of a specific mode of structural failure [2]. However, Code Case N-47 is based primarily on the approach of design by analysis, as opposed to the simpler design-by-formula approach offered (if only as an option) in other portions of the Code. As a result, there are few "off-the-shelf" design solutions for elevated-temperature Class I LMR components, and the structural adequacy of such components is very much dependent on the skill and judgement of the designer/analyst.

Not all elevated-temperature portions of an LMR can or should be designed solely in accordance with Code Case N-47. Supplementary and/or alternative structural design guidance and criteria are needed to accommodate special features (e.g., expansion bellows), unusual loading conditions (e.g., overpressure, pre-operational testing) and other classes (Class 2, Class 3) of components. No less than nine Code cases (Table 1) have been developed to supplement Code Case N-47 in particular instances. These cases are, in general, less extensive and less demanding than Code Case N-47, but nonetheless extend the guidance for structural design of high-temperature components found in N-47 to particular instances and applications.

TABLE 1

CODE CASES RELATING TO STRUCTURAL DESIGN OF NUCLEAR PLANT COMPONENTS IN ELEVATED-TEMPERATURE SERVICE

N-47	Class 1 Components in Elevated Temperature Service, Section III, Division 1
N-48	Fabrication and Installation of Elevated Temperature Components, Section III, Division 1
N-49	Examination of Elevated Temperature Nuclear Components, Section III, Class 1
N-50	Testing of Elevated Temperature Components, Section III, Division 1, Class 1
N-51	Protection Against Overpressure of Elevated Temperature Components, Section III, Division 1, Class 1
N-80	Elevated Temperature Piping Penetration Assemblies in Section III, Containment Vessels
N-201	Class CS Components in Elevated Temperature Service, Section III, Division 1
N-253	Construction of Class 2 or Class 3 Components for Elevated Temperature Service, Section III, Division 1
N-257	Protection Against Overpressure of Elevated Temperature Components, Section III, Division 1, Classes 2 and 3
N-290	Expansion Joints in Class 1, Liquid Metal Piping, Section III, Division 1

The Code structural design guidance for elevated-temperature components is further supplemented by standards developed under the cognizance of the U.S. Department of Energy (DOE) for the specific purposes of the U.S. LMR program. Two of these standards, NE F9-4T and NE F9-5T, deal particularly with the subject of structural design criteria [3, 4]. NE Standard F9-4T defines structural evaluation paths and criteria which are specific to LMRs. It should be noted that these modifications to the precepts of Code Case N-47 are minimal in scope and level. The DOE standard is not a replacement for the Code, but a modest supplement. It is simply a way of treating selected, special issues.

Standard NE F9-5T provides guidance for satisfaction of the various provisions of Code Case N-47 and standard NE F9-4T. This standard provides an increased understanding of both the complete structural analysis process and the reasons for specific modeling, analysis, and evaluation recommendations. Such guidance is required by the reliance on (often inelastic) analysis for the structural design of elevated-temperature LMR components and the corresponding dependence on the skill and judgement of the designer/analyst. Moreover, while analysts have a wide range of choices and options within the recommendations of NE F9-5T, those recommendations do eliminate many significant potential differences among individual analyses and thereby enhance the commonality, comparability, and reproducibility of such work.

Structural design guidance and criteria for elevated-temperature LMR components, whether in the Code or supplementary standards, are largely based on results from the U.S. DOE LMR Base Technology Program, and particularly from the materials and structures development work performed as part of the High-Temperature Structural Design Base Technology task. Much of the Base Technology work has been explicitly directed toward development of structural codes and standards and has been rapidly assimilated into design guidance documents. Other work, however, while pertinent to structural design, has been directed toward failure modes and service effects not explicitly covered by design codes and standards. The results of this work have not been translated into explicit design guidance, nor even compiled in any single, complete form. The work is recorded in topical reports, progress reports and other semi-formal documents. This work is not as directly accessible as are existing codes and standards, and the results are not cast in the form of explicit design guidance and criteria. It does, however, significantly extend and supplement existing codes and standards. Issues treated by existing codes and standards may be more extensively addressed with the supplementary information, issues not covered by existing codes and standards may be similarly addressed.

Materials behavior models play a fundamental role in the structural analysis of elevated-temperature LMR components and are therefore central to the design process for these components. A large number of individual aspects of material behavior must, in general, be modeled. The results of any given analysis are dependent (in a very complex and interactive way) on the particular material models used. The use of appropriate, consistent models of pertinent facets of material behavior is thus essential. While some material properties are defined in existing codes and standards, many of the material behavior models required for LMR structural design are defined in the Nuclear Systems Materials Handbook (NSMH), which is maintained for DOE [5]. The material models contained in this handbook are based on analyses of available test data and thus embody the best available technical information at any given point in time. The NSMH also eases the analyst's overall modeling task significantly and enhances the consistency, comparability, and reproducibility of structural analyses.

It is important to recognize that the high-temperature structural design technology developed for the LMR program was not without an initial basis. Guidance for the design and evaluation of structures operating in the elevated-temperature regime (although not in nuclear power service) has long been available in other portions of the Code. Pressurized systems designed and evaluated in accordance with this guidance have given satisfactory service in fossil fuel, petrochemical, and other types of plants. Components for such service are not, of course, required to exhibit the same extreme degree of demonstrable structural adequacy as are those for nuclear service. However, the economic consequences of structural failure (down-time, lost production, repair costs, etc.) are certainly severe enough to motivate significant efforts to assure satisfactory structural service. Non-nuclear service does not imply tolerance of structural failures. The Code design guidance for non-nuclear structures in elevated-temperature service is thus considered an appropriate basis on which to build the more elaborate guidance for elevated-temperature LMR components.

The full range of structural design guidance and criteria applied to the CRBRP and the supporting elevated-temperature structural design technology are relatively new, but some significant service experience has nonetheless been developed with both liquid sodium systems and relatively small-scale LMRs. These earlier systems were designed to different criteria and for different purposes than was the CRBRP, but they did exhibit generally satisfactory structural performance during their design lives. While this experience does not directly confirm the suitability of recent structural design criteria, it does provide practical background to be factored into the design criteria and also demonstrates the fundamental soundness of the basic approach to elevated-temperature structural design which has subsequently evolved.

3.0 PROJECT POSITION ON HIGH-TEMPERATURE STRUCTURAL DESIGN

The position of the CRBR Project with respect to high-temperature structural design was one of confidence at the outset of the licensing process for the CRBRP. Structural design for elevated-temperature service was not a completely new and untried idea, but a proven concept with successful service history in a variety of non-nuclear applications, as well as some which were reasonably prototypic of LMR service. Significant effort had been expended for generation of experimental data and development of that data into guidance for the structural analysis and evaluation of elevated-temperature LMR components. Sufficiency of the design guidance and criteria employed for the CRBRP was further assured by lead experience from two smaller-scale, experimental reactors (EBR-II and FFTF). It was argued that any flaws or omissions in the CRBRP structural design basis would be first revealed by failure indications in a precedent experimental facility.

The material behavior data base developed in support of the CRBRP structural design was quite extensive at the beginning of the licensing process, although further investigations were (and still are) continuing. The scope of this data base is demonstrated by the fact that it is presently large enough to permit the legitimate application of statistical techniques for modeling some aspects of material behavior. Moreover, the design guidance and criteria contained in pertinent portions of the Code and the NE Standards were developed through a group consensus process by a body of knowledgeable experts, composed of representatives from virtually every organization involved in the domestic LMR program. This participatory method, conducted in a public forum, resulted in criteria having a wide base of informed support, significant balance among differing opinions, and a large degree of credibility.

Confidence in the general state of the high-temperature structural design technology used for the CRBRP was significantly enhanced by a number of supplementary considerations. Extensive quality assurance measures were imposed at every step of the procurement, fabrication, and construction processes to prevent any unacceptable deviation of structural materials and installed components from their specified norms. These measures assured that the structural design guidance and criteria would not be inadvertently used beyond their legitimate ranges of application.

Supplementary evaluations of selected critical components of the CRBRP were performed to define quantitative margins against failure beyond the design limits. These margins, even when conservatively evaluated, were typically very large. The margin evaluations provided additional confidence in the structural adequacy of the CRBRP and helped to demonstrate the high degree of conservatism embodied in the structural design guidance and criteria.

In general, the stated position of the CRBR Project with respect to high-temperature structural design was that all potentially significant structural problems had been fully addressed and adequately resolved. The structural adequacy of the CRBRP had been (or could be) assured, to the degree necessary to protect the health and safety of the general public, with a high degree of confidence. This position was, of course, simplistically optimistic to some degree. Successful structural evaluation of all components had not been accomplished, but eventual success was considered virtually certain. Some general problem areas (e.g., thermal striping) and local design features were yet unresolved, but were certainly expected to prove no worse than the problems already surmounted. High-temperature structural design technology was well-developed and well-established. Furthermore, additional development work was being carried out. The confident position of the CRBR Project with regard to this technology was largely justified.

4.0 LICENSING ISSUES

The licensing process for any commercial nuclear power plant, including the CRBRP, involves two independent reviews, one by the Nuclear Regulatory Commission (NRC) and one by the Advisory Committee on Reactor Safeguards (ACRS). The purpose of both reviews is to assure that the design of a nuclear power plant provides adequate (i.e., maximal) protection of the health and safety of the general public. Despite this rather focused common charter, each review raised its own set of concerns with respect to high-temperature structural design technology and its use in the design of the CRBRP.

At the outset of the CRBRP licensing process, the NRC had not been following the progress of the LMR program in any detail for some period of time. The NRC staff was thus not completely familiar with many of the unique considerations involved in high-temperature structural design, LMRs in general, or the specific design of the CRBRP. Faced with the need to evaluate a completely new nuclear plant concept without the usual level of in-house expertise or any established precedent, the NRC was effectively forced to rely on outside consultants.

The first step of the CRBRP licensing process consisted of an NRC review of the Preliminary Safety Analysis Report (PSAR). The results of this review, documented in the Safety Evaluation Report (SER), are a set of individual findings or concerns, for which the NRC required some Project resolution. Of the relatively large number of findings listed in the initial issue of the SER, fourteen were classified as major concerns related to the high-temperature structural design of the CRBRP [6]. These fourteen findings are listed in Table 2; the full text of these findings is given in Appendix A.

The fourteen findings fall into two categories, those related to high-temperature structural design technology and/or its application in general (Findings 1 through 11) and those related to specific components and design features of the CRBRP (Findings 12 through 14). Findings in the latter category merited (and certainly received) a full measure of attention from the Project. The findings in the general category, however, did not simply challenge a single component or design feature, but, in some cases, called

TABLE 2

MAJOR FINDINGS RELATED TO HIGH-TEMPERATURE STRUCTURAL DESIGN

Finding No. 1:	Weldment Safety Evaluation
Finding No. 2:	Leak-Before-Break in Hot Leg Piping
Finding No. 3:	Elevated Temperature Seismic Effects
Finding No. 4:	Safe Allowable Stresses and Strains
Finding No. 5:	Inelastic Design Analysis Methods
Finding No. 6:	Elastic Follow-up in Elevated Temperature Piping Components
Finding No. 7:	Creep Rupture Damage Evaluations
Finding No. 8:	Notch Weakening
Finding No. 9:	Flaw Sensitivity
Finding No. 10:	Creep-Fatigue Evaluation
Finding No. 11:	Acceptable Codes and Standards
Finding No. 12:	Reactor Vessel Transition Joints
Finding No. 13:	Intermediate Heat Transport System Transition Weld
Finding No. 14:	Steam Generator

into question the entire basis for structural design of nuclear components in elevated-temperature service. Design changes to a single component or feature, while likely to prove difficult and expensive, were feasible. Creation of a completely new structural design basis was not feasible. The findings in the general category thus became a key point in further interactions between the CRBR Project and the NRC.

The fourteen original findings were somewhat redundant, and the actions required for resolution of these concerns overlapped to some degree. The Project and the NRC therefore agreed to combine the fourteen findings into nine issues, shown in Table 3. This consolidation reduced the number of individual concerns to a more manageable size and thereby facilitated their resolution. However, the original finding number 12 was expanded from a specific to a general concern in the course of the consolidation process.

Licensing concerns of the ACRS were not stated as explicitly or concisely as those of the NRC, so the two regulatory positions cannot be easily compared on a point-by-point basis. However, a general (if somewhat lengthy) picture of the ACRS position on high-temperature structural design technology and its application to the CRBRP may be obtained from the verbatim transcripts of meetings between the ACRS and the CRBR Project. The ACRS position revealed in these transcripts is somewhat different from that of the NRC.

The ACRS appeared to be generally more comfortable with the state of high-temperature structural design technology and the overall structural design of the CRBRP than did the NRC. Of course, some part of the ACRS attitude may have been due to prior resolution of NRC concerns regarding the structural adequacy of the CRBRP. The ACRS nonetheless seemed to accept that the design guidance used for the CRBRP did provide adequate assurance of structural integrity within the self-imposed limitations of that guidance.

Much of the ACRS interest was focused on matters outside the design basis and margin beyond the design basis, areas where general analysis methods and evaluation criteria for elevated-temperature structures do not exist.

TABLE 3

NRC LICENSING ISSUES RELATED TO HIGH-TEMPERATURE STRUCTURAL DESIGN

Issue 1: Welding Integrity/Leak-Before-Break (Findings 1, 2, 9)

Issue 2: Elevated-Temperature Seismic Effects (Finding 3)

Issue 3: Design Analysis Methods, Codes & Standards (Findings 4, 5, 7, 11)

Issue 4: Elastic Follow-up in Elevated-Temperature Piping and Components Finding 6)

Issue 5: Notch Weakening (Finding 8)

Issue 6: Creep-Fatigue Evaluation (Finding 10)

Issue 7: Plastic Strain Concentration Factors (Finding 12)

Issue 8: Intermediate Heat Transport System Transition Weld (Finding 13)

Issue 9: Steam Generator (Finding 14)

Consequently, ACRS attention was largely focused on the structural features required for core shutdown, heat removal, containment integrity, and other aspects of accident mitigation. In keeping with this focus, ACRS concerns addressed topics not covered by design criteria (e.g., thermal aging, delta-sigma transformation), conservatism and categorization of loading conditions, safety systems and procedures (e.g., loose parts detection and collection, leak detection and proposed reaction to detected leaks, in-service inspection), and special loading conditions (e.g., flow-induced vibrations, thermal striping, flow stratification, fluid-structure interactions).

Although the specific concerns of the ACRS and the NRC regarding elevated-temperature structural design overlap to some degree, the postures of the regulatory agencies were distinctly different. The underlying concern of the NRC was that of demonstrating a sufficient degree of structural adequacy to assure that any structural failure was an event with a very low probability of occurrence. The basic approach of the ACRS was to accept the structural design basis of the CRBRP as being generally sound, but to assure that the design basis had been conservatively applied and that the design was sufficiently flexible and tolerant to mitigate any consequences of error or accident (structural or otherwise) to the degree necessary to protect the health and safety of the general public.

Initial CRBR Project efforts toward resolution of high-temperature structural design licensing issues were directed toward the concerns of the NRC, as this was the first regulatory authority to interact with the Project. In the early stages of this interaction, the Project believed many of the NRC concerns about high-temperature structural design to be largely attributable to past NRC inattention to the breeder reactor program in general and the high-temperature structural design area in particular. The Project considered its structural design licensing stance to be adequately sound, and issues raised to be indicative of difficulties in perception of that stance more than with its credibility.

Based on this assessment of the licensing issues, the initial Project approach to resolution was essentially one of exposition. The purpose of this approach was to familiarize the NRC with both the scope of the high-temperature structural design technology development effort and the application of the results. The approach was implemented with a series of very extensive, extremely thorough presentations to the NRC covering all major topics in the area of high-temperature structural design and a variety of special topics as well [7-13].

The Project's exposition of the structural design base for the CRBRP was certainly warranted under the existing circumstances. The NRC was familiarized with the state-of-the-art in high-temperature structural design and assured that an extensive defensible structural design basis for the CRBRP did actually exist. Further discussions between the NRC and the Project were greatly facilitated, as both groups were dealing with the same body of basic information. The NRC nonetheless remained adamant on the major issues. General information, however extensive, was not accepted as a resolution of these issues, and the Project was forced to address them directly and individually. Resolution of these issues became the crux of the CRBRP licensing process in the area of high-temperature structural design.

5.0 PROJECT/NRC APPROACH TO RESOLUTION

In the course of developing its statement of major issues regarding high-temperature structural design, the NRC outlined a series of Project actions which it would regard as acceptable for resolution of each issue. The scope of the defined resolution effort varied somewhat from issue to issue, but was quite large in several cases. The Project considered the resolution effort excessive and argued that the majority of the work, although of recognized value to the furtherance of high-temperature structural design technology, was not necessary to assure the structural adequacy of the CRBRP. While some of the NRC issues did point to a lack of complete understanding of every detailed structural phenomenon known, suspected, or postulated to occur in elevated-temperature service, the Project argued that the concerns were generic, rather than specific to the design of the CRBRP. The Project contended that, while the existing state-of-the-art for high-temperature structural design technology was incomplete in some respects, a combination of design practices, operating conditions, and system features negated any adverse impact of technology uncertainties on the structural adequacy of the CRBRP.

The Project position on the NRC structural design issues was not without merit in most instances. However, the use of a technology still under active development and the lack of licensing precedent for LMRs were strong inducements to a very conservative regulatory stance. The confidence of the Project in the structural adequacy of the CRBRP was not enough to dispel NRC uncertainties and reservations. Direct and convincing evidence was needed.

It was necessary for the Project and the NRC to develop a mutually acceptable course of action for resolution of NRC concerns. Two different vehicles were employed in this resolution process. In some cases, the Project was able to resolve or largely resolve an issue by providing a very detailed and explicit explanation of analysis and evaluation methods to the NRC staff and consultants. Issues 2, 4, 6, 7, and 8 were addressed in this manner with a fair degree of success. There were, however, some points in these issues which were not completely resolved by the explanation process. Resolution of these points and the remaining issues was accomplished through the alternative approach of confirmatory programs.

Confirmatory programs were intended to be relatively long-term (approximately five-year) development efforts, to which the Project would commit as a condition for issuance of the CRBRP construction permit. All confirmatory programs were to be completed prior to issuance of the CRBRP operating license. Definition of these programs, through negotiation between the Project and the NRC became the focus of the licensing process in the area of high-temperature structural design.

It was fairly apparent at the outset of the negotiation process (and rapidly became more so) that the process would be somewhat unilateral. The Project had no realistic alternative to compliance with NRC dictates. Relatively modest initial Project proposals for confirmationary programs were dismissed as inadequate and replaced by NRC proposals of much larger scope. For example, the Project initially proposed a confirmatory program with a (rather generous) estimated scope of 77 man-months and $20K computer costs for resolution of Issue 1. The NRC rejected this proposal as inadequate and offered a counter-proposal with a scope of 192 man-months and $137K computer costs. In the course of further discussions, the Project negotiated a reduced scope of 122 man-months and $127K for computer costs [14]. The last level of effort was stated to be the minimum effort acceptable to the NRC for resolution of Issue 1. This escalation scenario was repeated for every confirmatory program negotiated.

Several factors contributed to escalation of the confirmatory programs. The issues requiring confirmatory programs for resolution were not well-defined, so there was no clear-cut limit on the specific resolution actions required. Residual points from other issues were occasionally added to existing major confirmatory programs. The single largest factor, however, was simply the issues themselves. The issues for which confirmatory programs were required were directed more toward the development of high-temperature structural design technology than toward specific concerns about the structural adequacy of the CRBRP. The issues were more directed toward the course of future work in the development of high-temperature structural design technology than of resolving specific design concerns. The very broad and general scope of these issues led to extensive confirmatory programs. The result was a series of confirmatory programs which could rival, if not supplant, the DOE High-Temperature Structural Design Base Technology Program in selected areas.

A major deficiency of the confirmatory programs was the open-ended manner in which they were defined. There was no clear statement of completion criteria, deliverables, or measures of success. There was thus no way to firmly establish the specific actions or results which would be mutually accepted as completion of any confirmatory program commitment. The absence of a known end result naturally precluded definition of any planned path to completion.

6.0 FINAL RESOLUTION

Prior to final issue of the CRBRP Safety Evaluation Report (SER), the Project and the NRC had reached agreement on a path for resolution of the NRC issues related to high-temperature structural design. The agreement reached consisted, in most cases, of Project commitments to perform future work, so the issues themselves were not actually resolved. Indeed, there was some question as to just what outcome of the agreed-upon tasks would constitute resolution of a given issue. The agreements were nonetheless considered sufficient resolution of the issues to prevent their appearance as open items in the SER [15].

While resolution of each of the nine major structural design issues was addressed individually in the SER (see Appendix B), the major portion of the resolution effort was concentrated in four major confirmatory programs. Three of these programs, for Issues 1, 5, and 9, are described in the SER; the fourth confirmatory program, for Issue 3, grew out of the NRC actions defined in the SER for resolution of this issue. In most cases, the scope and content of these programs were defined more explicitly and in far more detail than is given in the SER [16, 17]. It is the detailed program definitions which were negotiated and agreed to by the Project and the NRC. While the more general descriptions given in the SER do indeed bound the more detailed explanations, actual performance of the required programs is much more constrained than the general descriptions indicate.

The confirmatory programs required for resolution of Issues 1, 3, and 5 were primarily directed at creation of further advances in the development of high-temperature structural design technology, not at the confirmation of structural adequacy of the CRBRP. Many facets of NRC Issues 1, 3, and 5 had already been recognized as significant issues by the high-temperature structural design community and were already embodied in the long-range plan for the DOE Base Technology program on high-temperature structural design. The issues did not represent new concerns, but rather a revision of development priorities previously defined by general consensus.

The Project recognized that some large part of the confirmatory program requirements was in fact a revision of the DOE Base Technology work and that the required efforts were more appropriate to the scope of the Base Technology program than the specific Project work. These same facts were also apparent to the personnel of the Base Technology program. Attempts to adapt planned and current Base Technology work to satisfy confirmatory program requirements were therefore initiated. Base Technology work which contributed or could be altered to contribute to confirmatory program subtasks were identified [17]. Further pursuit of this approach was, however, forestalled by termination of the CRBR Project.

7.0 IMPLICATIONS FOR FUTURE LMRs

Although LMRs have been previously constructed and operated within the United States, the CRBRP was the first to be subjected to the full licensing process. While that process was not completed beyond the construction permit stage, it was carried far enough to establish some precedents for the licensing of future LMRs. It is certain that the CRBRP experience will be the initial basis for licensing evaluation of the next LMR.

The high-temperature structural design licensing issues raised in connection with the CRBRP, while often directed more toward general technology issues than toward the specific CRBRP design, are now a matter of record. They have not yet been resolved beyond commitments to perform further work. Despite a current enthusiasm for new construction alloys and/or design concepts to resolve these issues, they are far too general to be so simply dismissed. Complete resolution of the high-temperature structural design issues, before the fact or at the time, will be required for licensing of the next LMR.

Current LMR technology development efforts are modest and declining. The technical resources needed to address the high-temperature structural design licensing issues and to resolve those issues with the licensing authorities are significant. Careful planning and sustained focus will be needed to achieve satisfactory resolution of the issues within the existing and expected levels of effort. While an active LMR project will certainly provide the necessary focus at some future point in time, it is prudent to resolve the issues outstanding from the CRBRP before embarking on the licensing process for a new plant.

8.0 CONCLUSIONS

At the outset of the CRBRP licensing process, high-temperature structural design technology was highly developed. Extensive design guidance and criteria were in use, elaborate models of material behavior were defined, and substantial experience with structural (though not necessarily nuclear) service in the elevated-temperature regime was available. There was general consensus among the various corporate and individual parties involved in LMR structural design that the existing technology, while neither all-encompassing nor perfect, was more than sufficient to ensure the structural adequacy of the CRBRP, when applied to the specific CRBRP design conditions.

The CRBR Project entered the licensing process with a high level of confidence in its position on structural adequacy. Some questions were expected from the regulatory authorities. However, the existing technical state of the art, the pertinent plant design features, and the specified plant operating conditions together assured a high margin of safety against any significant structural failure (i.e., one which could imperil public health and safety). The degree of assurance against structural failure was particularly impressive in a prototypic plant.

The issues raised by the NRC came to dominate Project licensing activities in the area of high-temperature structural design. These issues were generally directed at the more vulnerable facets of the technical position on structural design established by the Project. They were, however, more concerned with the state and direction of development for high-temperature structural design technology than with the structural adequacy of the CRBRP per se.

Though not carried through to completion, the CRBRP licensing process illustrates that a sound technical position is certainly a prerequisite to the licensing process, but does not insure ultimate success. In retrospect, the Project may have placed too much confidence in the overall safety of the CRBRP and failed to realize the significance of uncertainties in specific details.

The CRBRP licensing experience also demonstrates that the Project (and, indeed, every first-of-a-kind nuclear licensing applicant) enters the licensing arena at a distinct disadvantage, as there are no precedents to establish areas of concern or means and scope of resolution. Precedent is a significant mechanism for focusing the very broad powers of the regulatory agencies and the attention of the applicant on specific issues. This mechanism is necessary to bound the range of issues raised and establish predictable paths for resolution. If the regulatory authorities cannot be persuaded to abandon an issue, the applicant must comply with whatever regulatory requirements are imposed for resolution unless a precedent has been established. The Project expended substantial amounts of time and effort seeking a path for rebuttal of the structural design licensing issues. There was (and is) no such path. It may have been more efficient for the Project to have simply accepted the inevitable when its nature first became apparent.

Although the Project was able to negotiate some matters (e.g., combination of concerns) with the NRC, it met with only limited success in negotiating confirmatory programs. Continued negotiations often resulted in the Project being ratchetted upwards to an increased scope of effort. It would have been advantageous to the Project to have concluded these negotiations at an earlier stage.

The Project apparently intended the negotiated confirmatory programs to be performed under the Base Technology program. However, Base Technology management was not significantly involved until the programs had been fully negotiated. Earlier and more extensive involvement of Base Technology would have been appropriate.

The licensing process did highlight some known vulnerable points (e.g., weldments, notch effects) in current high-temperature structural design technology. The need for further development in these areas had already been recognized and work was in progress. <u>The actual impact of these points on the structural adequacy of the CRBRP was never fully defined</u>. While further effort in these areas is certainly warranted, it is not clear that such effort should have been defined or pursued in the context of licensing.

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APPENDIX A

LIST OF MAJOR FINDINGS IN HIGH TEMPERATURE AREAS

LIST OF MAJOR FINDINGS IN HIGH TEMPERATURE AREAS

Finding No. 1. Weldment Safety Evaluation

There is a concern for potential cracking problems in weldments of the materials of interest operating at the elevated temperatures of interest. There are a number of important factors which do not appear to have been included in current design practice and Code criteria for weldments for elevated temperature service. The structural integrity of weldments for elevated temperature service has not yet been satisfactorily demonstrated by the Applicant. The following additional factors must be taken into account:

(i) Early crack initiation is known to occur at the inside wall surface in the heat-affected zone (HAZ) where the weldment is exposed to fluid temperature cycling. This should be considered in the fatigue evaluation and flaw-sensitivity evaluation.

(ii) Metallurgical notch effects are much more pronounced at elevated temperature where the creep rates of adjacent materials at the same stress levels may differ by an order of magnitude. These effects do not appear to have been considered.

(iii) The ductility and fracture toughness of the materials may be reduced by the strains imposed during the welding process, cyclic hardening, irradiation effects, carbon migration, carbide formation, sigma phase formulation and other thermal aging effects. The effect of these degraded properties should be evaluated. Some of these effects are being investigated under the ongoing program: "Development Program for Long-term Thermal Aging Effects on CRBRP Welds" as described in WARD-D-0185 Report.

(iv) The strength of the weld metal or HAZ may be lower than that of the base metal. Cracking is common in the fusion zone where the yield strength is relatively high and creep rupture strength is relatively low.

(v) Residual stresses introduced by welding are known to cause damage in the subsequent elevated temperature operation. Residual shear stresses at the weld metal/base metal interface are of particular concern. Triaxial residual stresses are a known contributor to cracking.

(vi) The weld process itself and any stress-relief annealing is known to cause damage. Cooldown between these processes cannot be avoided and should be included in the damage evaluation.

Detailed finite element analysis of weldments including material property variations could be used to answer many of the above questions for CRBR weldments. A more reliable alternative would be to develop a program of tests and analyses which would provide quantitative evaluations of the above open questions for the range of parameters of interest in LMFBR design. Such a program should be designed to obtain these quantitative results prior to plant startup.

In addition, consideration should be given to performing elevated temperature weld prototype testing for the critical weldments in the CRBR plant where additional assurance of structural integrity is needed and where in-service volumetric examinations are not feasible or reliable.

Finding No. 2 Leak-Before-Break in Hot Leg Piping

Leak-before-break has been satisfactorily demonstrated for the cold leg piping. However, leak-before-break in the hot leg piping requires additional justification by the Applicant. WARD-D-0185 provided considerable justification for this important safety concept and included consideration of the relevant factors. Therefore WARD-D-0185 is used in the following description of the items remaining to be resolved.

Three Levels of Mechanistic Evaluation were included in WARD-D-0185:

Section 4.1 ASME Code Evaluation

The use of low temperature elastic stress indices for piping components in the creep regime has not been justified. While there are large safety margins against the Primary stress limits, the Applicant has not shown how other Code safety margins are met using inelastic analyses. The Applicant should commit to use acceptable methods and criteria to satisfy all Code stress and strain limit safety margins prior to issuance of a CP.

Section 4.2 Fatigue Crack Growth

For hot leg piping, initial cracks 25 percent of the wall thickness are shown to grow to only 39 percent of the wall thickness. However, the analyses were based on linear elastic fracture mechanics correlations with test data largely restricted to cyclic conditions which were rapid compared to the times associated with reactor operating cycles. Some experts believe that creep rupture damage associated with hold times will greatly accelerate the crack propagation rate per cycle. Moreover, worst case mean stress corrections should be used in FSAR calculations for crack growth evaluations of weldments because of residual stresses.

Section 4.2 Verification by Test

The actual thermal transient conditions in the plant may result in significant thermal shock stresses at the inside wall surface of elbows, piping and components. Such stresses are conducive to the initiation and propagation of long shallow surface cracks in the plant hardware. The elbow tests tended to include higher linearized stresses associated with forcing the cross section into an elliptical shape. Axial bending tests on elbows and piping tend to produce leaks associated with tearing before gross rupture. Bursting tests do the same. Thus, leak-before-break was more likely to occur in the Applicant's laboratory tests than in the actual plant hardware.

Long circumferential cracks tend to be initiated very early in heat-affected zones (HAZ) of weldments exposed to cyclic fluid temperatures. Such cracking has been observed in nonnuclear hardware constructed of materials and subjected to temperatures and loading conditions similar to CRBR. Similar results have been obtained in laboratory specimen testing and in component testing. Such cracks are less likely to leak before a break occurs than cracks which propagate from local defects. Leak-before-break is an important safety concept; and, additional technical justification is needed to verify the concept for weldments (see Finding No. 1) in Elevated Temperature Service. NRC will evaluate the acceptability of the criteria which will be used by the Applicant to determine whether this concept can be adequately demonstrated for the design, fabrication and operating conditions of interest. Acceptable criteria for such a justification is required prior to issuance of a CP if the Applicant intends to rely on the leak-before-break safety concept for the hot-leg piping.

Finding No. 3 Elevated Temperature Seismic Effects

Code Case 1592 imposes limits on various inelastic strains accumulated within the life of a component. The life history is described by grouped cycles of certain limited intensities. The consequence of varying the loading sequence is not important below the creep regime, and stresses are classified into stress-controlled primary and strain-controlled secondary values. These stress values are then used to perform structural analyses of the cyclic life of the structure.

Seismic events impose high short-term primary stresses on the structure. The seismic loads affect the inelastic strain accumulation by changing the residual stresses which produce enhanced creep. Seismic loads also produce plastic strain accumulation generated within each motion if the intensity of the shake is high enough to cause plastic ratcheting. The relaxation of high residual stresses which exist after a seismic event produces enhanced creep during subsequent elevated temperature operation. Consequently, the sequence of loading becomes important in the creep regime.

The Applicant should commit to use acceptable methods and criteria to account for creep enhancement of strain accumulation and creep rupture damage due to potential realistic sequences of seismic events prior to issuance of a CP.

For elevated temperature OBE seismic loads are evaluated based on code limits for upset conditions. For these conditions Code evaluation include limits on primary stresses based on y time-independent S_m and time-dependent S_t allowables. Since the duration of the seismic event is very short, the seismic stresses are generally not limited by the corresponding high S_t allowables. Seismic stresses are therefore usually limited only by the time-independent S_m allowables determined for virgin materials. The S_m value is related to the yield and ultimate strength of the material obtained in a conventional tensile test. These properties can be reduced by prior creep rupture damage and thermal aging under stress during long operation at elevated temperature. Thus, the design safety margin is reduced if the seismic event occurs at the end of component life. Applicant should commit to use acceptable methods and criteria to effects prior to issuance of a CP and to apply these methods and criteria in the FSAR.

Finding No. 4 Safe Allowable Stresses and Strains

A major consideration in elevated temperature structural integrity is the complex interface between engineering design and materials considerations. There are several such areas where the applicant has not yet provided adequate justification of structural integrity. These are:

(i) Heat transport system structural materials data show large heat-to-heat variations in properties. The data also show that room temperature tensile data does not correlate well with the elevated temperature properties. Since room temperature test data was used as a basis for material acceptance when purchased for components, the applicant needs to provide additional assurance that the material has the minimum creep properties used for designing critical structures, systems and components where such properties are essential to structural integrity. Where elevated temperature properties were measured, the results should be described. European material specifications are considerably tighter than the U.S. practice. The Applicant should indicate where tighter specifications than required by Code were used.

(ii) Thermal aging accompanied by service-induced strain can cause a very significant reduction in ductility and fracture toughness. The Applicant should provide assurance that such effects were taken into account when evaluating the structural integrity of all elevated temperature equipment.

(iii) Carbon surface redeposits may introduce a brittle surface condition conducive to crack initiation. The Applicant needs to demonstrate that this will not result in unacceptable cracking.

(iv) Austenitic materials with grain boundary porosity pass all acceptance specifications but have very low creep fatigue strength and very low ductility in the creep regime. The Applicant needs to provide assurance either that there will not be any of this material in the plant or that it will not compromise minimum safety margins where it could exist in the plant.

(v) Where the loading conditions are so severe that the structure or component cannot meet the basic bounding Tests of Code Case N-47 and detailed inelastic analysis has to be used to pass the Code, justification for use of the methods of RDT-Standard F9-5T is required. The latter methods have not yet had the benefit of independent review as National Consensus Standards or NRC review. NRC will review RDT Standards F9-4T and F9-5T and will inform the Applicant of any revisions or further technical justification which may be required based on the results of this review.

(vi) Anywhere the nonmandatory criteria of Code Case 1592 (now Code Case N-47) are not used, and alternate criteria are used per agreement with the owner as allowed by the Code, such alternate criteria must be justifed since they will not have had the benefit of independent review as National Consensus Standards.

Finding No. 5 Inelastic Design Analysis Methods

The preliminary Code evaluation of WARD-D-0185 Report, "Integrity of Primary and Intermediate Heat Transport System Piping in Containment," has been based on elastic analyses. For some locations the results of elastic analyses given in the report do not satisfy Code limits. Moreover, the Code does not have any applicable methods for discontinuities. In some cases accumulated inelastic strains are evaluated using the simplified method given in Item 6.2 of RDT-5T Standard for insignificant creep. The Standard limits application of this method only by the conditions that the maximum metal temperature is always below the value corresponding to the point where $S_m > S_t$ for 10^5 hours. For hot-leg piping this condition is satisfied. However, this condition is not sufficient to ensure that the creep effects are insignificant. Primary membrane plus bending stresses are allowed to reach 1.5 S_m but are limited to 1.25 S_t or less.

The Applicant in the report, WARD-D-0185 indicates that full inelastic analysis will be employed for locations where elastic analysis results do not meet Code limits. These locations should be listed by the Applicant. The Applicant should also describe the logic and sequence of events applied to establish the conservative duty cycle for such analyses. Inelastic analysis is usually performed for a limited number of cycles and then the results are extrapolated for the entire life of the component. Methods of extrapolation should be described. Inelastic analysis introduces consideration of strain hardening, cyclic hardening and thermal annealing, creep hardening and thermal cycling. Using strength properties which are too weak may underpredict creep rupture damage. Using strength properties which are too high may underpredict ratcheting. Thus sensitivity of the analysis to the material characteristics should also be evaluated.

RDT Standards F9-4T and F9-5T, which were used by the Applicant for performing inelastic design analyses will be reviewed by NRC. The Applicant will be advised of any revisions or further technical justification which may be required based on the results of this review.

Finding No. 6 Elastic Follow-up in Elevated Temperature Piping and Components

At elevated temperatures, during creep relaxation, a portion of the elastic strain is converted to creep strain. Higher-stressed areas are subjected to additional strain accumulation due to elastic follow-up from structurally coupled adjacent elements of the component. Thus in order to provide for adequate safety margins, the Code requires that stresses with elastic follow-up be classified as primary. The Applicant, however, performed an inelastic analysis for the thermal loading to show that the thermal stresses can be classified as secondary. Thus in the Applicant's definition there is no follow-up since there is no essential redistribution of resultant moments and the inelastic analysis accounts for local strain concentrations.

An open question remains as to how the Applicant's method satisfies the safety margins of the Code for superposed mechanical and thermal loading. The Applicant should provide the method and criteria used to account for creep redistribution of stress in the piping system and the criteria for including the redistributed loads in combination with the elastically calculated loads in sizing the supports for NRC review and concurrence.

Finding No. 7 Creep Rupture Damage Evaluations

Creep rupture damage at stress raisers was evaluated by the ratios of the time at stress to the minimum time to rupture at that stress. Since the elastically-calculated thermal stresses at stress raisers are well above yield, the yield strength properties were used to calculate local stresses. Average rather than minimum yield strength values were used to evaluate creep rupture damage per RDT Standard F9-4T so as not to underestimate the stresses and damage. However, cyclic hardening can more than double the yield strengths of austenitic materials, thereby increasing the local stresses and creep rupture damage. Since creep rupture damage is such a highly nonlinear function of stress, the damage occurring after cyclic hardening can be orders of magnitude higher. At all locations where the local cyclic stresses exceed yield, these effects should be included in the creep rupture damage evaluation. The Applicant should commit to the use of acceptable criteria for evaluating this damage prior to issuance of a CP.

Finding No. 8 Notch Weakening

The basic allowable stress limits of the Code are based on unnotched creep specimen test data. Stress raisers influence the creep behavior of the entire wall in two basic ways. They introduce a constraint against inelastic flow by inhibiting slip line development. This is manifested in a reduction in the average stress intensity in the net section (a notch strengthening effect). Stress raisers also introduce a site where creep rupture damage could cause early crack initiation and more rapid crack propagation (a notch weakening effect). While the combined effect is notch strengthening in most cases, an evaluation is needed to determine what geometric, loading and material parameters could cause significant notch weakening, particularly for long-term loading at elevated temperature. Loading conditions such as transverse shear do not introduce any notch strengthening and have contributed to weldment cracking at structural discontinuities.

Applicant should commit to an acceptable program for conducting a parametric study of geometric notches, loading conditions and material properties within the range of interest in LMFBR design is needed to quantify the extent and seriousness of the problem prior to issuance of a CP. This study should examine long-term loadings where the material ductility may be minimized by prior cyclic and monotonic straining and thermal aging. Geometric configurations with low inelastic flow constraint and high local stress concentrations should be considered in this evaluation.

Finding No. 9 Flaw Sensitivity

The flaw sensitivity of the hardware should be evaluated considering the reduced fracture toughness and ducitility due to temper embrittling, carbon surface deposits, prior creep rupture damage and irradiation effects where relevant.

At weldments, early crack initiation should be anticipated in the heat-affected zones on surfaces exposed to cyclic sodium temperatures. The subsequent crack propagations should be evaluated considering "worst case" mean stress effects to account for residual welding stresses. The effects of creep rupture damage during hold-times at temperature should be included.

Surface crazing cracks caused by thermal striping has recently been found to be more deleterious than previously anticipated. While the crazed surface acts as insulation, a small number of cracks continue to propagate, essentially acting as isolated deep cracks without benefit of the thermal stress-relieving benefits from neighboring cracks. The equipment must be shown to be safe with this type of behavior.

The Applicant should commit to an acceptable program to evaluate flow sensitivity prior to issuance of a CP.

Finding No. 10 Creep-Fatigue Evaluation

The applicant has modified the creep-fatigue damage rules given in Code Case 1592 when applied to austenitic stainless steels SS304 and SS316.* These rules assume that in compressive hold, the creep damage is 20% as damaging as that caused by the same sustained stress in tension. The Applicant states that test evidence shows that compressive stresses have little damaging effect for austenitic stainless steels. By the same token, however, there are some studies that indicate shear stress to be a good creep rupture criteria for stainless steels. If this were the case, compressive stresses would cause more damage than that obtained by the Applicant's modified rules. Thus, the Applicant should provide some justification for arriving at the 20% factor.

Stainless steel materials subjected to high cycle thermal fluctuations and flow-induced vibrations require a fatigue strength evaluation beyond the CC 1592 curve limit of 10^6 cycle. Thus for high cycle fatigue evaluation of stainless steels beyond the Code Case limit of 10^6 cycles, the Applicant has extrapolated the fatigue curve using a slope of -0.12 on cycles for load-controlled situations and developed special purpose high-cycle fatigue criteria for strain-controlled situations. For temperatures below 800°F, the Code adopted a new high cycle fatigue design curve up to 10^{11} cycles in 1982. The allowable stress for 10^9 cycles is reduced to 14,000 psi, whereas previously the Code did not go beyond 10^6 cycles where the allowable stress was 28,200 psi. Similar data beyond 10^6 cycles also is available for temperatures above 800°F. The Applicant should confirm that his criterion is in agreement with the new data.

For 2-1/4Cr-1Mo new fatigue design curves which account for environmental effects have recently been approved by Code Committees. The Applicant should ensure that his criteria are consistent with the new curves.

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*only for components which are not Code stamped

Finding No. 11 Acceptable Codes and Standards

The general criteria of 10 CFR 50 and the "CRBRP Preliminary Design Criteria," Site Suitability Report DOCKET 50-537, dated March 4, 1977, were used as the basis for this review. The PSAR for the CRBRP has been written following the Standard Review Plan (SRP) for Light Water Reactors (LWR). The SRP for LWRs, however, has no review procedures and acceptance criteria that are applicable for elevated temperature equipment where creep is governing. The only national consensus or NRC approved Codes and Standards are the ASME Code Cases 1592-3, -4, -5, and -6 for Components in Elevated Temperature Service and Regulatory Guide 1.87, "Guidance for Construction of Class 1 Components in Elevated Temperature Reactors." The Applicant is not required to incorporate Code changes after the 1974 Addenda to the ASME Code. Numerous revisions to Code Case 1592 for Class 1 Components in Elevated Temperature Service have since been made and are included in the current version of Code Case N-47, which is the successor to Code Case 1592. The Applicant should determine what areas of elevated temperature systems and components do not meet current N-47 Code Case requirements, identify these areas, and provide justification that these areas satisfy the general safety margins of the ASME Code.

The time-dependent response of structural materials in the creep regime increases the complexity of design by an order of magnitude. The formulation of constitutive relations for the plastic and creep properties, including hardening effects, required extensive materials' testing and the development of reliable models. Such work has been carried out in the base technology program of the Department of Energy and used to develop RDT Standards F9-4T and F9-5T.

RDT Standard F9-4T "Requirements for Construction of Class 1 Elevated Temperature Nuclear System Components," and RDT Standard F9-5T, "Guidelines and Procedures for Design of Class 1 Elevated Temperature Nuclear System Components," have not had the benefit of independent review as national consensus standards nor review by NRC.

Therefore they could not be treated as validated acceptance criteria for the conduct of this review. NRC is conducting detailed review of these Standards prior to issuance of the CP and will inform the Applicant of any revisions or further technical justification which may be required in the design analysis methods, constitutive relations or design acceptance criteria therein.

Component Specific Major Findings

Finding No. 12 Reactor Vessel Transition Joints

The Reactor Vessel transition joints were analyzed by the Applicant in accordance with the ASME Section III criteria. The peak local stress intensity range including the stress concentration factor exceeds 3 S_m for the lower reactor vessel transition weld, which was analyzed in accordance with NB3228.3 of the Code. However, the Code includes no plastic strain concentration effects and is not conservative when the nominal stress range is under 3 S_m and the local stress range exceeds twice yield. In this important weldment, the Applicant should commit to the use of acceptable criteria to account for inelastic strain concentrations when the peak stress range exceeds twice yield prior to issuance of a CP.

Component Specific Major Findings

Finding No. 13 Intermediate Heat Transport System Transition Weld

The Intermediate Heat Transport System (IHTS) transition joints were analyzed in accordance with the ASME Code and applicable RDT Standards. A detailed inelastic analysis showed that the hot joint could meet the ASME Code criteria for only fifteen-year life. Also, the Applicant's conclusion is based on an anticipated minimum carbon content which does not fall below 0.05%. This joint is expected to see 936°F. The reviewer's comments under Finding No. 1, Weldment Safety Evaluation, apply. In addition, the variation in expansion properties between the different materials may be critical and the difference in properties should be carefully examined. Unless the Applicant can provide assurance that all of the carbon cannot be depleted from the worst cross-section, the structural integrity with zero carbon should be examined. The creep rupture damage may be higher than expected due to the higher yield properties produced by hardening in a multi-pass welding process.

The Applicant should provide an acceptable plan regarding justification of the joints for thirty years' service or for replacement after fifteen years prior to issuance of CP.

Component Specific Major Findings

Finding No. 14 Steam Generator

The present steam generator/superheater design differs from that shown in the PSAR. The changes and the impact they have on the safety of the system will be evaluated as soon as the required detailed information is received from the Applicant.

Large thermal stresses arise in the outer region of the perforated area of the steam generator tubesheet adjacent to the rim. Creep rupture damage combined with fatigue due to relaxation of high residual stresses limits the life of the component. The ASME Code does not provide acceptance criteria for the design of the perforated plates in elevated temperature service.

The tube-to-tubesheet joint is a critical location in the steam generator for which there are no approved acceptance criteria and evaluation methods.

Accelerated proof testing of scale models has been proposed to verify the structural integrity of various elements of the steam generator. Since creep is a major concern in certain areas, the thermal and mechanical loading used must be carefully selected to provide the proper load and time ratios. The test plans and the justification for the adequacy of the proposed testing will be reviewed by NRC.

NRC is planning to provide a position on the acceptance criteria for:

(1) perforated tubesheet operating in elevated temperature service,
(2) tube-to-tubesheet joints, and
(3) accelerated scale model testing.

NRC will clarify acceptable criteria prior to issuance of a CP.

APPENDIX B

HIGH-TEMPERATURE STRUCTURAL DESIGN CONFIRMATORY PROGRAMS DEFINED IN THE CRBRP SAFETY EVALUATION REPORT

3.9.9 Elevated-Temperature Mechanical Integrity

3.9.9.1 Area of Review

The CRBR systems, components, and supports operating at elevated temperature were reviewed with regard to potential failure modes. Design for operation at elevated temperatures requires consideration of additional failure modes, the increased cyclic thermal loads associated with the higher temperatures, and the reduced material strength and ductility in the creep regime. Time-dependent nonlinear response of the material at elevated temperatures must be considered in the analysis.

Systems and components in service at elevated temperatures are subjected to larger temperature variations and differentials than LWR hardware. Moreover, the materials have lower strength at elevated temperatures. The resulting higher thermal strain ranges and increased inelastic strain concentrations tend to accelerate fatigue damage. In addition, the materials are susceptible to creep-rupture damage that results from both applied and residual stresses persisting after transient conditions. Relaxation of such stresses tends to cause ratcheting on subsequent load cycles. The effective microscopic ductility of many of the materials and product forms is reduced by concentration of creep strains in grain boundaries. Consequently, cracking can occur at accumulated strain levels that would cause no problems at temperatures below the creep regime.

The review covers the mechanical engineering aspects of safety for CRBR systems, components, and supports that will operate at elevated temperatures are are included in Chapters 3, 4 (portion), and 5 of the PSAR.

3.9.9.2 Review Evaluation and Acceptance Criteria

3.9.9.2.1 Finding No. 1--Weldment Safety Evaluation Confirmatory Program Required

Potential cracking problems in weldments of the materials of interest operating at the elevated temperatures of interest are a cause for concern. A number of important factors apparently have not been included in the CRBR application for weldments in service at elevated temperatures. The structural integrity of weldments in service at elevated temperatures has not yet been satisfactorily demonstrated by the applicants. The following additional factors must be taken into account:

A. Consideration of crack initiation in the heat-affected zone (HAZ) of the weldment exposed to cyclic sodium temperatures at the inside surface.

B. Consideration of the creep-fatigue and creep-rupture damage peculiar to the material property variations or metallurgical notch effects at weldment.

C. Consideration of time rate, cyclic rate, and hold-time effects on the HAZ of the weldment in the presence of long shallow cracks.

D. Consideration of the enhanced creep in the remaining uncracked wall thickness caused by residual stresses and thermal cycling.

E. Evaluation of stability of remaining uncracked wall ligament for operation in the creep regime.

A confirmatory program of test and analyses is required to provide quantitative evaluation of the above open questions for the parameters of interest in the CRBRP. Quantitative results are required before an operating license is issued.

Resolution

Resolution consists of the applicants carrying out a program to confirm the structural adequacy of the CRBRP design with regard to weldment integrity. This program will encompass the major elements described herein to quantify the safety margins of the weldments in service at elevated temperatures and hot-leg piping.

The applicants in their February 10, 1983 letter (J. R. Longenecker to J. N. Grace) agreed to provide additional confirmatory testing during the OL review. The details of the confirmatory testing program have not been finalized but the staff has provided a typical confirmatory testing plan in EG&G Report EA-6150 (January 1983). The staff finds this acceptable for the CP review.

3.9.9.2.2 Finding No. 2--Elevated-Temperature Seismic Effects

ASME Code Case 1592 imposes limits on various inelastic strains accumulated within the life of a component. The life history is described by grouped cycles of limited intensities. The consequence of varying the loading sequence is not important below the creep regime, and stresses are classified into stress-controlled primary and strain-controlled secondary values. These stress values are then used to perform structural analyses of the cyclic life of the structure.

Seismic events impose high short-term primary stresses on the structure. The seismic loads affect the inelastic strain accumulation by changing the residual stresses that produce enhanced creep. Seismic loads also produce plastic strain accumulation generated within each motion if the intensity of the shake is great enough to cause plastic ratcheting. The relaxation of high residual stresses that exist after a seismic event produces enhanced creep during subsequent operation at elevated temperatures. Consequently, the sequence of loading becomes important in the creep regime.

Resolution

The applicants are committed to take into account any enhanced creep (ratcheting) and any creep-rupture damage resulting from residual stresses at local stress raisers following seismic events. This necessarily includes consideration of the sequence of the seismic events with respect to the operating transients. Since the methods used by he applicants have been supplemented by DOE's RDT, F9-4T, "Requirements for Construction of Class 1 Elevated Temperature Nuclear Systems Components," and F9-5T, "Guidelines and Procedures for Design of Class 1 Elevated Temperature Nuclear System Components," this issue is considered resolved with the NRC review of RDT F9-4T and F9-5T as described in Section 3.9.9.2.3, and the resolution of any relevant findings resulted from the review.

3.9.9.2.3 Finding No. 3--Design Analysis Methods, Codes and Standards (Open--Subject to NRC Review of RDT F9-4T and F9-5T Design Methods and Criteria and Resolution of Findings by Applicants)

The CRBRP Principal Design Criteria were used as the basis for this review. The PSAR for the CRBRP has been written following the Standard Review Plan (SRP) for light-water reactors (LWRs). The SRP, however, contains no review procedures and acceptance criteria that are applicable for equipment in service at elevated temperatures where creep is occurring. The only national consensus of NRC-approved codes and standards are ASME Code Cases 1592-3, -4, -5, and -6 for components in service at elevated temperatures and RG 1.87, "Guidance for Construction of Class 1 Components in Elevated Temperature Reactors." However, numerous revisions to Code Case 1592 for Class 1 components in service at elevated temperature have been made and are included in the current version of Code Case N-47, which is the successor to Code Case 1592 which was used by the applicants.

Creep-rupture damage at stress raisers was evaluated by the ratios of the time at stress to the minimum time to rupture at the stress. Since the elastically calculated thermal stresses at stress raisers are well above yield, the yield strength properties were used to calculate local stresses. Average rather than minimum yield strength values were used to evaluate creep-rupture damage according to RDT F9-5T so as not to underestimate the stresses and damage. However, cyclic hardening can more than double the yield strengths of austenitic materials, thereby increasing the local stresses and creep-rupture damage. Since creep-rupture damage is such a highly nonlinear function of stress, the damage occurring after cyclic hardening can be orders of magnitude higher. These effects should be included in the creep-rupture damage evaluation at all locations where the local stress exceed yield.

The preliminary Code evaluation of report WARD-D-0185, "Integrity of Primary and Intermediate Heat Transport System Piping in Containments," is based on elastic analyses. For some locations the results of elastic analyses given in the report do not satisfy Code limits. Moreover, the Code does not have any applicable elastic analysis criteria for discontinuities. In some cases accumulated inelastic strains were evaluated using the simplified method only by the condition that the maximum metal temperature is always below the value corresponding to the point where $S_m > S_t$ for 10^5 hours. For the hot-leg piping this condition was satisfied. However, this condition is not as limiting as ASME Code Cae N-47 wherein primary membrane plus bending stresses are allowed to reach 1.5 S_m but are limited to 1.25 S_t.

The applicants in the PSAR and in the report WARD-D-0185 indicate that full inelastic analysis will be used for locations where elastic analysis results do not meet Code limits. The inelastic analysis will be performed in accordance with the RDT F9-5T. Since both RDT F9-4T and F9-5T have not had the benefit of independent review as national consensus standards nor review by the staff, they could not be treated as validated acceptance criteria for the conduct of this review.

Resolution

The proposed resolution consists of the following actions:

A. The applicants commit to keep abreast of the developing design technology for operation at elevated temperatures and to assess the potential safety implications of new developments for CRBRP.

B. The staff will conduct a review of RDT F9-4T and F9-5T and will identify any revisions or further technical justification that may be necessary to meet national consensus and NRC safety standards. On the basis of an initial review, the applicants' calculated creep-rupture damage may be too low when compared with the considerable strain and cyclic hardening that occurs during fabrication and operation. The applicants' calculated fatigue damage and accumulated strains may be too low if the actual yield strength will be below the average value used in the design analyses.

3.9.9.2.4 Finding No. 4--Elastic Follow-up in Elevated-Temperature Piping

At elevated temperatures, during creep relaxation, a portion of the elastic strain is converted to creep strain. Areas of piping that are more highly stressed are subjected to additional cyclic strain and strain accumulation resulting from elastic follow-up. To provide for adequate safety margins, the Code requires that stresses with elastic follow-up be classified as "primary stresses." The applications, however, performed an inelastic analysis for thermal loading and did not include any portion of the thermal expansion stresses as primary. This approach circumvents the demonstration of adequate safety margins for creep-rupture and creep-fatigue damage.

Resolution

The staff and the applicants agreed to satisfactory methods and criteria for quantifying thermal expansion stresses in piping systems that must be considered to be primary (meeting of November 22, 23, and 24, 1982 held at Westinghouse-ARD). The resolution of this item consists of the applicants using these methods and criteria.

3.9.9.2.5 Finding No. 5--Notch Weakening

The basic allowable stress limits of the Code are based on unnotched creep specimen test data. Stress raisers influence the creep behavior of the entire wall in two basic ways. They introduce a constraint against inelastic flow by inhibiting slip line development. This is manifested in a reduction in the average stress intensity in the net section (a notch strengthening effect). Stress raisers also introduce a site where creep-rupture damage could cause early crack initiation and more rapid crack progagation (a notch weakening effect). Although the combined effect is notch strengthening in most cases, an evaluation is needed to determine what geometric, loading, and material parameters could cause significant notch weakening, particularly for long-term loading at elevated temperatures. Loading conditions such as transverse shear do not introduce any notch strengthening and have contributed to weldment cracking at structural discontinuities.

The applicants should commit to an acceptable program for conducting a parametric study of geometric notches, loading conditions, and material properties in the CRBRP design that is needed to quantify the extent and seriousness of the problem before a construction permit is issued. This study should examine long-term loadings where the material ductility may be minimized by prior cyclic and monotonic straining and thermal aging. Geometric configurations with low inelastic flow constraint and high local stress concentrations should be considered in this evaluation.

Resolution

Resolution consists of the applicants carrying out a program to confirm their creep-fatigue and creep-rupture damage criteria for geometric notches, including local stress and strain concentration effects at structural discontinuities. The applicants in their February 10, 1983 letter (J. R. Longenecker to J. N. Grace) agreed to provide additional confirmatory testing during the OL review. The details of the confirmatory the testing program have not been finalized but the staff has provided a typical confirmatory testing plan in EG&G report (EA-6150, January 1983). The staff finds this acceptable for the CP review.

3.9.9.2.6 Finding No. 6--Creep-Fatigue Evaluation

The applicants have modified the creep-fatigue damage rules given in Code Case 1592 when applied to austenitic stainless steel types 304 and 316 for components that are not Code stamped. These rules assume that the compressive hold, the creep damage is 20% as damaging as that caused by the same sustained stress in tension. The applicants have presented test data showing that compressive stresses have little damaging effect for austenitic stainless steels. By the same token, however, there are some studies that indicate that shear stress is a valid creep-rupture criterion for stainless steels. If the latter were the case, compressive stresses would cause more damage than that obtained by the applicants' modified rules. Thus, the applicants should provide documented justification for arriving at the 20% factor, rather than using 100% damage as required by the Code.

Stainless steel materials subjected to high cycle thermal fluctuations and flow-induced vibrations require a fatigue strength evaluation beyond the Code Case 1592 curve limit of 10^6 cycles. Thus, for the high cycle fatigue evaluation of stainless steels beyond the Code Case limit of 10^6 cycles, the applicants have extrapolated the fatigue curve using a slope of -0.12 on cycles for load-controlled situations. The applicants have also developed special purpose high cycle fatigue criteria for strain-controlled situations. For temperatures below 800°F, ASME Code Committees adopted a new high cycle fatigue design curve up to 10^{11} cycles in 1982. The allowable stress for 10^9 cycles is reduced to 14,000 psi, whereas previously the Code did not go beyond 10^6 cycles, where the allowable stress was 28,200 psi. Similar data on cycles beyond 10^6 also are available for temperatures above 800°F. The applicants should confirm that Code safety margins are met with the new fatigue design curves as extended to 10^{11} cycles.

For 2-1/4 Cr-1Mo new fatigue design curves that account for environmental effects have recently been approved by ASME Code Committees. The applicant should ensure that Code safety margins are met with the new fatigue design curves.

Resolution

The applicants have provided sufficient technical justification to resolve this issue but have not yet submitted the required documentation of:

A. The use of the 20% (or less) damage factor for compressive hold times in types 304 and 316 stainless steels.

B. The evaluation of the effects of the reduced fatigue design curves for austenitic materials cycled beyond 10^6 cycles, which will be issued in the Winter 1982 Addenda of the ASME Code.

C. the evaluation of the effects of the 2-1/4 Cr-1Mo elevated-temperature fatigue design curves currently proposed by ASME Code Committees.

This issue is considered resolved subject to the applicants' completion of the required documentation.

3.9.9.2.7 Finding No. 7--Plastic Strain Concentration Factors

This issue concerns the use of the plastic strain concentration factors, K_e, in performing fatigue evaluations. The simplified methods of the ASME Code, used by the applicants (e.g., in the core support structure--support cone weld analysis), allow this factor to be unity until the primary plus secondary stress range exceeds 3 S_m. Actually, this factor begins to exceed unity when the local maximum stress range, including the elastic stress concentration factor, exceeds 2 S_y. Moreover, strain multipliers for the concentration of plastic strain on the weaker side of a product form or materials interface is not included in existing formulas for Ke in the Code. The lack of conservatism in the simplified elastic-plastic method of the ASME Code has been pointed out in the published literature.

Resolution

The applicants are committed to performing a satisfactory evaluation of actual or conservative plastic strain concentration effects and the resulting fatigue design life wherever the local maximum stress range exceeds 2 S_y.

3.9.9.2.8 Finding No. 8--Intermediate Heat Transport System Transition Weld

The transition joints of the intermediate heat transport system (IHTS) were analyzed in accordance with the ASME Code and applicable RDT standards. A detailed inelastic analysis showed that the hot joint could meet the ASME Code criteria for only a 15-year life. Also, the applicants' conclusion is based on an anticipated minimum carbon content that does not fall below 0.05%. This joint is expected to see 936°F. The variation in expansion properties between the different materials may be critical, and the difference in properties should be carefully examined. The increased creep-rupture damage resulting from the higher yield properties produced by hardening in a multipass welding process should be evaluated.

Resolution

Resolution consists of the applicants commitment to perform analyses using the methods and criteria to be developed in the confirmatory program described in Section 3.9.9.2.1 in order to evaluate the structural integrity of the critical IHTS transition joints for 30 years' service. If these transition welds cannot be shown to be adequate for 30 years' service, the applicants must provide an acceptable plan for earlier replacement before an operating license is issued.

3.9.9.2.9 Finding No. 9--Steam Generator

The steam generator design is supported by existing test results and by several planned tests which are outlined in the PSAR. Tests of mechanical properties are included in the planned program to verify and supplement methods in the ASME Code and RDT standards and design information for ensuring the structural adequacy of the steam generator. Prototype and other steam generator tests are included in the program. These test programs are designed to verify assumptions and to provide quantitative data to confirm the adequacy of design analyses.

Simplified elastic design methods for steam generator tubesheets are given in Section III of the ASME Code. These methods are only applicable where plastic deformations are not significant and temperatures remain below the creep regime. The CRBR steam generator tubesheets will be subjected to severe thermal cycling beyond the elastic regime. Significant thermal stresses will arise in the outer region of the perforated area of the steam generator tubesheet adjacent to the rim. Creep rupture damage combined with fatigue resulting from relaxation of high residual stresses limits the life of the component. The ASME Code does not provide similar simplified methods for the design of perforated plates in service at elevated temperatures.

The applicants have stated that Code Case 1592 supplemented by RDT F9-4T will be used for the tubesheet. Although the general criteria for elevated-temperature design may be found in Code Case 1592, the application of these criteria to the three-dimensional stress variations in ligaments operating at elevated temperatures is difficult. The steam generator tubesheet is a complex and unique structure that requires special design analysis methods to handle creep effects. Elastic follow-up is known to occur in ligaments subjected to cyclic straining under creep conditions. Thermal stresses with elastic follow-up must be classified as "primary stresses."

During thermal transients the temperature of the ligaments closely follows the temperature of the fluid in the tubes. The thermal response of the unperforated rim, however, is significantly delayed. Significant in-plant stresses arise between the rim and perforated part of the tubesheet. Moreover, the transient results in severe radial temperature gradients at the interface the rim and perforated part of the tubesheet. Thus, the outer ligaments remain at significantly higher temperatures than those in the remaining area of the tubesheet. These higher temperatures result in lower strength, which makes the outer ligaments potentially more vulnerable to inelastic deformation. Contraction of the perforated portion of the tubesheet may be sufficiently high to cause unacceptable inelastic flow in the outer ligaments. Thus, special analysis methods are needed to bound the deformation, strain ranges, a maximum stress in order to obtain reliable fatigue and creep-rupture damage evaluations.

The applicants have indicated that a detailed inelastic finite element analysis will be performed for a sector of the tubesheet. However, difficulties arise with such an analysis in the modeling of ligaments because of the complex thermal structural interaction with the rim and the tubes. The elastic response of the perforated plate is isotropic; the inelastic response is anisotropic. Moreover, the tubesheet being a discrete structure is particularly sensitive to the use of minimum versus average properties for inelastic analysis. Thus, the boundary conditions and effective properties of the perforated region for inelastic analyses under creep conditions have to be very carefully modeled in order to achieve the required structural integrity.

The tube-to-tubesheet junction is a critical location in the steam generator. The design methods should account for the local stresses at the junction of the tubesheet plate and standpipes.

Resolution

The resolution of this item consists of the following actions:

A. The applicants are committed to perform test of mechanical properties tests of mechanical properties to verify and supplement the methods in the ASME Code and the RDT standards and design information for ensuring the structural adequacy of the steam generator. Prototype steam generator tests will be run to verify certain performance characteristics. Hydraulic test model, large-leak tests, few-tube tests, DNB, (departure from nucleate boiling) tests, tube support wear tests, modular steam generator tests, single-tube performance tests, stability and interaction tests, tube-to-tubesheet weld tests, scale hydraulic model feature tests, and flow-induced vibration tests will also be conducted. The tests are needed to confirm the structural adequacy of the tubes.

B. The applicants will carry out a program to confirm the adequacy of the methods and criteria used to ensure the structural adequacy of the tube-sheet for its intended lifetime. The specific tasks involved are.

(1) Develop effective properties of perforated region for use in inelastic design analyses.

(2) Evaluate effects of thermal gradients and equivalent material property variations on ligaments near the periphery of the perforated region.

(3) Extend existing Appendix A-8000 Code methods for calculating the linearized membrane, shear, and in-plant bending* stresses in the ligaments using the equivalent solid plate stresses. Include all of these nominal stresses in the comparison with allowable primary membrane plus bending, and primary plus secondary allowables.

(4) Develop methods of evaluating local cyclic plastic strain concentration effects based on equivalent solid plate stresses for use in the fatigue evaluation.

(5) Develop methods of evaluating local cyclic creep strain concentration effects based on equivalent solid plate stresses for use in the fatigue evaluation.

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*In-plant bending occurs on either side of a minimum ligament section creating a "kinking" type of failure mechanism.

(6) Evaluate elastic follow-up into outermost ligaments

Reclassify portion of discontinuity stresses caused by pressure and mechanical loads as "primary" in accordance with the associated amount of elastic follow-up that occurs during thermal transients.

Reclassify portion of thermal stresses as "primary" in accordance with the amount of elastic follow-up that occurs during thermal transients.

(7) Develop ratcheting evaluation methods for outermost ligaments based on elastic equivalent solid plate stresses reclassified according to Item A(6) and including nominal membrane, shear, and in-plant bending stresses.

(8) Develop creep rupture damage evaluation methods for outermost ligaments based on equivalent solid plate stresses. The effects of elastic follow-up will reduce the amount of stress relaxation and increase the creep-rupture damage.

(9) Perform detailed tube-to-tubesheet joint analysis for tubes in high radial thermal transient region at periphery of the perforated region and include local thermal effects.