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THE U.S. FAST REACTOR MATERIALS AND STRUCTURES PROGRAM

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MATERIALS AND STRUCTURES PROBLEMS are central to many critical issues concerning the economic competitiveness, reliable performance, and safety of liquid metal fast breeder reactor (LMFBR) power plants.

Like their predecessors the light water reactors (LWRs), LMFBRs are expected to operate with high availability for 30 to 40 years in potentially corrosive environments and high radiation fields. In addition, LMFBRs, unlike LWRs, must operate at temperatures where creep effects are significant and under conditions which subject plant components to large and repeated thermal transient loadings. Information on the adequacy of design methods and the behavior of structural materials under these conditions is being generated from the construction experience and operation of LMFBR plants worldwide. Although operation has generally been successful, several reported failures of structural components have pointed the way for additional needed work. For example, creep-fatigue cracking due to severe cyclic thermal transient loadings has occurred in several components, usually in weldments. Steam generator failures have been particularly prominent, and these have been attributed to structural design deficiencies, fatigue, improper heat treatment, and undetected defects in welds from less-than-optimum fabrication procedures.

In view of these considerations, the U.S. Department of Energy has sponsored for many years a national LMFBR materials and structures program. This program currently is administered by the DOE Office of Breeder Technology Projects and the DOE Oak Ridge Operations Office; it is technically managed

by the Materials and Structures Technology Management Center at the Oak Ridge National Laboratory. Seven industrial and governmental contractor organizations directly participate in the program.

The objectives of the program are (1) to provide the technological basis for assuring that LMFBR components and systems will be free from significant structural failures during their design lifetimes and (2) to develop materials, design methods and criteria, materials property data, and procedures — all aimed at providing for broad flexibility in LMFBR component and system design and operation.

Technology areas included in the program are high-temperature structural design; seismic design; mechanical properties design data; fabrication; tribology (friction, wear, and self-welding); coolant technology (sodium and steam/water); advanced structural alloys; and nondestructive testing. The first two areas are categorized as Structural Design Technology and the remaining six as Materials Technology.

The products of the program are reports, design guideline documents, contributions to codes and standards, and a repository for design data which contains data correlations and supporting information covering a wide range of mechanical and physical properties. Many aspects of the technologies being developed are applicable to other advanced energy systems, including the high-temperature gas-cooled reactor, fusion energy devices, coal conversion plants, and solar energy conversion equipment.

It is the purpose of this paper to indicate briefly for each of the program's technology areas the objective, the scope, and some significant accomplishments. Future directions for the program are also discussed.

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STRUCTURAL DESIGN TECHNOLOGY

Included in this category are high-temperature structural design (HTSD) technology and seismic design technology.

HIGH-TEMPERATURE STRUCTURAL DESIGN TECHNOLOGY — The HTSD technology area is being developed to more effectively respond to the unique challenges presented by the LMFBR. These challenges arise not only from the high operating temperatures where creep occurs, but also from the large, rapid temperature changes that can occur in liquid-metal systems. Particular emphasis is placed on guarding against failure by the accumulation of creep and fatigue damage, creep-enhanced ratchetting, creep buckling, and environmental enhancement of low-temperature failure modes. Predictive methods and associated verified design criteria are being developed to provide adequate margins of safety against rupture or excessive deformation by the identified failure modes.

The HTSD technology area consists of three interrelated task elements: (1) development of new or improved design analysis methods and criteria for guarding against structural failures, (2) experimental validation of the products of the development task element, and (3) application of validated design methods and criteria through transfer of the technology in the form of guidelines, procedures, and standards to ASME code bodies, consensus standards organizations, reactor and component manufacturers, and utility organizations.

Development — This task is multifaceted, embracing the areas of (1) inelastic design analysis methods, which is primarily concerned with the development and validation of mathematical descriptions (constitutive equations) of inelastic material behavior for use in design analysis; (2) design criteria and rules for precluding time-dependent failures; (3) weldment design methodology; (4) confirmatory structural assessments for judging the adequacy of the developing high-temperature structural design methodology; and (5) simplified design methods for reducing the cost and time of design analysis. A number of high-temperature test facilities have been built and measurement devices developed for accurately obtaining structural test data and exploratory uniaxial and multiaxial material behavior data.

In the area of inelastic design analysis methods, constitutive equations, and guidelines for their use, have been formulated and periodically improved for the three major LMFBR alloys — types 304 and 316 stainless steel and 2-1/4 Cr-1 Mo steel. The capability of these equations to describe the key behavioral trends has been confirmed by a significant body of uniaxial and multiaxial exploratory test data. However, some relatively minor behavioral features are not

described as well as desired. Accordingly, efforts are underway to further improve the existing equations and to establish improved alternative equations. For example, a non-linear, combined kinematic/isotropic plasticity theory has been formulated and is being assessed as an improvement within the existing framework. In addition, a new unified theory, which does not distinguish between rate-dependent plasticity and time-dependent creep, has been developed.

In the time-dependent failure criteria area, damage accumulation rules and multiaxial strength theories have been developed for use in establishing criteria for precluding failure by creep rupture or creep-fatigue. Two interpretive reports on creep-fatigue have been published, and a significant body of uniaxial and multiaxial creep-rupture and creep-fatigue data has been generated for the three major LMFBR alloys. These data have been used to assess both current and potential damage accumulation rules and multiaxial strength theories. They have also provided a basis for developing improved theories and identifying problem areas and future directions.

More than 50 high-temperature structural assessment tests have been performed on geometries ranging from simple beams and plates to nozzle-to-spherical shell attachments. Test conditions have ranged from constant temperature and monotonic loading to highly transient thermal-mechanical cyclic loads. Data from the tests have been compared with deformation predictions and, where appropriate, with failure predictions from inelastic structural analyses using existing and proposed constitutive equations and failure criteria. The comparisons have shown that, in most cases, inelastic analyses are capable of predicting essential behavioral features in the relatively simple test structures. An example of an acceptable correlation is shown in Fig. 1 for a type 304 stainless steel nozzle-to-spherical-shell experiment at 593°C (1100°F). All test data are available to computer code developers and design analysts for use in benchmark problem calculations for code verification and qualification.

In the weldment area, emphasis is placed on the development of simple and explicit design rules that will assure structural integrity. Design factors for welds are being evaluated through fatigue and creep-rupture tests of welded structures. A long-term test of a welded pipe subjected to repeated thermal transient loadings is included in the testing program.

Existing simplified design methods are largely based on either the inelastic response of a symmetrically loaded thick-walled cylinder or the elastically predicted response of the actual structure being evaluated. The current effort is aimed at further

Fig. 1 - Experimental setup for determining the inelastic behavior of a nozzle-to-spherical-shell junction with results showing measured and predicted values for type 304 stainless steel at 593°C (1100°F).

improving the existing procedures and providing a demonstration of the applicability and limitations of the methods for specific design situations, which are usually three dimensional, not symmetrically loaded, and inelastic. Bounding methods are also being pursued.

Validation - The validation task is based on assessments made from testing and analyzing specimens that range from simple configurations, used to separate out the effects of specific assumptions, to full-size components and prototypic conditions, used to validate the total design analysis procedure. The principal objectives are to (1) validate methods for predicting crack initiation and

propagation under cyclic thermal loading conditions and steady creep at elevated temperatures; (2) validate the proposed constitutive equations under multiaxial states of stress; (3) determine the factors affecting strain at failure in base metal and welds, including the effects of geometrical discontinuities; (4) validate detailed and simplified methods for predicting deformation and buckling of piping components; (5) develop benchmark data for the validation of analysis methods for pipe clamps; (6) characterize elastic followup behavior in piping loops; and (7) validate analysis procedures for creep-ratchetting in two- and three-dimensional configurations for pressure vessel components.

Application - This task is devoted to technology transfer. This important function is performed in close association with ongoing LMFBR design activities so that design needs can be identified and fed back into the HTSD development and validation tasks. The principal effort is on piping design methods, design codes and standards, methods for structural evaluation, and computer programs for design.

Participants in this task have contributed to about 20 revisions of ASME Section III Code Case N-47 (ref. 1) and to several issues of supplemental design rules and guidelines for the three major LMFBR alloys.

SEISMIC DESIGN TECHNOLOGY - In this technology area improved seismic design methods, procedures, and criteria are being developed and validated. The purpose is to more accurately predict earthquake-induced motions and loads imposed on building structures and shells, piping and supporting structures, reactor and in-core components, vessels, and other items of plant equipment. The work includes ground motion definitions and soil/structure interaction models.

Early in the 1970s when design activities were underway for the Fast Flux Test Facility (FFTF)*, it became clear that the seismic design technology used for LWR nuclear plants was not directly applicable to LMFBR plants. In fact, the seismic design rules used for LWRs have been shown in some cases to be overly conservative for LMFBR designs. For example, it has been established that the thinner wall, more heavily insulated LMFBR piping has inherently higher damping properties than LWR piping. In addition, guidelines have been developed for locating seismic restraints on LMFBR pipelines that lead to a reduction of about 50% in the number of snubbers required for new LMFBR plants relative to those designed in the 1970s.

*The FFTF is a 400 MW(th) sodium-cooled test reactor at the Hanford Engineering Development Laboratory in Hanford, Washington.

MATERIALS TECHNOLOGY

The following areas are covered in the Materials Technology portion of the program: mechanical properties design data, fabrication, tribology, coolant technology, advanced structural alloys, and nondestructive testing.

MECHANICAL PROPERTIES DESIGN DATA — The objective of this technology area is to develop an elevated-temperature mechanical properties data base for the design of LMFBR power plants with service lifetimes to 40 years.

The data base provides direct support to designers and to developers of high-temperature structural design methods. It also serves as the basis for licensing and regulatory decisions, safety analyses, operation and maintenance procedures, and in-service inspection criteria. The data are provided to the ASME Boiler and Pressure Vessel Code for use in Code Case N-47 of Section III (ref. 1) and to nuclear energy standards.

The scope of the testing program ranges from exploratory tests for determining characteristic deformation and failure behavior to the full-scale acquisition of engineering mechanical properties data. The materials tested are those used in the reactor system (excluding fuel cladding and duct alloys), the reactor enclosure system, the primary and intermediate heat transport systems, and the steam generator system. Specifically, they are as follows: wrought types 304, 316, and A-286 stainless steels; CF-8 and CF-8M stainless steel castings; types 308, 316, and 16-8-2 stainless steel weld metals; 2-1/4 Cr-1 Mo and modified 9 Cr-1 Mo steels; and alloy 718.

The principal emphasis is on the development of a reference data base in air. In a less extensive testing effort, the effects of system environments — sodium, water/steam, and radiation — on the reference data base are assessed. The mechanical properties determined are tensile, creep, creep-rupture, fatigue, creep-fatigue, corrosion fatigue, fatigue crack growth rates, and fracture toughness.

Reference Data Base — The reference data base consists of mechanical properties determined in air. It includes the effects of important metallurgical variables such as heat-to-heat variations, forming method, and thermomechanical treatment. Some highlights of the results of work performed in developing the reference data base are summarized below.

It has been shown that small reductions in grain size can dramatically improve the creep, creep-rupture, high-cycle fatigue, and creep-fatigue properties of austenitic stainless steels. In the case of the high-cycle fatigue of type 316 stainless steel, a decrease in grain size corresponding to a difference of only one in the ASTM grain size

number (i.e., 3.5 vs 4.5) resulted in an order-of-magnitude improvement. It has also been determined that for high-cycle fatigue applications of type 316 stainless steel the maximum allowable strain is substantially greater than that previously specified.

An elevated-temperature fracture toughness test program has been carried out on type 304 stainless steel. It was determined that the requirements for inspection and surveillance of austenitic stainless steel structures and components can be significantly reduced relative to those previously established on the basis of room-temperature Charpy impact properties.

A much-needed data base for the long-term use of the stainless steel casting alloys CF-8 and CF-8M is being developed. These alloys are not included in ASME Code Case N-47 primarily because of the lack of mechanical properties data. In recent studies it was shown that the extensive fatigue-crack growth data base for wrought austenitic stainless steels is applicable to the CF-8 and CF-8M alloys.

An excellent review of the relation of microstructure and mechanical properties for types 304 and 316 stainless steels was published recently by Horak et al.²

Effects of Sodium and Water/Steam Environments — For sodium conditions relevant to LMFBR operation it has been shown that the mechanical properties of the LMFBR structural alloys in sodium are equal to or superior to the same properties in air.^{3,4} Examples are shown in Fig. 2 for the fatigue life of type 316 stainless steel and in Fig. 3 for the fatigue life of 2-1/4 Cr-1 Mo steel.

The considerable amount of information that has been developed in the fossil energy and LWR programs on the effects of water/steam environments on the mechanical properties of 2-1/4 Cr-1 Mo steel is applicable to LMFBR steam generator systems as well.^{5,6}

Fig. 2 — Fatigue life of type 316 stainless steel in air and in sodium at 593°C (1100°F). (Ms in the legend stands for megaseconds). (Ref. 3)

Fig. 3 - Total strain range versus cycles to failure for isothermally annealed 2-1/4 Cr - 1 Mo steel in sodium at 482°C (900°F) and 593°C (1100°F) and in air at 427°C (800°F) and 593°C (1100°F). (Ref. 4)

Effects of Irradiation - The effects of neutron irradiation, both atomic displacements and neutronically produced helium, are being determined to establish neutron exposure limits for out-of-core structures and components such as the core-support and core-former structures, the upper internals system, flow control modules, in-vessel fuel handling equipment, the reactor vessel, and the reactor guard vessel.

It has been demonstrated that significantly higher neutron exposures are permissible for out-of-core structures of types 304 and 316 stainless steel than those previously allowed. The higher limits are based on data from tests with low-energy neutrons prototypical for these applications; the previous limits were based on results from irradiations with high-energy neutrons more typical of those encountered by fuel cladding and ducts. These results have led to the specification of longer component lifetimes and reduced shielding requirements.

Current emphasis in this task area is on (1) the tensile and creep-rupture properties of irradiated types 304 and 316 stainless steels and their weldments at temperatures where helium can agglomerate to produce significant reductions in strength and ductility, and (2) the tensile and fracture toughness properties of these materials after irradiations corresponding to end-of-life values of displacements per atom ranging from 0.5 to 5.

FABRICATION TECHNOLOGY - There are two major tasks in this technology area: (1) the development of stainless steel weld metal with improved creep strength and ductility, and (2) the development and evaluation of austenitic-to-ferritic steel transition joints. The products are process specifications and improved procedures and equipment for commercial applications.

Improved Stainless Steel Welds - To minimize the tendency of fully austenitic stainless steel welds for hot-cracking and microfissuring, the weld-metal compositions

for this class of alloys are generally modified to produce small amounts of delta ferrite (usually 3 to 7 vol %). However, after extended periods at high temperatures, the ferrite phase can transform to brittle intermetallic phases, the presence of which significantly impairs the creep-rupture properties (both strength and ductility) of the welds relative to those for wrought-annealed material.

The ASME Boiler and Pressure Vessel Code, Section III, requires substantially lower allowable design strains for austenitic stainless steel welds and specifies stress reduction factors for these welds in components where creep effects are significant. In view of this situation, a program with a commercial manufacturer was initiated several years ago to develop austenitic stainless steel weld metals with improved creep properties. These alloys contain controlled residual elements (CRE) - specifically, higher-than-normal concentrations of titanium, boron, and phosphorous. The improvement in creep-rupture properties of type 308 stainless steel welds containing CRE is shown in Fig. 4.

Fig. 4 - Comparison of average rupture life at 649°C (1200°F) of type 308 controlled residual element (CRE) weld metal and conventional type 308 weld metal for gas tungsten-arc (GTA) and submerged-arc (SA) welds.

Transition Joint Development - Ferritic/austenitic dissimilar metal weld (DMW) failures have been regarded as a serious problem in the fossil-fired power industry since the early 1950s. However, evidence from service experience and laboratory testing has not conclusively shown why DMWs fail sooner than similar metal welds and base metal.⁷ The LMFBR program has also recognized the importance of this problem and several years ago initiated a base technology effort to develop a stainless steel to 2-1/4 Cr-1 Mo steel transition joint capable of serving out the design life of an LMFBR plant. A transition joint involving an alloy 800H spool piece was designed to minimize the

probable causes of failure, and this configuration was evaluated in an extensive life testing program. It was observed that failures always occurred in the heat-affected zone of the ferritic alloy. It was concluded that (1) the prime cause of failure is thermally induced stress and (2) inelastic failure analysis can provide a reasonable life prediction.

Observations based on these and other DMW tests have led to the hypothesis that early failure in DMWs is due primarily to the difference in thermal coefficient of expansion of the dissimilar materials, and far less to other factors such as carbon depletion, oxide notching, and brittle-phase formation. A program is underway to test this hypothesis.

TRIBOLOGY TECHNOLOGY — In the design of a safe and reliable sodium-cooled fast reactor system, one of the most critical problem areas involves the tribological phenomena of friction, wear, and self-welding at component interfaces that experience intermittent or continuous sliding or static contact during service. Tribological behavior in a sodium environment is vastly different from that in more common environments because in sodium, conditions are present that can lead to high friction coefficients, severe galling and self-welding, and high wear rates. The inaccessibility and induced radioactivity of many component bearing surfaces place even further demands for reliable tribological performance over the plant lifetime.

Examples of components and equipment for which good tribological performance is important are valves, heat exchangers, coolant pumps, fuel components, control rod drive systems, upper core internal structures, piston-ring seals, nozzle liners, and fuel handling equipment. Some potential failures of concern are leakage of heat exchanger tubes, excessive deformation stresses in structural members, seizure of rubbing parts, and even rupture of components.

A major objective of the test programs underway is to develop a base list of well-characterized wear-resistant materials, practical and economic fabrication processes, and qualified sources of supply. Five categories of materials are being qualified: (1) coatings such as those applied by diffusion, detonation-gun, plasma spray, and spark deposition processes; (2) weld-deposited hard-facing alloys; (3) candidate structural alloys; (4) specialty bearing materials; and (5) other special materials, generally non-metallic, such as lubricants and elastomers. Examples of important developments in the first category are presented below.

Electro-Spark Deposition Coatings — One of the most significant advances in coating technology for reactor components has been the application of electro-spark discharge (ESD) coatings. These coatings are produced by discharging stored energy from high-voltage capacitors through an electrode of

the material to be deposited. In the resulting spark, a small amount of material is melted, removed from the electrode, and welded to the substrate material.

A principal advantage of the ESD process is that thin (usually 25 to 50 μm layers) of nearly any electrically conductive material can be fused to a metal surface with such a low heat input that thermal distortions or changes in metallurgical structure of the substrate do not occur. It also produces a coating that is metallurgically bonded and therefore significantly more resistant to damage and wear than the mechanically bonded coatings applied by other low-heat-input processes such as detonation-gun, plasma spray, and electrochemical deposition. Also, compared to other types of coatings, ESD coatings are relatively inexpensive, and they dramatically reduce turnaround times and schedule delays. Designers now have the flexibility of substituting ESD-coated ferritic steels or other low-cost alloys for more expensive alloys that would otherwise be required for some applications.

Fully automated ESD processes have been developed and qualified for coating valve-guide fingers, fuel ducts, and control-rod ducts.

Detonation-Gun and Plasma-Sprayed Coatings — A number of detonation-gun (D-gun) and plasma-sprayed coatings have been tested extensively under breeder reactor operating conditions, and selected D-gun coatings have been qualified for several specific applications.

Process parameters, including substrate preparation techniques, must be optimized for each D-gun coating/substrate combination if adequate coating adherence and reliability are to be achieved. The present reference coating for core component load pads is a mixture of chromium carbide and 15 vol.% nichrome (80Ni-20Cr).

Roughening of the substrate can significantly improve the bonding of D-gun coatings. A new grit blast material has been qualified for this purpose.

COOLANT TECHNOLOGY — The purpose of research and development in this area is threefold: (1) to provide data on sodium and water/steam corrosion effects in LMFBR systems for design applications; (2) to develop and assess methods and equipment for handling, purifying, and chemically analyzing LMFBR coolants; and (3) to develop and evaluate methods for the measurement and control of radioactivity in the coolant circuits.

Corrosion — An extensive program has been conducted over the past two decades to characterize the corrosion of materials in flowing sodium. Stainless steels, low-alloy steels, and nickel-base alloys have been shown to be acceptably compatible with sodium.

A phenomenon known as mass transfer occurs in flowing sodium systems when certain

allowing elements dissolve in one region of the system and deposit in another. This phenomenon is of particular importance in the intermediate heat transport system where carbon tends to transfer from the ferritic to the austenitic portions of the system. An intermediate system mockup loop was designed and operated to evaluate this effect under realistic operating conditions. It was observed that carbon transport occurred, but that it essentially stopped after 30,000 h. It was concluded that decarburization of the 2-1/4 Cr-1 Mo steel was not solid-state-diffusion controlled and that the loss of strength resulting from decarburization of even the hottest portions of the ferritic steel in the steam generator would not be deleterious to plant operation.

It has also been observed that impurities or corrosion products sometimes combine with sodium to form particulates which can be transported and deposited in various parts of the system. For example, a flow impedance that developed in the primary system of an operating test reactor has been attributed to the combining of silicon and sodium and the subsequent formation of crystalline deposits of the reaction product on the surface of inlet orifices. Although it has been concluded that flow impedances of this type will not impose major design, cost, or operating penalties for commercial LMFBRs, further development work is needed to establish the roles of oxygen level, orifice design, and iron particulates as variables in this process.

Experience at the FFTF has shown that sodium vapor deposits, or "frost," can form in cover-gas spaces and impair the performance of components with moving parts. Studies of this phenomenon have led to methods for predicting and controlling the transport of sodium vapor so that it is now possible to prevent operational problems, such as the malfunctioning of rotating equipment, that frosting can cause in LMFBR systems.

Treatment, Control, and Monitoring of Impurities — Current work on sodium purification is directed toward the development of improved cold trapping techniques to remove oxygen and hydrogen. A computer code developed for design optimization has led to an improved cold trap with a fourfold increase in life and considerable savings in operating costs. It was also demonstrated that cold trap lifetimes can be further extended by in situ regeneration through vacuum decomposition of impurities. However, gases produced by this regeneration process are contaminated with tritium and require special

handling. An oxidation process using a reactant bed has been developed for this purpose, and work on this process is currently underway.

Multipurpose samplers and on-line meters for oxygen, hydrogen, carbon, and tritium have been developed.^{8,9} These devices contribute to more consistent reactor operation and signal maintenance requirements, and prevent unnecessary reactor downtime.

Radioactivity Control — In this task area methods are developed to predict, measure, control, and remove radioactive materials in sodium-cooled reactor systems. Control of the transport of radioactive species promotes improved plant economics because it can lead to significant reductions in plant downtime, personnel exposure, and tritium release.

Analysis of data on the buildup of radionuclides in the FFTF primary system has led to the development of an analytical model for predicting and controlling this effect in future reactors. To prevent the undesirable buildup of radioactivity in the system, nuclide traps have been developed for collecting fission and corrosion products such as ⁵⁴Mn, ⁶⁰Co, and ¹³⁵Cs.

Radiation from deposits of radioactive species in the primary system can severely limit access for maintenance operations. Methods have been developed for the removal of these deposits so that even hands-on maintenance can be performed on components removed from the system. In hot-leg areas where radioactive species diffuse into the material to a depth of about 25 m (0.001 in.) or more, a process using a 2-1/2% hydroxyacetic (glycolic) acid — 2-1/2% citric acid as the decontaminating reagent has been found to be effective provided dissolved oxygen in the solution is maintained at a level below 10 ppb.

ADVANCED ALLOY TECHNOLOGY — The objective of this technology effort is to develop to the state of commercialization an LMFBR structural alloy with better overall properties than the standard austenitic stainless and low-alloy ferritic steels. A thorough review of the situation was conducted by a national task force in 1974, and the following criteria were established for the advanced alloy: (1) design stress allowables equal to or greater than those for type 304 stainless steel at 593°C (1100°F); (2) better resistance to chloride and caustic stress corrosion cracking than austenitic stainless steels; (3) good resistance to radiation-induced swelling and helium embrittlement; and (4) good resistance to atmospheric rusting.

The alloy selected for extensive development and evaluation was a ferritic/martensitic alloy known as modified 9 Cr-1 Mo steel. The chemical composition specifications for this alloy are given below in weight percent.

Chromium	8.00-9.50
Molybdenum	0.85-1.05
Carbon	0.08-0.12
Manganese	0.30-0.60
Nickel	0.40 max
Vanadium	0.18-0.25
Niobium	0.06-0.10
Silicon	0.20-0.50
Nitrogen	0.03-0.07
Aluminum	0.04 max
Phosphorous	0.020 max
Sulfur	0.010 max

The main difference between this alloy and standard 9 Cr-1 Mo steel is the addition of vanadium and niobium for improved elevated-temperature strength. The modified alloy is recommended for use in the normalized and tempered condition.¹⁰

The normalizing treatment consists of heating the alloy to 1038°C (1900°F), holding for one hour, for thicknesses to 25 mm (1 in.), and then air-cooling to room temperature. This treatment produces a fully martensitic structure with a hardness of about Rockwell C40. Tempering is accomplished by heating to 760°C (1400°F), holding for one hour, for thicknesses to 25 mm (1 in.), and then air-cooling to room temperature. In the tempered state, the alloy is single phase with a fine grain size (ASTM 8-9) and a hardness of Rockwell B95. The microstructure of the matrix shows high-dislocation-density subboundaries which appear to be stabilized by the presence of fine $M_{23}C_6$ precipitates. Carbide precipitates are also found on prior austenite grain boundaries. Apparently, the effect of vanadium is to retard the growth of the fine $M_{23}C_6$ precipitates at service temperatures.

This alloy is considered to be particularly attractive because it affords the possibility of serving as the only alloy to be employed in the entire LMFBR system (including fuel cladding and ducts), thereby eliminating the troublesome dissimilar metal transition joints discussed above under Fabrication Technology. The use of modified 9 Cr-1 Mo steel as just an improved steam-generator alloy would allow higher design margins for ratchetting and creep-fatigue.

The ultimate goal of this effort is to obtain complete ASME Code approval for the advanced alloy. The status is as follows: Section I - approved in July 1983 as Code Case 1943; Section II - submitted to ASTM; Section VIII - expected approval in June 1984; Section III - initial data package submitted in December 1983.

Commercial Melting and Fabrication - Eleven commercial heats ranging in size from 0.5 to 65 tons have been melted and cast by domestic and foreign steel companies. The melting practices used are electric furnace, electroslag remelt, argon-oxygen decarburization, air induction, vacuum induction, and vacuum oxygen deoxidation.

Plate material has been fabricated by hot forging and hot rolling; flat and round bars by hot forging and hot extrusion; and tubing by hot extrusion, centrifugal casting and cold pilgering, and cold drawing.

Mechanical and Physical Properties - Data for eight heats of this alloy have shown conclusively that the criterion cited above regarding design allowable stresses has been met (see Fig. 5). Mechanical properties tests in air and in sodium are in progress to obtain data for the various ASME Code packages. These include tensile, creep, fatigue, creep-fatigue, fracture toughness, and crack growth tests. The physical properties being measured are Young's modulus, Poisson's ratio, thermal expansion coefficient, thermal conductivity, and heat capacity.

Structural Design Methods - Efforts here are focused on (1) identification of constitutive equations that can be used for inelastic design analyses on an interim basis and

Fig. 5 - Design stress as function of temperature for modified 9 Cr-1 Mo steel compared with standard 9 Cr-1 Mo steel, 2-1/4 Cr-1 Mo steel, and type 304 stainless steel.

(2) preliminary assessments of failure criteria. Exploratory elastic-plastic and creep tests are being performed to provide a basis for assessing the adequacy of the general framework of the currently used breeder reactor constitutive equations.

Failure activities are initially focusing on creep-rupture damage accumulation under variable loads and temperatures. Thermal transient tests to failure of thick-walled cylinder specimens are planned. The results of these tests will serve as the basis for assessing the applicability of the overall design methodology to modified 9 Cr-1 Mo steel.

Joining - Filler metal development and studies of the effects of process variables and welding parameters on the properties of modified 9 Cr-1 Mo steel weldments are underway. Weld procedures are being developed for gas-tungsten-arc, submerged-arc, and shielded-metal-arc processes, and the susceptibility of welds to hot or cold cracking is being determined.

Corrosion - The corrosion behavior of modified 9 Cr-1 Mo steel is being evaluated in air, steam, aqueous hydroxide and chloride solutions, and sodium.

As shown in Fig. 6, the air oxidation resistance of the advanced alloy is dramatically better than that of 2-1/4 Cr-1 Mo steel. Based on 30,000 h tests, the behavior of the advanced alloy in steam is similar to that of 2-1/4 Cr-1 Mo steel at 482° and 593°C (900° and 1100°F).

Fig. 6 - Comparison of weight gain as function of time in air at 593°C (1100°F) for 2-1/4 Cr-1 Mo steel and three heats of modified 9 Cr-1 Mo steel.

In slow strain rate caustic stress corrosion cracking tests at 232°C (450°F), the behavior of modified 9 Cr-1 Mo steel was found to be similar to or better than that of 2-1/4 Cr-1 Mo steel. When cracking occurred, the failure mode was different from that in annealed 2-1/4 Cr-1 Mo steel. In caustic solutions, 2-1/4 Cr-1 Mo steel cracks intergranularly or by a mixed mode of intergranular initiation and intergranular/transgranular propagation. Modified 9 Cr-1 Mo steel, on the other hand, has failed transgranularly in all cases evaluated thus far.

Operating Experience - Tubes of modified 9 Cr-1 Mo steel are currently being operated in steam generator sections of various fossil power plants in the United States, the United Kingdom, and Japan. The longest time (about 4 yr) has been reached on tubes in the superheater of the TVA Kingston Steam Plant. A recent examination during a scheduled plant shutdown showed no unusual deposits on these tubes, as compared with the adjacent stainless steel tubes, and no change in wall thickness.

An 8-in. diam. test section of modified 9 Cr-1 Mo steel pipe was tested in the Sodium Components Test Loop (SCTL) at the Energy Technology Engineering Center (ETEC). The modified 9 Cr-1 Mo steel pipe was safe-ended with type 304L spool pieces using ERNiCr-3 filler wire. The test section was operated for 5245 h at 520°C (950°F) during a sodium pump test in the SCTL. It is being retained at ETEC for further exposure in SCTL tests. In addition, a 16-in. diam. pipe section was installed in the Sodium Pump Test Facility at ETEC in March 1984, and a 12-in. diam. steam generator sodium-inlet section was inserted in the EBR-II test reactor plant at Argonne-West in April 1984.

NONDESTRUCTIVE TESTING TECHNOLOGY -

Tasks in the nondestructive testing technology (NDT) area are aimed at the inspection of materials and components during manufacture and in-service, including situations requiring remote, automated testing methods. Advanced radiography, ultrasound, and eddy-current techniques are being developed, especially for stainless steel welds, ferritic steel tube-to-tubesheet welds, and high-temperature in-service inspection (ISI).

Methods for Specific Components - In an LMFBR steam generator, heat is transferred from sodium to water/steam with the tubing serving as the barrier between these highly incompatible fluids. Three important NDT technologies have been developed to ensure the integrity of the steam generator: (1) microfocus rod-anode x-ray technology for manufacturing inspection of tube-to-tubesheet welds, (2) ultrasonic technology for ISI of tube-to-tubesheet welds and ferritic/austenitic transition joints, and (3) borehole eddy-current technology for ISI of steam generator tubing.

The rod-anode x-ray technique was found to be significantly more sensitive in identifying defects in tube-to-tubesheet welds than standard isotopic methods based on thulium and ytterbium. The use of this highly sensitive technique during fabrication of a prototypic LMFBR steam generator resulted in the development of improved welding procedures and higher quality tube-to-tubesheet welds.

Although there is substantial evidence that austenitic stainless steel piping containing sodium at low pressures in the primary system of an LMFBR would leak before a catastrophic break occurred, the ability to perform ISI of austenitic stainless steel welds in this piping is important to provide added confidence in system integrity. These welds are difficult to inspect because of their coarse grain size and anisotropic dendritic structure. A major accomplishment in this area has been the development and demonstration of remote-examination ultrasonic techniques and equipment for detection of cracks of sizes corresponding to 25 to 50% of the wall thickness of sodium-filled stainless steel pipe at 200°C (392°F).

Multicomponent Applications — Generic NDT technology is being developed for use in a number of LMFBR component applications. Included in this category are advanced sensors and instrumentation for quantitative flaw detection and characterization.

As indicated in Fig. 7, the results of in-service eddy-current examinations can be difficult to interpret because of the large number of test parameters that can affect the signal output. To distinguish which parameters are causing which signal variations, the eddy-current instrument must normally make as

many independent readings as there are variations. In an effort to simplify matters, a systematic process was developed for multifrequency eddy-current examinations based on theoretical models and the design of electronic instrumentation and computer programs.¹¹ Using this process, volumetric inspection of steam generator tubing by eddy currents can now be performed in compliance with the ASME Code in much less time than that required for ultrasonic examination.

Ultrasonic transducers capable of operating at high temperatures were developed because of strong incentives to perform ISI at temperatures of 200°C (392°F) or above with liquid sodium in the system. Conventional transducers designed for use at room temperature generally will not perform at 200°C (392°F) and above, largely because of the limitations of the piezoelectric element. A lithium niobate element was selected for the higher-temperature application, and a prototype transducer using it has been successfully tested. The application of these transducers to phased annular arrays with electronic focusing will allow more accurate flaw sizing and better penetration of weld metal in the ultrasonic inspection of stainless steel piping.

SUMMARY AND FUTURE DIRECTIONS

The U.S. Department of Energy has sponsored a vigorous breeder reactor materials and structures program for about 15 years. Important contributions to breeder development have resulted from this effort in the areas of design (inelastic rules, verified methods, seismic criteria, mechanical properties data); resolution of licensing issues (technical witnessing, confirmatory testing); construction (fabrication/welding procedures, nondestructive testing techniques); and operation (sodium purification, instrumentation and chemical analysis, radioactivity control, and in-service inspection).

The national LMFBR program currently is being restructured to address base technology needs for innovative designs that are economically competitive and demonstrably safe. It is anticipated that in support of this activity the Materials and Structures Program will focus its efforts in the following areas: (1) removal of anticipated licensing impediments through confirmation of the adequacy of structural design methods and criteria for components containing welds and geometric discontinuities, the generation of mechanical properties for stainless steel castings and weldments, and the evaluation of irradiation effects; (2) qualification of modified 9 Cr-1 Mo steel and tribological coatings for design flexibility; (3) development of improved inelastic design guidelines and procedures; (4) reform of design codes and standards and engineering practices,

Fig. 7 - Illustration of eddy-current probe in a steam generator tube indicating system variables that can affect the signal output.

leading to simpler, less conservative rules and to simplified design analysis methods; and (5) incorporation of information from foreign programs through exchange and collaboration.

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