PHILOSOPHICAL TRANSACTIONS OF THE ROYAL SOCIETY A MATHEMATICAL, PHYSICAL & ENGINEERING SCIENCES

Structural Integrity of Fast Reactor Components [and Discussion]

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Structural integrity of fast reactor components

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[Plate 1]

The paper focuses on the generic aspects of the main structural integrity issues in the liquid-sodium-cooled fast reactor. The choice of sodium as a coolant has important consequences for the deformation and failure process in the materials used for the main plant components. For example, its high boiling point means that the primary and secondary circuit containment operates at ambient pressure and the system loading is dominated by thermal stress. The resultant low primary stresses make leak-before-break a viable integrity criterion for all sodium boundary components. Sodium coolant operates at comparatively high temperatures and this, together with the good heat-transfer properties, means that thermal fatigue and creep are of concern, particularly in the hotter parts of the plant. A third factor concerns the steam generators, where the integrity of the sodium–water boundary is particularly important. The paper will consider the failure processes that must be addressed in relation to these conditions and the development of the integrity assessment arguments.

1. INTRODUCTION

There is an impressively wide measure of agreement on the basic science, technology and design principles of liquid-sodium-cooled fast reactors, and major programmes are under way notably in Europe, Japan and the U.S.S.R. aimed at commercial demonstration of the technology. Structural integrity of key components is an important issue governing both safety and economic performance of fast reactors. On economics, we are concerned with both operational reliability and protection of investment, in that failure of key inaccessible components could prove to be life-limiting. Design and construction codes which have evolved for use in the design of elevated-temperature plant (ASME 1986; RCC MR 1987) include procedures to provide a margin against structural failure by perceived materials-failure processes. These provide the basis for consideration of the integrity of a specific component.

In this paper structural integrity of fast reactor components is considered in relation to three main areas of the plant:

(*a*) core support path and containment, the integrity of which are crucial to safety, although clearly protection of investment is also involved;

(*b*) hot pool structures, where the key issue is protection of investment;

(*c*) steam generator units, where again the economic factors are crucial but where safety is also important.

The present discussion is related particularly to the European Fast Reactor (EFR), and following a brief description of the plant highlighting the above three areas the related structural integrity issues are addressed, following the approach shown schematically in figure 1.

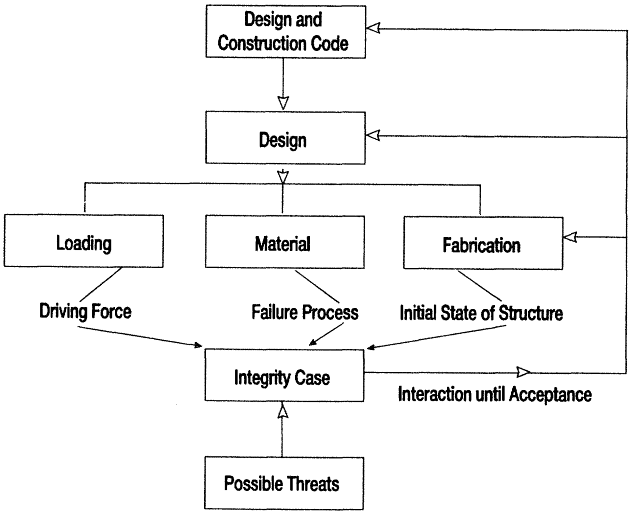

FIGURE 1. Factors influencing the structural integrity case for a component.

2. PLANT DESCRIPTION

The use of sodium as a coolant in both the primary and secondary circuits is a major factor influencing structural integrity of key components. The main operating parameters and the physical characteristics of the sodium coolant are listed and compared with parallel data for a typical large pressurized water reactor (PWR) in table 1. There are two main factors distinguishing fast reactors from water reactors.

TABLE 1. PRIMARY CIRCUIT PARAMETERS

	characteristic	fast reactor	PWR
1	reactor thermal power	3600 MW	3410 MW
2	coolant	sodium	water
3	core inlet temperature	390 °C	290 °C
4	core outlet temperature	550 °C	325 °C
5	boiling temperature (at operating pressure)	900 °C	345 °C
6	coolant pressure	3 bar[a]	158 bar[a]
7	thermal conductivity	500 W (m °C)$^{-1}$	0.68 W (m °C)$^{-1}$
8	specific heat	1260 J (kg °C)$^{-1}$	4220 J (kg °C)$^{-1}$
9	density	820 kg m^{-3}	960 kg m^{-3}
10	reactor vessel containment material	316 LN	A508 Cl. III
11	vessel thickness	25 mm	200 mm

[a] 1 bar = 10^5 Pa.

1. The high boiling point of sodium, which is *ca.* 300 °C above the maximum normal operating temperature and 200 °C above the estimated maximum temperature under the worst-design-basis fault transient. Moreover, the sodium coolant is at virtually atmospheric pressure during operation, and all sodium-containing vessels are therefore relatively thin.

2. The high thermal conductivity of sodium (more than 100 times higher than water) resulting in high rates of heat transfer from coolant to structural components. This, coupled with the relatively high specific heat and density, causes rapid changes of surface temperatures

in components during fast transients such as occur during reactor trips or loss of secondary circuits. This can result in large through-thickness temperature differences, particularly in the hot pool structural components.

As indicated in figure 1, a second major factor influencing structural integrity is the choice of materials for structural components. Type 316LN stainless steel has been selected for virtually all of the primary circuit components and the secondary circuit sodium pipework. Exceptions are the roof structure closing the primary containment vessel, for which a carbon–manganese steel will be used, and the wrapper and cladding in the fuel element assemblies, which will be manufactured from a void-resistant austenitic alloy. The material selected for the steam generator units is at present a 9Cr–1Mo-type ferritic steel, although the specification has yet to be endorsed. Important factors determining the choice of 316LN are its high ductility and fracture toughness coupled to good creep properties required particularly for the components exposed to hot sodium, although its poor thermal conductivity exacerbates the through-thickness temperature differences and associated thermal stress. The choice of a 9Cr–1Mo-type steel for the steam generator units in EFR is because of its high mechanical strength and thermal conductivity. The economic benefits (in terms of material savings) from optimizing strength have resulted in a 9Cr–1Mo steel strengthened with niobium and vanadium being adopted as the favoured option.

A general aspect of the design relating to structural integrity is that in the thickness used, particularly for the sodium containment vessel, no post-weld heat treatment is required. Thus residual welding stresses require special consideration in assessing fracture characteristics.

3. Safety-related structural components

The major safety requirement is that there must not be any failures that could lead to a rapid reactivity transient triggering a whole-core accident. In this context it is essential that the core does not drop away from the control rods and that there is not a large-scale loss of coolant. To meet such criteria requires a high level of confidence in the structural integrity of the components making up the core-support path, coupled to a fail-safe approach being adopted in their design. The components making up the core-support path from the core through the diagrid and strongback reactor vessel to the ferritic steel roof are shown in figure 2. Inherent structural integrity is achieved by a high-quality structural unit fabricated in ductile materials suited to the operating environment and enhanced by a number of key design conditions. Important among these are the adoption of a pool concept with an unpenetrated low-pressure primary vessel, and location of the core-support path components in a comparatively low-temperature environment below the range for significant creep and long-term degradation of material properties due to ageing. Thus the dominant potential failure process in all of these components is classical ductile fracture. The main sources and levels of stress are summarized in table 2. The system-loading-related stresses are very low, and potentially the highest stresses are weld residual stresses, which can be up to yield stress level. Overall, failure of any components in the core-support path that could trigger off a whole-core accident is beyond the design basis and individually must meet a probability of less than 10^{-7} per reactor year.

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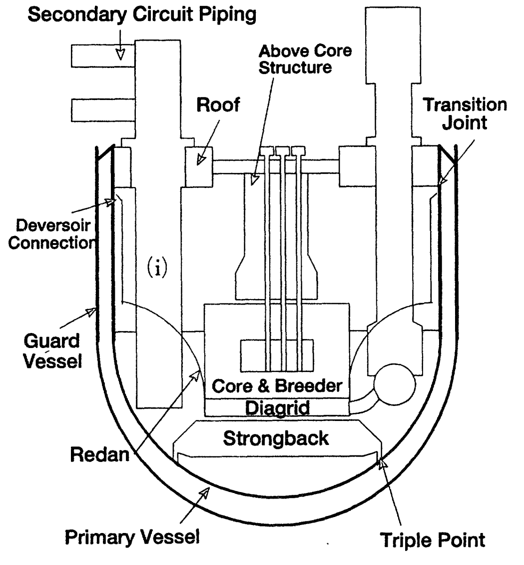

FIGURE 2. Core-support path and hot pool structures in EFR design. ((i) is intermediate heat exchanger.)

TABLE 2

component	source of stress	level and type[a]
primary containment vessel	self-loading	very low, primary
	residual stress at welds	up to yield stress, secondary
	thermal stress	low, cyclic secondary
	seismic	moderate, primary
core support: diagrid and strong back	self-loading	very low, primary
	residual stress at welds	up to yield, secondary
	hydraulic	low, primary
	thermal	low, secondary
	seismic	low, primary
roof	self-loading	very low, primary
	residual stress at welds	up to yield, secondary
	seismic	low, primary

[a] Primary stresses are long-range stresses in equilibrium with applied loads. Secondary stresses are usually short range and are self-equilibrating within the structure (e.g. thermal, residual).

3.1. *Diagrid and strongback*

(*a*) *Design*

The diagrid and strongback are welded box structures fabricated from an inherently ductile material (316LN) and have considerable rigidity and redundancy. The main loading is a steady compressive dead-weight load, independent of plant operating conditions and transmitted to the lower end of the pressure vessel via the triple-point weld. Normal operational transients produce insignificant stress, and the only possible thermal transient loading introducing thermal stress is a secondary circuit trip resulting in hotter sodium being introduced into the cold pool from the IHXs for a short period. Mechanical transient stress may be introduced via the reactor during a seismic event.

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(*b*) *Structural integrity case*

The first line of defence is the adoption of high standards during fabrication coupled to rigorous quality assurance. Since the diagrid, strongback and triple-point weld are generally loaded in compression they are classed as defect-tolerant structures. The redundancy in the box-welded structure and the support afforded to the diagrid by the strongback introduces an element of fail-safe into the integrity case. Lastly, the integrity case rests on having significant margins to failure at all times, and this is provided by the low system and operational stress. It can be reinforced by continuous or periodic monitoring of deflections in the key structural elements.

(*c*) *Threats*

The main threat to the integrity case could come from flaws introduced during fabrication, and this must be countered by rigorous pre-service inspection coupled with in-service monitoring. On the question of degradation in material properties, the diagrid is subject to neutron irradiation, but for the lifetime dose of 1–2 DPA the decrease in ductility and fracture toughness is not significant. As already emphasized, thermal ageing effects at cold pool temperatures (*ca.* 390 °C) are also expected to be insignificant.

3.2. *Reactor vessel*

(*a*) *Design*

The reactor vessel presents the most vulnerable part of the load path. It is a large (17 m diameter) vessel, which, because it is unpressurized, is thin walled (25–35 mm with local reinforcement to 60 mm). Although the primary system loads are low, the vessel is left in the stress-unrelieved condition, resulting in short- and long-range residual stress (table 2) which can be complex, particularly at the triple-point weld joining the strongback to the vessel. An important consequence of the design is that the vessel is maintained uniformly at the cold pool temperature (390 °C) up to the sodium level. This is achieved by a 'deversoir' or weir arrangement which protects the upper part of the vessel from contact with the hot pool sodium by maintaining a constant level and flow of sodium at the core inlet temperature. As well as ensuring that the vessel remains below the creep range, the 'deversoir' also protects the vessel from thermal fatigue. A further important feature of the design is the provision of a guard vessel around the reactor vessel, thereby ensuring that the sodium level is kept above the core in the event of a leak in the latter.

(*b*) *Structural integrity case*

The first line of defence is again the adoption of high fabrication standards during vessel manufacture, coupled to rigorous quality assurance. Nevertheless, the vessel can be assumed to contain a population of flaws below the pre-service inspection limits. Catastrophic failure of the vessel with the possible consequence of the core dropping away from the central rods is beyond the design basis, and thus the probability of any such undetected defect triggering such an event must be demonstrated to be extremely low, i.e. less than 10^{-7} per reactor year.

The low operational loadings, coupled with the high toughness of stainless steel, have led to a leak-before-break (LBB) integrity argument being adopted for the primary vessel (Tomkins 1988). This is a fail-safe argument that should minimize the need for detailed in-service inspection. However, confidence in its validity demands a substantial margin between expected flaw sizes and those that are tolerable; that breaks can be reliably detected; and that the consequences of a break are acceptable in terms of plant operational safety.

The essential features of the LBB argument can be illustrated on a failure assessment diagram (Munz 1984) shown schematically in figure 3. Curve A represents the boundary sizes of flaws which are expected to be detected to a specified high probability during pre-service inspection. Although some small amount of growth of flaws above the threshold limit represented by curve B may occur during service, due for example to thermal fatigue, the integrity case is concerned with those flaws that can break through the wall by ductile tearing. This is illustrated by curve C and, allowing for some in-service crack growth, the start of life-allowable flaw size is represented by curve D. Curve C can be deduced from an analysis using the CEGB R6 procedure (Harrison *et al.* 1980), while the margin between curves D and C is determined from thermal fatigue data. A source of uncertainty is the start of stable ductile tearing, but for the large length-to-depth ratios at the right-hand side of the diagram the stress-intensity factors are increasing so rapidly with crack depth that the choice of precise failure conditions is of second order. The critical crack size for unstable growth of a through-wall crack is denoted by point F after growth along a trajectory F′F. Point E represents the minimum through-wall crack size for leaks to be reliably detected.

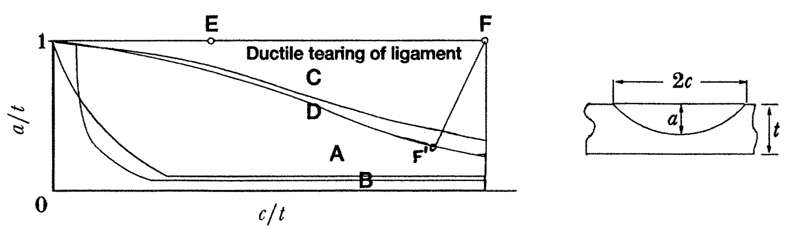

FIGURE 3. Leak-before-break. A, Pre-service flaw detection limit; B, threshold for flaw growth; C, boundary for unstable failure of ligament; D, start of life-allowable flaw size; E, minimum through-wall crack size for leak detection; F, critical through-wall crack size for unstable growth.

Returning to the specific case of the fast reactor vessel, the operational conditions are such that the main crack-growth mechanism is classical ductile fracture. The ductility and toughness of the vessel material and welds (figure 4, plate 1) are such that the critical through-crack length for unstable fracture (point F on figure 3) is typically very much greater than 1 m. This is very much larger than the reliable detection limits for flaws, denoted by curve A in figure 3. Thus, even allowing for a small amount of crack growth due to a low number of thermal cycles during service, the margin between detectable flaw and tolerable flaw size is extremely large, and provided this can be established beyond any doubt by extensive testing, an 'improbability of failure' argument can be proposed. The LBB argument requires that the minimum detection size for both sodium leaks and gas leaks (in the cover gas region of the vessel) is below the critical size, i.e. point E must lie to the left of F by a considerable margin on figure 3. Various techniques for leak detection are under consideration and being tested, but it is confidently expected that there will be a considerable margin between the detection levels and critical sizes.

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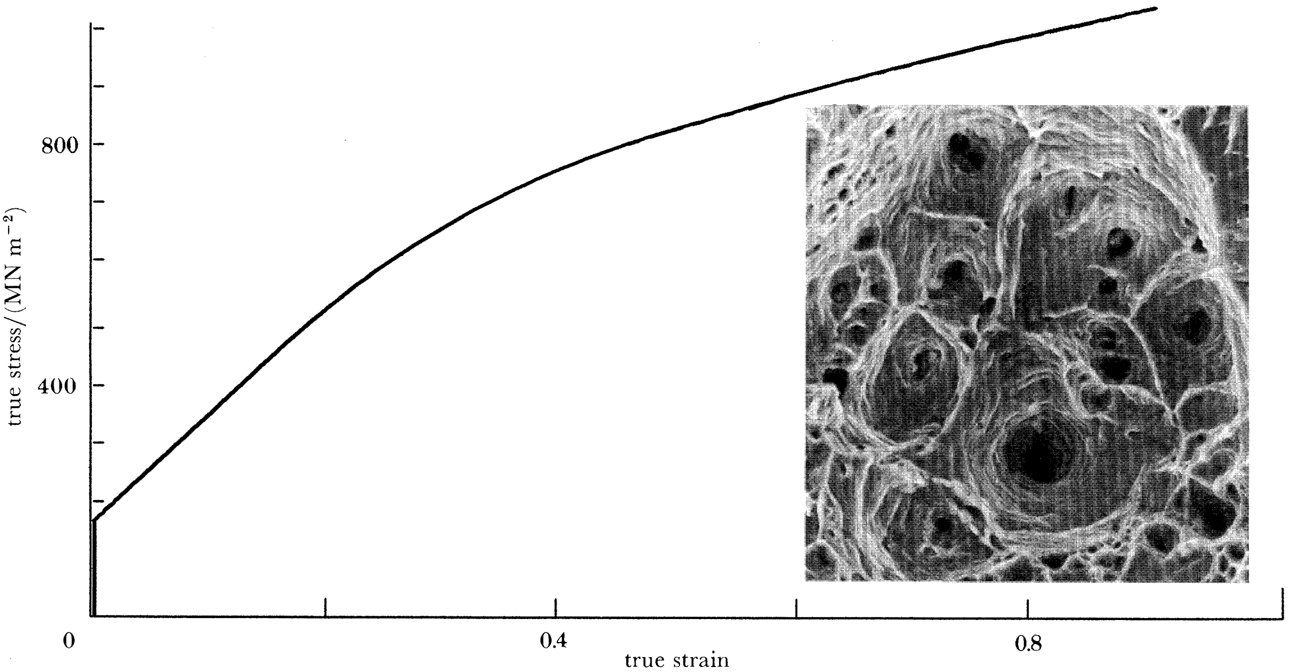

FIGURE 4. Tensile failure characteristics of stainless steel.

400 °C	ϵ_f	J_i/(kJ m^{-2})	$J\Delta a = 1$ mm
plate	0.9	90	274
weld	0.5	35	224

(*Facing p.* 424)

(*c*) *Threats*

It is essential to establish firmly the margins between tolerable and critical defect size and to ensure that these margins are not seriously eroded during service. A programme of large-scale tests is continuing, particularly on plate specimens containing stress-unrelieved welds. This programme will be supplemented by tests on the Structural Fracture Test Facility currently being built at the UKAEA's Risley Laboratory. Using this machine it will be possible to test wider plates (up to 2.5 m) of full service thickness, at operational temperatures (390 °C) and under distributed and multiaxial loading that will closely match the in-service loading condition. Thus the analytical methods used to calculate failure-assessment diagrams will be validated to a much higher level than has been possible hitherto, and any uncertainties are expected to be extremely low.

With regard to metallurgical factors, and particularly the possible deterioration of ductility and toughness during service, there are three main aspects to consider. First, the welds have a higher inclusion content than the parent plate material resulting in lower toughness, emphasizing the need for rigorous quality assurance in the selection of weld consumables and weld procedures. Secondly, for austenitic materials such as 316LN steel and the weld material used for the reactor vessel, the main embrittlement process of concern is intergranular precipitation, particularly of more brittle intermetallic phases such as σ and χ (I. J. O'Donnell, F. W. Noble and B. L. Eyre, unpublished work). However, the temperatures of concern here (up to 390 °C) are below the range for such precipitation-induced embrittlement to occur. Thirdly, 316LN and the associated weld metal can undergo dynamic strain ageing at temperatures in the range 300–600 °C, and this can result in some loss of ductility at low strain rates (O'Donnell *et al.* 1989). However, the fracture mode remains ductile-void coalescence, and the decrease in toughness and crack size is not likely to pose a significant threat to the LBB integrity case. Again, this will be confirmed by the structural-features test programme.

Thus, in summary, the integrity case based on LBB is very robust and will be further strengthened by the programme of testing. It should not be necessary, therefore, to carry out detailed volumetric in-service inspection of the reactor vessel, but there is sufficient space between the primary and guard vessels to allow remotely controlled inspection systems to be used should some periodic monitoring of the state of the vessel be required.

3.3. *Reactor roof*

(*a*) *Design*

As shown in figure 2, the primary reactor vessel is closed at the top by a flat roof, fabricated as an annular box-type structure from a high-toughness carbon–manganese steel. It is penetrated by and supports the rotating plug and major components such as the intermediate heat exchanger (IHX) and primary pumps, as well as reactor instrumentation and other services. The whole of the roof is maintained at a temperature of *ca.* 120 °C by the circulation of air through the box structure.

Like the diagrid and strongback, the roof is a highly redundant structure subject to low stress levels under normal operating conditions. The most sensitive region of the structure is where it is attached to the austenitic primary vessel via a transition weld. In EFR the current design is such that the principal load-bearing transition weld is in tension.

(*b*) *Structural integrity case*

The basic argument is similar to that for the diagrid and strongback and rests on redundancy, together with the large margins between design and failure loads. Having the transition weld carrying the weight of the primary vessel in compression means that it is defect-tolerant. Unstable fracture in the roof-to-vessel attachment region, including the lower austenitic weld that is in tension, is beyond design basis. This is substantiated by an LBB argument together with periodic in-service inspection to detect any flaws that may grow during operation.

(*c*) *Threats*

Structural integrity of the roof could be threatened by poor material properties and flaws introduced during fabrication. The carbon–manganese steel selected is aluminium-grain-refined to give high toughness, and both materials and fabrication procedures will be subject to rigorous quality assurance and pre-service inspection. Overheating of the roof due to failure of the air cooling system or insulation could take the steel into the creep range (*ca.* 400 °C) with the associated risk of dosage accumulation. However, this will be quickly detected by temperature monitoring. There could also be some risk of stress corrosion cracking, particularly from the accumulation of caustic in the cover gas due to reaction of moisture with the primary sodium, and from refluxing of zinc and barium from the sodium. This threat will be countered by careful monitoring of the roof environment.

4. Hot pool structures

Design

The hot pool of the fast reactor contains a number of large complex structures, all fabricated from type 316LN steel, and operating in an ambient temperature of *ca.* 550 °C. These structures would be difficult to remove and/or repair, so their failure is a major threat to plant investment. The main structures are identified in figure 2 as the above-core structure (ACS), IHX and redan. The secondary circuit pipework should also be considered in this group as it is fabricated from the same steel, although it operates at a slightly lower temperature of *ca.* 525 °C. At temperatures in excess of 480 °C, creep deformation and possibly failure processes are operative, and must be taken into account in the design. This has necessitated the development of special design codes (ASME 1986; RCC MR 1987). As with the core-support structures, applied mechanical loads are low, but because of the high temperature, thermal loads can be high.

The ACS comprises a number of guide tubes, through which pass control and shut-off rods as well as instrumentation. The tube array is contained within an outer shell that also contains a baffle to deflect sodium leaving the core. The temperature of sodium leaving the core varies from channel to channel, particularly in the vicinity of the core–breeder boundary, thus exposing the bottom of the ACS to rapid temperature fluctuations, known as thermal striping. In addition, because sodium is such a good heat-transfer medium, large coolant temperature changes, for example on trip, are rapidly experienced by all hot pool structures including the ACS, giving temperature changes at rates of up to 10–15 $^{\circ}C\,s^{-1}$. In the thicker sections of stainless steel these rates can induce significant temperature gradients with yield-level surface stresses (figure 5).

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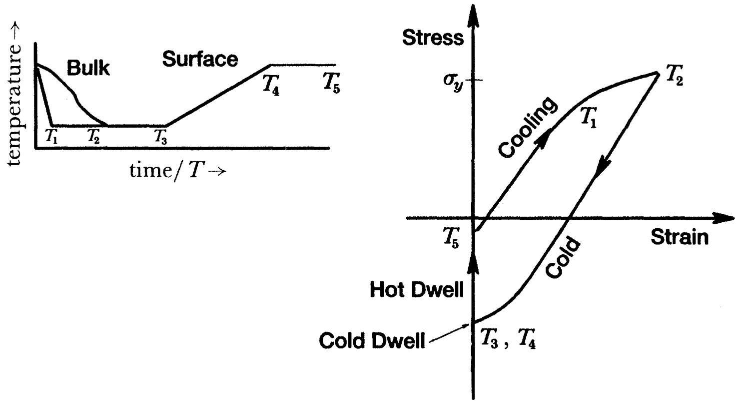

FIGURE 5. Reactor trip transient and resulting surface hysteresis loop.

The function of the IHX is to transfer the heat produced by the reactor from the primary to the secondary sodium. Mechanical loadings are modest, the principal load being the secondary-side internal-tube pressure of 5 bar (5×10^5 Pa). Good thermal hydraulic design will minimize internal temperature differentials during operation despite an overall unit temperature differential of 150 °C. Thermal shock loading can occur on trip and is highest at the lower tube plate.

The redan provides the hot-cold pool boundary and is subject to the most significant thermal stress. In fact the through-wall temperature differential can give rise to a surface-stress difference of almost 300 MPa. Although not subject to thermal shocks, parts of the redan can undergo significant thermal cycling, for example with movement of the sodium level, and during start-up. The thermal hydraulic performance of the hot pool is of crucial importance to the integrity of the redan, and both computer analysis and physical modelling are being used to optimize the design.

Structural integrity case

An important feature of the structural integrity case for all of the hot pool structures is the response to thermal loading. Where this results in local yielding, as shown in figure 5, a subsequent dwell period at elevated temperature leads to stress relaxation, during which creep deformation and possibly damage can occur with associated damage accumulation. Repeated thermal loading throughout the life of the plant, up to a few hundred cycles for major transients and millions of cycles for thermal striping, can lead to fatigue damage. The integrity case for hot pool structures relies on such fatigue and/or creep damage being within limits set by the design code. Confidence in these limits is gained by operating-plant experience, combined with an awareness that they adequately describe the relevant failure process. Operating experience of thermally driven plant such as the fast reactor is limited, so it is essential to understand the failure processes; creep–fatigue interaction and thermal fatigue (thermal striping).

(*a*) *Creep–fatigue interaction*

Figure 6 shows how strongly the fatigue life of a material can be degraded by the presence of a dwell period such as that shown in figure 5 (Wareing *et al.* 1986). The reduction in life is a result of creep cavitation and cracking damage induced by creep strain during the stress relaxation periods (Hales & Tomkins 1982). This damage is cumulative, and can produce failure before the conventional transgranular fatigue-cracking failure process has had time to

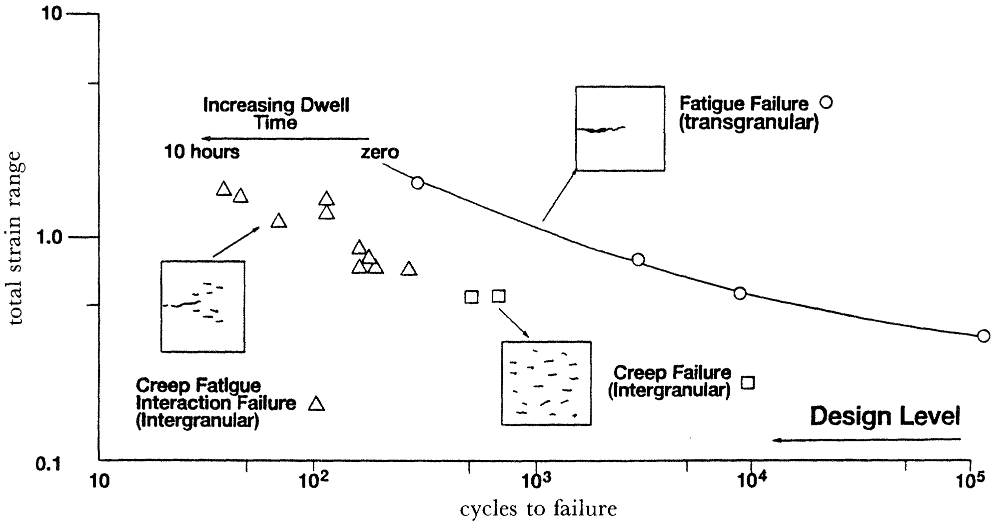

FIGURE 6. Creep-fatigue failure in type 316 stainless steel at 570 °C.

TABLE 3. KEY PARAMETERS FOR HOT POOL STRUCTURES

component	main loadings	predominant damage processes
redan	low primary stress flow-induced vibrations seismic loading temperature gradients temperature changes	creep/creep–fatigue
secondary pipework	low primary stresses seismic loadings thermal expansion thermal shock sodium–water reaction	creep-fatigue ductile fracture
ACS	low primary stress flow-induced vibrations seismic loading temperature gradients thermal shock temperature fluctuations	thermal striping creep/creep-fatigue (exacerbated by irradiation)
IHX	low primary stress flow-induced vibrations seismic loading temperature differentials thermal shock	creep-fatigue

establish itself; it is particularly effective at lower levels of cyclic strain. The design rule used in the integrity case attempts to sum creep and fatigue damage and define a limiting interaction condition. In the present codes this is done on the basis of a time fraction to assess creep damage and a cycle fraction to assess fatigue damage. Data such as that shown in figure 6 have been found to be more compatible with a strain rather than time fraction to describe creep damage, a description consistent with the operative mechanisms. A new strain-based rule is currently being examined as a possible alternative to the existing time-based rule for design use in EFR.

(*b*) *Thermal striping*

If repeated thermal loading is fast enough, creep effects cannot develop because the strain rate is above that which can cause damage. In such cases, fatigue would dominate, and the one

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structure in the hot pool which is open to such damage is the ACS. Its lower end is 'thermally striped' by blobs of sodium exiting the core channels at different temperatures. Thermal striping is both spatially and temporally random in nature, making it extremely difficult to analyse in terms of thermal-loading history. A typical striping frequency is of the order of 1 Hz, and a first estimate of material resistance to thermal striping is given by high cycle fatigue curves, derived at the peak temperature. These indicate a limit of 60–70 °C to avoid crack initiation by fatigue (Lloyd & Wood 1979).

The secondary circuit pipework which links the IHXs to the steam generators is extensive, with straight runs, elbows and nozzle connections. The operating temperature of 525 °C is within the creep range for deformation, although this has been shown to be insignificant at the stress levels present. Creep damage is also not thought to occur to any significant degree at this temperature. The main integrity issue with this large-diameter (700 mm), thin (11 mm) piping is the prevention of a large leak, such as would occur with a double-ended guillotine failure (DEGF) leading to a large release of sodium and subsequent fire hazard. Thus leak-before-break is seen as a desirable feature for the piping. As with the primary vessel, the failure mechanism is ductile fracture, but at 525 °C properties are likely to be more degraded by ageing effects. The low internal normal operating pressure of 10 bar (10^6 Pa) gives rise to low primary stresses, but general thermal stress and transient thermal loadings will occur, giving rise to long- and short-range secondary stresses. Also, higher primary stresses could be generated by steam pressurization following a steam generator fracture. Fracture and LBB testing is in hand on secondary piping in both 'as fabricated' and aged conditions and no difficulty is foreseen in establishing a robust LBB case.

Threats

The transient strain ranges induced in hot pool structures are likely to be less than 0.5 %, thus making the most likely failure process cyclically accumulated creep damage. Although this occurs without any net extension of the component, the greatest threat is loss of creep ductility. An adequate level of ductility can be estimated simply by multiplying the maximum number of design transients (say 300) by the maximum amount of creep relaxation straining induced in a cycle (say 0.03 %). This gives a required ductility level of 9 %. Parent plate creep ductility is well above this value, 30–40 % being readily achievable. However, manual metal arc (MMA) weldments in austenitic steels can have ductilities below 10 %, and attention is being given to ensuring adequate ductility in welds, for example by use of the hot-wire TIG process.

Ductility can also be reduced by ageing and neutron irradiation. In long-term creep-fatigue testing, ageing is taken into account, but examination of the effect of neutron irradiation requires the complex creep-fatigue testing of irradiated material. Tests to date show a significant reduction in ductility due to helium formation in grain boundaries as a result of the (n, α) reaction. Helium contents as low as 10^{-4} a.p.p.m. have been shown to have an effect, and associated levels of irradiation would be experienced by hot-pool structures adjacent to the core such as the lower part of the ACS.

In the case of the redan, the threats are minimized because the maximum tensile stresses are generated on the cold side, with creep effects present only on the hot side, which is in compression. Under these circumstances, creep damage cannot develop.

As with all fatigue strength limits, a major threat is the presence of surface damage, which can provide a ready crack-initiation site. Concern that cracks could be initiated by the thermal shock–creep-fatigue process or at pre-existing flaws has led to consideration of a defect-tolerant approach to thermal striping. Both computational and experimental thermal striping work using a major new facility, SUPERSOMITE, is under way to demonstrate the adequacy of the integrity case for the lower areas of the ACS.

5. Steam generator units

Design

The steam generator of a fast reactor is a highly rated component called upon to remove heat effectively from sodium. It provides the boundary between hot, low-pressure sodium and pressurized steam–water and its failure can result in an exothermic sodium–water reaction. The subsequent pressurization of the secondary sodium circuit threatens both the integrity of the circuit and subsequent operation of the unit. If failure of the primary–secondary boundary within the IHX were to result, reactor safety could be threatened, while severe damage to a tube bundle would impose a large economic penalty.

The preferred design of steam generator for EFR is shown schematically in figure 7. It comprises a bundle of 1386 straight tubes welded at either end to thick tubeplates and supported along their length by a number of grid plates fixed directly to the main shell. At the top and bottom of the units the shell diameter increases to form the sodium inlet and outlet annuli respectively. The toroidal sections of the shell in this region are thinned locally to provide flexibility to accommodate displacements induced by tube-bundle–shell-temperature differences. The main operating characteristics of the steam generator are also given in figure 7.

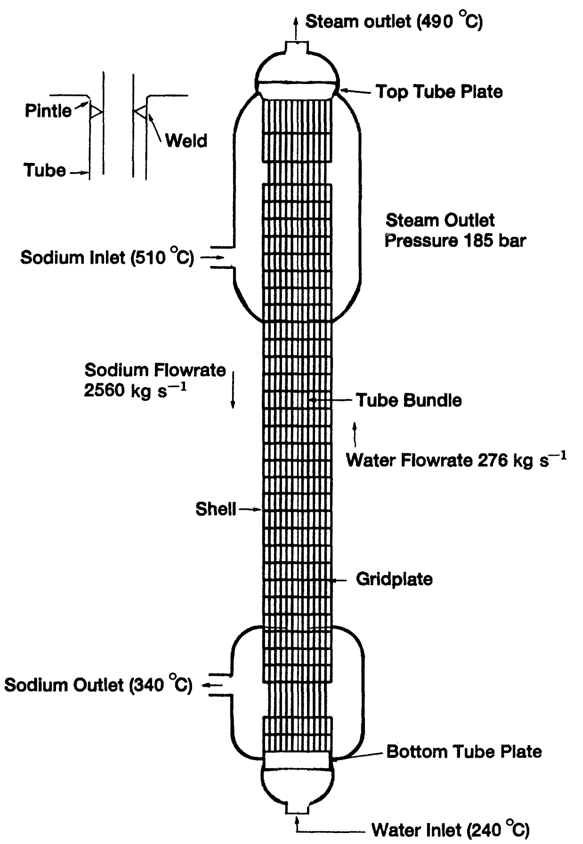

FIGURE 7. Straight tube steam generator.

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For EFR, tubes, tubeplates and shells will be fabricated from a 9Cr–1Mo steel, chosen for its high-temperature creep strength and corrosion resistance. A high-strength variant of the steel containing Nb and V has been chosen to give maximum economic benefit. Potential loading of the steam generator components comes from several sources; mechanical from the water–steam pressure of 185 bar, thermal from the temperature differences within the unit and vibrational from the high sodium flow through the unit. Design primary stress levels are well below the design limit of $\frac{2}{3}$ yield for the tubes and within the design allowable level for the shell under the maximum envisaged pressurization during a sodium–water reaction. Thermal and flow-induced vibrational loadings are minimized by good design with thermal hydraulic modelling support.

Integrity case

Several possible failure modes have been considered for the steam generator, particularly the sodium–water boundary.

1. *Tube buckling*. This could occur if the temperature difference between adjacent tubes rises to *ca*. 30 °C. Operation within this limit should be achievable with the existing design even when some tubes are plugged.

2. *Creep fatigue*. As with the IHX, thermal shocks which occur on trip can impose a creep-fatigue condition on the bottom section of tubes and bottom tubeplates. As with the austenitic steel, creep-fatigue design rules have been generated for the ferritic 9Cr–1Mo steels.

3. *Flow-induced vibration and fretting*. Careful attention is being paid in design to the tube–grid contact region to minimize vibration. The use of low-friction bush bearings or spray coatings should minimize fretting wear. Experience on prototype reactors has given considerable confidence in this area.

4. *Corrosion*. The high rating of fast reactor boilers induces heavy growth of magnetite on the inside tube wall. This is a strongly protective film, and good water chemistry control is specified to ensure protection throughout life.

5. The construction of a steam generator involves the welding of many thin-walled (2 mm) tubes to two very thick (300 mm) tubeplates. These are the only tube welds made, and provide potentially the most vulnerable location in the units. The current proposal is to weld tubes to a small annular upstand or pintle, on the tubeplate; this would give in effect a local 'tube–tube' weld which could be stress-relieved and inspected following fabrication. In operation, the top welds are above the sodium level in the unit, while an eggbox wastage plate has been devised to protect the bottom tube–tubeplate welds which are under sodium.

Threats

The choice of high-strength 9Cr–1Mo steel is a possible threat because it is an unproven material in service. There are indications of a loss of creep strength and ductility after long ageing times which could jeopardize creep and creep-fatigue performance, particularly if the ductility drops below 10%, as with austenitics. The steel will also be carefully monitored during development for adequate weldability and susceptibility to long-term weld failure by heat-affected zone cracking, the so-called type IV low-ductility cracking encountered in some ferritic steam generating plant. A lower-strength variant of the material, which has proven high integrity in service, is seen as a back-up, while the helical, austenitic steam generator design used in Superphénix is a second, but rather more expensive back-up.

A second threat concerns corrosion. In a single-unit steam generator such as that proposed for EFR, the conditions in a single tube change from water to superheated steam, with a region of the tube providing the water–steam boundary, or 'dry-out' zone. It is in this region that lack of strict control of water chemistry could give rise to the deposition of salts, giving enhanced corrosion.

Despite all precautions, it is expected that tube leaks will occur over the design life of a unit because of the large number of tubes present. Experience with PWR steam generators indicates a tube failure rate of one every three years per unit. If failure of a single tube can be detected before escalation to other tube failures, the unit can be shut down and the tube plugged, thus minimizing economic penalties. EFR steam generators are therefore to be equipped with leak-detection systems capable of ensuring this restricted failure pattern. Two systems are feasible; hydrogen detection and acoustic detection. Hydrogen detection is used on existing prototypes and picks up, in the gas space or under sodium, the hydrogen produced when steam leaking from a tube interacts with sodium. With current sensors, a significant leak (greater than 0.01 g s^{-1}) must have been present for at least 1 min for hydrogen detection to give a reliable signal. Acoustic detection senses the noise generated by an under-sodium leak, and can effectively detect larger leaks (greater than 50 g s^{-1}) within a few seconds.

A large under-sodium leak presents the greatest threat to the integrity of both the steam generator and secondary circuit. The meeting of steam and sodium produces an exothermic reaction which, when combined with the pressure difference, results in a flame jet at the leak site. This jet impinges on adjacent tubes, grids or shell and gives rise to several effects. The surface of adjacent cooled tubes in the flame can rise to more than 1000 °C, with the sodium-rich area of the flame being extremely corrosive. This produces local corrosion wastage that can penetrate the adjacent tube wall. The time for penetration is dependent on leak rate but is typically 10 s for a leak of 10 g s^{-1} and several hours for a leak of 0.01 g s^{-1}. The injection of high-pressure steam into the low-pressure sodium side of the unit increases the pressure on that side and throughout the secondary circuit. If large amounts (several kilograms) of steam were injected within a fraction of a second, the pressure rise would be so great as to threaten the circuit integrity, including the primary–secondary boundary in the IHX. In practice, such a scenario is very unlikely and has led to the definition of a design-basis accident (DBA). The currently proposed DBA is a single instantaneous guillotine failure of a tube, and the secondary circuit is designed to withstand the associated pressure rise.

Conclusions

The use of a high-boiling-point, high-thermal-conductivity coolant in fast reactors has a major influence on structural integrity. On the one hand the primary circuit is unpressurized and a robust structural integrity case can be made for the containment and key structural components in the core-support load path.

On the other hand, the high thermal conductivity of sodium coupled to the comparatively low thermal conductivity of austenitic stainless steel means that induced secondary thermal stresses are important, and it is essential to design the hot pool components against the combined effects of creep and fatigue and against thermal striping. Good progress is being made in developing and validating the codes to achieve this.

A further important factor is the sodium–water boundary in the steam generator units and the need to minimize exothermic sodium–water reactions. Here the choice of material still has

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to be fully validated, but the design margins are large. These margins and the use of sensitive instrumentation to detect any leaks of water–steam into the sodium at an early stage, allowing the plant to be safely shut down, are important components of the structural integrity case.

The lessons we have learned from existing plant emphasize the importance of achieving high standards of engineering and materials fabrication coupled to rigorous quality assurance. These are the basis of confidence in the structural integrity and operational reliability of the fast reactor.

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Discussion

J. D. Lewins (*Cambridge University, U.K.*). It has become general folklore that the capital cost of fast reactors will always be higher than other systems because they require an intermediate heat exchanger, diagrid and strongback. Is this still true or are there other features of the system with cost consequences?

M. Köhler (*Interatom, F.R.G.*). If I may answer that question. These specific components do not govern the overall cost of the system. The nuclear island of a fast breeder is indeed more sophisticated and therefore more costly as compared with those of an LWR (e.g. secondary loop, material, auxiliaries). On the other hand, the cost of the balance of plant and the fuel cycle have the potential of cost saving. So in total the power-generation cost of an FBR might be equivalent to those of an LWR in the long term.

J. Spence (*University of Strathclyde, U.K.*). Dr Eyre classified stresses as primary or secondary. Sometimes it is relatively easy to make this distinction. However, can it not be rather more difficult where the stress field is complex or if there is significant elastic follow-up? For example, in the region of the triple-point weld?

B. L. Eyre. It is easy to define the primary and secondary stresses though it may be very difficult to estimate the values, particularly of the latter.

B. Tomkins (*UKAEA, Risley, U.K.*). A good deal of consideration is being given to this question. Elastic follow-up does give rise to long-range effects such that secondary stresses can become quasi-primary. It is therefore important that validation of analyses be performed on sufficiently large pieces of the structure.

K. Q. Bagley (*UKAEA, Risley, U.K.*). It was indicated in the conclusions of the paper that the use of high-strength 9Cr–1Mo steels necessitated the highest standards of design and quality assurance. Does this imply a need for mandatory post-weld heat treatment in the construction of steam generators and other welded components?

B. L. Eyre. Further work needs to be done to validate the use of this material for various designs of steam generator. It is easier to heat-treat the pintle design which I showed.

D. Broadley (*NNC, Risley, U.K.*). I can confirm that all welds in the EFR steam generator will be post-weld heat treated.

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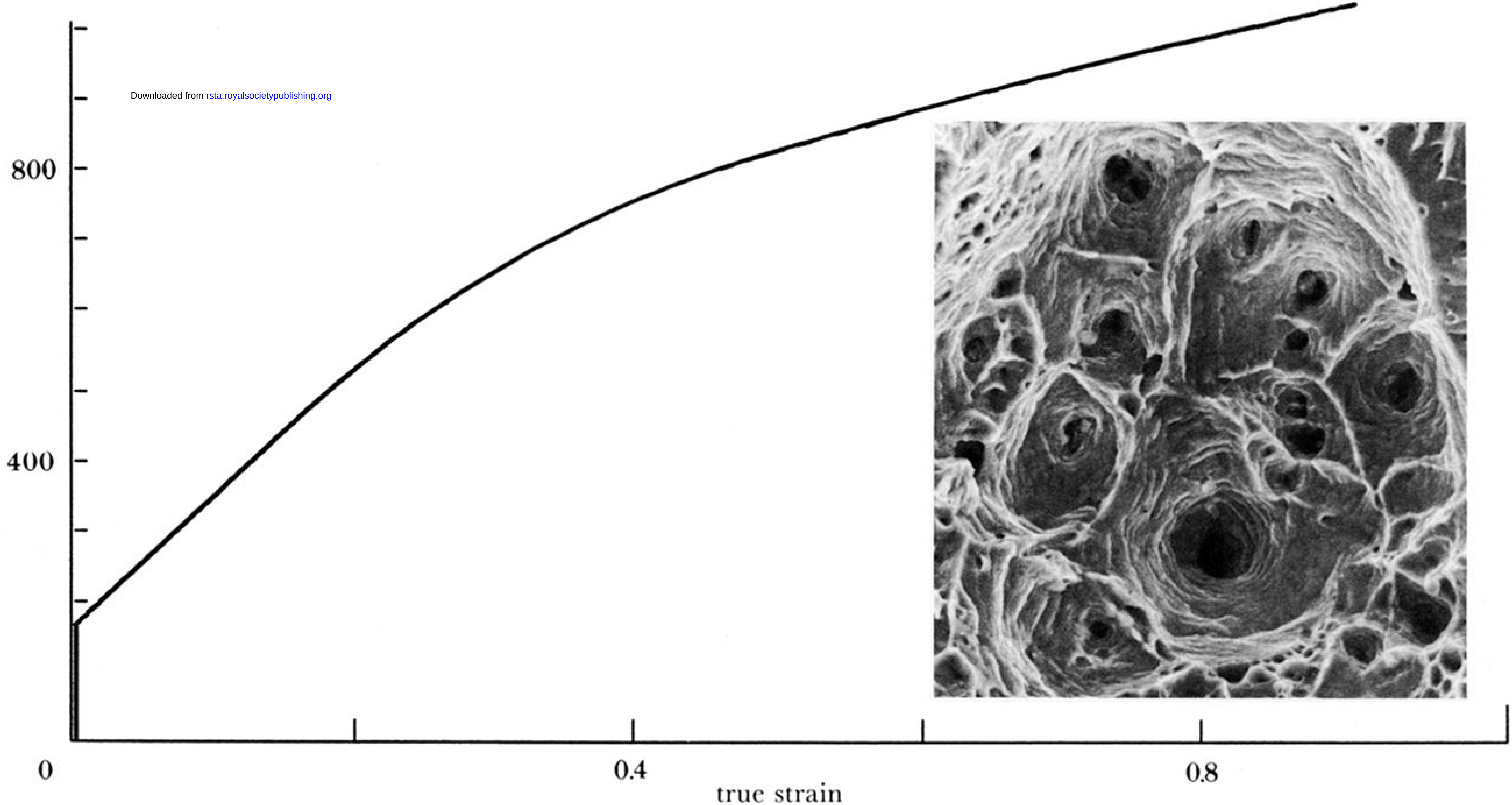

FIGURE 4. Tensile failure characteristics of stainless steel.

400 °C	ϵ_f	$J_i/(\mathrm{kJ\ m^{-2}})$	$J \Delta a = 1$ mm
plate	0.9	90	274
weld	0.5	35	224

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