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Structural integrity of Sizewell B—the way forward*

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Introduction

Sizewell B is Britain's first commercial pressurized water reactor (PWR). It was chosen following a review of reactor systems in the 1970s and manufacture and construction on-site commenced in 1987. Construction is now complete and the station is in commercial operation.

All the systems and components involved in or connected to the nuclear part of the station are designed and manufactured to the ASME Code Section III.[1] This in itself demands a high degree of design and analysis to assure structural integrity. It requires the manufacturer of a component to provide a design report which must show that the designed and constructed component will be capable of withstanding all the loads imposed on it. This must include all conditions, i.e. normal operation as well as large loads coming from anticipated 'upset' conditions. What is not required in the code is an assessment to show that the components remain intact if defects were present that might have remained undetected. Reviews by several groups have pointed out that for the Reactor Pressure Vessel (RPV), for example, to be built in the UK, they would expect such a safety case. The case was first put by the Light Water Reactor Study Group[2] (known as the 'Marshall Group') and then taken up as a requirement by the Nuclear Installations Inspectorate.[3]

Sizewell B can be considered to be an advanced PWR since it involves some of the latest and most advanced technology. It is the basis of the Nuclear Electric/Westinghouse bid to construct a similar station at Lungmen in Taiwan. Nuclear Electric also intends currently that the design would be used as a basis for any future power station project. The Sizewell B Structural Integrity Safety Case will remain as a basic building block of the overall safety case. It is hoped, however, to make some improvements and changes which will in no way undermine the case.

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Incredibility of failure

The Incredibility of Failure (IOF) approach has been used on Sizewell B. This involves demonstrating that under all conditions, the failure of certain components is so remote as to be considered incredible. It means that the integrity of these components is such that they must remain intact assuming other plant items do fail. For example, a complete failure of the main loop pipework (Fig. 1) must be withstood by the RPV and steam generator shell.

The IOF components are mainly those of the reactor coolant circuit and the purpose of this paper is to describe the approach adopted for the RPV, the steam generator shell, pressurizer and reactor coolant pump casing. The two arms of the safety case are achievement and demonstration of integrity.

Achievement of integrity

The measures taken include

(*a*) procurement of materials
(*b*) fabrication methods
(*c*) mechanical testing
(*d*) non-destructive examination.

These will be considered in detail for the RPV; however, the measures apply to the other components as well.

Procurement of material

The Sizewell B RPV is shown in Fig. 2. It comprises five ring courses, two heads and eight welded-in nozzles. The measures taken on procuring these materials are as follows.

(*a*) All the courses are made from ring forgings rather than welded plates. This has the advantage of eliminating vertical welds, a feature of many previous RPVs.
(*b*) The material chosen must be well-proven and possess good resistance to fracture. The steel-making, forging

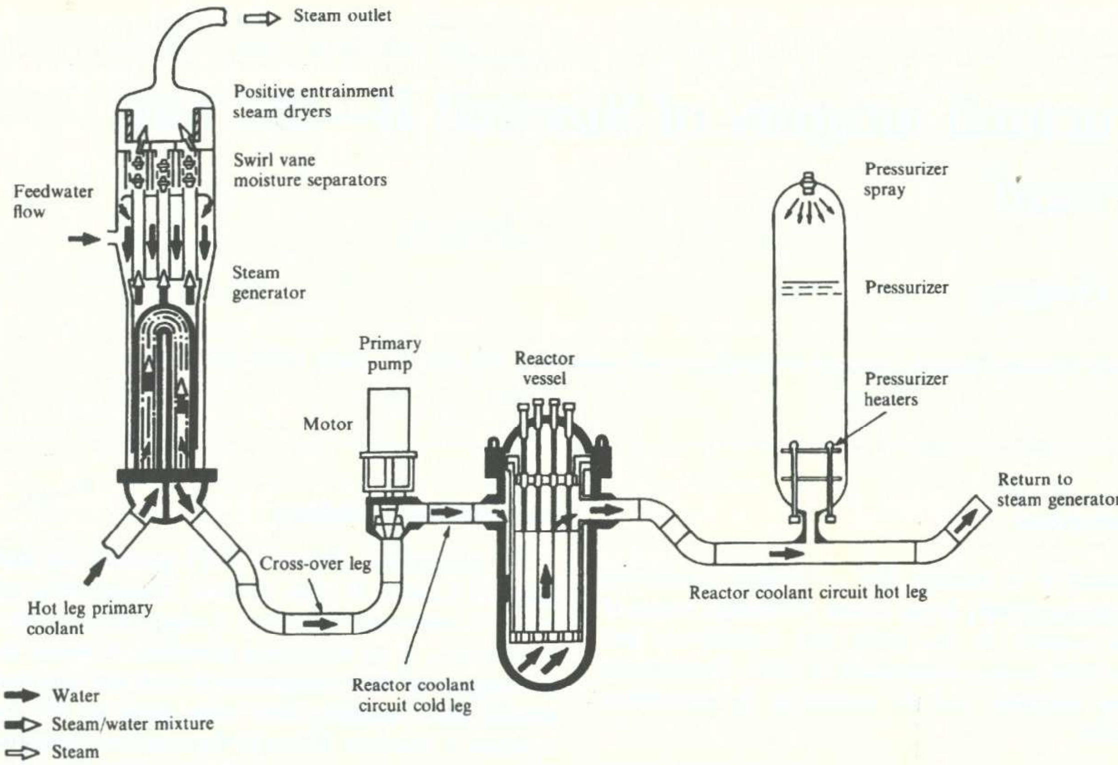

Fig. 1. Reactor coolant system

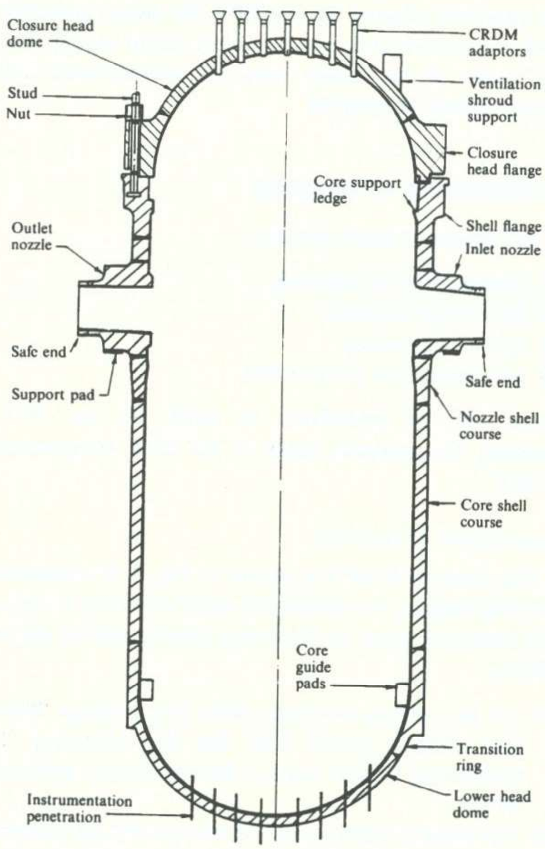

Fig. 2. Reactor pressure vessel—general arrangement

and heat treatment procedures to produce these properties had to be submitted for approval. Hence, the material chosen was to the forging specification ASME SA 508 Class 3.

(*c*) The chemical composition of the materials was further restricted from the ASME Code limitations, as shown in Table 1. These limits were to ensure good mechanical properties and resistance to time-dependent degradation of properties from either irradiation or temperature.

Table 1. Limitations on chemical composition of reactor pressure vessel

		SA 508 Class 3 forgings
Element	ASME composition: %	Sizewell B composition: %
Carbon	0·25 max.	0·20 max.
Nickel	0·4–0·1	0·4–0·85
Sulphur	0·025 max.	0·008 max.
Phosphorus	0·025 max.	0·008 max.
Silicon	0·4 max.	0·3 max.
Chromium	0·25 max.	0·15 max.
Vanadium	0·05 max.	0·01 max.
Antimony	—	0·008 max.
Arsenic	—	0·015 max.
Cobalt	—	0·02 max.
Tin	—	0·01 max.
Aluminium	—	0·045 max.
Hydrogen	—	1 ppm (product)

Fabrication methods

Well-proven fabrication methods were required and naturally, the first requirement was a fabricator with a proven track record in RPV manufacture. Hence the contract for the Sizewell B RPV was placed with Framatome who had already manufactured many RPVs for both the French programme and overseas. Their normal welding methods were used, namely submerged arc narrow gap structural welds and 60 mm wide strip cladding, also using submerged arc.

As additional measures to ensure high fabrication integrity, the following safety requirements were imposed.

(*a*) Weld procedure qualification tests were performed on welds which were representative of the size and geometry of the welds in the actual reactor pressure vessel.
(*b*) Examinations of the weld procedure qualification test specimens were carried out to demonstrate a weld of high integrity, free from defects. These examinations included non-destructive examinations. Weld procedure qualification tests for cladding are particularly important, in demonstrating freedom from under-cladding cracking.
(*c*) Levels of pre-heat were imposed to avoid the possibility of heat-affected-zone (HAZ) hydrogen-induced cracking. The minimum levels employed were 175°C for welds and 150°C for cladding. Post-welding hydrogen removal treatments were also employed.
(*d*) All heat affected zones in ferritic RPV material were subjected to post-weld heat treatment.
(*e*) The locations of repair welds have been carefully documented.

Mechanical testing

It is, of course, conventional to test the materials of construction to show they possess the properties that we would expect. Normally tensile and Charpy impact testing would be employed. For Sizewell B two additional tests with stringent acceptance standards were applied.

(*a*) Fracture toughness *J*-testing of the ferritic forgings was performed using procedure ASTM E813 methods with some modification of the data analysis, see Fig. 3. These tests were also performed on the weld procedure qualification test pieces and the acceptance values requires are shown in Table 2. It is believed this is the first time such tests have been specified for a pressure vessel. In practice, the values obtained comfortably exceeded the minimum values required. This provided an input to the detailed fracture analysis that was performed as part of the demonstration of integrity.
(*b*) The ductile–brittle transition temperature was measured using the ASME RT_{NDT} requirement. The maximum allowed values were −22°C for nozzle forgings and −12°C for other forgings and welds. These temperatures are sufficiently low to allow for shifts in transition temperature due to irradiation or thermal ageing.

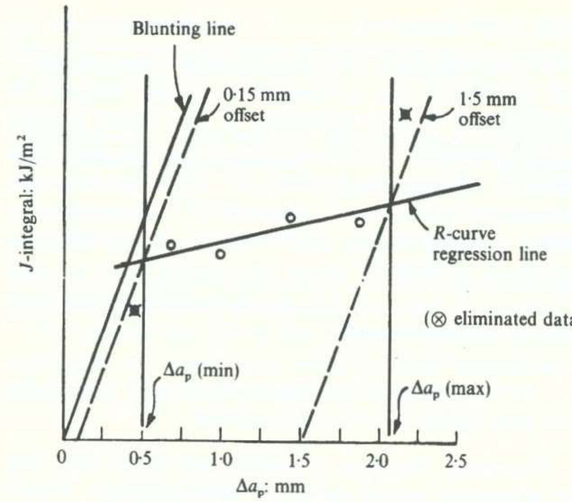

Fig. 3.

Table 2. RPV fracture toughness testing acceptance standards

Test temperature	43°C		293°C	
Material	Forging	Weld	Forging	Weld
J_{IC}: kJ/m^2	156	156	136	136
K_{JC}: MPa$\sqrt{m}$	187	187	169	169
J_{a2}: kJ/m^2	720	500	400	250
K_J: MPa$\sqrt{m}$	300	300	—	—

J_{IC} is the initiation value of J
K_{JC} is the value of K computed from J_{IC}
J_{a2} is the value of J at the intersection of the linear $J\Delta a$ regression line with $\Delta = 2$ mm (Δ is crack growth)
K_J is the value of K computed from J below which no specimen may show cleavage instability.

Non-destructive examination

An important stage in achieving a vessel of high integrity is to show by non-destructive examination methods that the RPV is free from defects. The extent of inspection performed is shown in Table 3. This is far in excess of any inspection programme carried out on any other pressure vessel; to complete such a programme is in itself a remarkable achievement. The parties involved in these inspections were:

(*a*) Framatome and their subcontractors who performed the required manual ultrasonic inspections set down in the specification, as well as the other required inspections, namely radiography, dye penetrant, magnetic particle and eddy current
(*b*) Babcock Research who carried out the additional automated inspections of the vessel
(*c*) Rolls Royce and Associates who carried out an inspection of the vessel at site prior to it entering

Table 3. RPV inspections

	Forgings	Circumferential welds	Nozzle welds
Framatome	Pre-stress* relief	Pre-stress relief	Pre-stress relief
Framatome	—	Post-stress relief	Post-stress relief
BEL	Pre-stress relief	—	Post-stress relief
BEL	—	Post-hydro	Post-hydro
RR&A	Nozzle radii pre-service	Pre-service	Pre-serivce

* Performed by Framatome Subcontractor, Creusot-Loire. Core shell inspection performed by OIS.

service using the equipment which will have to be used during in-service inspections.

The actual ultrasonic inspection procedures used had to incorporate several significant features.

(*a*) Ultrasonic examination of the forgings and welds had to be carried out during manufacture. (Note that the ASME code does not require any ultrasonic examination of welds until in-service inspection begins.)
(*b*) The ultrasonic examination had to be performed by the RPV fabricator to laid-down acceptance standards which were built into the contract.

These acceptance standards emphasized the requirement for the ultrasonic technique to be capable of not only finding defects but also determining their size in both the length and through-thickness directions. This through-thickness defect acceptance size ranges from 3 mm near the surface to 12 mm near the centre of the wall.

It is, of course, important to have confidence that the ultrasonic procedures being used are, in fact, capable of finding the defects of interest. On Sizewell B this led to the concept of inspection validation.[4] The part it plays in the safety case for Sizewell B is significant and will be discussed later in the Demonstration of Integrity.

Other IOF components

Very similar requirements were imposed on the other IOF components; the only effect not needing consideration was that of potential irradiation embrittlement. Comparisons are shown in Table 4: steam generator compositions; Table 5: steam generator and pressurizer pre-heats; and Table 6: steam generator inspections.

Table 4. Limitations on chemical composition of steam generator

Element	Composition limitations of forgings and welds: % max.
Carbon	0·20
Sulphur	0·010
Phosphorus	0·012
Antimony	0·010
Arsenic	0·020
Tin	0·015
Chromium	0·15 (Primary side only)

Table 5. Minimum pre-heat requirements

Component	Welds	Cladding
RPV	175°C	150°C
Steam generator	170°C	150°C
Pressurizer	150°C	150°C

The reactor coolant pump casing is an austenitic stainless steel casting and therefore has different requirements. The same principles however, have been applied, i.e. limits on chemical composition, control of fabrication, mechanical testing and even an ultrasonic inspection.

Demonstration of integrity

Once a component of high integrity had been achieved for Sizewell B, it was an additional requirement to show that under all design basis conditions, the vessel would remain intact. It must be demonstrated that defects that may have gone undetected during manufacture will not cause failure during the lifetime of the plant. For the Sizewell B RPV a detailed study of the Demonstration of Integrity has been presented at a previous conference.[5] The essential steps involved in the process are listed below.

(i). Consideration of RPV inspections

Assumptions are made about the size of defect that can reliably be detected and sized. This is known as the validation defect size, which the inspection validation has shown can be detected with a high reliability, using the correct procedures, equipment and actual NDT operators. This was carried out for the Sizewell B RPV at the

Table 6. Steam generator inspection

	Forgings	Welds	Tubing
Supplier	Pre-stress relief		UT + ET
OIS	Post-stress relief (×2)		
BEL		Pre-stress relief	
BEL		Post-stress relief	
OIS		Pre-stress relief	
OIS		Post-stress relief	
OIS		Pre-service	
Conam Nuclear			Pre-service (ET)

Inspection Validation Centre (IVC). The requirements were that capability should be measured in test pieces of appropriate size and geometry representative of parts of the vessel and containing defects representative of the real defects that might occur in actual vessels. The types and orientations of the defects to be detected, with the sizes, were specified to the IVC.

This procedure again represents a major additional requirement for Sizewell B compared with all other previous reactor pressure vessels. All the inspections detailed in Table 3 were validated in this way and it was an expensive, time-consuming exercise.

(ii). Validation factor

To ensure that an adequate margin exists between the tolerable defect size and the size of defect which may be present in the vessel, account is taken of the tendency for the assumed defect (the validation defect size) to grow owing to cyclic loadings during service. A fatigue crack growth calculation was therefore performed for the validation defect size which was 25 mm through-thickness in RPV welds and 10 mm through-thickness for forging material. The margin between the tolerable defect size and the validation size (including calculated fatigue crack growth) is termed the validation factor. It is defined by the equation:

$$\text{Validation factor} = \frac{\text{Limiting defect size}}{\text{end-of-life defect size}}$$

where the end-of-life defect size is the validation size with calculated fatigue crack growth added, and the limiting defect size is the size of defect which is calculated by fracture mechanics as being capable of causing failure of the vessel under the most onerous loading conditions. A target value of 2 has been assumed for this factor.

(iii). Vessel loadings

To calculate the above limiting defect size, one first needs to know the loadings that will be applied to the vessel including those due to pressure, fluid temperature changes, mechanical loadings and welding residual stress. These loadings are normally calculated by the primary circuit designer since it requires knowledge of the other components, including pipework, that can impose such loads on the vessel. For Sizewell B this was Westinghouse.

(iv). Stress analysis

A stress analysis must be performed to show that the vessel can safely withstand the imposed loads from (iii) above. Most of the stress analysis used in this safety case was generated by Framatome as part of the design stress report required by the ASME Code. These stresses formed the basis of the fracture assessment for the more likely vessel loadings. The fracture assessment for other vessel loadings required the generation of some further stress analyses.

(v). Fracture analysis

The fracture analysis is a key part of the demonstration of integrity, taking stress analysis and calculating the limiting defect size at end-of-life. This is the limiting defect size required by the equation for validation factors shown in (ii) above. The method used for this fracture analysis was the R6 Revision 3 assessment procedure, produced by the CEGB.[6] This permits account to be taken of both fracture and plastic collapse in assessing the integrity of the structure.

The analysis has shown that validation factors for all loadings in the vessel, even under the most onerous and improbable loading conditions, are acceptable.

Other components

The same procedure as above has also been used on the other ferritic IOF components. The analysis for the austenitic reactor coolant pump bowl was somewhat different in detail but used the same principles.

The way forward

In considering the way forward, the first thing to examine closely is the advantage of not moving forward at all or maintaining the status quo. The Sizewell B structural integrity safety case took considerable effort and not insignificant amounts of cash to put together. It has been vigorously reviewed and by way of their endorsement of the Sizewell B Pre-Operational Safety Report (POSR), it has been approved by the NII. If nothing were to change, the maximum benefit that could be gained from the Sizewell B case is that most of the work would not have to be repeated. It is only really the time-dependent approvals such as NDE operator approvals and validations that would need some further attention.

Replication

Replication is a way forward. Replication could be employed to gain the maximum advantages over maintaining the status quo. However it must be said that replication is neither completely possible or completely desirable.

It is not completely possible because the loads and transients imposed on the station structure cannot be identical. Even if another station were to be built on land adjacent to Sizewell B, the soil properties and seismic assumptions would not be exactly the same. Some revisiting of the stress and fracture analysis is therefore inevitable.

In an increasingly commercial world it becomes essential to show that nuclear power is competitive with other energy forms. Hence there is an enormous incentive to reduce costs and this would result in changes being made from Sizewell B. There is also significant effort being put into improving fuel performance, giving increased output and longer periods between outages. This again will necessitate further revisits to the stress and fracture analysis.

The extent to which replication is required or what in fact we mean by replication on a more detailed level is not completely clear. If we take the case of the reactor pressure vessel what would have to be repeated? Clearly the overall dimension and basic materials would have to remain the same. It is, however, possible to make improvements in manufacturing that would reduce costs.

The Sizewell B RPV was completed in 1991, now 4 years ago. Possible fabrication changes could be considered and clearly the impact of such changes would have to be assessed. For example, if we change the weld procedure to place more reliance on one-sided welds, can we make the same assumptions on materials properties and the likelihood of producing defect-free welds? Also, if we change the cladding process can we still expect the same inspection performance when examining structural welds from the inside of the vessel, i.e. through the cladding?

It is, therefore, important that the effects of all changes to the Sizewell B design are assessed in terms of their effect on the structural integrity safety case, otherwise what appears to be a cost saving could prove to be completely the opposite.

Materials

We will definitely be making some changes to the materials of construction brought about mainly by operational experience of current PWRs. Top of the list is Inconel Alloy 600 which is continuing to give corrosion type problems. There has been an extensive history of stress corrosion cracking in Inconel 600 steam generator tubing. For Sizewell B measures were taken to avoid such problems, most notably the use of Inconel 690 instead of Inconel 600 as the tubing material. More recently, stress corrosion cracking has been observed in other PWR primary circuit components, i.e. penetrations in the RPV and pressurizer. There are other components made from Inconel 600 materials and its associated weld metals Inconel 82 and 182.

We have consequently carried out an assessment of the stress corrosion susceptibility of these materials, taking into account component processing history and the resultant microstructure, residual stresses introduced during fabrication and the component operating conditions, in particular temperature.[7] This report derives a Relative Susceptibility Index: the higher the number, the more likely is stress corrosion to occur. Some of the results are shown in Table 7 which effectively compares the components to CRDM penetrations in which some cracking has already been observed. We will use this work to assess the need or desirability to replace Inconel 600 components.

Table 7. Assessment of stress corrosion susceptability of PWR components

Item	Component	RSI
Outlet nozzle/safe end weld metal	Pressurizer	219
CRDM penetrations	RPV	100
Channel head stub runner/divider plate	Steam generator	68
Nozzle dam mounting ring	Steam generator	62
Vent pipe	RPV	43
Channel head tubesheet cladding	Steam generator	22
Tube to tubesheet weld	Steam generator	12
Instrument penetrations	RPV	4
J-tubes	Steam generator	2

Non-destructive examination

As can be seen from Tables 3 and 6, the amount of ultrasonic inspection applied to the Sizewell B IOF components was extensive, and we consequently conducted an in-depth review of these inspections considering the cost-effectiveness together with their contribution to overall integrity.[8] This considered the results of the inspections on Sizewell B and the results of their validations. Conclusions can be drawn on the effectiveness of each inspection which was not always consistent with the time and cost taken to perform the inspection. The work concludes that some changes could be made that would result in significant financial savings and a 30 week reduction in the construction programme with no adverse effect on plant structural integrity.

Inspection qualification and validation

The concept of inspection validation came with the Sizewell B Project. Since then it has been noticeable that the concept of assurance of inspection capability has grown and is now evident in other countries, see for example ASME XI Appendix 8.[9] In Europe the European Network for Inspection Qualification (ENIQ) has been formed with active Nuclear Electric participation. The lessons learnt from the Sizewell B project are considerable and are being fed into this network. The most striking lesson is that blind trials of NDT procedures and operators, the most expensive part of inspection validation, might not always be necessary. Emphasis is now being placed on technical justifications being produced for each inspection explaining how the defects of concern will be detected. Such justifications can be assessed theoretically, and if successful can be confirmed by a smaller amount of practical testing which need not even be conducted blind. This is the direction being pursued in Europe via ENIQ and it will be followed for future inspection qualifications and validations conducted in the UK.

Conclusions

The IOF approach has been used for several of the key components of the Sizewell B Nuclear Power Station. The manufacture of these components has been carried out to achieve a high level of integrity which has been measured by mechanical testings and a programme of proven non-destructive examinations.

An analysis has been performed to demonstrate that even if defects were present, the components will remain intact for the expected lifetime of the plant. In all cases, acceptable validation factors have been obtained, generally in excess of two. It can therefore be concluded that the failure of the components is so unlikely as to be deemed incredible.

Aspects of the structural integrity case for IOF components are now being examined for future power station construction. Changes will be proposed which will be advantageous and cost-effective without in any way diminishing the well-proven case for Sizewell B.

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