Health and Safety
Executive

HEALTH AND SAFETY EXECUTIVE
HM NUCLEAR INSTALLATIONS INSPECTORATE

New Reactor Generic Design Assessment (GDA) - Step 2

Preliminary Review Assessment of:
Structural Integrity Aspects of AREVA/EdF EPR

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FOREWORD

Structural integrity here means the integrity of metal pressure boundary components and their supports; it also includes vessel internals.

Due to resource limitations within NII, this preliminary review was conducted by a member of staff of Division 1 on behalf of Division 6. It was only possible to devote a short period to this preliminary review. In the time available, it was not possible to produce an Assessment Report in the usual format. Instead, this Assessment Report consists of the text of a Summary and the Notes made during the preliminary review.

Although only a short time was available for this preliminary review, it was sufficient to:

> Review the main Claims in the safety case sequence of Claims, Arguments and Evidence and take note of the nature of the Arguments and Evidence to support the Claims. For Step 2 this is sufficient. Testing Arguments and Evidence in detail would be the subject of any later Steps in the GDA process;
>
> Indicate where there may be particular areas for review and assessment during any later Steps in the GDA process;
>
> Highlight where there may be areas for review and assessment in any later Steps in the GDA process that have the potential to affect long lead-time items (i.e. components that need to be ordered early in any construction sequence).

This preliminary review has used the GDA Guidance (see Summary). The review is also based on the NII Safety Assessment Principles (SAPs). In addition, for potential future more detailed assessment, regard has been taken of the following Licence Conditions:

> 14 - Safety Documentation, paragraph 1. Adequate arrangements for the production and assessment of safety cases.
>
> 23 - Operating Rules, paragraph 1. In respect of any operation that may affect safety, produce an adequate safety case to demonstrate the safety of that operation.

SUMMARY for EPR
Structural Integrity of Metal Components and Structures

For Step 2 of GDA (ref 1), HSE's review of design concepts and claims for the integrity of metal components and structures includes aspects of:

2.19 The safety philosophy, standards and criteria used

2.21 The Design Basis Analysis / fault study approach

2.23 The overall safety case scope and extent

2.24 An overview of the claims in a wide range of areas of the safety analysis

A fundamental aspect of the NII Safety Assessment Principles for integrity of metal components and structures (pressure vessels and piping, their supports and vessel internals), is the identification of those components where the claim is gross failure is so unlikely the consequences can be discounted from consideration in the design of the station and its safety case.

For the EPR, implicit in the submission is that gross failure of the Reactor Pressure Vessel is discounted, together with discounting gross failure of any of the four Steam Generators and the Pressuriser. By comparison, gross failure of certain piping is explicitly discounted (a claim) based on a set of arguments and evidence referred to as 'Break Preclusion'.

The NII SAPs encompass the claim that gross failure of a component is so unlikely it can be discounted (SAPS paragraph 238 to 279, and in particular paragraphs 238 to 253). Then the emphasis falls on the arguments and evidence to support the claim that gross failure is so unlikely it can be discounted. Similar claims have featured in safety cases for operating nuclear stations in the UK and the supporting arguments and evidence have been considered by NII. NII would assess the strength of arguments and evidence on the basis that gross failure was discounted. The structural integrity assessment would not be modified because certain consequences of gross failure are considered elsewhere in the safety case.

The Step 2 review has not examined in detail the arguments and evidence to support claims on structural integrity of metal components and structures. However some of the items in question are long lead-time components and relevant general matters which would likely arise in Step 3 / 4 assessment are:

material specification for ferritic forgings and welds to be used in main vessels;

design of the RPV which as submitted for Step 2 indicates a circumferential weld at core mid-height;

nature of the arguments and evidence to support integrity claims for some piping ('Break Preclusion').

Overall, we conclude AREVA/EdF have provided an adequate overview of the claims made for structural integrity of metal components and structures. But for Step 3/ 4 there would need to be an explicit listing of those components where gross failure is claimed to be so unlikely it can be discounted. AREVA/EdF have also provided some coverage of the type of arguments and evidence to support the claims. Subsequent review of arguments and evidence against the NII SAPs may reveal areas where a different emphasis or modification to the arguments and evidence is needed.

Ref 1. Nuclear Power Station Generic Design Assessment - Guidance to requesting Parties. Version 2 (16 July 2007). HSE.

AREVA/EdF - UK-EPR

NOTES OF PRELIMINARY REVIEW OF THE FUNAMENTAL SAFETY OVERVIEW - STRUCTURAL INTEGRITY

11 JANUARY 2008

These notes are the outcome of my preliminary review of the structural integrity aspects of the “Fundamental Safety Overview” documentation provided by AREVA/EdF for the UK-EPR.

Structural integrity here means the integrity of metal components and structures; this includes metal pressure boundary components and their supports and reactor internals. In this preliminary review I have concentrated on metal pressure boundary components; I have not looked at reactor internals.

I have looked in particular at the following parts of the Fundamental Safety Overview:

Volume 1 Chapter A and parts of Chapter H;

Volume 2 Chapters C4, C6, E2, E3, E4, J3, J5 and P0.

The notes below are in the form of 9 extended, numbered bullet points. The notes include comments and questions. The comments may be as significant for the future as the questions. The questions are highlighted.

1. Pipework

Break Preclusion. This is acceptable in principle but we will need to see and be content with the detailed arguments and evidence that support the claim of Break Preclusion. I assume for the primary circuit pipework that Break Preclusion includes both sides of the safe ends at vessel nozzles. In principle, Break Preclusion could be acceptable for both the main primary loop pipework and the main steam lines, as set out in the Overview.

For assessment of the detailed structural integrity arguments and evidence supporting the claim of Break Preclusion, we would ignore the use elsewhere in the safety case of “2A” breaks (e.g. for the capability of the Safety Injection System and stability of major components on their supports). From a structural integrity point of view, the claim is either Break Preclusion or it is not, there is not ‘halfway’ point.

NII SAPs 2006 Edition, paragraphs 238 to 253 summarise NII’s approach to such safety claims (with paragraphs 254 to 257 explaining the overall difference in approach when the claim is less than “Break Preclusion” or similar concepts).

I note that “Leak Before Break” (LBB) plays only a supporting role in the overall integrity argument for pipework designated as Break Preclusion (Sub-Chapter E.2

Table 1). I am not sure what contribution LBB can make in the case of the main steam lines, unless there is leak detection close to the main steam lines, especially outside containment in the valve houses.

For pipework, reference defects of 20mm long by 5mm deep (though wall extent) are mentioned (C4.2, 2.3.1.1.1 and J5 2.1.1.2.3). Fatigue crack growth over life is claimed to extend the depth of such a crack to no more than 25% of wall thickness. This is for wall thickness of between 76 and 96mm for the primary circuit main loop pipework. Qualification of inspection methods (e.g. ultrasonic inspection) to justify a high confidence of detecting and sizing defects 5mm x 20mm could be challenging in our experience, especially for stainless steel - even if it is forged.

Another approach is to determine 'limiting defect depths', apply inverse fatigue crack growth to determine the start-of-life defect (that could grow to the limiting defect depth by end-of-life) and then seek to justify the ability of the manufacturing / in-service inspection techniques to detect and size something notably smaller than this predicted start-of-life defect.

For the main steam lines, there is the matter of the material of construction, the choice of which needs to be consistent with the claim of Break Preclusion. This is particularly the case at complex locations such as branch points for the relief valves, containment penetrations, and restraint point downstream of the Main Steam Isolation Valves.

For pipework that is not covered by the claim of Break Preclusion, I note the criteria for postulating locations for leaks and breaks are covered in C4.2 2.2.1.2. These appear similar to, but not the same as those in USNRC Branch Technical Position BTP 3-4 which is referenced from NUREG 0800 Standard Review Plan SRP 3.6.2.

QUESTION: What are the differences between the leak and break location criteria in the Fundamental Safety Overview and USNRC Branch Technical Position BTP 3-4 (Revision 2 March 2007)?

2. Main Pressure Vessels

Main pressure vessels include: Reactor Pressure Vessel, Steam Generator primary side channel head and secondary side shells, Pressuriser, Safety Injection System Accumulators.

The integrity of these main vessels is essential for nuclear safety. Gross failure of a Steam Generator secondary side shell or a SIS Accumulator would be a substantial internal hazard. There is a good deal of coverage of Break Preclusion for pipework in the Overview and some repetition. By comparison, there is rather less coverage of the integrity of these main vessels.

The integrity claim for these main vessels amounts to the statement in Volume 2 Chapter C.4, 4.1 that breaks are discounted because the components are designed to RCC-M.

If gross failure of the main vessels is discounted (another form of 'break preclusion' claim) there will need to be detailed arguments and evidence to support that claim for each relevant vessel.

It may be that the RCC-M code is sufficient to provide the arguments and evidence to support discounting gross failure of a main vessel shell. NII will need to understand the current (or relevant) version of the RCC-M code. We know RCC-M is similar to ASME III. But from our experience of applying ASME III to a PWR, we know we required additional and more restrictive requirements. These were mainly in the areas of materials specification, material toughness, fracture mechanics analyses and manufacturing inspections (extent and capability).

Looking across the UK nuclear industry, there is a degree of familiarity with ASME III. We understand the EPR design is likely to be offered in different regulatory jurisdictions around the world.

QUESTION: As a practical matter for the UK, could the metal pressure boundary components of the EPR for the UK take ASME III as the basic design code?

SIGNIFICANT POINT: NII has a strong preference for avoiding welds adjacent to the reactor core, even if the neutron dose is lower than in earlier designs. The drawings and text in the Overview show a circumferential weld in the Reactor Pressure Vessel more or less level with the mid-height of the core. We have a strong preference for a single forging to span the core so there is no such circumferential weld. Our understanding of forging technology is that a single forging for the cylinder section of the RPV body is reasonably practicable.

The overview expresses neutron dose to the RPV in terms of nvt (presumably E>1MeV). We think it good practice also to express the dose in terms of Displacements per Atom (dpa). This allows better account of overall neutron energy spectrum. Shielding will reduce total irradiation, but will also tend to increase the relative proportion of thermal neutrons.

I note for the RPV the design shows set on nozzles. Manufacturing and in-service inspectability of the inner fusion faces of the nozzles welds needs clarification.

QUESTION: For ultrasonic inspection of the RPV nozzle welds, is inspection from both sides of the weld possible? In particular, is ultrasonic inspection from the inner fusion face side of the weld possible in manufacturing inspection and in-service inspection? Once the vessel is clad on the inner surface, is ultrasonic inspection of the inner fusion face side of the nozzle welds still a sensitive means of inspection?

3. Overpressure Protection

Volume 1 Chapter A Table 1-1.

Primary Side. As usual for PWRs (and other reactor designs), the overpressure protection depends on a combination of reactor trip and relief valves.

Secondary Side. The Overview states a new approach is used. The new basis for steam side overpressure protection also depends on a combination of reactor trip and relief valves.

There is nothing in the NII SAPs to preclude this. NII SAPs 2006 Edition para 236 mentions the combination of relief valves and an active protection system to terminate generation of energy or mass input.

The UK Pressure Equipment Regulation (PER 1999) and the Pressure Systems Safety regulations (PSSR 2000) do not preclude the proposed approach.

NII will expect Overpressure Protection Reports to be part of the detail supporting the structural integrity claims of the primary and secondary side pressure boundaries. The Overpressure Protection reports must include the active system aspects, reactor trip system, as well as the relief aspects to show how the overall overpressure protection is achieved.

QUESTION: Overview document (Volume 1 Chapter A 3.3.2) indicates the following steam side relief capacity:

1x discharge line 50% of full flow

2x discharge lines each 25% capacity

Is there 100% flow relief capacity, assuming all relief paths fully open?

4. In-Service Pressure Tests

Periodic in-service hydraulic tests are proposed for the primary circuit (fuel removed) to 1.2x design pressure.

UK practice (nuclear and non-nuclear) is not as a routine matter to subject pressure equipment to such a high pressure test.

The ability to do such test could be useful if major repair work was ever needed during service.

Our view is that overpressure tests (above design pressure) are useful in confirming the combination of wall thickness and material strength (e.g. yield strength) is adequate; but such test are not a good indicator of the presence of cracks, especially in ductile materials. The concern is such tests in service will add an increment of damage to a sub-surface crack, but not reveal the crack.

We know there have been occasions when such overpressure tests in service have revealed leakage, but taking the overall balance or risks and benefits, we would prefer routine pressure testing in-service not to exceed design pressure.

5. Main vessels ferritic forging material

The Overview document states the forging material will be 16 MN 15.

We know 16 MN 15 is similar to ASME A508. In the UK, A508 has been used, but with extra and more restrictive conditions compared with the ASME A508 specification. The additional requirements in the UK for A508 included:

> restrictions on the maximum Carbon content, maximum Copper content (for RPV) and lower allowed quantities of impurity elements;
>
> procurement specification which included minimum material fracture toughness in terms of K and J, not just Charpy Impact Energy.

QUESTION: Can AREVA/EdF supply the chemical composition of 16 MN 15 as intended to be used to fabricate EPR main pressure vessels? What fracture toughness requirements are included in the material supply specification for 16 MN 15 as used for EPR main vessels? What are the corresponding requirements for welds?

6. Load Combinations

The Overview document, Volume 2 Sub-Chapter C6 Table 1 shows load combinations. A footnote to the table and the second column under the "Emergency Condition" heading shows LOCA plus Design Earthquake (DE) as a combination.

Elsewhere in the document, mention is made of a Safe Shutdown Earthquake (SSE) and Design Earthquake (DE). I am not sure whether there are 2 levels of earthquake loading defined for different aspects of the design / safety justification process.

QUESTION: For analysis of pressure boundary components, can AREVA/EdF confirm the earthquake loading to be used and what internal and external hazard loadings are combined in the analysis? Alternatively, are internal and external hazard loadings only considered one by one, combined with normal operation loads?

7. In-Service Inspection

The Overview is somewhat vague on this topic; overall there are statements that in-service inspection requirements will be determined during design.

NII's non-prescriptive approach can deal with different ways of achieving safety. We do not specify a particular code or method for determining in-service inspection requirements. However we would expect to take a close interest in the specific intent, however it was derived.

I note the sentence in the Overview (C.4, 2.3.1.2.2):

The In-Service inspection must also cover the lower risk areas in terms of the defence in depth principle."

I believe NII would concur with that statement. In addition, from experience, simple in-service inspection (visual) has been notably useful in detecting hitherto 'unexpected' degradation.

8. Reactor Coolant Pump

The Overview document states the RCP bowl will be cast stainless steel.

From a structural integrity point of view (including inspectability), NII would prefer a forged pump bowl.

The pump bowl is a complex shape. If a casting manufacturing route is used, it is likely the structural integrity justification (presumably discounting gross failure) will be challenging. It is likely that one or more out of 4 cast pump bowls would contain major manufacturing repair weld zones and these could be of order half wall thickness in some cases. The justification of the integrity of such repairs could be challenging (material toughness, residual stress, inspectability of the repair zones).

The integrity of the RCP flywheel needs to be covered. Failure of a flywheel could lead to missile generation. Either the consequences of failure of a RCP flywheel need to be dealt with in the design, or a structural integrity argument made to justify discounting gross failure.

QUESTION: For the Reactor Coolant Pump Bowl, is a forged stainless steel component manufacturing route available? Is gross failure of the Reactor Coolant Pump flywheels discounted or is the potential for missiles generated from such failure included in the design?

9. Steam Line Valve Houses on Top of Safeguard Buildings 1 and 4

Volume 1 Chapter A (page 127) of the overview contains the following:

> Main Steam and Feed Water Valve Stations
>
> “Thus physical separation and the main steam valve house enclosure provide protection against external hazards.”

QUESTION: Does the valve house enclosure provide the same level of protection against external hazards as the protection around Safeguard Buildings 2 and 3 and the Fuel Building? Alternatively, is the plant designed to cope with 2 simultaneous Main Steam Line Breaks that cannot be isolated (and possibly loss of 2 lines of the Main Feedwater system)?