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 Westinghouse AP1000

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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 AP-1000

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 AP-1000, 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, the Pressuriser, the Core Makeup Tanks and the Accumulators. By comparison, gross failure of certain piping is explicitly discounted (a claim) based on one of two sets of arguments and evidence referred to as 'break exclusion zone' (piping in the vicinity of the containment wall) or 'leak before break evaluation procedure'.

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 (Reactor Pressure Vessel, Steam Generators, Pressuriser);

materials used for the reactor coolant pump bowl and the weld joining the bowl to the steam generator channel head;

nature of the arguments and evidence to support integrity claims for some piping.

Overall, we conclude Westinghouse has 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. Westinghouse has 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.

Westinghouse - AP1000

NOTES OF PRELIMINARY REVIEW OF THE UK APPLICATION - STRUCTURAL INTEGRITY

19 JANUARY 2008

These notes are the outcome of my preliminary review of the structural integrity aspects of the UK application documentation provided by Westinghouse for the AP1000.

Structural integrity here means the integrity of pressure boundary components and their supports, and reactor internals. In this review, I have concentrated on metal pressure boundary components; i.e. I have not considered their supports or the reactor internals. The scope of this review excludes civil structures which may perform a pressure boundary function (e.g. the containment building). To be clear, the containment system includes a large metal shell; in this preliminary review, I have not considered the integrity of this shell.

I have looked in particular at parts of the following documents:

Executive Summary. UKP-GW-GL-720.

UK Compliance Document for AP1000 Design. UKP-GW-GL-710:
Section A.

UK AP1000 Safety, Security, and Environmental Report. UKP-GW-GL-700 Rev 1:

- Chapter 1, Subchapters 1.1, 1.2
- Chapter 3, Subchapters 3.2, 3.6, 3.9, Appendices 3B, 3C, 3E
- Chapter 5
- Chapter 6, Subchapters 6.3, 6.6
- Chapter 10, Subchapter 10.3.

I made the selection above for my sample review based on an overview of all the documentation provided. Examples of parts of documents I explicitly excluded from this preliminary review are:

UKP-GW-GL-710 Section F
UKP-GW-GL-700 Subchapter 1.9, Subchapter 3.1.

The notes below are in the form of 15 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.

In the following, when I refer to Chapters, Sub-chapters, pages tables or figures the document is UKP-GW-GL-700 Revision 1, unless shown otherwise.

I note the following from the Executive Summary UKP-GW-GL-720 on page 10.

“Six aspects of the AP1000 design contribute to defense-in-depth:

• Stable operation – In normal operation, the most fundamental level of defense-in-depth ensures that the plant can be operated stably and reliably. **This is achieved by the selection of materials**,……..”

We agree selection of materials is a fundamental aspect of achieving safety.

1. Pressure Boundary Components - Achievement of Integrity as a Contribution to the Overall Safety Justification

The primary and secondary circuits of the reactor system have a crucial role in achieving safety, in two general ways:

1. the potential for a pressure boundary failure to be an initiating event;

2. the need for continued pressure boundary integrity overall given an initiating event occurs (including a pressure boundary component failure causing a hazard to other pressure boundary components).

The AP1000 documentation deals at length with the integrity claims for piping. However, there is no explicit coverage of basic integrity claim for vessels inside containment. The starting point of NII’s Safety Assessment Principles (2006 Edition) for structural integrity is identification of the target reliability claim for pressure boundary components. In particular, a basic distinction is made between:

> 1. components where the consequences of their gross failure are considered, and steps taken to deal with those consequences;
>
> 2. components where the consequences of gross failure are not considered, instead arguments and evidence are provided that for these components the likelihood of failure is so low it can be discounted. That is there is no designed protection for the consequences of gross failure but such gross failure could lead to a significant release of radioactivity.

In principle, it is acceptable to claim that gross failure of certain pressure boundary components is so low that such failure can be discounted. Such a claim has to be supported by arguments and evidence. However, the starting point must be the explicit acknowledgement in the safety case that such failures are being claimed as so low as to be discounted. Such safety claims and the supporting arguments and evidence are the subject of the NII Safety Assessment Principles (SAPs), 1996 Edition; see SAPs 2006 paragraphs 238 on and especially paragraphs 243 on.

There are a number of vessels within containment which are likely to fall under the second category above, namely:

> Reactor Pressure Vessel
>
> Pressuriser

Steam Generator primary side (channel head)

Steam Generator secondary side (secondary side shell)

Core Makeup Tanks (Passive Core Cooling System)

Accumulators (Passive Core Cooling System)

Reactor Coolant Pump Bowls

Reactor Coolant Pump - Stator shell and associated components.

QUESTION: For which pressure boundary components inside containment is the claim made that the likelihood of gross failure is so low that it can be discounted?

2. Requirements Additional to Code Requirements

In the UK, for metal pressure boundary components where the claim is that gross failure is so low it can be discounted, requirements over and above standard ASME III requirements have been applied. Important areas include:

Active efforts to minimise the number of welds;

A preference for forged material over plate or casting;

In the case of low alloy forging material (e.g. A508) more restrictive limits on chemical composition than the latitude allowed by the code. For A508 forging material this included more restrictive limits on carbon content and impurity elements;

Volumetric inspection of welds during manufacture using qualified ultrasonic inspection methods;

Materials supply specifications which include a minimum fracture toughness requirement in terms of the K or J parameter, not just Drop Weight and Charpy impact energy;

Fracture mechanics analyses for postulated defects at several key locations in the component for the full range of design loading conditions. The overall purpose of these analyses being to show that with the minimum toughness specified for supply of material, the limiting defect sizes would be larger by a margin than the size of defect which could be detected and sized with high confidence by the qualified inspection method.

3. Reactor Pressure Vessel

I note the overall design of the RPV is familiar to NII, the main differences are no bottom head penetrations, correspondingly more penetrations in the top head and a slight vertical offset in the inlet and outlet nozzles.

I note the RPV is substantially manufactured from forgings.

I note for the RPV body, there is no weld in the cylindrical section adjacent to the core i.e. a single cylindrical forging is used which spans the height of the core.

From Table 5.2-1 of UKP-GW-GL-700, the forging material specified is SA-508 Grade 3 Class 1 (former designation in ASME II, SA-508 Class 3). From studies done in the UK some years ago (1970s-1980s), this was the recommended material for the RPV. However, those studies recommended additional restrictions on chemical composition. For instance, compared with ASME II, reduced maximum carbon content and extra restrictions on impurity elements. There were also specific restrictions on chemical elements which play a role in irradiation embrittlement, for material closest to the core.

Page 5.3-4 states:

> "In addition to the ASME Code, Section III nondestructive examination, full penetration ferritic pressure boundary welds in the reactor vessel are ultrasonically examined during fabrication. This test is performed upon completion of the welding and intermediate heat treatment but prior to the final postweld heat treatment.
>
> After hydrotesting, full penetration ferritic pressure boundary welds in the reactor vessel, as well as the nozzle to safe end welds, are ultrasonically examined. These inspections are performed in addition to the ASME Code, Section III nondestructive examination requirements."

Table 5.3-4 indicates 1/2T compact tension specimens are included in the surveillance programme capsules. There are apparently six 1/2T CT specimens in each capsule, two for each of 3 material forms. We have found CT specimens a useful part of surveillance programmes; six specimens in each capsule seems a low number.

4. Steam Generators

There are 2 Steam Generators and the plant has an electrical output of about 1100MWe. The Steam Generators are therefore larger than for an equivalent electrical output plant with 4 Steam Generators. However this is not a fundamental issue and there are precedents for similar size Steam Generators.

UKP-GW-GL-700 Table 5.2-1 shows the material of the channel head to be SA-508 Grade 3 Class 2 (former designation in ASME II, SA-508 Class 3A). This has higher strength than SA-508 Grade 3 Class 1 (as used for the RPV). This is achieved during manufacture by SA-508 Grade 3 Class 2 being tempered at a lower temperature than SA-508 Grade 3 Class 1: the nominal chemical compositions being the same.

UK practice for PWR Steam Generators has been to use SA-508 Grade 3 Class 1 for the channel head.

UKP-GW-GL-700 Table 5.2-1 shows the material of the secondary shell also to be SA-508 Grade 3 Class 2. Again, UK practice for PWR Steam Generators has been to use SA-508 Grade 3 Class 1 for secondary shells.

UKP-GW-GL-700 Table 5.2-1 also lists for the Steam Generators plate material, SA-533 Type B Class 1. In general, forgings are preferred as these reduce the number of welds, especially axial welds.

Table 3.2-3 of UKP-GW-GL-700 shows for the Steam Generators the primary side as Class A, Principal Construction Code ASME III Class 1, and for the Steam Generator shells Class B, Principal Construction Code ASME III Class 1. Text on page 5.4-10 states:

> "The ASME Code classification for the secondary side is specified as Class 2. The pressure-retaining parts of the steam generator, including the primary and secondary pressure boundaries, are designed to satisfy the criteria specified in Section III of the ASME Code for Class 1 components."

This suggests both the primary side and secondary side of the Steam Generators are designed to ASME III Class 1, even though the AP1000 Classes are different. This would be welcome.

QUESTION: Are the Steam Generator secondary shells designed, manufactured, inspected and tested in full compliance with ASME III Class 1 (i.e. Subsection NB of the Code)? If so, does this mean the scope and extent of pre- and in-service inspections using ASME XI would be as for other ASME III Class 1 vessels (e.g. the RPV)?

I note the specification of Thermally Treated Alloy 690 for the Steam Generator tubes. This is the material used in the UK for PWR Steam Generators.

Each of the two Reactor Coolant Pump bowls is welded directly to the associated outlet nozzle of Steam Generator channel head. I believe the pump bowl is made from austenitic stainless steel (see later) and the SG channel head is ferritic steel. This means a dissimilar metal joint. UKP-GW-GL-700 page 5.4-11 includes the following text:

> "A high-integrity, nickel-chromium-iron (Alloy 690) weld is made to the nickel-chromium-iron alloy buttered ends of these nozzles."

QUESTION: For the weld joining the Reactor Coolant Pump bowl to the Steam Generator channel head, do the geometry, accessibility and materials of construction, allow volumetric examination (including ultrasonic examination) from both sides of the weld; during manufacture and during service?

QUESTION: Is gross failure of a channel head divider plate part of the design basis?

5. Pressuriser

Comments in terms of materials selection are similar to those for the Steam Generators.

UKP-GW-GL-700 Table 5.2-1 shows the material of the Pressuriser shell to be SA-508 Grade 3 Class 2. Again, UK practice for this component has been to use SA-508 Grade 3 Class 1.

UKP-GW-GL-700 Table 5.2-1 also lists for the Pressuriser plate material, SA-533 Type B Class 1. In general, forgings are preferred as these reduce the number of welds, especially axial welds.

I note the inside surface will be clad using austenitic stainless steel.

I note the statement on page 5.4-28 of UKP-GW-GL-700 that:

> “Nickel-chromium-iron alloys are not used for heater wells or instrument nozzles.”

QUESTION: What material is used for the Pressuriser heater wells and instrument nozzles?

6. Reactor Coolant Pumps

UKP-GW-GL-700 Table 5.2-1 lists for the Reactor Coolant Pump materials for both pressure forgings and pressure castings. Both are austenitic stainless steels. It is not clear which materials are used for the parts of the Reactor Coolant Pump pressure boundary.

For the Reactor Coolant Pump bowl, there are merits in a forging rather than a casting.

QUESTION: What are the materials of construction of the Reactor Coolant Pump pressure boundary, including the pump bowl, the stator shell, closures and bolting systems?

I note the claims in UKP-GW-GL-700 subsection 5.4.1 (page 5.4-1) that:

> “The reactor coolant pump pressure boundary shields the balance of the reactor coolant pressure boundary from theoretical worst-case flywheel failures. The reactor coolant pump pressure boundary is analyzed to demonstrate that a fractured flywheel cannot breach the reactor coolant system boundary (impacted pressure boundary components are stator

> closure, stator main flange, and lower stator flange) and impair the operation of safety-related systems or components."

I also note the statement on page 5.4-18 that:

> "Pipe rupture overspeed is based on a break of the largest branch line pipe connected to the reactor coolant system piping that is not qualified for leak-before-break criteria."

7. Core Makeup Tanks and Accumulators

UKP-GW-GL-700 contains outline information for these vessels. There are 2 of each type of vessel and each has a volume (Table 6.3-2) which exceeds (Core Makeup Tanks 2500cuft) or is slightly less than (Accumulators, 2000cuft) the Pressuriser (2100cuft, Table 5.4-9).

The Core Makeup Tanks operate at the pressure of the primary circuit, being connected by an open line to a hot leg; they have a design pressure of 2485psig, the same as the Pressuriser. The Core Makeup Tanks are vertical, cylindrical vessels, the Accumulators are spherical.

The Accumulators are isolated from the primary circuit during normal operation, and would only become connected once primary circuit pressure was low. The Accumulators are pressurised with Nitrogen gas and their design pressure is 800 psig (Table 6.3-2).

During normal operation (by far the most likely condition) the Core Makeup Tanks and Accumulators contain water whose temperature is essentially that of containment, say about 40^{o}C.

The lower temperature of operation by itself does not provide a decisive argument for a claim that gross failure of these vessels is so unlikely that it can be discounted. Perhaps gross failure of one of these vessels can be tolerated within the safety case. That argument would need to be made. Otherwise the points outlined in bullet points 1 and 2 above would need to be considered.

I note from UKP-GW-GL-700 Table 6.3-2 that the material of construction of these 4 vessels is "carbon steel, stainless steel clad".

I note from UKP-GW-GL-700 Table 3.2-3 (sheet 14) that for the Core Makeup Tanks the principal construction code is ASME III Class 1 while for the Accumulator Tanks it is ASME III Class 3.

I note the following with regard to check valves, although this is not part of pressure boundary matters:

> "One change in the definition of active failures has been incorporated into the passive core cooling system design. The system has been specifically designed to treat check valve failures to reposition as active failures. More specifically, it is assumed that normally closed check valves may fail to open

and normally open check valves may fail to close. Check valves that remain in the same position before and after an event are not considered active failures.

There are two exceptions to this treatment of check valve failures in the passive core cooling system. One exception is made for the accumulator check valves, which is consistent with the treatment of these specific check valves in currently licensed plant designs. The other exception is made for the core makeup tank check valves failure to re-open after they have closed during an accident. The valves are normally open, biased-opened check valves. This exception is based on the low probability of these check valves not re-opening within a few minutes after they have cycled closed during accumulator operation."

8. Piping - General Comments

Two sorts of claims are made for piping integrity:

A. gross failures (guillotine breaks or large axial splits) are postulated and the consequences of such failure are included in the design basis. Consequences include the dynamic effects of such a break, e.g. pipe whip, and the thermal-hydraulic effects of a sudden loss of fluid from the system;

B. gross failures (guillotine breaks or large axial splits) are discounted and certain consequences are not included in the design. This claim is made for sections of the primary loop pipework (6-inch nominal internal diameter or above, page 3.6-1) and the Main Steam Lines from the Steam Generator outlets to just downstream of the Main Steam Isolation Valves. The consequences discounted (i.e. not included in the design basis) are so-called 'local dynamic' effects such as pipe whip.

When gross failure is not postulated (for the defined sub-set of all consequences), two sorts of structural integrity arguments are used, one is relatively recent, the other of longer standing.

The more recent basis for not postulating gross failure is referred to as a 'Leak Before Break Evaluation Procedure'. This uses the approach set out in USNRC Standard Review Plan Section 3.6.3. SRP 3.6.3 was issued as Revision 0 March 1987 and as Revision 1 March 2007 (USNRC web site). SRP 3.6.3 was draft status for some years and UKP-GW-GL-700 on page 3.6-31 and on page 3B-1 refers to "Draft Standard Review Plan 3.6.3" and the version which was published in the Federal Register on 28 August 1987 for public comment.

The older basis for not postulating gross failure in piping was established for piping in the vicinity of the containment penetration. As stated on page 3.6-12 of UKP-GW-GL-700:

"Breaks are not postulated in piping in the vicinity of containment penetrations. The portion of the piping that does not have postulated breaks is the break exclusion area."

The basis for not postulating breaks in the vicinity of containment penetrations is set out in section 3.6.2.1.1.4 of UKP-GW-GL-700. The measures are as set out in USNRC Branch Technical Position BTP 3-4 (which used to be an Appendix to SRP 3.6.2 and known as BTP MEB 3-1, but as of Revision 2 is a free-standing document, BTP 3-4). This basis is different from the Leak-Before-Break Evaluation Procedure; indeed leakage is not mentioned as part of the justification for excluding breaks. One feature of the arrangements of the break exclusion zone piping is ‘augmented’ examination (100% volumetric examinations in-service). Such examinations are required as part of BTP 3-4 and are dealt with in UKP-GW-GL-700 Chapter 6, subsection 6.6.8.

Figures 3E-1 to 3E-5 of Appendix 3E to UKP-GW-GL-700 show the piping that is:

candidate LBB piping

high energy break exclusion zone piping

other high energy piping

Note that according to UKP-GW-GL-700, page 3.6-31, piping designed to any of the 3 Classes of ASME III could have the Leak-Before-Break evaluation procedure applied. This means piping designed to any ASME III Class could be subject of a claim that sudden catastrophic failure of the pipe is not credible. I note that USNRC Standard Review Plan 3.6.3 only includes ASME III Class 1 and 2 piping, though other piping would be considered, but compared with ASME III Class 1 or 2 requirements.

In addition, for vessels such as the Reactor Pressure Vessel gross failure is not within the design basis (nor even mentioned as a beyond design basis event). But for such vessels, apparently ASME III code compliance is sufficient.

This set of bases may have developed as expedient responses to practical issues, but they do not have the appearance of coherence.

As stated in bullet point 1 above, in principle, it is acceptable to claim that gross failure of certain pressure boundary components is so low that such failure can be discounted. Such safety claims and the supporting arguments and evidence are the subject of the NII Safety Assessment Principles (SAPs), 1996 Edition; see SAPs 2006 paragraphs 238 on and especially paragraphs 243 on.

In the UK to date, for PWR Main Steam Lines outside containment, (between the containment penetration and the restraint downstream of the Main Steam Isolation Valves) a safety case has been accepted on the basis of a number of measures to achieve and demonstrate a level of structural integrity such that gross failure was discounted.

Whatever the detailed arguments and evidence for this level of structural integrity, the basic point from a structural integrity point of view is either gross failure can be discounted or not. The structural integrity test of ‘no break’ cannot be modulated by subsequently only some consequences being excluded.

UKP-GW-GL-700 Appendix 3B, Table 3B-1 lists the piping subject to leak-before-break analysis.

9. Piping - Leak Before Break Evaluation Procedure

The Leak Before Break evaluation procedure has two aspects:

1. satisfaction of a number of qualitative requirements, mostly to demonstrate certain in-service degradation mechanisms are not applicable to the particular segment of pipe in question;

2. a quantitative analysis based on through-wall cracks.

The first aspect is useful in terms of meeting the expectations of the NII SAPs for a pressure boundary component where the claim is the likelihood of gross failure is so low that it can be discounted. This is especially so if it influences choice of materials, methods of fabrication and design. However the focus appears to be on avoiding well-known, ‘common-cause’ (i.e. found in more than one plant) degradation issues which have arisen to date and been identified from operational experience. High reliability is not plausible if frequent, common-cause degradation is feasible. But eliminating such issues alone does not establish a level of reliability so high that failure can be discounted. In addition, some thought needs to be given to common-cause degradation which is at a lower rate of incidence than operational experience to date is likely to have revealed. And for very high reliability claims, it is also necessary to deal with the potential for occurrence of essentially random events.

However the NII SAPs for a level of integrity where gross failure can be discounted are seeking what might be termed a ’No Break’ or ‘No Leak or Break’ safety case. As a minor point, the latter appears more consistent as the aspiration in the Technical Specifications, Limiting Condition of Operation for RCS leakage of “no pressure boundary leakage” (UKP-GW-GL-700, Chapter 16 LCO 3.4.7 and Bases B 3.4.7).

In terms of fracture mechanics analyses, the emphasis in the UK has been on evaluating part-penetrating, that is ahead of the point at which they become through-wall defects and liable to result in leakage. Such fracture mechanics analyses have linked the resilience of the component in terms of limiting crack-like defect size with the qualified capability of examination methods to detect and size crack-like defects. This applies to manufacturing, pre-service and in-service examinations.

With this approach, claims on leakage detection or ‘leak-before-break’ have only a supporting role in the defence-in-depth arguments.

Regarding leak-before-break analyses, the method set out in UKP-GW-GL-700 is based on USNRC SRP 3.6.3 which has two aspects for each specific piping location analysed:

> 1. establish the length of through-wall crack which gives a predicted leak rate of 10 times the leak detection capability;

2. determine limiting through-wall crack lengths for a fault condition, namely the Safe Shutdown Earthquake in combination with normal operating conditions. The requirement is this limiting defect length should be at least 2x the leak crack length calculated in step 1 above. Demonstrate the leak crack length calculated in step 1 above is stable when subjected to a load of 1.4x(normal + SSE). An alternative is to use 1.0x(normal and SSE inertial and anchor loads added in absolute sum).

Regarding step 1 above, the leak rate through the postulated through-wall crack is determined by calculation using computational model of the physics of fluid flow though a crack. My understanding is the factor of 10 on leakage is at least in part to account for 'real-world' uncertainties which cannot be modelled in detail.

Regarding step 2 above, the fracture mechanics analysis is based on either an amount of torn toughness or a limit load analysis. The latter is only specified for austenitic stainless steel piping. For toughness, SRP 3.6.3 states in 11B(i):

> "The specimens used to generate the J-R curves should be large enough to provide crack extensions up to an amount consistent with J/T condition determined by the applicant's or licensee's analysis. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques may be used…."

The NII SAPs states in Paragraph 278:

> For fracture analyses of infrequent fault or hazard loading conditions, results using initiation fracture toughness may be supplemented with results using fracture toughness based on limited amounts of stable tearing. In this case, there must be valid materials fracture toughness data up to at least the limited extent of tearing used.

Thus, although the NII SAPs are consistent with using stable tearing for infrequent fault conditions (such as safe shutdown earthquake), they require valid materials fracture toughness data up to at least the amount of tearing used. My interpretation of this passage in the NII SAPs is that extrapolation for torn toughness is excluded.

The limit load analysis permitted for austenitic stainless steels in SRP 3.6.3 implies an indeterminate amount of tearing, and without some restriction appears to fall outwith the NII SAPs.

Perhaps the main point from an NII SAPs perspective is the Leak Before Break evaluation procedure contains no margin analysis for the through-wall crack determined in step 1 above under normal operational loads (Level A and B Service Limit loads). For normal loading conditions, the NII SAPs call for initiation toughness to be used and a margin demonstrated. The combination of infrequent fault load (SSE) with a relaxed fracture toughness condition (torn toughness) does not demonstrate an adequate margin under the combination of normal loading and initiation toughness. The following paragraph explains why normal operation loads are relevant.

In practice, if a through wall crack did develop and leak, in the first instance it would likely prompt an ‘engineering evaluation’ to come to a judgement whether Technical Specification Limiting Conditions of Operation were likely to be breached. In particular whether the ‘no pressure boundary leakage’ LCO was likely to be breached. Entry to the containment is not straightforward from an at-power state and the ‘engineering evaluation’ will have a limited range of input information. In practice, leakage is generally considered more likely to occur through seals, gaskets rather than the pressure boundary. Therefore, until more decisive information became available it would be natural for an ‘engineering evaluation’ to conclude any new leak or increment in leakage was from a seal or gasket and not from a through-wall crack in the pressure boundary. Hence, depending on the leak rate, a leak could persist for some period of normal operation.

The above is on the assumption that the overall logic of the Leak Before Break evaluation procedure is for the hypothetical through-wall crack to be detected by leakage during normal operation, before any ‘one-off’, more severe loading condition is likely to occur. This is a literal interpretation of Leak-Before-Break. A less literal interpretation of the procedure is the method simply demonstrates a general robustness from the point of view of fracture toughness; the determination of the length of the leaking through-wall crack being just a method to determine a representative crack dimension.

As a minor point, it is not clear whether both sides of safe ends are included in pipe segments.

10. Reactor Coolant Loop Primary Pipework and Surge Line

UKP-GW-GL-700 page 5.4-13 states the reactor coolant system piping is fabricated from austenitic stainless steel. Forged seamless piping is used and the piping system does not contain any cast fittings. The number of welds is minimised. NII, in the past has sought the use of forged stainless steel pipe for PWR primary circuit use.

The amount of primary loop pipework is reduced compared to a four-loop plant and the design of the reactor coolant pumps further reduces the amount of piping.

Page 5.4-24 states that the Pressuriser Surge Line has been specifically designed and instrumented to minimise the potential for thermal stratification.

The sloping of the Surge Line is a notable change from previous practice.
UKP-GW-GL-700 page 5.44-21 states the reactor coolant system piping is designed to ASME III Class 1. From the definition of the reactor coolant system given on page 5.1-3, it is inferred that the surge line is also designed to ASME III Class 1. This information does not appear in Table 3.2-3.

Page 5.4-26 states that radiographic examination is performed on circumferential butt welds and on branch connection nozzle welds exceeding 4-inch nominal pipe size.

For any primary circuit piping where catastrophic failure is discounted, NII would expect a range of examination methods to be used, including ultrasonic methods. Our understanding is forged stainless steel pipe is amenable to ultrasonic examination given a suitable grain structure is specified and achieved.

11. Main Steam Lines

Inside containment, breaks in the Main Steam Lines are precluded from the design based on applying the Leak Before Break evaluation procedure. From where the Main Steam Lines emerge from the containment (containment penetration) up to the restraint point beyond the Main Steam Isolation Valves, breaks are excluded based on BTP 3-4 (Figure 3E-1 of Appendix 3E to UKP-GW-GL-700).

Page 10.3-1 states the Main Steam Line from the Steam Generator up to and including the Main Steam Line Isolation Valves is Class B, while the Main Steam Lines from the MSIVs to the pipe restraint located at the auxiliary / turbine building wall is Class C.

The above implies the Class B part of the Main Steam Line is designed to ASME III Class 2 and the Class C part to ASME III Class 3. This is uncertain because I can find no definite statement in UKP-GW-GL-700 Chapter 10 subchapter 10.3 or Table 3.2-3.

QUESTION: What ASME III Code Classes are used for design of the Main Steam Lines from the Steam Generator to the pipe restraint at the auxiliary / turbine building wall?

The Main Steam Line piping from the Steam Generator to the auxiliary to turbine building wall is made from SA-333 Grade 6 seamless pipe (Table 10.3.2-3). The corresponding Main Feedwater Line piping is made from SA-335 Grade P.11 (Table 10.3.2-3). Significant lengths of Main Steam Line and Feedwater Line are covered by a safety case which basically discounts gross failure. Therefore the material properties of SA-333 Gr 6 and SA-335 Gr P.11 will be of special interest, particularly fracture toughness properties in terms of K or J type parameters. The Main Steam Line Isolation Valves are part of the continuous pressure boundary, we would expect the integrity of the MSIV valve body to be justified to a level of structural integrity consistent with that of the Main Steam Line pipework attached to the MSIVs.

The area of the Main Steam Lines where the Main Steam Safety Valves branch off is a complex geometry.

QUESTION: What materials / fabrication route is used to manufacture the Main Steam Line where the Main Steam Safety Valve branch connections are made?

From the auxiliary / turbine building wall to the containment penetration the Main Feedwater Lines are defined as break exclusion zones.

12. Overpressure Protection

UKP-GW-GL-700 in section 5.2.2 (page 5.2-5) states:

> "Reactor coolant system and steam system overpressure protection during power operation are provided by the pressurizer safety valves and the steam generator safety valves, in conjunction with the action of the reactor protection system."

For the primary circuit it is understood that overpressure protection is provided by a combination of the pressuriser safety valves and the reactor protection system - specifically reactor trip.

However for the secondary circuit, and in particular the steam system, UKP-GW-GL-700 in section 10.3.2.2.2 (page 10.3-6) states:

> Main steam safety valves with sufficient rated capacity are provided to prevent the steam pressure from exceeding 110 percent of the main steam system design pressure:
>
> - Following a turbine trip without a reactor trip and with main feedwater flow maintained
>
> - Following a turbine trip with a delayed reactor trip and with the loss of main feedwater flow

Table 10.3.2-1 shows the steam flow per Steam Generator as 7,490,000 lb/hr and Table 10.3.2-2 shows the relieving capacity per steam line to be 8,240,000 lb/hr.

The steam flow rates in the tables and the first bullet point from page 10.3-6 suggest the Main Steam Safety Valves can discharge a flow equivalent to normal steam flow at full power.

QUESTION: Does overpressure protection of the Main Steam System depend on the reactor protection system, or is the steam relief flow capacity sufficient to avoid exceeding 110% of design pressure without the aid of a reactor trip?

13. ASME Code Edition

UKP-GW-GL-700 on page 5.2-1 states:

> The baseline used for the evaluations done to support this safety analysis report and the Design Certification is the 1998 Edition, 2000 Addenda, except as follows:
>
> The 1989 Edition, 1989 Addenda is used for Articles NB-3200, NB-3600, NC-3600, and ND-3600 in lieu of later editions and addenda.

By the time a plant could be built in the UK the ASME Code 1998 edition with 2000 addenda will be quite old.

The above exception, the use of the ASME code 1989 edition for certain articles is consistent with 10CFR50.55a (b) (1) (ii) which states:

> (b) (1) (iii) Seismic design. Applicants or licensees may use Articles NB–3200, NB–3600, NC–3600, and ND–3600 up to and including the 1993 Addenda, subject to the limitation specified in paragraph (b)(1)(ii) of this section. Applicants or licensees may not use these articles in the 1994 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section.

The relevant parts of the referenced Articles in the ASME Code in the 1994 Addenda and later which led to their exclusion are concerned with the limits for reversing or non-reversing dynamic loads. Seismic loading is an example of a reversing dynamic load.

14. Load Combinations and Stress Limits

UKP-GW-GL-700 Subchapter 3.9, Table 3.9-5 lists the minimum design loading combinations for ASME Class 1, 2, 3 and CS systems and components. Table 3.9-8 lists the minimum design loading combinations for supports for ASME Class 1, 2, 3 piping and components.

It is noted that in Tables 3.9-5 and 3.9-8, the last mentioned load combination in the Level D Service Limit category includes design basis pipe break and safe shutdown earthquake.

Table 3.9-6 lists additional load combinations and stress limits for ASME Class 1 piping. For the Level D Service Limit seismic anchor motion loads are included with stress limits for axial and bending stress of respectively $1.0S_m$ and $6.0S_m$.

Table 3.9-7 lists additional load combinations and stress limits for ASME Class 2, 3 piping. For the Level D Service Limit seismic anchor motion loads are included with stress limits for axial and bending stress of respectively $1.0S_h$ and $3.0S_h$.

It is not clear whether the Level D Service Limits for seismic loads in Tables 3.9-6 and 3.9-7 are consistent with use of ASME Code articles prior to the 1994 Addenda (see bullet point 13 above).

QUESTION: Are the stress limits in UKP-GW-GL-700 Tables 3.9-6 and 3.9-7 for seismic anchor motion consistent with not using Articles NB–3200, NB–3600, NC–3600, and ND–3600 of the ASME III code later than the 1993 Addenda?

UKP-W-GL-700 Subchapter 5.2 on page 5.2-2 outlines the seismic load analysis which will be performed for the nonsafety-related CVS piping inside containment. For Level D Service Limit load combination the stress limit for ‘Equation 9’ of ASME III NB3650) is stated as:

> equal to the smaller of $4.5\ S_h$ and $3.0\ S_y$

QUESTION: Is there other nonsafety-related piping treated in the same manner as the CVS piping inside containment?

15. In-Service Inspection

The following from UKP-GW-GL-700 are noted.

Subchapter 5.2, page 5.2-31 (for ASME Class 1 components):

> “The … applicant will provide a plant-specific preservice inspection and inservice inspection program. The program will address reference to the edition and addenda of the ASME Code Section XI used for selecting components subject to examination, a description of the components exempt from examination by the applicable code, and drawings or other descriptive information used for the examination.”

Subchapter 6.6, page 6.6-3 (for ASME Class 2 and 3 components):

> “… applicants referencing the AP1000 certified design will prepare a pre-service inspection program (nondestructive examination) and an inservice inspection program for ASME Code, Section III Class 2 and 3 systems, components, and supports. The pre-service inspection program will address the equipment and techniques used.”

NII would likely take an interest in the scope and extent of the pre- and in-service inspection programmes and the qualification of examination processes, procedures and personnel. To date in the UK the basis of such qualification has been based on the European Network on Inspection Qualification (ENIQ) recommendations.