

# HEALTH AND SAFETY EXECUTIVE HM NUCLEAR INSTALLATIONS INSPECTORATE

## **New Reactor Generic Design Assessment (GDA) - Step 2**

### **Preliminary Review Assessment of: Structural Integrity Aspects of AECL ACR-1000**

## 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 ACR-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 ACR-1000, implicit in the submission is that gross failure is discounted for any of the four Steam Generators and the Pressuriser. The ACR-1000 does not have a Reactor Pressure Vessel. Instead the nuclear fuel is contained in a number of horizontal pressure tubes. Failure of a pressure tube is a design basis event.

For some areas of piping, the safety claim is not clear.

The NII SAPs encompass the claim that gross failure of a component is so unlikely it can be discounted (SAPS paragraphs 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.

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 (Steam Generators, Pressuriser);

basis of the safety case for some pipework.

Overall, we conclude AECL 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. AECL has provided some coverage of the type of arguments and evidence to support the claims. However, in this area and

at this step, more detail would have been useful. 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.

## **AECL ACR-1000**

### **NOTES OF PRELIMINARY REVIEW OF THE SAFETY, SECURITY AND ENVIRONMENTAL REPORT**

#### **STRUCTURAL INTEGRITY**

These notes are the outcome of my preliminary review of the structural integrity aspects of the "Safety, Security and Environmental Report" documentation provided by AECL for the ACR-1000 (Tracking Sheet 2, 21 September 2007).

Structural integrity here means the integrity of metal components and their supports. The scope of this review excludes civil structures which may perform a pressure boundary function (e.g. the containment building). In this review I have concentrated on metal pressure boundary components; i.e. I have not considered supports.

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

ACR 1000 Technical Description. Parts of Sections 1, 2, 3, 4, 5, 6 and 9. Document 10820-01377-TED-001;

ACR 1000 Events. Section 3.2. Document 10820-03600-ASD-001-H;

Safety Design Guide on Safety Classification of Structures, Systems and Components. Section 3, Table 1 to 4. Document 108-03650-SDG-001-H;

Design Assist Assessment of In-Core Damage in a Spontaneous Pressure Tube Rupture Event. Document 018UK-03560-220-002;

Systematic Review of Plant Design for Identification of Initiating Events. Section 4 up to 4.6 and Parts explaining Master Logic Diagram for Steam Pressure Reduction Events. Document 108UK-03660-ASD-001;

ACR-1000 Safety Basis. Document 108-03600-AB-003-H;

ACR-1000 Submission for Step 2 of UK Generic Design Requirement, Part 1 - HSE NII Requirements. Section 2.17. Document 10820-01321-ASD-004-H.

The notes below are in the form of 18 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. Pressure Boundary Components - Achievement of Integrity as a Contribution to the Overall Safety Justification**

The primary and secondary circuits of the heat transport 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).

From Table 2 of 10820-03600-ASD-001-H, it is clear that some pressure boundary failure events are included as design basis events, mainly 'pipe' failures. This implicitly suggests some pressure boundary failures are not included within the design basis (or even in beyond design basis events). For example, there is no mention of Steam Generator channel head (primary side) secondary side shell or Pressuriser shell gross failure as an initiating event.

According to 108UK-03660-ASD-001, page 2-1:

"No events identified are screened out of this document."

However, the Master Logic Diagram through Figures 1, 2, 7 and 10 ending at "SG Depressurisation Due to Breaks" does not include SG secondary shell gross failure.

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.

The structural integrity safety case for pressure boundary components (i.e. the arguments and evidence to support whatever safety claims are made) depend on the design of the components, materials selection, manufacturing methods, manufacturing inspections and subsequent in-service use, maintenance, testing and inspection. These aspects receive only cursory and variable coverage within the suite of documents I have reviewed. For these areas, some of the subsequent points below request more specific information.

## **2. Long Lead Items - Safety Justification**

Section 2.17 of 10820-01321-ASD-004-H deals with long lead-time items and section 2.17.4 lists the ACR-1000 Long Lead Items. Included in the list are the Steam Generators and HTS pumps. Both items have significant pressure boundary component aspects. Section 2.17.3 states:

“...we understand that the Regulators will perform an assessment of the Quality Assurance arrangements for the early manufacture of long lead-time items important to safety together with the safety case and design change control arrangements in place if changes to these items are subsequently required.”

Before we get to the matter of “Quality Assurance” arrangements, it will have been necessary for NII to satisfy itself with the overall design, materials selection and manufacturing routes chosen. Greater detail will be needed for those pressure boundary components where it is claimed the likelihood of gross failure is so low that it can be discounted.

### **3. Request for List of Metal Pressure Boundary Components Where it is Claimed the Likelihood of Gross Failure is so Low That Such Failure Can be Discounted**

**QUESTION:** For the ACR-1000 design, for which pressure boundary components is gross failure of the pressure boundary claimed to be so low that such failure can be discounted?

The list appears likely to include the channel heads of the Steam Generators (the primary side of the Steam Generator), the secondary shells of the Steam Generators, the Pressuriser shell. This list might include the bowls of the primary coolant pumps.

### **4. Status of Headers, Piping Joining Headers to Main Pumps and Piping Joining Pumps to Steam Generators**

Table 2 of 10820-03699-ASD-001-H, at 6.6 defines a Large-break LOCA as a rupture with a total cross-sectional area equal to or greater than that of the largest feeder. The use of the phrase “equal to *or greater than*” leaves the area unbounded. I have not found a list of components whose failure would cause a Large-break LOCA as defined.

**QUESTION:** Are full guillotine breaks within the design basis for the following components: the Headers, the pipework joining the Headers to the Main Pumps and the piping joining the Main Pumps to the Steam Generators?

**QUESTION:** If a full guillotine break in a Header is within the design basis, what are the consequences for the feeder pipes connected to the failed Header?

### **5. Design Codes for Metal Pressure Boundary Components**

For metal pressure boundary components, Table T-3 of 108-03650-SDG-001-H indicates that for Safety Class 1, the Code Class is CSA N285.0 PB Class 1, 2, 3 or 4. Table 2.1-4 of 10820-01371-TED-001 lists CAN/CSA Codes and Standards, and the first 8 are relevant to pressure boundary components. The footnote to Table 2.1-4 states that CSA standards may call up other International standards as applicable. Standards prepared by ASME and ANSI for pressure vessels and piping are applied in this manner.

NII would have to gain some level of understanding of the cited codes (e.g. CAN/CSA-N285 series). Particular areas of interest would be where the CAN/CSA codes modify, add to or delete requirements in cited codes.

Looking across the UK nuclear industry, there is a degree of familiarity with ASME III. We understand the ACR-1000 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 ACR-1000 for the UK take ASME III as the basic design code?

## **6. 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.

## **7. Loading Conditions Used in Design of Metal Pressure Boundary Components**

Section 3.2.1.1 of 108-03650-SDG-001-H describes the four Plant States:

Normal Operation (NO)

Anticipated Operational Occurrences (AOO)

Design Basis Accidents (DBA)

## Beyond Design Basis Accidents (BDBA).

Section 3.2 of 10820-03600-ASD-001-H gives the best-estimate occurrence frequency ranges (per reactor year) of initiating events to be assigned to the Plant States AOO, DBA1, DBA2 and LCDA (BDBA).

Tables 1 to 3 of 10820-03600-ASD-001-H list events falling under the Plant States NO, AOO and DBA and BDBA.

Section 3.3.2 of 108-03600-AB-003-H states the Acceptance Criteria for AOO, DBA1 and DBA2. These Acceptance Criteria include limits applicable to metal pressure boundary components, i.e.:

**AOO.** ASME Level B Service Limit not exceeded if 1st shutdown system acts as intended. Level C Service Limit not exceeded if 1st shutdown system fails to act but 2nd shutdown system does act.

**DBA1.** ASME Level C Service Limit not exceeded if 1st shutdown system acts as intended. Level D Service Limit not exceeded if 1st shutdown system fails to act but 2nd shutdown system does act.

**DBA2.** ASME Level C Service Limit not exceeded if 1st shutdown system acts as intended, and in exceptional cases case may be made for acceptance of Level D Service Limit. Level D Service Limit not exceeded if 1st shutdown system fails to act but 2nd shutdown system does act.

The above ASME Service Limit based criteria are stated to be in accordance with Canadian Nuclear Safety Commission Regulatory Policy Statement R-77.

I have been unable to find a list of design loading conditions for metal pressure boundary components. I would expect such design loading conditions to be classified under the headings of NO, AOO, DBA1, DBA2 and for the design loading conditions to be based on the initiating events included under each Plant State. An important part of such a load definition would be load combinations to be included in the design.

**QUESTION:** For the metal pressure boundary components of the ACR-1000, is there a specific definition for each component of the design loading conditions to be considered in their design?

In ASME III terms, I would expect such information to appear in the Design Specification for each component. At this stage, I am interested in this information for the main vessels and piping of the primary and secondary circuits.

## 8. Materials of Construction

Statements on materials of construction are limited in the documentation I have reviewed. To take some examples:

**Steam Generator.** In 10820-01371-TED-001, page 5-10 (5.1.4.2.2) states the tubesheet could be carbon or low-alloy material and the SG head is carbon steel. Apparently, the tubesheet is clad on the primary side, but there is no statement about whether the channel head overall is clad. No statement is made on the material of the channel head divider plate, nor how it is fixed into the channel head. For the SG secondary shell no material form is mentioned.

The U-tubes are mentioned in the design description, but there is no mention of the material of the U-tubes. 10820-03600-ASD-001-H in Table 2 at 6.4 (page T-7) and Appendix A at ID GE-65 (page A-30) mentions in passing that the SG tubes are made of corrosion-resistant and ductile material (e.g., Alloy 690, 800). This still leaves open whether the material is thermally treated and whether after bending the tubes, they are given an annealing heat treatment.

**Pressuriser.** No statement on the material of construction of the main shell. No statement on the material to be used for nozzles in the main shell. No statement about whether the inner surface of the shell is clad.

**Heat Transport Pumps.** Casing is stated to be cast steel to SA 216 Gr WCC. SA 216 is a carbon steel, not stainless steel. There is no statement about whether the inner surface is clad. No statement about how the required rotational inertia for coast down is achieved.

**Headers and Feeder Pipes.** According to 10820-01371-TED-001, 5.1.2.5 and 5.1.2.6 (page 5-4) the Headers and Feeder Pipes are made of SA-312 TP316N stainless steel.

**Piping Connecting Headers to HT Pumps and HT Pumps to Steam Generators.** I could find no statement on material of construction in the documentation reviewed.

**Main Steam Lines.** I could find no statement on material of construction in the documentation reviewed.

**QUESTION:** At this stage for ACR-1000, are the materials of construction defined? Or are they still to be determined by the AECL design process, or left as a matter of choice for the customer?

## **9. Main Heat Transport Pump Bowl**

10820-01371-TED-001 section 5.1.4.3.2 states the HT Pump bowls will be manufactured from cast steel, SA 216 Gr WCC. This is a carbon steel casting grade.

**QUESTION:** Has cast carbon steel been used for all previous CANDU HT pump bowls? Is a minimum fracture toughness of the manufactured casting material specified?

**QUESTIONS:** What experience is there of manufacture of cast carbon steel pump bowls for CANDU reactors - numbers manufactured and incidence of need for repair welding after initial casting?

**QUESTIONS:** Can meaningful volumetric examination to search for crack-like defects be applied to the basic pump bowl casting and any repair welds?

**QUESTION:** For significant repair welds in pump bowl castings, are fracture mechanics analyses done and if so what material fracture toughness is used and how are residual stresses in the repairs dealt with?

## **10. Heat Transport System Leak Detection**

10820-01371-TED-001, section 5.2.6 covers HTS leak detection. It states:

“The leak detection methods are based on diverse and independent detection principles and are designed to alert the operators to small leaks with sensitivity and speed consistent with regulatory requirements for early detection of HTS leaks and for leak before break design measures for a number of pipe segments.”

**QUESTIONS:** To which pipe segments are ‘leak before break design measures’ applied? For such pipe segments, are pipe breaks not included in the design basis? Does the term ‘leak before break design measures’ refer to a defined procedure or is this a generic term?

Section 5.1.3.5 states:

“The whole of the heat transport system and its auxiliaries are located within the pre-stressed concrete containment structure, and the majority of the systems are within the dry vault. Any leakage within this vault increases the dew point of the recirculating air and is detected.”

From the drawings included in the documents supplied, I could not determine the boundary of this ‘dry vault’.

**QUESTIONS:** What is the extent of the ‘dry vault’? Does the ‘dry vault’ extend to the top of the steam generators? Is the ‘dry vault’ a separate volume within the overall containment, such that the recirculating air has limited and designed inlet and outlet routes?

**QUESTION:** For the purposes of leak detection / leak before break, how is leakage through the pressure boundary (base metal, welds) distinguished from leakage through seals, glands, joints and other non-pressure boundary locations? And how is leakage from the main primary circuit (or those pipe segments covered by ‘leak before break measures’) differentiated from leakage from other locations?

## **11. Multiple Steam Generator Tube Failure**

10820-03600-ASD-001-H in Table 2 (page T-7) includes item 6.4, Steam Generator tube rupture - a complete severance of a single steam generator tube (Plant State DBA2). Table 3 (page T-16) includes item 1.3, Multiple SG tube ruptures - a simultaneous guillotine rupture of 10 or more steam generator tubes (Plant State LCDA or BDBA event).

**QUESTION:** How does the safety case address the simultaneous complete severance (guillotine failure) of between 2 and 9 SG tubes?

## **12. Postulated Feeder Pipe Failure**

10820-03600-ASD-001-H in Table 2 (page T-8) at 6.6.1 and 6.6.2 define LOCAs with reference to the cross-sectional area of an individual Feeder Pipe. Plant states are either DBA1 or DBA2. It is assumed this reference to LOCA break area in terms of a Feeder Pipe means a guillotine break of a Feeder Pipe is included in the design basis.

**QUESTION:** Assuming a guillotine break of a Feeder Pipe, how are the potential consequences for impact with adjacent Feeder Pipes dealt with in the safety case (i.e. pipe on pipe impact)?

## **13. Pressure Tube Rupture / Ending Fitting Failure**

10820-03600-ASD-001-H Table 2 at 6.7.2 to 6.7.4 (page T-10) shows design basis events involving Pressure Tube Rupture and End-Fitting failure. An axial breach would lead to a LOCA but in mechanical terms might not imply significant axial movement. However, a circumferential failure in the Pressure Tube or End-Fitting could lead to significant disruption in the axial direction. One candidate location for a circumferential failure might be the rolled joint between Pressure Tube and End-Fitting. Significant axial motion could lead to rapid movement of the Feeder Pipes at each end of the failed channel and each Feeder Pipe could presumably impact adjacent Feeder Pipes.

**QUESTIONS:** Does the safety justification include consideration of significant axial displacement due to a circumferential (guillotine) failure in a Pressure Tube or End-Fitting? If so, is the potential hazard of one Feeder Pipe on another considered?

## **14. Response of Calandria Tube to a Pressure Tube Rupture**

108UK-03560-220-002 states on pages 5:

“For ACR-1000, a CT rupture in response to a spontaneous PTR may be classified as a beyond design basis accident (BDBA), depending on the design and qualification of the CT. Regardless of its final classification, the event is analysed for the time being as a design basis accident (DBA)...”

**QUESTIONS:** Why is the response of a Calandria Tube (CT) to a Pressure Tube Rupture (PTR) uncertain? Is there uncertainty about the design and qualification of the CT, or is it a customer option to specify the design and qualification of the CT?

## **15. Fuelling Machine Head and Removable Fittings of the Fuel Channel Pressure Boundary**

**QUESTION:** What are the materials of construction and method of construction of the pressure boundary of the Fuelling Machine Head?

I understand some changes are proposed in the design of the Fuelling Machine Head for ACR-1000 compared with earlier CANDU designs.

Table 1-1 of 10820-01371-TED-001 lists CANDU reactors built or under construction.

**QUESTIONS:** Is the design of the removable closure at the end of the Fuel Channel changed in ACR-1000 compared with earlier designs? Is the method of removing and fitting the closure at the end of the Fuel Channel changed in ACR-1000 compared with earlier designs? Of the operational experience suggested by Table 1-1, how much is relevant to the design and in-service use of the removable closures of the Fuel Channels that are intended for ACR-1000?

## 16. Overpressure Protection

10820-01371-TED-001 section 5.1.3.3.2 states:

“Overpressure protection for the heat transport system is provided by liquid relief valves in conjunction with shutdown systems No. 1 and 2....”

Section 5.1.3.3.3 states:

“The heat transport system is provided with two 100% liquid relief valves connected to the reactor outlet header of each loop....”

For the Pressuriser, section 5.2.1.4 states:

“During shutdown, the pressuriser is isolated from the HT system circuit by motorised isolation valves in the surge line. Instrumented steam relief valves at the top of the pressuriser provide overpressure protection for the vessel when the pressuriser is isolated from the heat transport system.

For the steam side, the Main Steam Safety Valves (MSSVs) are described as follows in Section 5.2.5:

“The MSSVs are opened by the emergency core cooling system (ECCS) on detection of a LOCA to crash cool the secondary side....”

Section 5.2.5.3.3 states:

“A total of 12 spring loaded and pneumatic operated safety valves are provided, three per steam generator. The combined capacity of three MSSVs provides a capacity of 100% of the steam flow from each steam generator...”

**QUESTIONS:** Are the arrangements for primary and secondary side overpressure protection for ACR-1000 the same as for earlier CANDU designs? How much of the overall CANDU operating experience (Table 1-1 of 10820-01371-TED-001) is relevant to the ACR-1000 overpressure protection arrangements?

**QUESTION:** Do the HTS liquid relief valves combine a spring loaded function with remote Control & Instrumentation function? Or, is the control and instrumentation system local to the valves or a combination of local and remote?

**QUESTION:** Do the HTS liquid relief valves have upstream isolation valves to cope with a stuck open liquid relief valve?

**QUESTION:** For main steam relief, is the single failure criterion applied to each steam line?

**QUESTION:** In the first instance, I assume the overpressure protection arrangements meet CAN/CSA codes; do the overpressure protection arrangements meet ASME III code requirements?

## **17. In-Service inspection**

10820-01372-TED-001 in section 6.8.1 (page 6-45) states the following:

“The inspection program for pressure retaining components is based on a sample of the components subject to inspection, with inspection area selected on failure size and consequence, peak stress, fatigue usage factor, and the number of identical components.

Based on the CANDU standard that addresses the periodic inspection program the following criterion is applied:”

**QUESTION:** What is the CANDU standard that addresses the periodic inspection programme?

From the wording, I take it that although ASME III has some role in design and manufacture of metal pressure boundary components, ASME XI is not used for the in-service inspection programme.

On page 6-46 the following is stated:

“The criteria used to select the inspection areas are intended to identify areas where potential deterioration may affect a number of components on a generic basis, where failure of these components may challenge the Safety Systems in a way that was not intended. The extent of the inspection sample required is determined by taking into consideration the size of the failure (i.e. the consequence of failure) and the likelihood of failure in terms of stress level, the fatigue usage factor (where calculated) and the potential for erosion or corrosion.

For example, the shutdown system No 1 (SDS1),... all components subjected to low pressure....Failure of the pressure boundary does not cause any increase in pressure in containment...It is therefore excluded from the periodic inspection.

As another example, the steam generator secondary shells are the only components of the steam and feedwater system which are subject to inspection, because their failure could impair the capability to cool the fuel.....rupture of the steam and feedwater system piping would not jeopardize the integrity of the heat transport system (HTS), and the system piping is, therefore, excluded from inspection....

The design of the ACR systems ensures that parts of the systems subject to inspection are readily accessible to minimise the dose to operators...piping location allows a full 360-degree weld inspection.”

**OBSERVATION:** NII is likely to take an interest in the in-service inspection programme and how it is defined. This is an important but long time-scale matter. The important point for now is that in-service inspection options (notably accessibility) are not foreclosed by a design process that includes assumptions about a specific in-service inspection programme. In the UK, the steam and feedwater lines in areas accessible to personnel would fall under the Pressure Systems Safety Regulations (PSSR 2000) and may require a degree of inspection for ‘conventional safety’ reasons. The inspection programme required under PSSR 2000 Regulations would be a matter for the ‘Competent Person’ appointed for the particular station.

### **18. Main Steam Lines Outside Containment - Main Steam Safety Valves and Main Steam Isolation Valves**

10820-012371-TED-001 Figure 2.3-2 shows the Main Steam Safety and Isolation valves and associated main steam line pipework at the top of the Reactor Auxiliary Building between the Reactor Building and the Turbine Hall. Section 2.3.2 describes the location as “on the roof of the reactor auxiliary building”.

**QUESTION:** Are the Main Steam Safety and Isolation Valves and associated pipework protected from external events, including aircraft crash?

**QUESTION:** In the area at the top of the Reactor Auxiliary Building, are individual trains of the Main Steam Safety and Isolation Valves and associated main steam line pipework protected from gross failure of an adjacent steam line train?