

HEALTH AND SAFETY EXECUTIVE  
HM NUCLEAR INSTALLATIONS INSPECTORATE

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

**Preliminary Review Assessment of:  
Structural Integrity Aspects of GE-H ESBWR**

HM Nuclear Installations Inspectorate  
Redgrave Court  
Bootle  
Merseyside L20 7HS

## 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. This is sufficient for Step 2. 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 ESBWR

### **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 ESBWR, implicit in the submission is that gross failure of the Reactor Pressure Vessel is discounted. By comparison, gross failure of certain piping is explicitly discounted (a claim) on the basis of a set of arguments and evidence referred to as 'break exclusion zone' (piping in the vicinity of the containment wall).

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 the Reactor Pressure Vessel (RPV);

location of circumferential welds in the RPV body;

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

The ESBWR Reactor Pressure Vessel is large (27.5m high, 7.1m diameter) but this is not a fundamental issue, and represents an increment in size compared with

earlier BWR designs. NII would expect all main pressure welds of the RPV body to be completed before delivery to site.

Overall, we conclude GE-H has provided an adequate overview of the claims made for structural integrity of metal components and structures. However, 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. GE-H 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.

**ESBWR**

**NOTES OF PRELIMINARY REVIEW OF THE  
UK APPLICATION  
PRELIMINARY SAFETY REPORT (PSR)  
AND  
DESIGN CONTROL DOCUMENT (DCD)  
-  
STRUCTURAL INTEGRITY**

These notes are the outcome of my preliminary review of the structural integrity aspects of the UK application documentation provided by General Electric-Hitachi (GEH) for their ESBWR.

Structural integrity here means the integrity of metal pressure boundary components, 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.

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

ESBWR - UK Preliminary Safety Report 26A7403AA Rev 00. Sections 2.5 and 2.8;

ESBWR Design Control Document. Sections 1.1 to 1.3 and 1.9 in 26A6642AD Rev 03;

ESBWR Design Control Document Section 3.2 in 26A6642AJ Rev 03;

ESBWR Design Control Document Section 3.9.3 in 26A6642AK Rev 03;

ESBWR Design Control Document Sections 5.2.2, 5.2.3, 5.3, 5.4.4, 5.4.5, 5.4.9 in 26A6642AR Rev 03.

ESBWR Design Control Document Section 10.4.1 and 10A in 26A6642BF Rev 03.

The notes below are in the form of 11 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 circuit of the reactor system has 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 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.

The vessels and pipework inside the containment of the ESBWR which fall under the second category above appear to be limited to:

Reactor Pressure vessel;

Main Steam pipework 'in the vicinity of the containment wall' where break exclusion arguments are used (now set out in USNRC Standard Review Plan 0800, BTP 3-4, formerly MEB BTP 3-1 within SRP 3.6.2);

Main Feedwater pipework 'in the vicinity of the containment wall' where break exclusion arguments are used (now set out in USNRC Standard Review Plan 0800, BTP 3-4, formerly MEB BTP 3-1 within SRP 3.6.2).

**QUESTION:** Inside containment, is the claim that the likelihood of gross failure is so low that it can be discounted applied to components other than the Reactor Pressure Vessel and segments of the Main Steam and Feedwater Lines?

## 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

The Reactor Pressure Vessel is large (7.1m inside diameter, 27.5m high, 180mm wall thickness).

An RPV diameter of about 7.1m for a BWR was established with the ABWR design circa 1991 (US Patent 5213756) and was subsequently adopted for the SBWR (1550MWe) design (26A7403AA page 2.1-3) (original SBWR at 600MWe had an RPV diameter of about 6m). The ABWR RPV has a height of about 22m. Earlier evolutions of the BWR design have comparable vessel dimensions, e.g. BWR/4 and BWR/6 with height about 21.8m and diameter 6.4m (slide 4 in: <http://www.ne.doe.gov/np2010/pdfs/esbwrReactorCoreNeutronics.pdf>).

I understand ABWR RPVs have been constructed for the following plants:

Kashiwazaki-Kariwa 6 and 7 (operating)

Lungmen 1 and 2 (yet to operate)

The extra length of the ESBWR RPV is achieved by adding length to the cylindrical section of the vessel.

There is no fundamental issue with the size of the RPV proposed for the ESBWR.

I note the RPV is made mainly from forgings to specification 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.

26A6642AR, Figure 5.3-3 shows a general section view of the ESBWR RPV. However this section does not show the locations of the main seam welds of the RPV.

**QUESTION:** For the ESBWR Reactor Pressure Vessel, where are the main welds located? This might best be answered using a drawing.

26A6642AR section 5.3.1.6 includes a description of the specimens included in the materials surveillance programme. For the RPV materials surveillance programme, we would prefer inclusion of a reasonable number of fracture mechanics specimens, for instance Compact Tension (CT) specimens, for each material form of the pressure boundary.

26A6642AR section 5.3.3.5 (page 5.3-18) state:

“The completed reactor vessel is given a thorough cleaning and examination prior to shipment.....”

I assume this means the Reactor Pressure Vessel fabrication is completed in the fabrication shop before shipping to the plant site as a single entity. NII would not expect a final fabrication weld to be made on delivery to the plant site, for instance a final joining weld for the RPV body section.

**QUESTION:** For the ESBWR in the UK would the Reactor Pressure Vessel body be delivered to the plant site as a single entity (i.e. after completion of all main seam welds at the fabrication works)?

The Main Steam and Feedwater lines are ferritic steel and the Reactor Pressure Vessel is ferritic steel. It may be then that safe ends are not a feature of the main nozzles of the ESBWR RPV.

**QUESTION:** Do the Main Steam and Feedwater nozzles of the ESBWR Reactor Pressure Vessel have ‘safe ends’? Do other nozzles on the ESBWR Reactor Pressure Vessel have safe ends?

#### 4. 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 failures 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 Main Steam Lines and Main Feedwater Lines in the vicinity of containment penetrations.

The basis for not postulating breaks in the vicinity of containment penetrations is 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 in SRP 3.6.3. I note from 26A6642AD Table 1.9-20 (page 1.9-42) that for the ESBWR NRC SRP 3.6.2 and BTP MEB 3-1 are used but SRP 3.6.3 is not (for SRP 3.6.3, the Comment column states "Note credited").

**QUESTION:** 26A6642AD Table 1.9-20 (page 1.9-42) appears to state that NRC SRP 3.6.3 ("Leak Before Break Evaluation Procedure") is not used within the ESBWR design. Are the provisions in NRC SRP 3.6.3 not used in the ESBWR design?

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. These measures may go beyond those in SRP 3.6.2.

26A6642AJ on page 3.6-2 in the list of Assumptions for 3.6.1.1 Design Bases states:

"A whipping pipe is not capable of rupturing impacted pipes of equal or greater nominal pipe diameter, but may develop through-wall cracks in equal or larger nominal pipe sizes with thinner wall thickness."

**QUESTION:** What is the basis (arguments and evidence) for the above claim quoted from page 3.6-2 of 26A6642AJ? Where is this claim used in the plant design, i.e. pipe locations?

#### 5. Main Steam Lines and Main Feedwater Lines

From 26A6642AR page 5.1-1:

The RCPB includes all pressure-retaining components such as pressure vessels, piping, pumps, and valves, which are:

- part of the RCS, or
- connected to the RCS up to and including any and all of the following:
  - the outermost containment isolation valve in piping that penetrates containment;
  - the second of the two valves normally closed during normal reactor operation in system piping that does not penetrate containment; and
  - the RCS safety/relief valve (SRV) piping and the depressurization valve piping.

In the above, the clause:

“Connected to the RCS and including the outermost containment isolation valves in piping that penetrates containment”

applies to the Main Steam and Feedwater Lines.

26A6642AR, Figure 5.4-3 is a plan outline diagram of the routing of the Main Steam and Feedwater Lines from the Reactor Pressure Vessel, through the containment wall and up to the isolation valve outside containment.

This diagram does not provide a basis for understanding the detail of the:

- piping support arrangements (including pipe whip restraints);
- the locations of postulated pipe breaks for design purposes;
- containment penetrations for these lines.

**QUESTION:** For the Main Steam and Feedwater Lines from the Reactor Pressure Vessel nozzles to beyond the isolation valves outside containment:

What are the support arrangements, including pipe whip supports, for the pipe runs?

At what locations are pipe breaks postulated for design purposes?  
 What is the arrangement / design for the penetrations through the containment wall?

One or more diagrams might facilitate the answers to these questions. It would help if these could consist of a physical layout as well as a P&ID type diagram.

Parts of the Main Steam and Feedwater Lines are ‘break exclusion’ zones based on compliance with NUREG 0800 SRP 3.6.2. The DCD documentation does not provide a diagram showing the boundaries of these break exclusion zones.

**QUESTION:** For the Main Steam and Feedwater Lines what is the extent of the SRP 3.6.2 'break exclusion zones' inside and outside the containment? One or more diagrams might facilitate the answer.

The Main Steam Line piping is made from SA-333 Grade 6 seamless pipe (Table 5.2-4 of 26A6642AR, page 5.2-44). The corresponding Main Feedwater Line piping is made from SA-335 Grade P.22 (Table 5.2-4 of 26A6642AR, page 5.2-46). Significant lengths of Main Steam Line and Feedwater Line are covered by a safety case which basically discounts gross failure (break exclusion zone). Therefore the material properties of SA-333 Gr 6 and SA-335 Gr P.22 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 bodies to be justified to a level of structural integrity consistent with that of the Main Steam Line pipework attached to the MSIVs.

I note from 26A6642AK section 3.9.3.3 (page 3.9-23), the Main Steam Piping from the Reactor Pressure Vessel to and including the outboard Main Steam Isolation Valve is designed and constructed to ASME III Class 1. From 3.9.3.4, the Main Steam Line Isolation Valves, Safety Relief Valves and the Depressurisation Valves are also designed and constructed to ASME III Class 1. 26A6642AR on page 5.4-28 states the Feedwater Lines between the RPV to the outboard isolation check valves are ASME III Class 1 items. From the isolation shutoff valves to the seismic interface restraint, the Feedwater Lines are ASME III Class 2.

**QUESTION:** Can a diagram be provided which explains the change in ASME III Code Class along the length of the Main Feedwater Lines as described on page 5.4-28 of 26A6642AR? It would help if this could consist of a physical layout as well as a P&ID type diagram.

26A6642AR section 5.4.4 (page 5.4-1) deals with the Main Steam Line Flow Restrictors. Each steam line has a venturi flow restrictor in the outlet nozzle of the RPV. Page 5.4-1 states:

"The flow restrictor design limits flow during a main steamline break to twice the normal full power flow."

The first two safety design bases in 5.4.4.1 are:

- Limit the loss of coolant from the reactor vessel following a steamline rupture outside the containment;
- Withstand the maximum pressure difference expected across the restrictor following complete severance of a main steamline;

In the first bullet point above, reference to a rupture only outside the containment is puzzling; especially given the statement in 26A6642AD on page 1.2-11:

"The restrictor limits the coolant blowdown rate from the reactor vessel in the event a main steamline break occurs anywhere downstream of the nozzle."

**QUESTION:** Do the Main Steam Line Flow Restrictors in the RPV steam outlet nozzles limit fluid flow in the event of a Main Steam Line rupture inside as well as outside the containment?

## 6. Overpressure Protection

26A6642AR Table 5.2-2 (page 5.2-41) lists data for the Safety Relief Valves (SRVs) and Depressurisation Valves (DPVs), including rated discharge capacities. The total discharge capacity for all ADS (10 valves) and Non-ADS (8 valves) SRVs is 2501.6 kg/s. 26A6642AR Table 5.4-1 (page 5.4-33) states the rated steam flow for a Main Steam Isolation Valve (MSIV) is  $2.19 \times 10^6$  kg/hour, or 608 kg/s. For 4 MSIVs the reactor total steam flow is then 2422 kg/s.

The 8 DPVs have a total flow capacity of 1912 kg/s.

**QUESTION:** Do the Depressurisation Valves (DPVs) have any role in overpressure protection, or are they actuated only in response to a situation where depressurisation is required?

## 7. ASME Code Edition and Code Classes Applied to Components

26A6642AJ Table 3.2-1 lists the principal components of the ESBWR and the applicable Safety Class. The Safety Classes 1, 2 and 3 align with ASME III Code Classes 1, 2 and 3 (Table 3.2-2). Figures 3.2-1 and 3.2-2 in 26A6642AJ give schematic diagrams showing the physical extent of Quality Groups and Seismic Categories.

Table 1.9-22 of 26A6642AD (page 1.9-100) shows the following ASME Boiler and Pressure Vessel Code Section III Editions and Addenda as applicable to the ESBWR:

ASME III Section III Division 1 Subsections NCA and NE: 2004 Edition

ASME III Section III Division 1 Subsections NB, NC, ND, NF, NG: 2001 Edition including Addenda through 2003.

A footnote to the second entry for ASME III states all limitations and modifications specified in 10CFR 50.55a(b)(1) are required to be met.

Other relevant parts of the ASME Code listed in Table 1.9-22 are generally 2001 edition with Addenda to 2003.

**QUESTION:** Why for ASME III Division 1 NCA has the 2004 Edition been used, when most other ASME Code references are to the 2001 Edition with Addenda to 2003?

**QUESTION:** Are the stress limits in footnote 12 to Table 3.9-2 consistent with the requirement to meet 10CFR50.55a(b)(1), which for piping analysis requires use of

NB-3200, NB-3600, NC-3600 and ND-3600 to be ASME Code prior to the 1994 Addenda?

## **8. Load Combinations and Stress Limits**

Table 2.5-1 of 26A7403AA (page 2.5-10) lists abnormal events against the Event Classification of Anticipated Operational Occurrence (AOO), Infrequent Event (IE), Accident and Special Event.

Table 3.9-1 of 26A6642AK (page 3.9-53) presents the plant events to be considered for the design and analysis of all ESBWR ASME III Code Class 1, 2 and 3 components, component supports, core support structures and equipment. Table 3.9-1 includes numbers of cycles. A table on page 3.9-22 lists the frequency of occurrence (per reactor year) for plant conditions to which ASME Code Service Limits A, B, C and D are applied.

Table 3.9-2 of 26A6642AK (page 3.9-55) lists load combinations and acceptance criteria for ASME III Code Class 1, 2 and 3 components, component supports and Class CS structures. I note load combination 7 in Table 3.9-2 is Normal Operation + Large Break LOCA + Safe Shutdown Earthquake.

Table 3.9-9 of 26A6642AK (page 3.9-86) lists load combinations and acceptance criteria for ASME III Class 1 piping systems.

## **9. In-Service Inspection**

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.

## **10. Turbine Condenser System**

Chapter 10 of the DCD (26A6642BF) deals with the Steam and Power Conversion System and section 10.4.1 deals with the Main Condenser. For section 10A, the main part of Chapter 10 assumes the ultimate heat sink is provided by natural draft towers. Section 10A deals with an alternative ultimate heat sink of a lake. Therefore, Chapter 10 of the ESBWR DCD does not consider seawater as an ultimate heat sink. But in the UK, all currently operating nuclear power stations use seawater as the ultimate heat sink.

26A6642BF in section 10.4.1.1 (page 10.4-1) states:

“The main condenser does not perform, support or ensure any safety-related function, and thus has no safety bases.”

The integrity of the condenser tubes has a material influence on the water quality of the feedwater. In-leakage of ultimate heat sink water into the condenser could affect the quality of the feedwater, which ultimately passes through the reactor core and the primary pressure boundary (RPV, feedwater lines and main steam lines). If the

condenser has no safety-related function, then feedwater quality relies on feedwater treatment and quality detection systems.

The integrity of the condenser tubes influences the reliance placed on any feedwater quality detection system. NII would expect reasonably practicable measures to be taken to make the integrity of the turbine condenser commensurate with the ultimate heat sink water and so minimise challenges to safety-related water quality detection and treatment systems. Clearly if seawater is the ultimate heat sink, we would expect a high integrity condenser tube design.

26A6642BF on page 10.4-3 states:

“The condenser and water boxes are welded carbon steel or low alloy ferrite steel. The tubes are stainless steel or titanium with compatible stainless steel or titanium clad carbon steel tube sheets depending on circulating water chemistry.”

This suggests two possible condenser tub materials. However 26A6642BJ (Chapter 12 of DCD) in section 12.3.1 (page 12.3-1) states:

“Main condenser tubes and tubes sheets are made of titanium alloys to minimize the introduction of foreign material into the reactor system as a result of condenser tube leakage.”

**QUESTION:** For the UK, is the intent that ESBWR turbine condenser tubes will be titanium with titanium clad carbon steel tube sheets, or is the option retained to use stainless steel tubing and tube sheets?

## 11. Cobalt

The ESBWR recognises the role in radiation build-up within the plant of trace amounts of cobalt in steels within the primary coolant circuit. For instance 26A6642AR Chapter 5 section 5.2.3.2.2 (BWR Chemistry of Reactor Coolant) (page 5.2-11) states:

### “Radiation Field Buildup

The primary long-term source of radiation fields in most BWRs is Co60, which is formed by neutron activation of Co59. Corrosion products are released from corroding and wearing surfaces as soluble, colloidal, and particulate species. The formation of Co60 takes place after the corrosion products precipitate, adsorb, or deposit on the fuel rods. Subsequent re-entrainment in the coolant and deposition on out-of-core stainless steel surfaces leads to buildup of the activated corrosion products (such as Co60) on the out-of-core surfaces. The deposition may occur either in a loosely adherent layer created by particle deposition, or in a tightly adherent corrosion layer incorporating radioisotopes during corrosion and subsequent ion exchange. Water chemistry influences all of these transport processes. The key variables are the concentration of soluble Co60 in the reactor water and the characteristics of surface oxides. Thus, any reduction in the soluble Co60 concentration has positive benefits.

As a means to reduce radiation field buildup, cobalt content has been reduced in alloys to be used in high fluence areas such as fuel assemblies and control rods. In addition, cobalt base alloys used for pins and rollers in control rods have been replaced with non-cobalt alloys. Furthermore, cobalt content is restricted in stainless steel components in the reactor vessel and other selected stainless steel components that have large surface areas exposed to high flow rates toward the reactor vessel.

The Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system removes both dissolved and undissolved impurities, which can become radioactive deposits. Reduction of these radioactive deposits reduces occupational radiation exposure during operation and maintenance of the plant components.

Water quality parameters can have an influence on radiation buildup rates. In laboratory tests, the water conductivity and pH were varied systematically from a high purity base case. In each case, impurities increased the rate of Co60 uptake over that of the base case. The evidence suggests that these impurities change both the corrosion rate and the oxide film characteristics to adversely increase the Co60 uptake. Thus, controlling water purity should be beneficial in reducing radiation buildup.”

There is similar coverage in 26A6642AP (DCD Chapter 4) on pages 4.2-11 and 4.5-2.

**QUESTION:** Is any consideration given to reducing cobalt content for materials beyond the reactor vessel, for instance the main turbines?