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NGNP with Hydrogen Production RPV and IHX Pressure Vessel Alternatives

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Revision	Date	Pages/Sections Changed	Brief Description
000	04/03/08		Initial issue
001	04/30/08		INL's comments incorporated as well as further inputs from suppliers
		p. 11	Figure 2-2 updated
		p. 13	Section 3 revised
		p. 21	Section 4.2.1.4 revised
		p. 22	Section 4.2.4 revised
		p. 24	Section 4.3.1 revised
		p. 28-29	Section 4.3.3 revised
		p. 30	Section 5.1 revised
		p. 37-38	Sections 5.3 and 5.4 revised
		p. 39	Section 6.1 revised
		p. 42	Section 6.3 revised
		p. 47-49	Sections 6.3.1.5, 6.3.2 and 6.3.3 revised
		p. 53	Sections 6.3.4.5, 6.3.4.6 and 6.3.5 revised
		p. 54-59	Section 6.4.1 revised
		p. 64-66	Sections 6.4.2 and 6.4.3 revised
		p. 73-77	Sections 6.5.1 and 6.5.2 revised
		p. 89-90	Section 6.7 added
		p. 98-100	Section 7.3.1.4 revised
		p. 105	Section 7.3.3 revised
		p. 120	Sections 8 and 9 revised
		p. 122	Reference list updated
		p. B-5	Table 2 revised
		p. C1-C2	Appendix C added

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1.0 INTRODUCTION

This study is carried out in the context of the Next Generation Nuclear Plant (NGNP) project. It follows activities carried out in 2007 during the pre-conceptual design studies and is aimed at providing additional information to support the selection of key parameters and technologies for the NGNP (e.g., reactor power, gas outlet temperature, IHX design and materials, etc).

This study will evaluate alternatives for the Reactor Pressure Vessel (RPV) materials and design, the cross vessel, and IHX pressure vessel materials considering the range of potential design and initial operating conditions for NGNP and the required and achievable metallurgical and physical properties required for these operating conditions. This study will also consider acquisition, fabricability, and reliability factors.

This study will also identify and evaluate the advantages of options to provide cooling or other design features where desirable to permit use of traditional materials (e.g., SA508 used in similar applications for light water reactors) for these components that may reduce cost and schedule risk to the NGNP Project.

The study also considers the impact of varying NGNP design and initial operating conditions (power level, core inlet and outlet temperatures, primary pressure) on the conclusions of the evaluations.

2.0 BACKGROUND AND ASSUMPTIONS

The Pre-conceptual design Studies Report (PCDSR, ref.1) was prepared based on the Combined Cycle Gas Turbine (CCGT) concept adopted by the ANTARES project. A configuration was proposed using multiple tubular IHX with the objective of providing at the same time electricity and very high temperature heat. It was however acknowledged that the steam cycle could be the best path forward for near-term deployment of HTRs.

The present study is primarily based on the indirect steam concept which differs from the conventional steam cycle concept by the addition of an Intermediate Heat Exchanger between the Nuclear Heat Source (NHS) and the Steam Generator (see figure 2-1). The study is performed as previously assuming direct production of Helium at very high temperature (900-950°C) to feed a H_2 production facility.

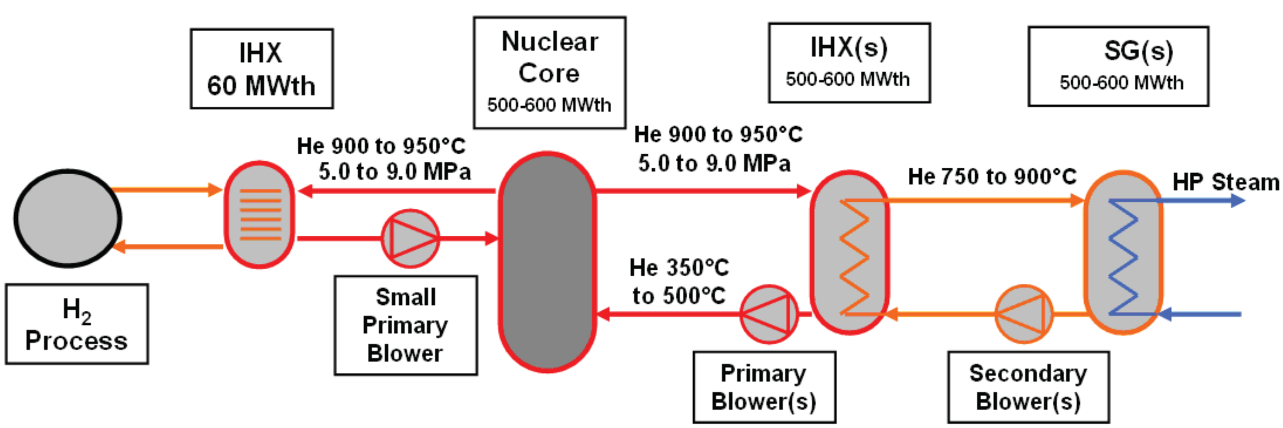


Figure 2-1: NGNP configuration considered in this study

Ref. 2 defines the configuration recommended by AREVA for the indirect steam concept. This configuration differs from the CCGT concept configuration by the fact that two loops can be envisioned on the Power Conversion side (instead of three for the CCGT concept). In the new configuration, the Reactor Pressure vessel is therefore surrounded by 2 tubular IHX vessels (with thermal power of 290 MWth each) and one compact IHX vessel (with thermal power of 60 MWth). Those vessels are located in an underground silo and are interconnected by cross-vessels.

The arrangement of Reactor Pressure Vessel, IHX vessels and cross vessels is shown in Figure 2-2. IHX vessels are themselves connected to Steam Generator vessels whose design and specific feasibility issues will not be discussed in the present document.

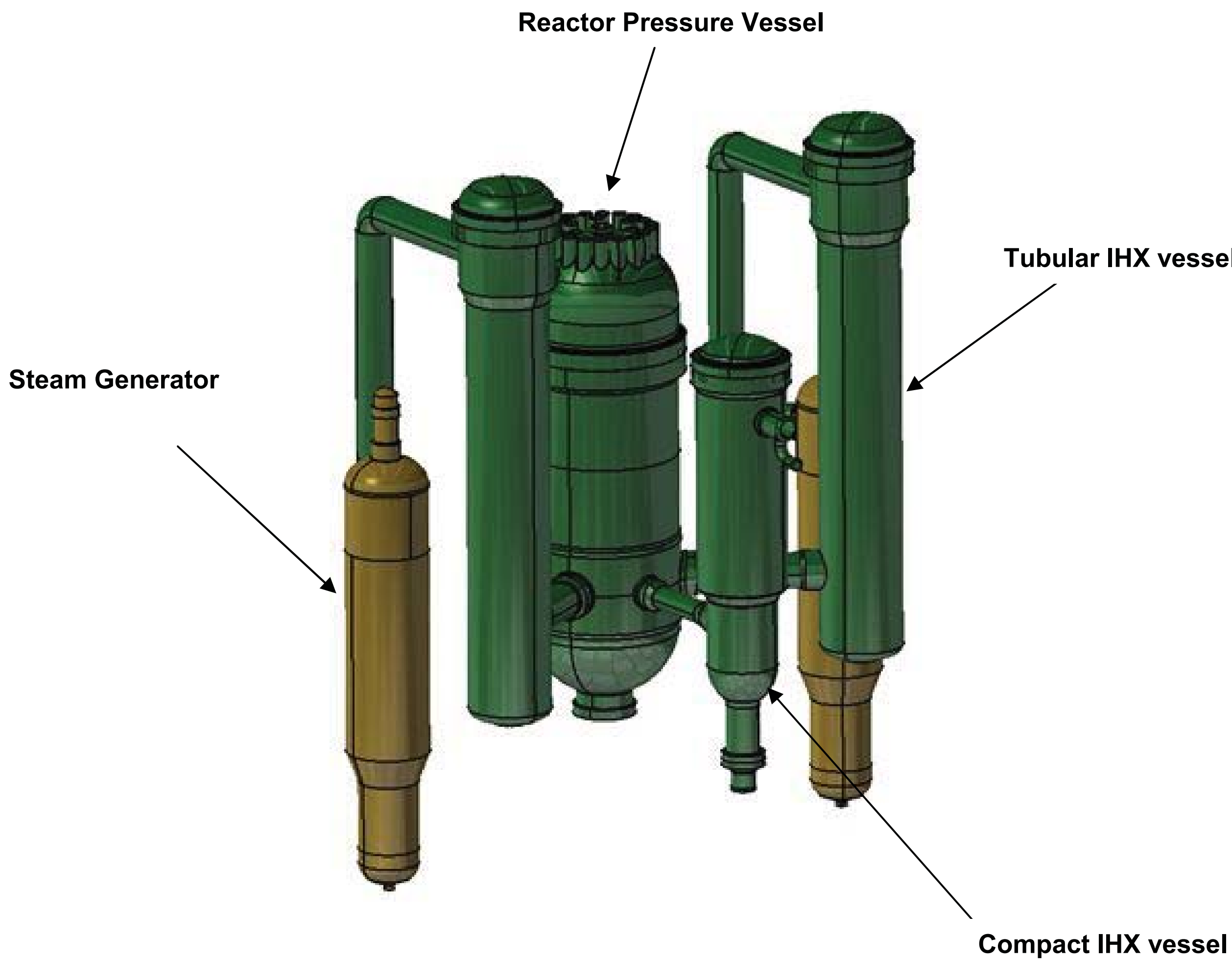


Figure 2-2: Vessels arrangement

The values of the normal operating parameters recommended as a result of Ref. 2 are indicated in Table 2-1.

The description of the Reactor Pressure Vessel, IHX vessels and cross vessels is provided in section 3.

One important assumption in carrying out this study is the objective of beginning initial operation of NGNP in 2018.

Table 2-1: Normal Operating Parameters

Parameter	Selection
Primary Side	
Primary Fluid	Helium
Reactor Power	565 MWt
Reactor Outlet Temperature	900°C
Reactor Inlet Temperature	500°C
Primary Coolant Flow Rate	272 kg/s
Primary Coolant Pressure	5 MPa at the circulator outlet
Heat transport to Hydrogen Production Plant	
Secondary Fluid	Helium
Heat Load	60 MWt
Heat transport to Power Conversion System	
Secondary Fluid	He
Heat Load	580 MWt (all electric mode)
Power Generation	
Power Generation System	Steam cycle

3.0 DESCRIPTION OF THE VESSEL SYSTEM

The Vessel System is composed of the vessels and supporting devices of the primary pressure boundary. This system is divided into the following subsystem:

- The Reactor Vessel
- The intermediate heat exchanger vessels (2 tubular IHX vessels and 1 compact IHX vessel).
- The cross-vessels (one for each IHX vessel)

The vessels are designed to contain the heat transport medium (helium) inventory within a leak tight pressure boundary and to maintain the integrity of this pressure boundary.

These vessels house and support the components of the Reactor Core, Reactor Internals, and the components of the Primary Heat Transfer System.

The Reactor Vessel and the IHX vessels are located in separate underground silo-type containment buildings and are interconnected by the cross-vessels, also located underground.

The preferred material for the vessels (based on pre-conceptual design work) is mod 9Cr1Mo steel. Section 4 identifies other possible alternatives. Section 5 discusses the strength that would be required, taking into account load conditions, geometrical constraints and fabrication issues. The emphasis is put in the current phase of the NGNP on feasibility issues, and section 6 discusses feasibility of procurement and manufacturing.

3.1 Reactor Vessel

The Reactor Vessel (see Figures 3-1 and 3-2) is approximately 25 meters high, 7.5 meters in diameter and 150 mm thick in the core belt line region (mod 9Cr1Mo option).

The upper closure head provides penetrations for the neutron control rod drives and fuel handling system. The closure head sealing device is ensured by means of 80 studs and a principle of metallic gaskets based on PWRs experience. For that, two concentric gaskets are fastened inside grooves machined into the top head flange.

The bottom head provides a single large opening for the shutdown cooling system blower and heat exchanger components.

The lower portion of the cylindrical vessel includes a local reinforcement because of the presence of the cross vessel nozzles and one lug welded at the level of the cross vessel axis which is used, together with two cross vessels, to support the reactor vessel.

Due to transportation limitations to INL site, the size of the Reactor Vessel will likely require that the vessel be assembled on the construction site. The current concept is that the vessel will be delivered on site in 4 packages + 1 for the cover head. Three circular welds will be required for final assembly of the Reactor Vessel on site (see Figures 3-3). The site welding could be performed in a dedicated on site workshop including the corresponding heat treatment, the final machining, the non destructive examination, the hydrotest and the cleaning facilities.

The total weight of the Reactor Vessel is 825 T (including 225 T for the cover head).

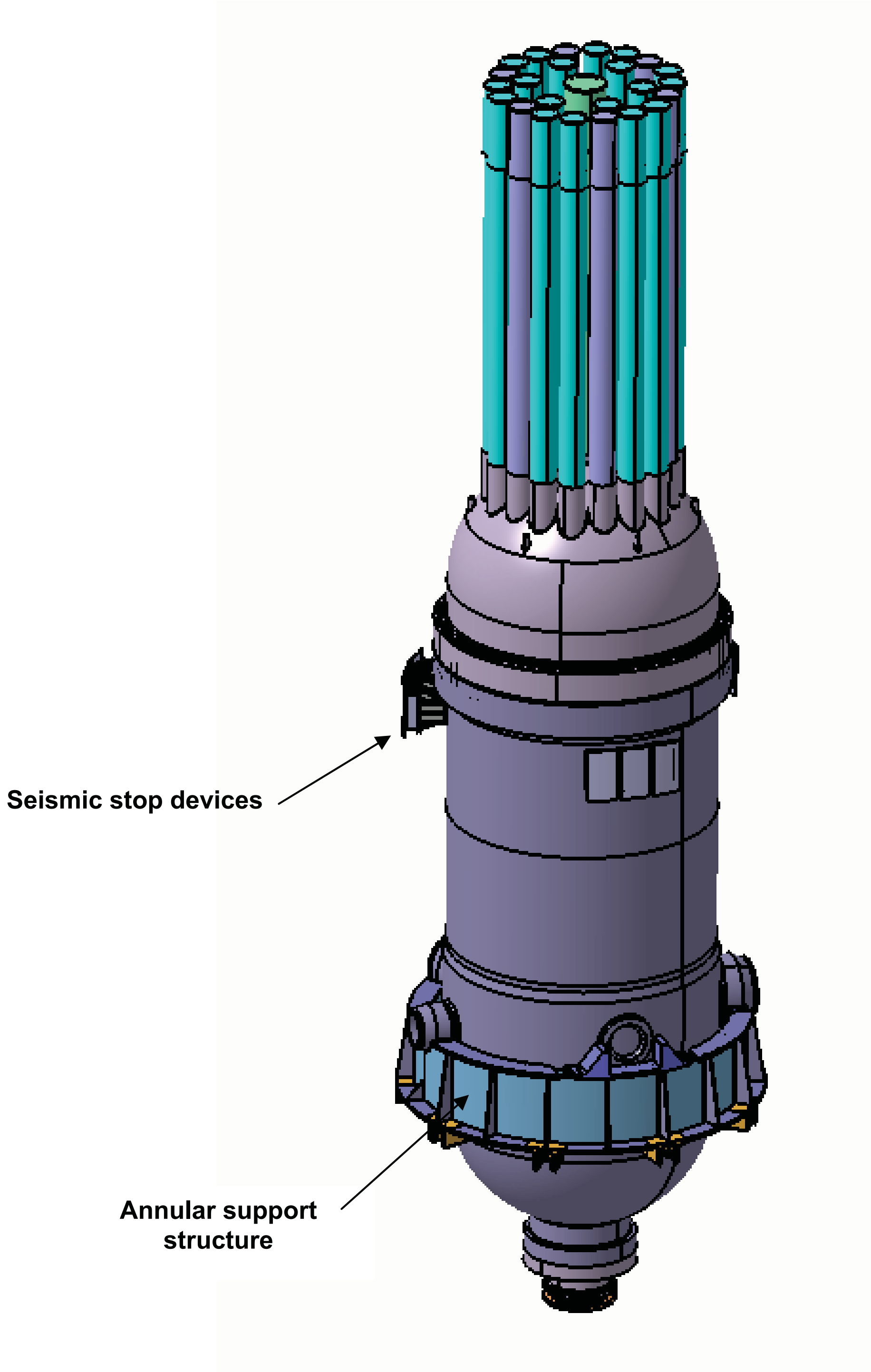
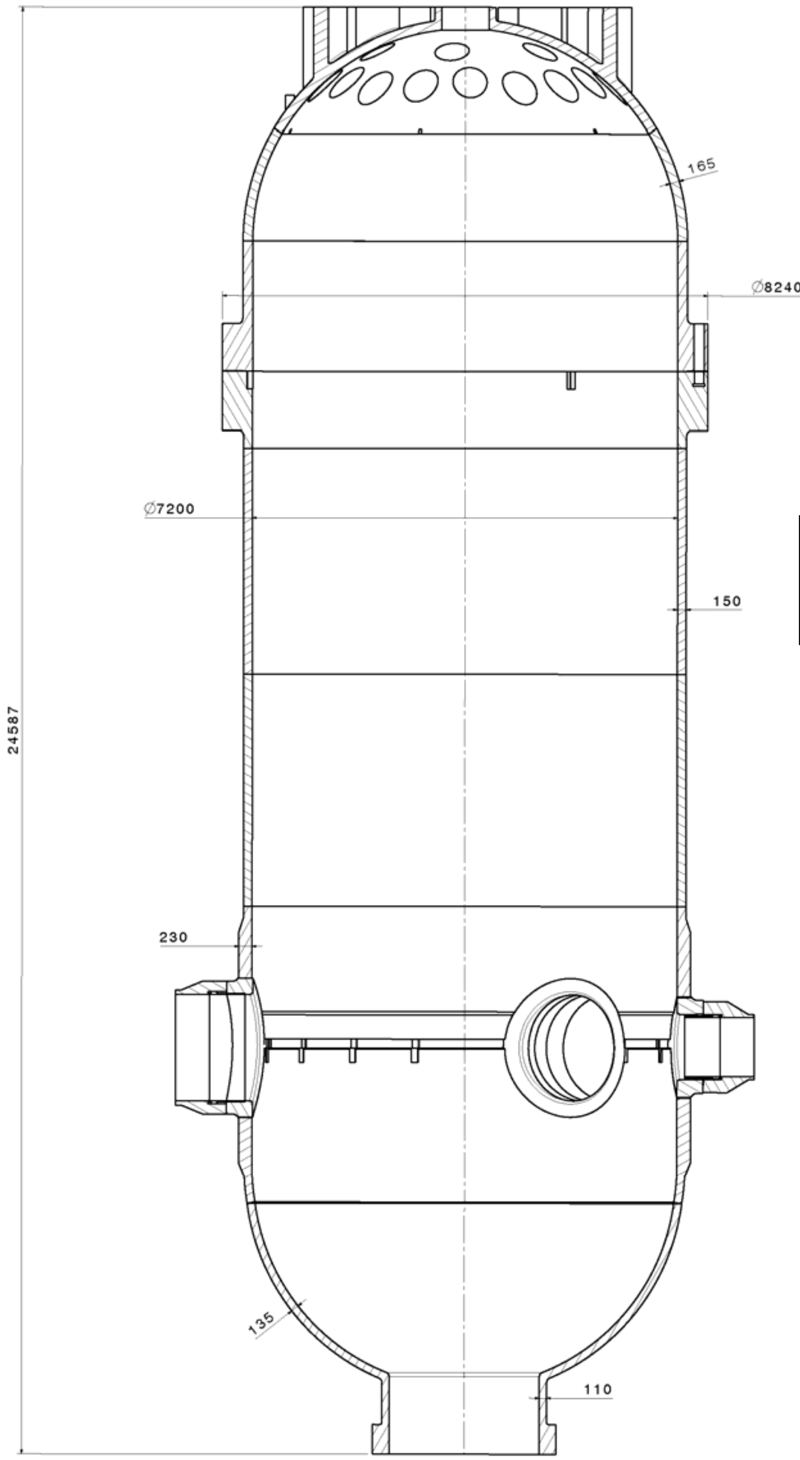


Figure 3-1: Reactor Vessel and Support System



Note: Drawing for mod 9Cr1Mo option

Figure 3-2: Reactor Vessel cross section

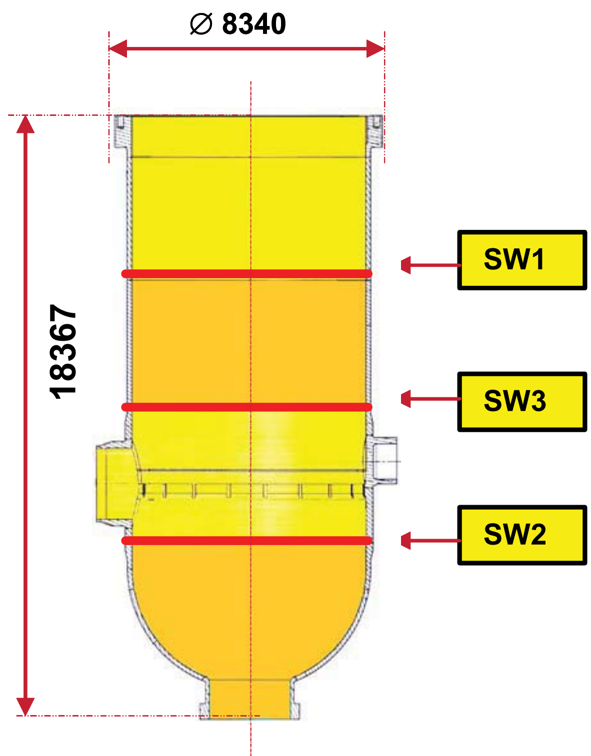


Figure 3-3: Reactor Vessel on Site Weld Locations

3.2 Cross-Vessels

The cross-vessels connect the lower portion of the Reactor Vessel to the lower portion of the intermediate heat exchanger vessels. The cross-vessels include a concentric duct (primary hot gas duct) that separates the hot (core exit) and the cold (core inlet) gas flow streams. The hot gas duct is insulated to reduce regenerative heat losses to the outer flow stream (core inlet cold gas). The cross vessel is a cylinder about 4 meters long, 85 mm thick with inner diameter of 1800 mm for the cross vessel to tubular IHX vessel. The cross vessel to the compact IHX vessel is very similar, except that the inner diameter of the cross vessel is reduced to 1100 mm.

The cross vessels are spread around the Reactor Vessel with an angle of 60°. Cross vessels and IHX vessels are clustered on one side of the Reactor Vessel to minimize the footprint impact.

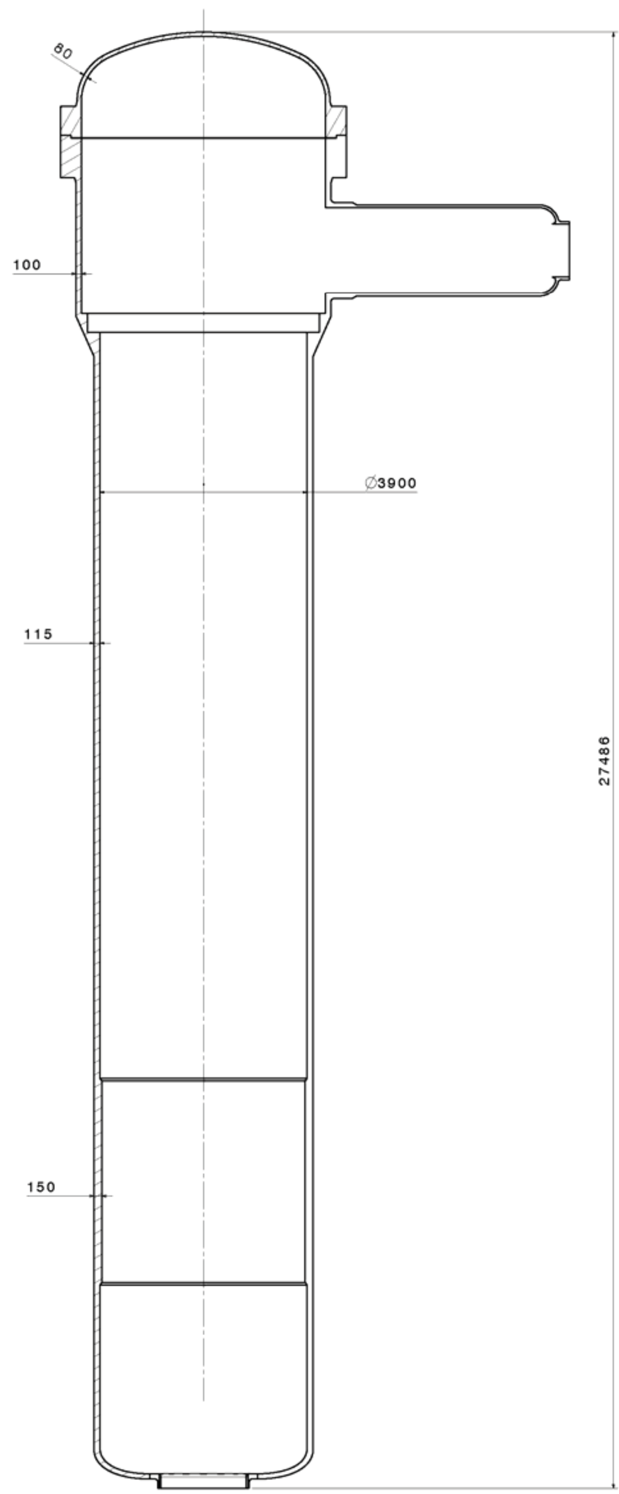
The welding of the cross-vessels to the Reactor Vessel and IHX vessels will be performed in the reactor cavity.

3.3 IHX Vessels

Figures 3-5 and 3-6 describe the IHX vessels. The sizes of the tubular and compact IHX vessels differ essentially by their height (about 27 m for the tubular vessel and about 21 m for the compact IHX vessel). The height of the tubular IHX vessel is reduced compared to the CCGT concept linked to the fact that the approach temperature recommended for the indirect steam cycle configuration allows a reduction of the tube bundle by 4 meters.

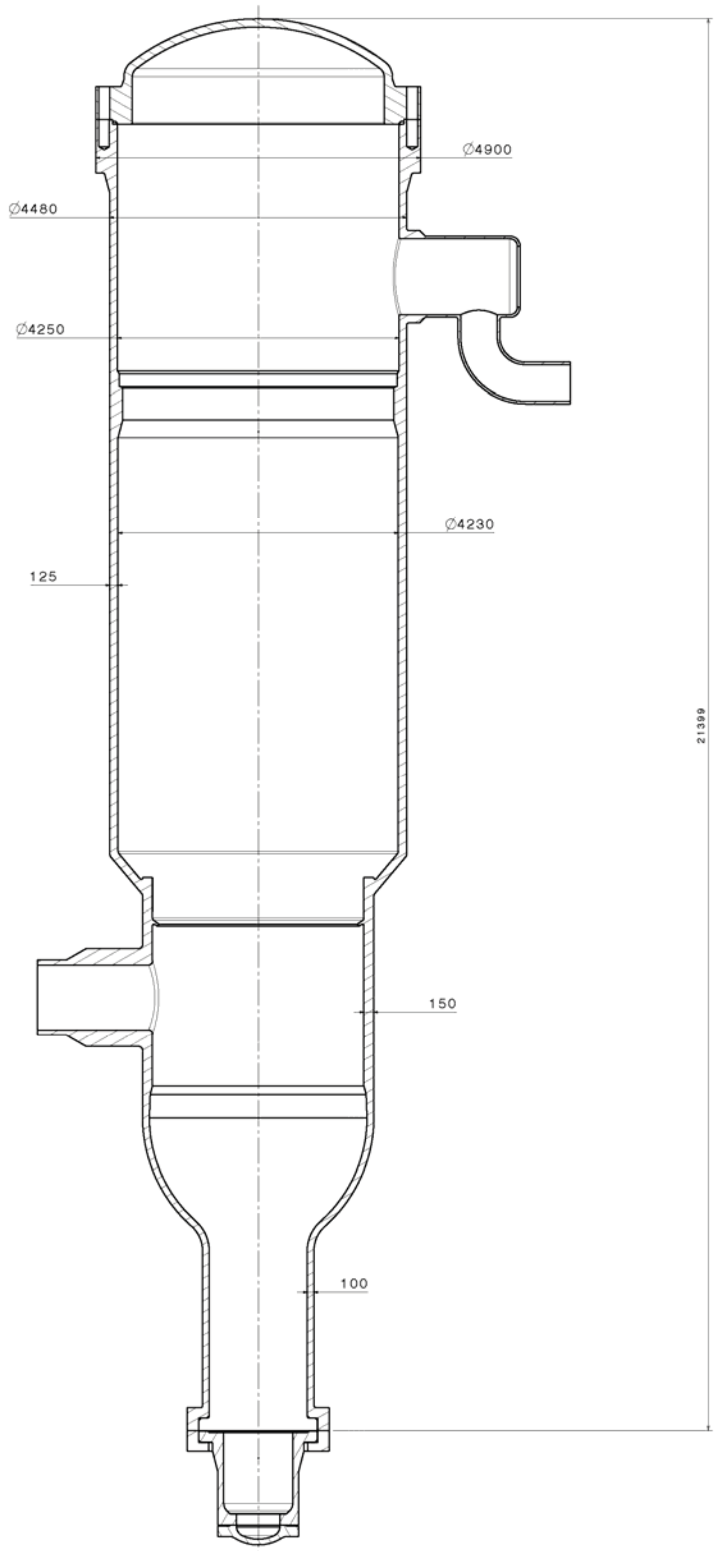
The outer diameter in the flange region is about 5 meters for both designs and, in contrast to the Reactor Vessel, it should be possible to fabricate these vessels in the workshop and transport them in one piece at INL site.

The IHX vessels should be thermally insulated in order to limit the heat losses and therefore increase the plant efficiency. As a result, the temperature should be very close to 500°C (except if active cooling were used or if thermal insulation was implemented inside instead of outside).



Note: Drawing for mod 9Cr1Mo option

Figure 3-4: Tubular IHX vessel



Note: Drawing for mod 9Cr1Mo option

Figure 3-5: Compact IHX vessel

4.0 IDENTIFICATION OF ALTERNATIVES

4.1 Scope of work

This section will identify potential alternatives for the design and material selection for these components.

The identification of design alternatives will provide comments on pros and cons of the different design options (e.g. forging design vs plate design, nozzle design with set-on vs set-in, etc) which might have ultimately an impact on material selection. Design options with cooling systems are covered in section 7.

The action on alternative materials will include a survey of the current practice in the non nuclear industry. The work will also provide a comparison of materials based on key selection criteria.

This work is based on the design developed by AREVA in the context of the PCDSR, adapted if necessary to cover new configurations envisioned for the conceptual design work.

4.2 Design alternatives

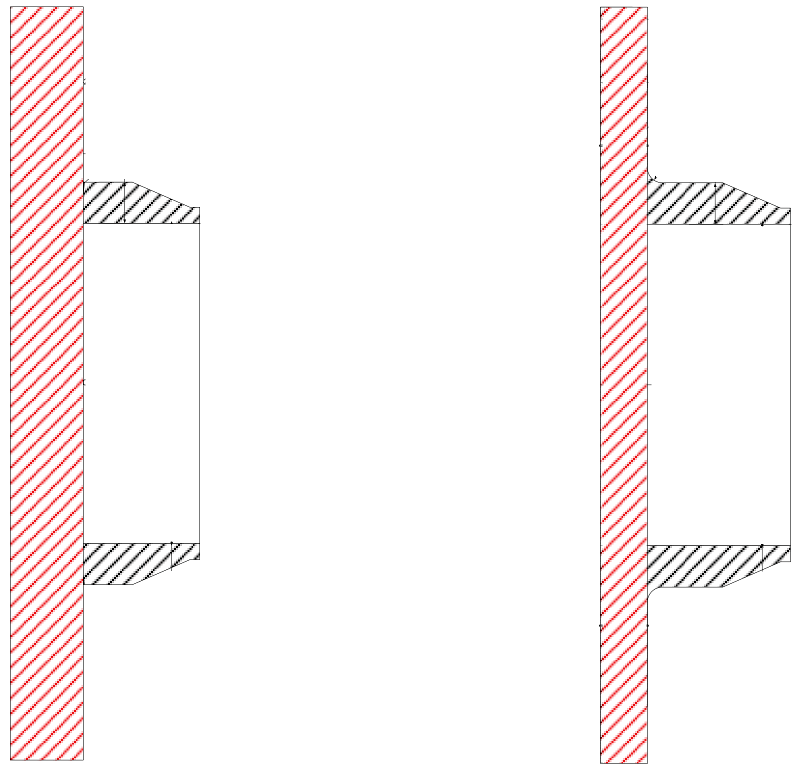
The following sections describe the design alternatives for the Reactor Pressure Vessel which is considered as more critical in terms of fabricability issues. The objective is to present the different options and discuss the pros and cons of the different alternatives. Section 6 discusses the feasibility of some of those alternatives, taking into account expected limitations from the suppliers and considering possible change of operating conditions.

4.2.1 Nozzle ring

4.2.1.1 Full forging with set-on nozzles

Nozzle ring with set-on nozzles correspond to current practice for PWR vessels. The advantage of the set-on nozzle is that it minimizes the risks of deformation of the nozzle ring as a result of welding operations.

On the other side, this option is more demanding in terms of forging size when compared to set-in nozzle option (see figure 4-1). The impact on the ingot mass could be in the order of 40% increase for nozzle ring with set-on nozzles.



(a) Set-on nozzle (b) Set-in nozzle

Figure 4-1: Forged ring with set-on or set-in nozzles

4.2.1.2 Full forging with set-in nozzles

Most PWRs and BWRs have been built in the past based on nozzle ring with set-in nozzles. The advantage of the set-in nozzle design is clearly a reduced size of the forged piece. This alternative might however require a tighter control of deformation during welding operation or could require increased thickness before welding to ensure that tolerances of inner diameter will be met.

4.2.1.3 Two-piece forgings with set-in nozzles

Figure 4-2 provides and example of two-piece forgings for the nozzle ring. This option was used for PWRs in the past when the capabilities of forging providers were limited. This was in particular the case for most of B&W 177 FA reactor vessels. This alternative would require necessarily a set-in nozzle to maintain a nozzle in one piece.

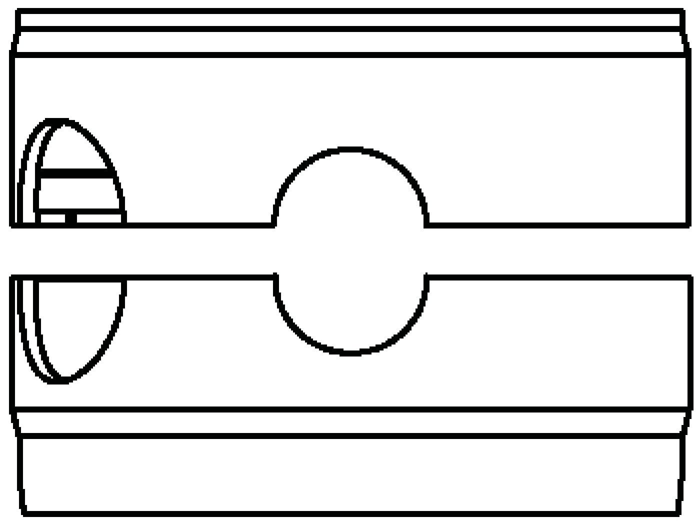


Figure 4-2: Two-piece nozzle ring forgings

4.2.1.4 Plates with set-in nozzles

The last alternative for the nozzle ring is the case based on rolled and welded plates with set-in nozzle. The limitation with this alternative is linked to the increased thickness of the reinforced area which would have to comply with welding process limitations as well as rolling capabilities. It would need to be checked what are the current capabilities of the suppliers in the US or overseas. The control of deformation during welding operation of the plates and the nozzles is also an issue that would need to be addressed.

4.2.2 Shell course

The current practice for PWRs is to rely on forgings in the core belt line region. The objective is three-fold:

- Minimize the number of welded joints to reduce costs of in-service inspections
- Limit the surveillance program to circular welds
- Avoid long seems which are subjected to hoop stresses two times greater than axial stresses.

Nothing prevents however to base the design on plates instead of forgings and older PWRs and BWRs were actually made of plates. It is also to be noted that irradiation is not considered as an issue for the HTR design and the expected fluence at the end of life is significantly lower than that in a PWR. The selection of mod 9Cr1Mo is also favorable due to improved irradiation behavior compared to conventional PWR steel.

The plate alternative (for both the nozzle ring and the shell course) would result in an increase of the length of welded joint.

4.2.3 Flanges

Figure 4-3 provides the detail of flanges envisioned for the NGNP. There is few alternatives available in terms of flange design and it can not be envisioned to provide the flange in several pieces due to the thickness of the flange which would make the welding challenging. The only alternative would be to suppress the flanges and replace them by a welded connection. This option would not be inconsistent with refueling operations which are performed through Control Rod Drive Mechanism (CRDM) penetrations. It might be however required to revisit the design of the non-metallic core support structure to ensure that pressure tests could be performed without graphite internals in place and that those internals could be loaded again through CRDM penetration after the pressure tests. This alternative would prevent easy access to the internals but would have the advantage of significantly reducing He leak through the metallic gaskets.

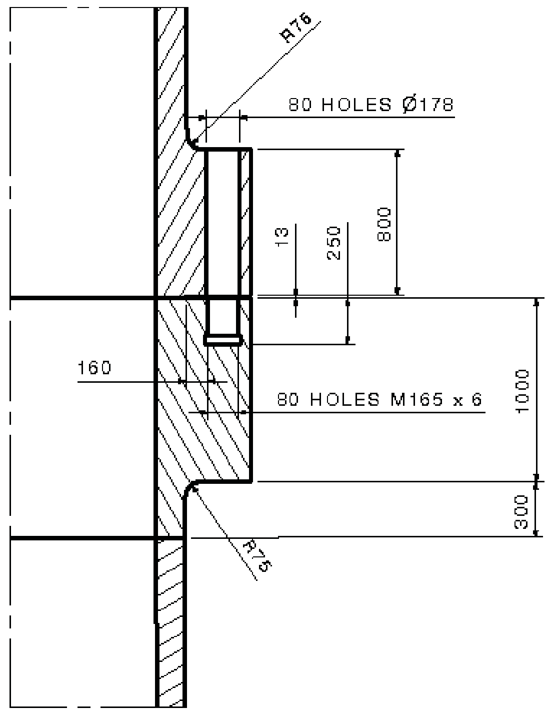


Figure 4-3: Detail of flange design

4.2.4 Bottom and upper head

The bottom head could be made indifferently of a forging or of plates. The only impact of plates would be to increase the number or welded joints to control during in-service inspection.

The top dome of the upper head (see figure 4-4) will have to be made of a forging so as to avoid interferences with the CRDM housing welds. The ingot required for this forging is likely to be significant if CRDM nozzles are integral to the upper head and it might be required to use set-in nozzles (which is the current practice in PWRs).

Alternatively, the top dome could also be envisioned based on two forged rings but non-destructive examination of the weld between the rings might be challenging due to the presence of the penetrations.

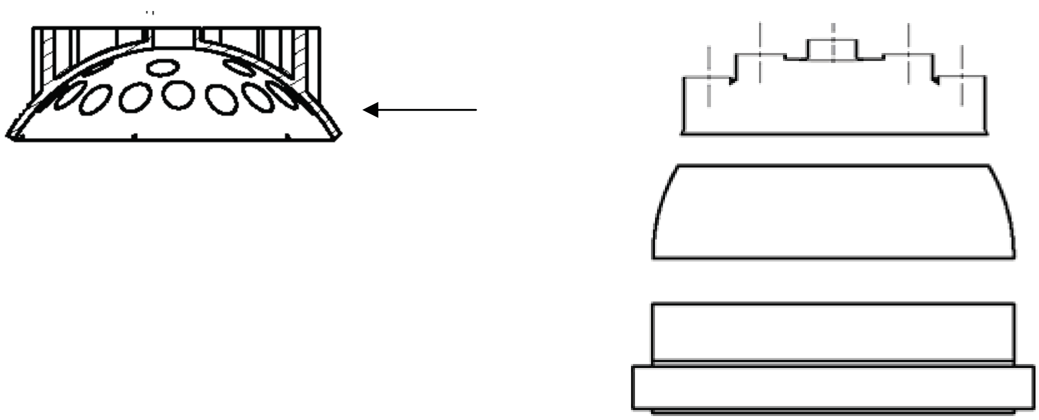


Figure 4-4: Upper head

4.3 Material alternatives

4.3.1 Identification of material alternatives

As discussed in section 3, the reference material selected for the vessel system (based on pre-conceptual design work) is mod 9Cr1Mo. This material was selected for its enhanced creep properties which would enable normal operation at a higher core inlet temperature (without having to rely on active cooling system) and would provide more margins to cope with high temperature transients. This material has also the advantage of a better behavior under irradiation compared to conventional PWR steel.

It is however recognized that there is issues associated to the fabrication of mod 9Cr1Mo vessels which remain to be solved and Table 4-1 identifies the complete list of material candidates which could be theoretically considered for such an application. It is to be noted that other grades of 2.25 Cr have been also developed in France for PWR or Sodium Fast Reactor applications but their use for the NGNP do not appear to be consistent with NGNP schedule.

Materials like grade 92, 12 or 122 developed for high temperature applications in the petrochemical industry are not considered as viable candidates for nuclear application as effort for codification is considered even more significant than that required for mod 9Cr1Mo and the objective is not to operate at high temperature (600°C or more) for long term operation. In the specific case of the NGNP vessels, the Reactor Vessel should be operated in the negligible creep regime. The IHX vessel could be operated in the significant creep regime but the operating temperature should be lower than that above which allowables are time-dependent.

Table 4-1 shows that the number of materials currently permitted for nuclear applications at low temperature (ASME III) or at elevated temperature (NH subsection) is limited to the following candidates:

- Mn Ni Mo low alloy steel (PWR grade)
- Mod 9Cr 1Mo
- 2.25Cr 1Mo annealed
- 2.25Cr 1Mo quenched / normalized and tempered
- 2.25Cr 1Mo with very high tensile strength (SA 541 grade 22 class 4)

It is not expected that a material not currently permitted by the ASME Code could be developed and qualified on time for a start-up by 2018.

Table 4-1: Material candidates

Material	ASME Designation	ASME III Class 1 Max Temp (°C)	ASME III Class 1 – NH Max Temp (°C)	ASME III Class 2 and 3 Max Temp (°C)	ASME VIII div. 1 Max Temp (°C)	ASME VIII div. 2 Max Temp (°C)
Mn Ni Mo low alloy steel (PWR grade)	SA 508 Grade 3 Class 1 SA 533 Grade B Class 1	371	NP (1)	371	427	371
Mod 9Cr 1Mo	SA 336 grade F91 SA 182 grade F91 SA 387 grade P91	NP 371 371	NP 650 650	NP 371 (t ≤ 3 in) 371 (t ≤ 3 in)	650 650 650	NP 482 482
2.25Cr 1Mo annealed	SA 336 grade F22 SA-387 Grade 22 Class 1	371	650	371	650	482
2.25Cr 1Mo quenched / normalized and tempered	SA-336 grade 22 class 3 SA-387 grade 22	371	NP	371	650	482
2.25Cr 1Mo with high tensile strength	SA 541 grade 22 class 3	NP	NP	NP	454	454
2.25Cr 1Mo with very high tensile strength	SA 541 grade 22 class 4	371	NP	NP	NP	NP
2.25Cr 1Mo V	SA-336 F22V	NP	NP	NP	482	482

Note: (1) Code Case N499-2 authorizes the use of this material up to 538°C under specific conditions
NP Not Permitted

Figure 4-5 provides a comparison of allowable stresses for the different material candidates. Notice that PWR steel and mod 9Cr1Mo have similar allowable stress around 370°C. 2.25Cr1Mo grades with high allowables show a significant drop in properties beyond 430°C. SA 541 grade 22 class 4 with even higher strength is permitted for use up to 371°C only and it is expected that it would follow the same trend as other 2.25Cr1Mo grades. It is also expected that fracture toughness properties would be low for this material. The annealed 2.25Cr1Mo material would require a significant increase of thickness compared to other candidates to compensate for the reduced tensile properties. On the other side 2.25Cr 1Mo V has similar allowable as PWR steel and mod 9Cr1Mo at low temperature and keeps its strength at higher temperatures with allowable even slightly above that of mod 9Cr1Mo. This material could therefore be envisioned for RPV application with expected reduced feasibility issues for welding but the time required to qualify it for the NGNP is not expected to be consistent with NGNP schedule.

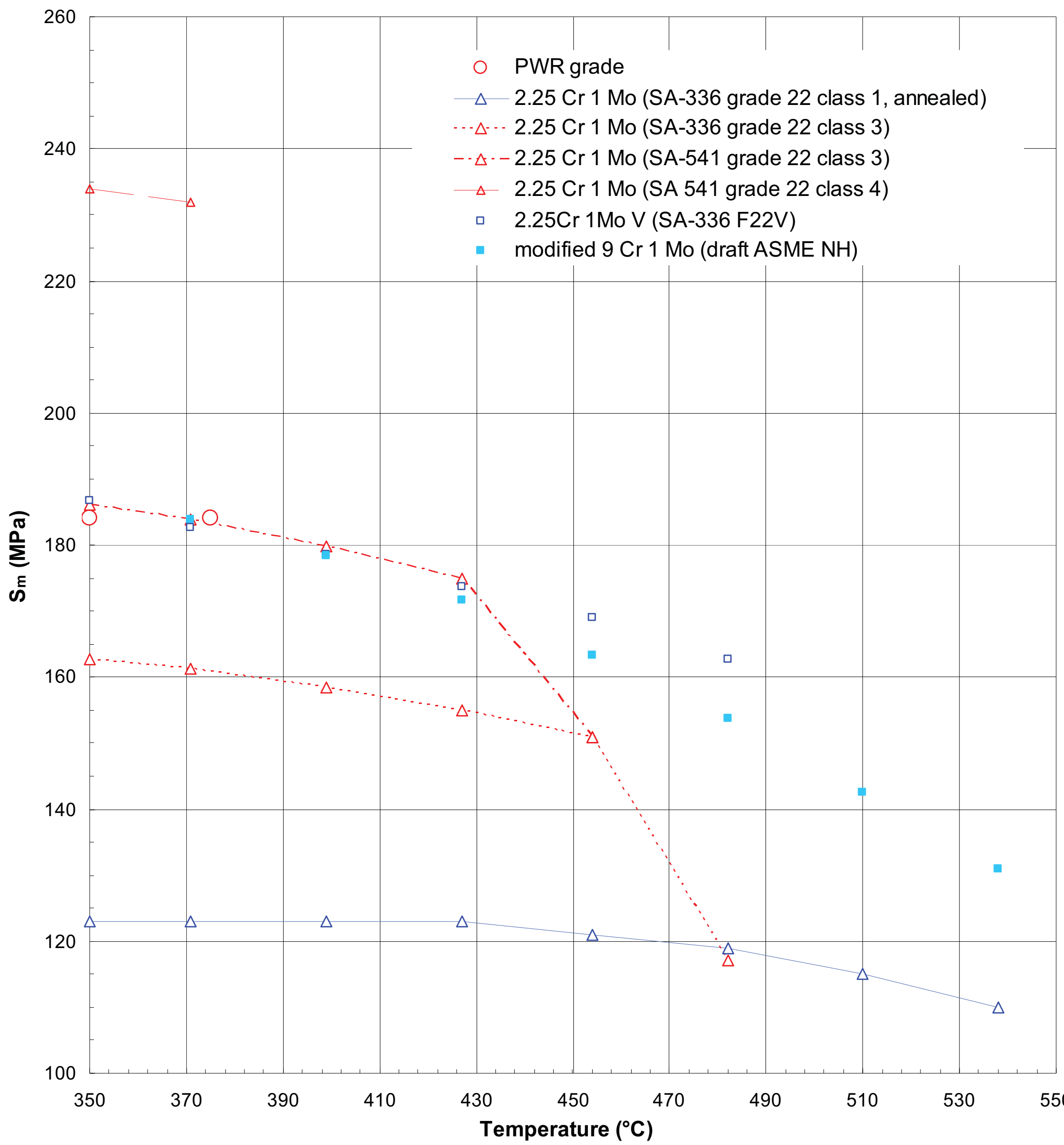


Figure 4-5: Comparison of Allowable Stress for Candidate RPV Materials

4.3.2 Survey of material used in Japan for pressure vessel applications

A survey has been performed of material used in Japan for pressure vessel applications. Table 4-2 provides a summary of materials used in the nuclear industry and Table 4-3 a summary of material used in the non nuclear industry for medium and high temperature applications.

Table 4-3 shows materials used at temperatures in the range 450 to 500°C but those material are not more creep resistant than PWR steel. Conversely, Cr-Mo alloy steel are used at about 400°C whereas operation at higher temperatures could be envisioned. It seems therefore that the material selection in the non nuclear industry is not necessarily based on creep resistance consideration.

Table 4-2: Material used in the nuclear industry

Material Name (JIS symbol)	Equivalent ASME code No.	Operating conditions (temperature, pressure)	Geometry (diameter, height thickness)	Adopted Component
Mn-Mo alloy steel (SQV2A, SFVQ2A)	SA 533B, SA508	343ºC, 17.2 MPa	~5.2m, ~13.6 m, ~255 mm	RPV, SG etc. for LWR
21/4Cr-1Mo alloy steel (SCMV4, SFVAF22B)	SA387 Gr22, SA 336 F22	395ºC, 3.9 MPa (1)	RPV : 5.5m, 13.2m, 122~160 mm	RPV&Heat Exchanger Vessels of HTTR

Note: (1) Design temperature and pressure of 440ºC and 4.8 MPa

Table 4-3: Material used in the non nuclear industry

Material Name (JIS symbol)	Equivalent ASME code No.	Operating conditions (temperature, pressure)	Geometry (diameter, height thickness)	Adopted Component
Mod 9Cr-1Mo (KA-SCMV28, KA-SFVAF28)	SA 387 Gr. 91, SA 182 Gr. F91, SA 335 Gr. P91	~600ºC		Main steam piping for Supercritical Pressure Boiler
Carbon steel for boiler and other pressure vessels (SB410,450,480)	SA 285/285M Gr. A,B,C	~450ºC	~ 200 mm (thickness)	Boiler and pressure vessel in medium and high temperature
Mo alloy steel (SB450M, SB480M)	SA 204/204 M Gr. A, B, C	~500ºC	~ 150mm (thickness)	Boiler and pressure vessel in medium and high temperature
Mn-Mo, Mn-Mo-Ni alloy steel (SBV1A,1B,2,3)	SA 302/302M Gr. A, B, C, D	~500ºC	~ 150mm (thickness)	Boiler and pressure vessel in medium and high temperature
Mn-Mo, Mn-Mo-Ni alloy steel (SQV1A,1B,2A,2B,3A,3B)	SA 533 Gr. A, B, C, D	~400ºC	~ 150mm	Boiler and pressure vessel in medium and high temperature
Cr-Mo alloy steel (SCMV1,2, 3,4,5,6)	SA 387/387M Gr. 2, 12, 11, 22, 22L,21, 21L, 5	~400ºC	SCMV1,2,3 : ~ 150mm , SCMV4,5,6 : ~ 300mm , (thickness)	Boiler and pressure vessel in medium and high temperature

4.3.3 Comparison of material candidates

Table 4-4 provides a comparison of material candidates on key selection criteria. The following materials are considered as possible candidates for start-up by 2018, mainly based on Codes and Standards considerations:

- Mn Ni Mo low alloy steel (PWR grade)
- Mod 9Cr 1Mo
- 2.25Cr 1Mo annealed (grade 22)

The detailed feasibility issues of the PWR grade and mod 9 Cr1Mo steel are further discussed in section 6.

The allowables of 2.25 Cr1Mo annealed are low and would require thicknesses about 150% larger than those required for other candidates. Section 6.5 also addresses feasibility of procurement of this material.

Table 4-4: Comparison of material candidates

Material	ASME III acceptance	Allowables	Negligible creep conditions	Procurement	Fabricability
Mn Ni Mo low alloy steel (PWR grade)	Permitted up to 371°C for normal operation and up to 538°C under specific conditions as per Code Case N499-2		No perspective of improving the negligible creep temperature of 371°C	Procurement of heavy section forgings to be clarified	No issue
Mod 9Cr 1Mo	Permitted up to 650°C but would required the acceptance of SA 336 grade F91 specification		To be defined but expected between 400 and 450°C	Procurement of heavy section forgings to be clarified	Welding qualification to be completed. Practicality of performing PWHT on site to be studied
2.25Cr 1Mo annealed	Permitted up to 650°C	Lower than PWR grade and mod 9 Cr 1 Mo which would require an increase of thickness by 150%	To be defined but expected around 400°C	Procurement of heavy section forgings to be clarified	Should be less a concern than for mod 9 Cr 1 Mo but welding qualification should be required
2.25Cr 1Mo quenched / normalized and tempered	Permitted for Class 1 components but not above 371°C	Lower than those for PWR grade and mod 9 Cr 1 Mo which would require an increase of thickness by 120%	To be defined but expected around 400°C	Procurement of heavy section forgings to be clarified	Should be less a concern than for mod 9 Cr 1 Mo but welding qualification should be required
2.25Cr 1Mo with high tensile strength (SA 541 grade 22 class 3)	Not permitted	Similar to those for PWR grade and mod 9 Cr 1 Mo but significant drop in properties beyond 430°C	To be defined but expected around 400°C	Procurement of heavy section forgings to be clarified	Should be less a concern than for mod 9 Cr 1 Mo but welding qualification should be required
2.25Cr 1Mo with very high tensile strength (SA 541 grade 22 class 4)	Permitted for Class 1 components but not above 371°C	Higher than thosefor PWR grade and mod 9 Cr 1 Mo but drop in properties expected as for SA 541 grade 22 class 3	To be defined but expected around 400°C	Procurement of heavy section forgings to be clarified	Should be less a concern than for mod 9 Cr 1 Mo but welding qualification should be required
2.25Cr 1Mo V	Not permitted	Similar to those for PWR grade and mod 9 Cr 1 Mo	To be defined but expected around 400°C	Procurement of heavy section forgings to be clarified	Should be less a concern than for mod 9 Cr 1 Mo but welding qualification should be required

5.0 REQUIRED MATERIAL PROPERTIES

5.1 Scope of work

This section will define the required dimensions and material physical properties of these pressure boundary components for the recommended operating conditions of the plant. This evaluation will summarize the stress levels, and other factors as required to assess the viability of the candidate materials for normal operating conditions, anticipated transients, abnormal events, and design basis events. This action will be mainly based on hand calculations and will address prevention of failure under primary stresses as well as creep-fatigue damage.

This action will be based on the design developed by AREVA in the context of the PCDSR, which means based on mod 9Cr1Mo option. Assessment based on 2.25Cr1Mo (grade 22) is discussed in section 5.3.

5.2 Preliminary assessment with mod 9Cr1Mo

A first investigation of the resistance of the vessels is performed. The precise areas considered are the following:

- ⇨ Current zone of the cross vessel
- ⇨ Current zone of IHX vessel
- ⇨ Current zone of reactor vessel

The vessels are examined,

- ⇨ Under design conditions
- ⇨ Under Level A and B
  - start-up / shutdown
  - Reactor trip conditions
- ⇨ Under Level C
  - Loss of heat sink
  - Depressurized Conduction Cooldown situation (only for RPV)

The ASME Code Subsection NH is used for this investigation. The stress level is estimated thanks to simplified formulations. No finite element model has been used except for thermal purpose in order to evaluate the gradient through the thickness of the vessels. This first evaluation shows that all the limits are fulfilled with comfortable margins except for the IHX vessel where the creep damage equals 17 (the limit to be respected is 1). A more detailed analysis should be performed in order to assess those results.

5.2.1 Material and description of the analyzed structures

The material chosen for vessels is mod 9Cr-1Mo. The following table sums up the diameter and the thickness of the different zones studied.

	Diameter (m)	thickness (m)
Cross vessel	1.885	0.085
IHX Vessel	4.015	0.115
Reactor Vessel	7.35	0.15

5.2.2 NGNP parameters

The NGNP parameters are sum up below:

- ⇨ Inlet/ outlet primary temperature: 500 – 900°C
- ⇨ Primary pressure: 50 bars
- ⇨ Power: 565MWth

5.2.3 Reactor Vessel

5.2.3.1 Stress level

The current part of the Reactor vessel is submitted to:

- Primary stresses computed by a simplified formula

[1] $$\sigma = \frac{PD}{2t}$$

D : diameter of the vessel (m)
t : thickness of the vessel (m)
P : pressure value (MPa)
σ : stress value (MPa)

- Secondary stresses due to a radial thermal gradient. secondary stresses are computed thanks to formula

[2] $$\sigma_{th} = \frac{E\alpha\Delta T}{2(1-\nu)}$$

E : Young modulus
α : thermal expansion coefficient
v : Poisson coefficient

5.2.3.2 Design limits

The calculated stress intensity value shall satisfy the limits below:

- $P_m \leq S_0$
- $P_l + P_b \leq 1.5S_0$

Only the first inequality needs to be verified in the current part (because there is no bending). The design pressure value is 6MPa.
The stresses are computed using simplified formulation:

- $$\sigma = \frac{PD}{2t}$$

D : diameter of the vessel (m)
t : thickness of the vessel (m)
P : pressure value (MPa)
σ : stress value (MPa)

σ (MPa)	Maximum design Temperature (°C)	S_0 (MPa)	margin %
147	460	161	9

The limits are satisfied.

5.2.3.3 Level A and B service limits

As there is no bending, only the following inequality needs to be verified:

- $P_m \leq S_{mt}$

The stress level is computed as in the previous paragraph. The pressure value is 5MPa.

σ (MPa)	Maximum Temperature (°C)	S_{mt} (MPa)	margin %
122.5	390	180	32

5.2.3.4 Level C service limits

The following inequalities need to be verified:

- $P_m \leq min\ (1.2S_m\ ;\ S_t)$
- $\sum \frac{t_i}{t_{im}} \leq 1$

t_i : the total duration of a specific loading
t_{im} : maximum allowed time under the load stress ;

The PCC situation is considered in order to verify the first inequality, because it is the most penalizing. Indeed, the pressure value equals 5.5MPa and the temperature is assumed to be 520°C during 2*500hrs.

σ (MPa)	Maximum Temperature (°C)	Min(1.2Sm ; St) (MPa)	margin %
135	520	165	18

The first inequality is verified.

In order to verify the second inequality, it is necessary to sum the contribution of loading during level A, B and C.

	P (MPa)	Maximum Temperature (°C)	t_i (hr)***	t_{im} (hr)	damage
Level A and B	122.5	390	$4.73\ 10^5$	∞	0
Loss of Heat Sink	135	391**	2*	∞	0
PCC transient	135	520	1000	20000	0.05

* Envelope value
** obtained by a thermal conduction computation
*** Time spent at the maximum temperature

The second inequality is verified.

5.2.3.5 Strain limits

Strain limits needs to be fulfilled for level A, B, C. Test N° A-2 is used and is validated when $X+Y \leq 1$ where :

⇨ $$X = \frac{P_m}{S_y}$$

⇨ $$Y = \frac{Q_r}{S_y}$$

S_y is the average value at the maximum and minimum wall averaged temperature during the cycle. In our case:

$$X = \frac{P_m}{S_y} = \frac{135}{362} = 0.373 \quad \text{and} \quad Y = \frac{Q_r}{S_y} = \frac{87}{362} = 0.24$$

Thus, $X+Y=0.613 \leq 1$. The limits are fulfilled.

5.2.3.6 Creep- fatigue evaluation

The creep-fatigue evaluation needs to be done on level A, B and C service loading. The following data are used for the analysis:

	P (MPa)	Q (MPa)	Maximum Temperature (°C)	t_i (hr)	Number of cycles
Level A and B	122.5	48	390	717*	660
Loss of Heat Sink	135	48	391	717	2
PCC transient	135	87	520	717	2

t_i= number of hours for one cycle for relaxation

* determined by considering the total service life at high temperature with 90% of availability divided by the number of cycles.
** Envelope value
It has to be noted that Level A and B takes into account start-up/ shutdown and Reactor trip. The distinction is not performed for the Reactor vessel because the stress level is the same for both transients.

It has to be underlined that the safety coefficient taken into account in the analysis is K'=0.9

	ε_t (%)	Number of allowable cycle	Allowable time	Creep damage	Fatigue damage
Level A and B	0.1	∞	∞	0	0
Loss of Heat Sink	0.1	∞	∞	0	0
PCC transient	0.15	$5*10^6$	$3*10^6$	0	0

The limits are fulfilled.

5.2.4 IHX Vessel

5.2.4.1 Stress level

The current part of the IHX vessel is submitted to:

- Primary stresses computed by the simplified formula [1]

- Secondary stresses but only during transients where a radial thermal gradient appears. Secondary stresses are computed thanks to formula [2]

5.2.4.2 Design limits

The methodology is the same as in § 5.2.3.2.

σ (MPa)	Maximum design Temperature (°C)	S_0 (MPa)	margin %
105	535	113	7

The limits are satisfied.

5.2.4.3 Level A and B service limits

As there is no bending, only the following inequality needs to be verified:

- $P_m \leq S_{mt}$

The stress level is computed as in the previous paragraph. The pressure value is 5MPa.

σ (MPa)	Maximum Temperature (°C)	S_{mt} (MPa)	margin %
87.3	490	138	37

5.2.4.4 Level C service limits

The methodology is the same as the one used in §5.2.3.4.

As only the RPV is concerned by PCC, simply the loss of heat sink is taken into account as a level C event.

Normal operating situation and Loss of heat sink are considered in order to verify the first inequality.

	σ (MPa)	Maximum Temperature (°C)	Total time	Min(1.2Sm ; St) (MPa)	margin %
Loss of heat sink	96	535*	2	158.4	39
Level A and B	87.3	490	$4.73*10^5$	138	37

* obtained by a thermal conduction computation

The first inequality is verified

In order to verify the second inequality, it is necessary to sum up the contribution of loading during level A, B and C.

	P (MPa)	Maximum Temperature (°C)	t_i (hr)***	t_{im} (hr)	damage
Level A and B	87.3	490	$4.73*10^5$	∞	0
Loss of Heat Sink	96	535**	2*	$4*10^5$	0

* Envelope value

** obtained by a thermal conduction computation

*** Time spent at the maximum temperature

The second inequality is verified.

5.2.4.5 Strain limits

Strain limits needs to be fulfilled for level A, B, C. Test N° A-2 is used and is validated when $X+Y \leq 1$ where :

⇨ $$X = \frac{P_m}{S_y}$$

⇨ $$Y = \frac{Q_r}{S_y}$$

S_y is the average value at the maximum and minimum wall averaged temperature during the cycle. In our case the most penalizing case is the reactor trip and the values are:

$$X = \frac{P_m}{S_y} = \frac{87.3}{306} = 0.285 \quad \text{and} \quad Y = \frac{Q_r}{S_y} = \frac{149}{306} = 0.487$$

Thus, $X+Y=0.77 \leq 1$. The limits are fulfilled.

5.2.4.6 Creep- fatigue evaluation

The creep-fatigue evaluation needs to be done on level A, B and C service loading by using the most penalizing combination in order to maximize the stress range. The following data are used for the analysis:

	P (MPa)	Q (MPa)	Maximum Temperature (°C)	t_i (hr)	Number of cycles
Normal operating conditions/ 0	87.3	0	490	717*	300
Reactor trip / 0	87.3	149	490	717*	358
Loss of Heat Sink / Reactor trip	9	245**	490	717*	2

t_i= number of hour by cycle for relaxation

* determined by considering the total service life at high temperature with 90% of availability divided by the number of cycles.

** corresponding to 149 MPa due to the Reactor trip plus 96MPa due to the Loss of heat sink.

It has to be underlined that the safety coefficient taken into account in the analysis is K'=0.9

	ε_t (%)	Number of allowable cycle	Creep damage	Fatigue damage
Normal operating conditions	0.05	∞	0	0
Reactor trip /0	0.14	∞	16.96 *	0
Loss of Heat Sink/ Reactor trip	0.15	∞	0.26	0

The limits are not fulfilled concerning the creep damage.

5.2.5 Cross Vessel

The cross vessel is only studied during normal operating conditions.

5.2.5.1 Stress level

The current part of the Cross vessel is submitted to:

- Primary stresses computed by the simplified formula [1]
- Secondary stresses due to a radial and axial thermal gradient. Secondary stresses are computed thanks to formula [2] and [3]

[3]

- $$\sigma_{th\_ax1} = \pm 0.353\sqrt{Rt}E\alpha\frac{\Delta T}{\Delta x}$$
- $$\sigma_{th\_ax2} = 0.195 \pm 0.106\sqrt{Rt}E\alpha\frac{\Delta T}{\Delta x}$$

E : Young modulus
α : thermal expansion coefficient
R : Radius of the vessel
t : thickness of the vessel
Δx : length of the thermal gradient

5.2.5.2 Design limits

The methodology is the same as in § 5.2.3.2.

σ (MPa)	Maximum design Temperature (°C)	S_0 (MPa)	margin %
67	535	113	41

The limits are satisfied.

5.2.5.3 Level A and B service limits

As there is no bending, only the following inequality needs to be verified:

- $P_m \leq S_{mt}$

The stress level is computed as in the previous paragraph. The pressure value is 5MPa.

σ (MPa)	Maximum Temperature (°C)	S_{mt} (MPa)	margin %
55	500	127	57

5.2.5.4 Strain limits

Strain limits needs to be fulfilled for level A, B, C. Test N° A-2 is used and is validated when $X+Y \leq 1$ where :

⇨ $$X = \frac{P_m}{S_y}$$

⇨ $$Y = \frac{Q_r}{S_y}$$

S_y is the average value at the maximum and minimum wall averaged temperature during the cycle. In our case, the most penalizing case is the reactor trip and the values are:

$$X = \frac{P_m}{S_y} = \frac{55}{342} = 0.161 \quad \text{and} \quad Y = \frac{Q_r}{S_y} = \frac{46}{342} = 0.135$$

Thus, $X+Y=0.3 \leq 1$. The limits are fulfilled.

5.2.5.5 Creep- fatigue evaluation

The creep-fatigue evaluation needs to be done on level A,B and C service loading. The following data are used for the analysis:

	P (MPa)	Q (MPa)	Maximum Temperature (°C)	t_i (hr)	Number of cycles
Level A and B	55	46	500	717*	660

t_i= number of hour by cycle

* determined by considering the total service life at high temperature with 90% of availability divided by the number of cycles.

** Envelope value

It has to be underlined that the safety coefficient taken into account in the analysis is K'=0.9

	ε_t (%)	Number of allowable cycle	Creep damage	Fatigue damage
Level A and B	0.06	∞	0	0

The limits are fulfilled.

5.3 Preliminary assessment with 2.25 Cr steel

Analyses carried out for the RPV in section 5.2 (design limits and Level A, B and C service limits) were run for 2.25 Cr steel (grade 22). It is shown that the thickness of the RPV in the core beltline region would have to be increased from 150 mm to 222 mm to maintain a similar margin of 9% for design limits. With such a thickness, a margin of 3% would be obtained in level C service limits.

If this thickness increase (+48%) is also applied to the thickness of the nozzle ring, the required thickness would be of 340 mm.

It would need to be checked if these thickness values would be acceptable from a procurement and manufacturing standpoint and a cost comparison between a mod 9 Cr1Mo vessel on the one side and a grade 22 vessel with increased thickness on the other side would need to be performed. Procurement of heavy section forging and plates made out of 2.25Cr1Mo is further discussed in section 6.5.

5.4 Conclusions

Preliminary evaluations based on simplified calculations and for mod 9Cr1Mo material have shown that the parts studied are acceptable except the IHX vessel for which the creep- fatigue damage is shown to be unacceptable. It is recommended that detailed transient analyses and Finite Element calculations be performed for the RPV, cross vessel and IHX vessel to confirm preliminary assessments.

Preliminary evaluations performed under the same conditions for 2.25 Cr steel (grade 22) show that the thickness would have to be increased by about 150 % and fabricability issues would have to be clarified.

6.0 EVALUATION OF ALTERNATIVE MATERIALS

6.1 Scope of work

This section will compare the feasibility issues associated to mod 9Cr1Mo and PWR grade (SA 508/SA 5333). This will include the following:

- Codes and Standards.
- Ability to procure the materials in the required for, (e.g., plate, forging), dimensions and thicknesses with acceptable through thickness metallurgical and material properties
- Ability to fabricate these components
- Initial and in-service inspection requirements and the practicality in successfully completing these inspections

For the ability to procure the materials in the required form, the following will be performed:

- Provide the initial key elements of material and component specifications for both mod 9 Cr1Mo and PWR grade options
- Identify alternative forging providers and assess the limitations in terms of ingot size
- Identify plate manufacturers and assess their capability to provide plates in the required dimensions and with the quality and properties expected.

The feasibility of procurement of forgings and plates will be focused on mod 9 Cr1Mo and PWR grade but will also discuss as to whether 2.25Cr1Mo steel could be procured in the required dimensions for the NGNP (with increased thickness compared to other material candidates as discussed in section 4.3.3).

It is to be mentioned that this study was performed in two steps. A preliminary assessment was carried out based on expected capabilities of suppliers. The work was then completed (as shown in the current revision of this report) with information obtained from suppliers (in particular Japan Steel Works and Industeel).

6.2 Codes and Standards

6.2.1 Modified 9Cr1Mo

6.2.1.1 Acceptability for the service under current ASME Code

Mod 9Cr1Mo is currently permitted for nuclear applications without restrictions up to 371°C and only for plates and small size forgings for temperatures up to 650°C.

Significant work has already been carried out on this material in the context of SFR projects and most of material properties are already available (even though update of some of the properties would need to be performed to take into account all test results available since then).

6.2.1.2 Effort required to extend or initiate new ASME Code cases

Actions required would be the following:

- Revision of Subsection NH to cover thick forgings (SA 336 grade F91) for RPV application
- Update of allowables for both base and weld material (covering the effect of heavy section forgings and plates)
- Improvement of the definition of negligible creep conditions
- Improvement of creep-fatigue damage rules.
- Validation of using fracture toughness properties of ASME Section XI appendix G for mod 9Cr1Mo or definition of specific properties for this material

Negligible creep conditions for mod 9Cr1Mo have been studied in reference 3 and reference 4 proposes a test program necessary to assess negligible creep conditions.

The conservatism of ASME creep fatigue rules is discussed in reference 5. The following has been identified:

- The values of the predicted stresses at beginning of hold times are far too high. Results could be improved by modifying the procedure for calculating the stress at the beginning of the hold time by taking into account the cyclic softening and symmetrization effects.
- The prediction of stress relaxation using the isochronous stress-strain curves is too conservative (the stress relaxation is underpredicted as compared with the experimental results). Conservatism could be reduced by performing systematic cyclic stress relaxation analyses using a creep strain law. The ASME Code may need also to be improved to provide recommendations on how to address elastic follow-up effects.
- The high safety factor used in the calculation of the creep damage (1/K'=1/0.67) is not justified. In the case of elastic analyses, at least, it would be more justified to use a value of 0.9 instead of 0.67 for K'. A proposal for modifying ASME subsection NH in such a way has been made and approved.
- The ASME NH creep–fatigue damage envelope is very conservative in the case of mod 9Cr-1Mo (bi-linear damage lines with (0.1, 0.01) intersection) and a bi-linear damage lines with (0.3, 0.3) intersection would seem more reasonable

Reference 6 proposes a test program to validate improved creep-fatigue rules.

It is also to be noted that LBB approaches could provide useful arguments in the frame of defense-in-depth analyses which are aimed at demonstrating the robustness of the design. LBB methodologies have been developed for PWRs and FRs but their application to HTRs would require further investigations. In that context, it would be required to develop the necessary fracture mechanics methods and material properties to perform crack growth and stability calculations.

It is finally to be noted that environmental effects will need to be addresses and section 6.2.3.1 defines R&D actions which could be needed for mod 9Cr1Mo.

6.2.2 PWR grade

6.2.2.1 Acceptability for the service under current ASME Code

As discussed in section 4.3.1, the PWR grade is permitted for nuclear applications up to 371°C under normal operation and up to 538°C for abnormal situations under the following conditions defined by Code Case N499-2:

- 3000 h maximum duration between 371 and 427 °C
- 1000 h and no more than 3 events between 427 and 538°C.

This material is fully qualified under PWR conditions but issues associated to the HTR environment will need to be addressed (see section 6.2.3.2).

6.2.2.2 Effort required to extend or initiate new ASME Code cases

The background of Code Case N 499-2 would need to be reviewed with the objective to offer the designer with more flexibility for short duration exposure.

In particular, it should be checked to which extent it could be allowed to exceed the present 538°C limit. The limits in terms of maximum duration for a given temperature range and number of occurrences could be also reviewed. Finally, the possibility of introducing different limits for level C and D events may be discussed.

6.2.3 R&D needs due to the environment

6.2.3.1 Modified 9Cr1Mo

6.2.3.1.1 Irradiation Effects

The effect of irradiation on a NGNP RPV made of modified 9Cr1Mo is expected to be small for three reasons:

- The end of life fluence on HTR RPV is expected to be smaller than in the case of PWR RPV by a factor of the order of 1/8.
- A higher service temperature results in a smaller deterioration of toughness for a given fluence.
- Modified 9Cr1Mo is free of nickel addition, nickel being a detrimental element for irradiation embrittlement.

A preliminary testing program has indicated that irradiation effects are not significant in the case of modified 9Cr1Mo HTR vessel. This result can be confirmed in the future by tests on more representative test coupons and toughness specimens.

6.2.3.1.2 Effect of Thermal Aging

The information on the effect of thermal aging on the mechanical properties has been provided by test programs at temperatures going from 482°C to 650°C. They have covered duration up to 75 000 h. At 482°C the effects of aging on tensile and toughness properties are very limited. At more elevated temperature the effects are dependent on the silicon content of the steel and as the toughness data indicate both aging and recovery effect on ductile–

brittle transition temperature (DBTT), they cannot be used to predict long term effect in the range 425–475°C. A specific aging program is required in that range of temperature.

Thermal aging at service temperature of NGNP RPV is not expected to destroy the tensile and creep resistance of modified 9Cr1Mo. But there are some indications that cyclic loading can accelerate significantly the microstructure evolution and load bearing capacity of the grade: this point needs to be clarified in representative conditions (temperature, loads and number of cycles).

6.2.3.1.3 Effect of Helium Environment

No particular effects are expected at service temperature of NGNP RPV from helium and from its impurity. This point would concern internals in service at more elevated temperatures.

6.2.3.1.4 Emissivity

Emissivity measurements in thermally treated conditions, after oxidation in air and after exposure to helium are necessary to reduce the uncertainty on this property which is used in design assessment.

6.2.3.2 PWR grade

Since the environment is different from PWR, the following points need to be checked:

- Thermal aging (end of life) in relation with accident conditions (short and medium term properties, 1000hrs at 540°C), followed by tensile tests and impact transition curves determination,

- Emissivity measurements similar to those for mod 9Cr1Mo but with reduced temperatures due to different service temperature.

- Effect of oxidation in helium environment (especially under hot transient conditions) will need to be checked (risks of C, N, O transfer between the fluid and the metallic surface).

6.3 Procurement specifications of alternatives candidates

This section identifies the requirements to be imposed during the procurement of mod 9 Cr1Mo and Mn-Ni-Mo low alloy steel. It provides in particular the background of the draft forging specifications provided in Appendices A and B. These specifications were discussed during a meeting with Japan Steel Works (JSW) and Appendix C summarizes comments made by JSW. Further meetings would be required to achieve a full consensus on the specifications.

6.3.1 Procurement of forged parts in grade F91

6.3.1.1 ASME-ASTME initial standards

The starting requirements are those of ASME-ASTM SA 336 for grade F91 which covers forged parts without limitation of mass. Unlike in SA 336, in SA 182 devoted to pipe flanges, forged fittings and valves, the mass is limited to 4 250kg. It is to be noted that subsection NH of ASME III concerning components in elevated temperature service allows grade F91 according SA 182 forgings but not according SA 336 forged parts.

The specified chemical analyses are quoted in Table 6-1. They are very similar in SA 336 and in SA 182. Differences are found for maximum phosphorus and sulfur contents only: The requirements for P and S will be discussed in 6.3.1.2.1.

The required mechanical properties are quoted in table 6-2. They are similar in SA 336 and SA 182. The unique difference is a maximum for ultimate tensile strength S_u of 760 MPa in SA 336; no maximum limit for S_u in SA 182.

Both SA 336 and SA 182 grades F91 are normalized and tempered with the same temperature requirements (Table 6-2).

Table 6-1: Specified chemical compositions (grade 91)

	Heat analysis		Product analysis	
%	**SA 336 grade F91**	**SA 182 grade F91**	**SA 336 grade F91**	**SA 182 grade F91**
C	0.08 – 0.12	0.08 – 0.12	0.06 – 0.15	0.06 – 0.15
Mn	0.30 – 0.60	0.30 – 0.60	0.25 – 0.66	0.25 – 0.66
P	0.025 max (1)	0.020 max	0.025 max (1)	0.025 max (1)
S	0.025 max (2)	0.010 max	0.025 max (2)	0.012 max
Si	0.20 – 0.50	0.20 – 0.50	0.18 – 0.56	0.18 – 0.56
Ni	0.40 max	0.40 max	0.40 max	0.43 max
Cr	8.0 – 9.5	8.0 – 9.5	7.90 – 9.60	7.90 – 9.60
Mo	0.85 – 1.05	0.85 – 1.05	0.80 – 1.10	0.80 – 1.10
V	0.18 – 0.25	0.18 – 0.25	0.16 – 0.27	0.16 – 0.27
Nb/Cb	0.06 – 0.10	0.06 – 0.10	0.05 – 0.11	0.05 – 0.11
N	0.03 – 0.07	0.03 – 0.07	0.025 - 0.080	0.025 - 0.080
Al	0.04 max	0.04 max	0.04 max	0.04 max
Ti				
Zr				
(1)	To be reduced to 0.020 or less in specification for RPV forged parts			
(2)	To be reduced to 0.010 or less in specification for RPV forged parts			

Table 6-2: Specified mechanical properties at room temperature and Heat Treatment for Mechanical Properties for grade 91

	SA 336 grade F91	**SA 182 grade F91**	**SA 387 grade P91**
S_y	≥ 415 MPa (60 KSI)	≥ 415 MPa (60 KSI)	≥ 415 MPa (60 KSI)
S_u	585–760 MPa (85–110 KSI)	≥ 585 MPa (85 KSI)	585–760 MPa (85–110 KSI)
A %	≥ 20	≥ 20	≥ 20
Normalize	1040–1095°C (1900–2000°F)	1040–1095°C (1900–2000°F)	1040–1080°C (1900–1975°F)
Temper	≥ 730°C (1350°F)	≥ 730°C (1350°F)	730-800°C (1350-1470°F)

6.3.1.2 Complementary requirements for chemical composition

6.3.1.2.1 Heat analysis

Starting the analysis specified by SA 336, it is worthwhile to clarify if some improvements in mechanical properties can be obtained through more severe limitations of phosphorus and sulfur contents (as in SA 182 as indicated in Table 6-1 or even more severe, S ≤ 0.005 for example). It is expected that more severe limitation of sulfur content should result in a lower transition temperature RT_{NDT} and in improved toughness. The sulfur contents of steels of grade similar to P91 or F91 produced and characterized in Europe and Japan are much less than the SA 336 specified maximum of 0.025% (less than 0.010%).

Following reduction in maximum phosphorus content, improvements in weldability are expected. The phosphorus contents of steels of grade similar to P91 or F91 produced and characterized in Europe and Japan are less than the SA 336 specified maximum of 0.025%. Their P contents are less than 0.012% with one exception at 0.019%.

6.3.1.2.2 Product analysis

Product analysis shall be performed in each coupon for mechanical tests. ASME/ASTM SA 961standard indicates the product analysis tolerances for alloyed steels. The corresponding extended chemical analysis to be specified in products, is given in Table 6-1 for some elements.
Due to the lack of data on the shift of composition from heat to products in the case of grade 91, it is proposed for the time being, to specify the same product analysis as in heat analysis.

6.3.1.3 Supplementary mechanical tests to be included in purchase specifications

6.3.1.3.1 Location of test specimens and tests after simulated post weld heat treatment

SA 336 standard requires two tensile tests in longitudinal direction. The tension tests shall be offset 180° from each other, except if the forging height exceeds 3.7 m. In such case, they shall be taken at each end of the forging. The practice for test specimens in heavy or complex pressure vessel components is to follow ASME III NB 2223:

- In the RPV beltline forgings, the specimen axis position should be t/4 from the heat treated surface, t being the wall thickness of the heat treated forging.

- In large and complex forgings (RPV nozzles, RPV flanges), specimen axis should be at least at 19 and 38 mm from the two nearest heat treated surfaces.

Sketches showing the exact locations shall be approved by the purchaser of the forged part. Longitudinal direction means that the longitudinal axis of the specimen is parallel to the major working of the forging. The corresponding direction will be approved by the purchaser as part of the approval of the Technical Manufacturing Program. For RPV beltline forgings, the longitudinal direction is the circumferential direction following the past PWR experience.

During fabrication of the RPV, the forged parts will undergo some post weld heat treatments. To cover a possible effect of these heat treatments on the mechanical properties of the parts, the tensile tests shall be repeated on coupons which have undergone a simulated post weld heat treatment. This treatment is dependent of the fabrication operations and will be defined by the purchaser. As a first proposal the holding conditions will be 20 h at 745 – 760°C (1375°F to 1400°F) ; the heating and cooling rates above 425°C (800°F) will be limited to 50°C/h (90°F/h). The specified properties shall be the same as for the tests after Heat Treatment for Mechanical Properties.

6.3.1.3.2 Tensile tests in transverse direction

As supplementary requirements, transverse tension tests should also be performed with tangential test specimens from the same test coupons as for longitudinal tension tests. The purpose of these tests is to verify the properties in the direction corresponding to the hoop stress of the cylindrical shell and to check that the forging operation gives similar mechanical properties in transverse and longitudinal directions. The required values for transverse tests are indicated in SA 336 : identical requirements for S_y and S_u in both directions, slightly lower required elongation in transverse case (19% instead of 20%).

6.3.1.3.3 Tensile tests at elevated temperature

Tensile tests at temperature close to the normal service temperature shall be performed in order to validate the tensile properties used in the design work. The specimens will be taken from the same coupons as for tests at room temperature. They will be in longitudinal or transverse directions whichever has shown the lower tensile strength at room temperature.

- Suggested test temperature : 425°C or 800°F
- Required tensile strength value : 455 MPa or 66 KSI
- Required yield strength value : 357 MPa or 52 KSI
- Elongation for information purpose

The specified yield strength is in agreement with ASME subsection II D table Y-1 which indicates S_y = 50.4 KSI at 800°F.

The specified tensile strength value is lower than that of ASME subsection II D table U which indicates S_u = 74.7 KSI at 800°F. According our statistical data, the 66 KSI level corresponds to average - 2 standard deviations. Such minimal values are probably acceptable for the supplier in opposition to the level of 74.7 KSI. Anyway, the agreement of the supplier is needed to put the acceptance level at S_u values. In addition, the test is performed after simulated PWHT which can reduce a little bit tensile strengths.

6.3.1.3.4 RT_{NDT} transition temperature

The determination of RT_{NDT} transition temperature can be useful for the assessment of the components against non ductile fracture. The procedure of determination is described in ASME III NB 2331. The determination of RT_{NDT} transition temperature will be performed on coupons which have undergone simulated post weld heat treatment.

The length of specimens for Pellini tests will be parallel to the axial direction. According to our past experience, RT_{NDT} as low as – 20°C can be obtained in heavy section forged parts (200 mm) but this probably requires a limitation in sulfur content as low as ≤ 0.010% or perhaps ≤ 0.005%.

The Cv specimens shall have their axis in axial direction, the notch tip being parallel to the radial direction (perpendicular to the surfaces of the forging). In order to draw the Charpy V impact transition curve, 3 specimens shall be broken at each of the test temperatures. At least 6 sets of specimens should be tested at 6 temperatures with one set on the upper shelf (100% shear).

6.3.1.4 Technical manufacturing program and inspections

6.3.1.4.1 Technical manufacturing program

The Technical Manufacturing Program submitted to the approval of the purchaser shall include at least the following data:

- Steel making and ingot pouring process
- Ingot size weight and discards
- Sketch of the forging after each operation with achieved dimensions and with indication of the main working direction
- Heat treatment cycles applied on the forging with dimensions at these stages
- The locations of the thermocouples on the forging during heat treatment for mechanical properties
- Sketch showing test coupons on the forging

- Sketch showing test specimens on the tests coupons

6.3.1.4.2 Inspections

Magnetic particle examination will be performed after machining of the as forged parts and after Heat Treatment for Mechanical Properties at the Forging's Supplier shop. Magnetic particle examination shall be performed in accordance with the paragraph NB 2545 of ASME code section III division 1

Ultrasonic examination will be performed after machining of the as forged parts and after Heat Treatment for Mechanical Properties at the Forging's Supplier shop. Ultrasonic examination shall be performed in accordance with the paragraph NB 2542 of ASME code section III division 1. The supplier shall submit the UT procedure to the purchaser approval.

6.3.1.4.3 Defective area removal

The surface defects shall be removed in accordance with the paragraph NB 2538 of ASME code section III division 1. No repair by welding shall be allowed at the Forging's Supplier shop.

6.3.1.4.4 Archive materials

The supplier shall deliver at the same time as the forging:

- One archive coupon for each coupons for delivery tests, the size of archive material being approximately the same as the test coupon.

- The circular prolongations of the test coupons not machined in test specimens.

Archive material shall have received the same heat treatment for mechanical properties as the forged part. They will be cut by mechanical means. They shall be marked in such a way that the main forging direction can be identified. They will be kept free from simulated post-weld heat treatment.

6.3.1.5 Key material properties for grade F91

The temperature of 425°C (800°F) proposed for tension tests at elevated temperature in 6.3.1.3.3 is related to the normal service temperature of the RPV. This selected value is linked to the assumption that any creep effect is negligible for the full life of the RPV at this temperature. The effort to validate this assumption is a key point for the use of grade 91 for the RPV. It seems possible to validate a limit of 425°C (800°F) ; extension of the service temperature up to 450°C (840°F) would be more difficult to assess.

6.3.2 Procurement of grade 91 plates

The bottom and the upper head of the RPV will be more or less spherical. For a head, there is no reason to prefer circular welds to meridian welds. Consequently heavy section plate can be preferred to forged parts if it is easier to fabricate and, furthermore, the use of plate is not expected to significantly affect the mechanical properties of the heads. The standard for grade P91plates is ASME-ASTM SA 387. The specified chemical analysis and the required mechanical properties are given in Tables 6-1 and 6-3, the minimum normalize and temper temperatures of Heat Treatment for Mechanical Properties being the same as for the forged parts.

In SA 387 of 2007 edition of ASME code, for grade 91, there are more severe limitation on Al content (0.02 max instead of 0.04 max) and new limitations on Ti and Zr. These new requirements were not present in 2004 ASME code: for them, we have no experience about neither the industrial feasibility nor the benefit in term of properties.

Table 6-3: Specified chemical compositions for grade 91 plates

	Heat analysis		
%	**SA 336 grade F91**	**SA 182 grade F91**	**SA 387 grade P91**
C	0.08 – 0.12	0.08 – 0.12	0.08 – 0.12
Mn	0.30 – 0.60	0.30 – 0.60	0.30 – 0.60
P	0.025 max (1)	0.020 max	0.020 max
S	0.025 max (2)	0.010 max	0.010 max
Si	0.20 – 0.50	0.20 – 0.50	0.20 – 0.50
Ni	0.40 max	0.40 max	0.40 max
Cr	8.0 – 9.5	8.0 – 9.5	8.0 – 9.5
Mo	0.85 – 1.05	0.85 – 1.05	0.85 – 1.05
V	0.18 – 0.25	0.18 – 0.25	0.18 – 0.25
Nb/Cb	0.06 – 0.10	0.06 – 0.10	0.06 – 0.10
N	0.03 – 0.07	0.03 – 0.07	0.030 – 0.070
Al	0.04 max	0.04 max	0.02 max
Ti			0.01 max
Zr			0.01 max
(1)	To be reduced to 0.020 or less in specification for RPV plates		
(2)	To be reduced to 0.010 or less in specification for RPV plates		

6.3.3 Filler metals for grade 91

For the range of thicknesses constituting the vessel system, the weldability of the grade 91 has to be demonstrated. For non nuclear applications, the modified grade 91 was welded by a number of manufacturers using the three classical electric arc process applied for nuclear applications: gas tungsten arc (GTAW), submerged arc (SAW) and shielded metal arc (SMAW) with manual electrodes.

During preliminary works for HTR, no difficulties coming from the base metal up to 200 mm thick were encountered during welding of test coupons in flat position. All the effort was directed to the selection and optimization of the filler metal for the following:

- Avoidance of hot cracking in weld metal when deposited with heat inputs appropriate to the welding process under industrial conditions
- Toughness of weld metals similar to base metal
- Creep strength of the welded joints (including the effect of the Heat Affected Zone).

The first point is met by using filler metals with severe limit in sulfur content ($S \leq 0.002\%$). The second and third point are difficult to meet together as the R&D for non nuclear application considers the third point as a priority in relation to the creep strength of the base material. In our opinion, the second point is as important as the creep strength for the acceptance of a material for a nuclear vessel in service at moderated high temperature.(425 – 475°C) as is the NGNP RPV in grade 91 steel.

R&D need to be carried on for improved specified chemical analysis for wire flux, tungsten inert gas wire and more specifically for coated electrode to be developed as none commercially available one was found fully satisfactory.

Welding qualification tests and subsequent characterization of welded joints properties need to be done.

The introduction of filler metals in the codes and standards (AWS/ASME) needs to be managed by comparing the optimized analyses with specifications (grade EB9 of SFA 5.23 for bare electrode for SAW, grade ER 905-B9 of SFA 5.28 for wire for gas shielded procedure including GTAW and grades E9015-B9, E9016-B9 or E9018-B9 of SFA 5.5 for covered electrode for SMAW).

The technological development for the welding operations of the NGNP RPV will then needs similar studies as in the case of SA 508 grade 3 base material.

6.3.4 Procurement of forged parts in SA 508 grade 3

6.3.4.1 ASME-ASTM initial standard

The starting requirements are those of ASME-ASTM SA 508 for grade 3 which covers forged parts without limitation of mass.

The specified chemical analyses are quoted in Table 6-4. Particular limitations are found for maximum Al, Ca B and Ti: These limitations as well as the requirements for P and S will be discussed in 6.3.4.2.

The required mechanical properties for SA 508 grade 3 class 1 are quoted in Table 6-5.

SA 508 grade 3 class1 is quenched by immersion or by spraying and tempered with the temperature requirement indicated in Table 6-5.

Table 6-4: Specified chemical compositions (SA 508 grade 3)

	Heat analysis		Product analysis
%	SA 508 Grade 3 Class 1	SA 533 type B Class 1	SA 508 Gade 3 + A 788
C	0.25 max	0.25 max	0.17 max
Mn	1.20 – 1.50	1.15 - 1.50	1.14 – 1.56
P	0.025 max	0.035 max	
S	0.025 max	0.035 max	
Si	0.40 max	0.15 – 0.40	0.45 max
Ni	0.40 – 1.00	0.40 – 0.70	0.37 - 1.03
Cr	0.25 max		0.28 max
Mo	0.45 – 0.60	0.45 – 0.60	0.42 – 0.63
V	0.05 max		0.06 max
Nb	0.01 max		
Cu	0.20 max		0.23 max
Ca	0.015 max		
Ti	0.015 max		
Al	0.025 max		0.035 max
B	0.003 max		

Table 6-5: Specified mechanical properties at room temperature and Heat Treatment for Mechanical Properties for SA 508 grade 3 class1

	SA 508 Grade 3 Class 1	SA 533 Type B Class 1
S_y	≥ 345 MPa (≥ 50 KSI)	≥ 345 MPa (≥ 50 KSI)
S_u	550 – 725 MPa (80 – 105 KSI)	550 – 690 MPa (80 – 100 KSI)
A %	≥ 18	≥ 18
Austenitizing before quenching		845 – 980°C (1550 – 1800°F)
Temper	≥ 635°C (1175°F)	≥ 595°C (1100°F)

6.3.4.2 Complementary requirements for chemical composition

6.3.4.2.1 Heat analysis

Some improvements in mechanical properties (RTNDT) and weldability can be obtained through more severe limitations of phosphorus and sulfur contents respectively down to 0.012% and 0.015% as suggested in supplementary requirements S9 of the SA 508 standard. Even more severe limitations down to 0.008 for both P and S can be industrially applied.

More severe limitations in Al content (0.025 max) and new requirements for Ca, B, and Ti were introduced in SA 508 grade 3 class 1 in 2007 ASME Code. These new requirements were not present in 2004 ASME code (limitation for All was 0.04 max). For these requirements, we have no experience about neither the industrial feasability nor the benefit in term of properties.

6.3.4.2.2 Product analysis

Product analysis shall be performed in each coupon for mechanical tests. The product analysis provision of Specification A 788 will be used. Application of A 788 to as low levels as specified for Nb, Ca and Ti on heat analysis seems to be inefficient.

6.3.4.3 Supplementary mechanical tests to be included in purchase specifications

6.3.4.3.1 Location of test specimens and tests after simulated post weld heat treatment

SA 508 standard requires two tensile tests in longitudinal direction. The tension tests shall be offset 180° from each other, except if the forging height exceeds 3.7 m. In such case, they shall be taken at each end of the forging. The practice for test specimens in heavy or complex pressure vessel components is to follow ASME III NB 2223:

- In the RPV beltline forgings, the specimen axis position should be t/4 from the heat treated surface, t being the wall thickness of the heat treated forging.
- In large and complex forgings (RPV nozzles, RPV flanges), specimen axis should be at least at 19 and 38 mm from the two nearest heat treated surfaces.

Sketches showing the exact locations shall be approved by the purchaser of the forged part.

Longitudinal direction means that the longitudinal axis of the specimen is parallel to the major working of the forging. The corresponding direction will be approved by the purchaser as part of the approval of the Technical Manufacturing Program. For RPV beltline forgings, the longitudinal direction is the circumferential direction following the past PWR experience.

During fabrication of the RPV, the forged parts will undergo some post weld heat treatments. To cover a possible effect of these heat treatments on the mechanical properties of the parts, the tensile tests shall be repeated on coupons which have undergone a simulated post weld heat treatment. This treatment is dependent of the fabrication operations and will be defined by the purchaser. As a first proposal the holding conditions will be 16 h at 595 – 620°C (1103 – 1148°F) ; the heating and cooling rates above 425°C (800°F) will be limited to 50°C/h (90°F/h). The specified properties shall be the same as for the tests after Heat Treatment for Mechanical Properties.

6.3.4.3.2 Tensile tests in transverse direction

As supplementary requirements, two transverse tension tests should also be performed with tangential test specimens from the other prolongation than for longitudinal tension tests. The purpose of these tests is to verify the properties in the direction corresponding to the hoop stress of the cylindrical shell and to check that the forging operation gives similar mechanical properties in transverse and longitudinal directions. The required values for transverse tests are indicated in SA 508: identical requirements for S_y, S_u and A% in both directions

6.3.4.3.3 Tensile tests at elevated temperature

Tensile tests at temperature close to the normal service temperature shall be performed in order to validate the tensile properties used in the design work. The specimens will be taken from the same coupons as for tests at room temperature. They will be in longitudinal or transverse directions whichever has shown the lower tensile strength at room temperature.

- Suggested test temperature : 350°C or 660°F
- Required tensile strength value : 497 MPa or 72KSI
- Required yield strength value : 300 MPa or 44 KSI
- Elongation : for information purpose.

The specified yield strength is in agreement with ASME subsection II D table Y-1 which indicate S_y = 41.5 KSI at 600°F.

The specified tensile strength value is 10% lower than that of ASME subsection II D table U which indicate S_u = 80 KSI at 800°F as well as at room temperature. According our statistical data, the S_u -10% level corresponds to average - 2 standard deviations. Such minimal values are probably acceptable for the supplier in opposition to the room temperature level of 80 KSI. Anyway, the agreement of the supplier is needed to put the acceptance level at S_u values. In addition, the test is performed after simulated PWHT which can reduce a little bit tensile strengths.

6.3.4.3.4 RT_{NDT} transition temperature

The determination of RT_{NDT} transition temperature can be useful for the assessment of the components against non ductile fracture. The procedure of determination is described in ASME III NB 2331. The determination of RT_{NDT} transition temperature will be performed on coupons which have undergone simulated post weld heat treatment.

The length of specimens for Pellini tests will be parallel to the axial direction. According our past experience, RT_{NDT} as low as – 30°C (-20°F) can be obtained in heavy section forged parts (200 mm) but this probably requires a limitation in sulfur content as low as ≤ 0.010% or perhaps ≤ 0.008%.

The Cv specimens shall have their axis in axial direction, the notch tip being parallel to the radial direction. In order to draw the Charpy V impact transition curve, 3 specimens shall be broken at each of the test temperatures. At least 6 sets of specimens should be tested at 6 temperatures with one set on the upper shelf (100% shear).

6.3.4.4 Technical manufacturing program and inspections

6.3.4.4.1 Technical manufacturing program

The Technical Manufacturing Program submitted to the approval of the purchaser shall include at least the following data:

- Steel making and ingot pouring process
- Ingot size weight and discards
- Sketch of the forging after each operation with achieved dimensions and with indication of the main working direction
- Heat treatment cycles applied on the forging with dimensions at these stages
- The locations of the thermocouples on the forging during heat treatment for mechanical properties
- Sketch showing test coupons on the forging
- Sketch showing test specimens on the tests coupons

6.3.4.4.2 Inspections

Magnetic particle examination will be performed after machining of the as forged parts and after Heat Treatment for Mechanical Properties at the Forging’s Supplier shop. Magnetic particle examination shall be performed in accordance with the paragraph NB 2545 of ASME code section III division 1

Ultrasonic examination will be performed after machining of the as forged parts and after Heat Treatment for Mechanical Properties at the Forging’s Supplier shop. Ultrasonic examination shall be performed in accordance with the paragraph NB 2542 of ASME code section III division 1. The supplier shall submit the UT procedure to the purchaser approval.

6.3.4.4.3 Defective area removal

The surface defects shall be removed in accordance with the paragraph NB 2538 of ASME code section III division 1. No repair by welding shall be allowed at the Forging’s Supplier shop.

6.3.4.4.4 Archive materials

The supplier shall deliver at the same time as the forging:

- One archive coupon for each coupons for delivery tests, the size of archive material being approximately the same as the test coupon.

- The circular prolongations of the test coupons not machined in test specimens.

Archive material shall have received the same heat treatment for mechanical properties as the forged part. They will be cut by mechanical means. They shall be marked in such a way that the main forging direction can be identified. They will be kept free from simulated post-weld heat treatment.

6.3.4.5 Key material properties for SA508 grade 3 class 1

The use of this grade for RPV, is subject to the assumption that the normal service temperature of the RPV does not exceed 371°C or 700°F (for example: a temperature of 350°C (660°F) which has been chosen for the tension tests in 3.3.3) and that the number and durations of hot transients follow the requirements of Code Case N-499-2. This implies that the mechanical properties of the forged parts are in accordance with the material data included in the Code Case.

6.3.4.6 Procurement of Mn-Mo-Ni low alloy plates

The standard SA 533 Type B covers plates of the same grade as SA 508 Grade 3. The specified chemical analysis is given in Table 6-4. SA 533 Type B Class 1 has the same level of mechanical properties as SA 508 grade 3 Class 1 (Table 6-5). There is no limitation in thickness for Class 1 and Class 2 plates. SA 533 Type B plates are quenched and tempered as SA 508 Grade 3 forged parts. Slightly lower tempering temperature is allowed. In Table 6-4, it can be seen that the purity requirements added in 2007 edition of the code for SA 508 grade 3 have not been extended to SA 533 Type B plates. In the case of mechanical properties, there is a difference between SA 533 Type B Class 1 plates and SA 508 Grade 3 Class 1 for the maximum limit on the tensile strength.

6.3.5 Filler metals for Mn-Mo-Ni steel

For welding this material, the welding parameters and weld consumable characteristics are well known due to PWR and other heavy wall pressure vessel experience.

No basic R&D for development of new filler metals is expected.

Nevertheless technological adjustment to the size and geometry of NGNP RPV needs to be studied. The first welding process for PWR RPV is automatic submerged arc which is applicable only in flat position. The fabrication of NGNP RPV will likely require welding in other position (circular welds in corner position for the shell, connection of cross vessels to the nozzles). In order to reach the goal of zero defects in welds, efficient automatic processes are highly desirable.

6.4 Ability to fabricate the components

This purpose of this section is to present the vessel manufacturing issues for the NGNP and to identify where additional R&D is required.

6.4.1 Size of the large forged pieces to procure

This section provides a preliminary estimate of the required size of the forgings. It is aimed at identifying the design options which seem the most relevant to minimize procurement risks and identify the effect of primary pressure on feasibility issues. Both mod 9Cr1Mo and SA 508 materials are considered.

6.4.1.1 Nozzle ring

Mod 9Cr1Mo steel

The following table provides the estimated required vessel's thickness in the core belt line region for mod 9Cr1Mo for a design temperature of 460°C and different values of primary pressure. It is however to be noted that the maximum temperatures during normal operating conditions are not expected to exceed 425°C.

Pressure (MPa)	Thickness
4	120mm
5	150 mm
6	180 mm

The thickness has to be reinforced due to the presence of nozzles (see figure 6-1) and the following table provides the corresponding thicknesses in the reinforcement area in the case of a 2 cross vessel design.

Pressure (MPa)	Vessel and nozzle reinforced thickness
4	190 mm
5	230 mm
6	270 mm

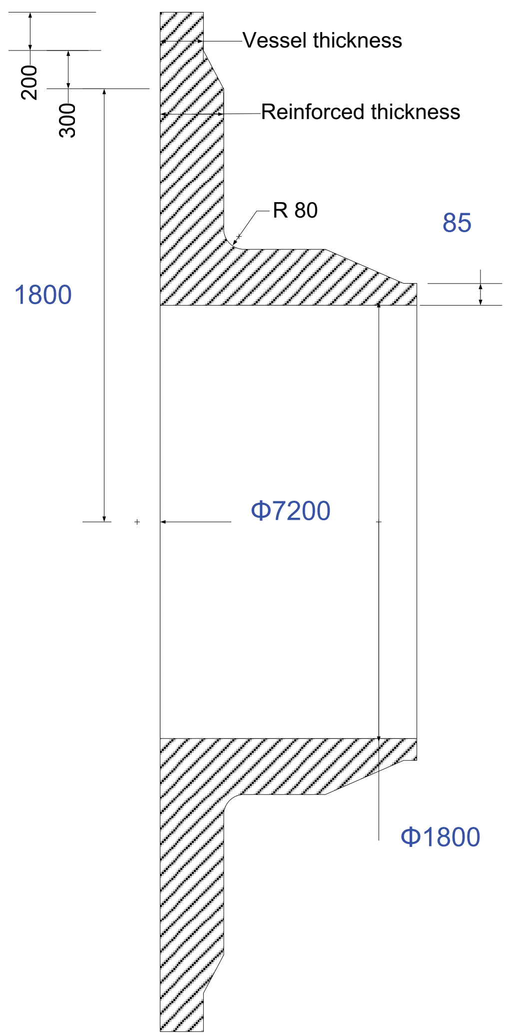


Figure 6-1: Nozzle ring

Based on this preliminary design, it is possible to estimate the mass of the nozzle ring (assumed in one piece) before final machining. Two options are considered, namely set-in and set-on design (see figure 6-2)

Pressure (MPa)	Nozzles ring mass with set-in design	Nozzles ring mass with set-on design
4	160 tons	230 tons
5	200 tons	265 tons
6	230 tons	300 tons

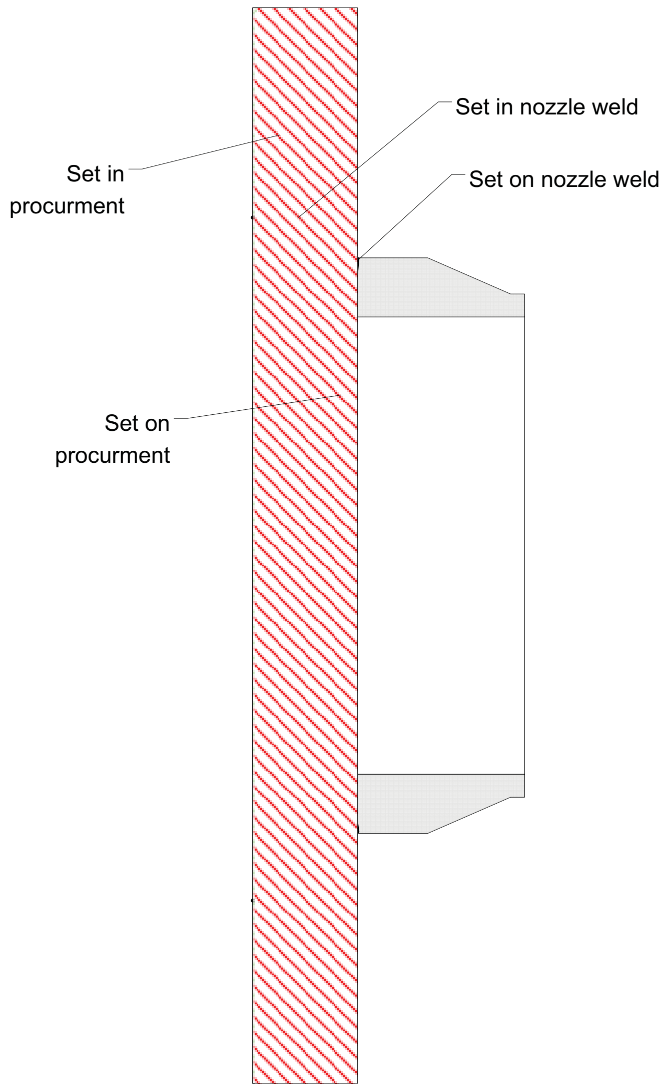


Figure 6-2: Nozzle ring procurement needs with set-in or set-on nozzle

Consequently, the ingot procurement masses for these rings are roughly estimated (assuming a factor 3 between the ingot mass and the machined product):

Pressure (MPa)	Ingot mass with set-in design	Ingot mass with set-on design
4	480 tons	690 tons
5	600 tons	795 tons
6	690 tons	900 tons

The set-in nozzle procurement could be expected similar to the sketch in figure 6-3. The procurement mass is estimated as follow:

Pressure (MPa)	2 cross vessels design
4	14 tons
5	16 tons
6	18 tons

Therefore, the set in nozzles ingots would have a mass between 15 and 20 tons depending on the design pressure.

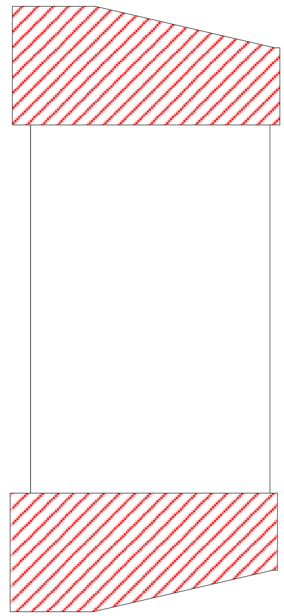


Figure 6-3: Set-in nozzle procurement need

SA 508 grade 3 class 1 material

The following table provides the estimated required vessel's thickness in the core belt line region for SA 508 grade 3 class 1 for a design temperature of 350°C and different values of primary pressure

Pressure (MPa)	Thickness
4	110 mm
5	135 mm
6	160 mm

The thickness has to be reinforced due to the presence of nozzles (see figure 6-1) and the following table provides the corresponding thicknesses in the reinforcement area in the case of a 2 cross vessel design.

Pressure (MPa)	Vessel and nozzle nominal thickness
4	175 mm
5	200 mm
6	250 mm

Based on this preliminary design, it is possible to estimate the mass of the nozzle ring (assumed in one piece) before final machining. Two options are considered, namely set-in and set-on design (see figure 6-2)

Pressure (MPa)	Nozzles ring mass with set-in design	Nozzles ring mass with set-on design
4	145 tons	215 tons
5	170 tons	240 tons
6	210 tons	280 tons

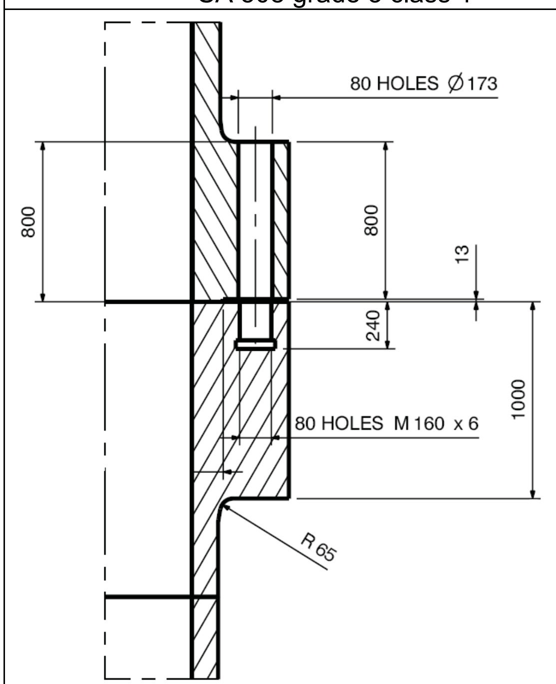
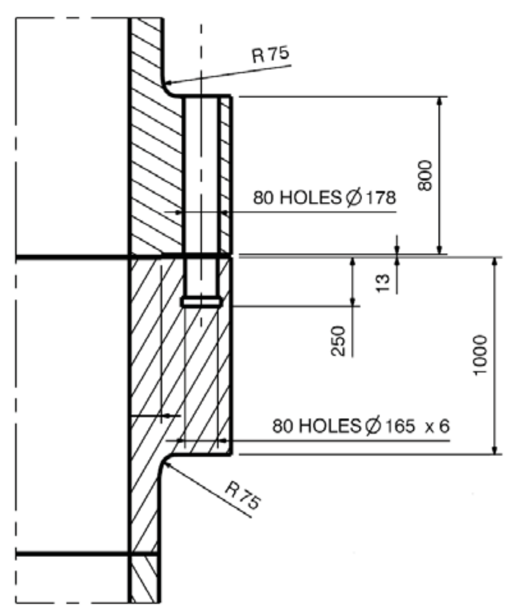
Consequently, the ingot procurement masses for these rings are roughly estimated (based again on an assumed factor 3 between the ingot mass and the machined product):

Pressure (MPa)	Ingot mass with set-in design	Ingot mass with set-on design
4	435 tons	645 tons
5	510 tons	720 tons
6	630 tons	840 tons

The set-in nozzles procurements are expected to be similar to those for mod 9Cr1Mo.

6.4.1.2 Flange procurement

The mass estimate is based on the following designs:

SA 508 grade 3 class 1	Grade 91
	

In both cases, the flange outer diameter is about 8200 mm and inner diameter is 7200mm.

Thus the mass of the shell used to machine the flange is about 130 tons and the ingot mass about 390 tons.

6.4.1.3 Procurements analysis

This section summarizes expected procurement issues depending on the different design options and based on preliminary investigations. Detailed procurement issues and supplier capabilities are discussed in section 6.5.

The following table summarizes mod 9Cr1Mo design feasibility for the nozzle ring.

Pressure (MPa)	Ingot mass with set-in design	Comments	Ingot mass with set-on design	Comments
4	480 tons	Out of supplier capacity	690 tons	Out of supplier capacity
5	600 tons	Out of supplier capacity	795 tons	Out of supplier capacity
6	690 tons	Out of supplier capacity	900 tons	Out of supplier capacity

It is therefore expected that the set-on design would not be feasible. The set-in design option may be envisioned with nozzle ring in two pieces but would require at least 300 ton ingot capability from the supplier. The ingot required for the RPV flange could be even larger than that.

The following table summarizes SA 508 design feasibility.

Pressure (MPa)	Ingot mass with set-in design	Comments	Ingot mass with set-on design	Comments
4	435 tons	Expected feasible	645 tons	Out of supplier capacity
5	510 tons	Expected feasible	720 tons	Out of supplier capacity
6	630 tons	Out of supplier capacity	840 tons	Out of supplier capacity

Again, the set-on design with nozzle ring in one piece is not feasible. The design should be based on the set-in option and it should be discussed with Japan Steel Works as to whether the nozzle ring in one piece could be envisioned.

As noted from the tables above, the pressure effect is significant. The selection of primary pressure is a trade-off between reactor vessel fabricability issues and circulator feasibility (and total required house load) and it is recommended for the time being to retain a value of 5 MPa.

6.4.1.4 Estimated Weight for NGNP Forgings

As a result of this analysis, forging breakdowns have been defined for the sake of discussions with forging suppliers. Figure 6-4 provides the reference breakdown proposed for discussions with Japan Steel Works. This figure is generic for SA-508 Grade 3 Class 1 and SA-336 F91 (mod 9Cr1Mo). The thickness dimensions are somewhat larger for the mod 9Cr1Mo material primarily due to the temperature assumptions for the design. The mod 9Cr1Mo material is also slightly lower strength even at the same temperature. As discussed previously, it has however to be noted that, for procurement of SA 508 from JSW, it could be envisioned to merge forgings 2 and 3 in one piece. Figure 6-5 provides another breakdown which would be less demanding in terms of ingot mass.

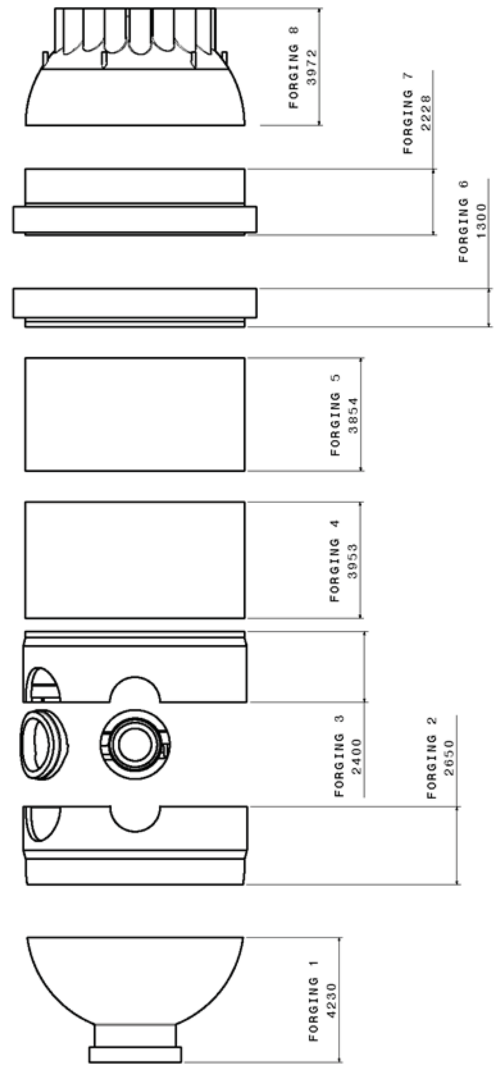


Figure 6-4: Reference Forging breakdown for NGNP Reactor Pressure Vessel

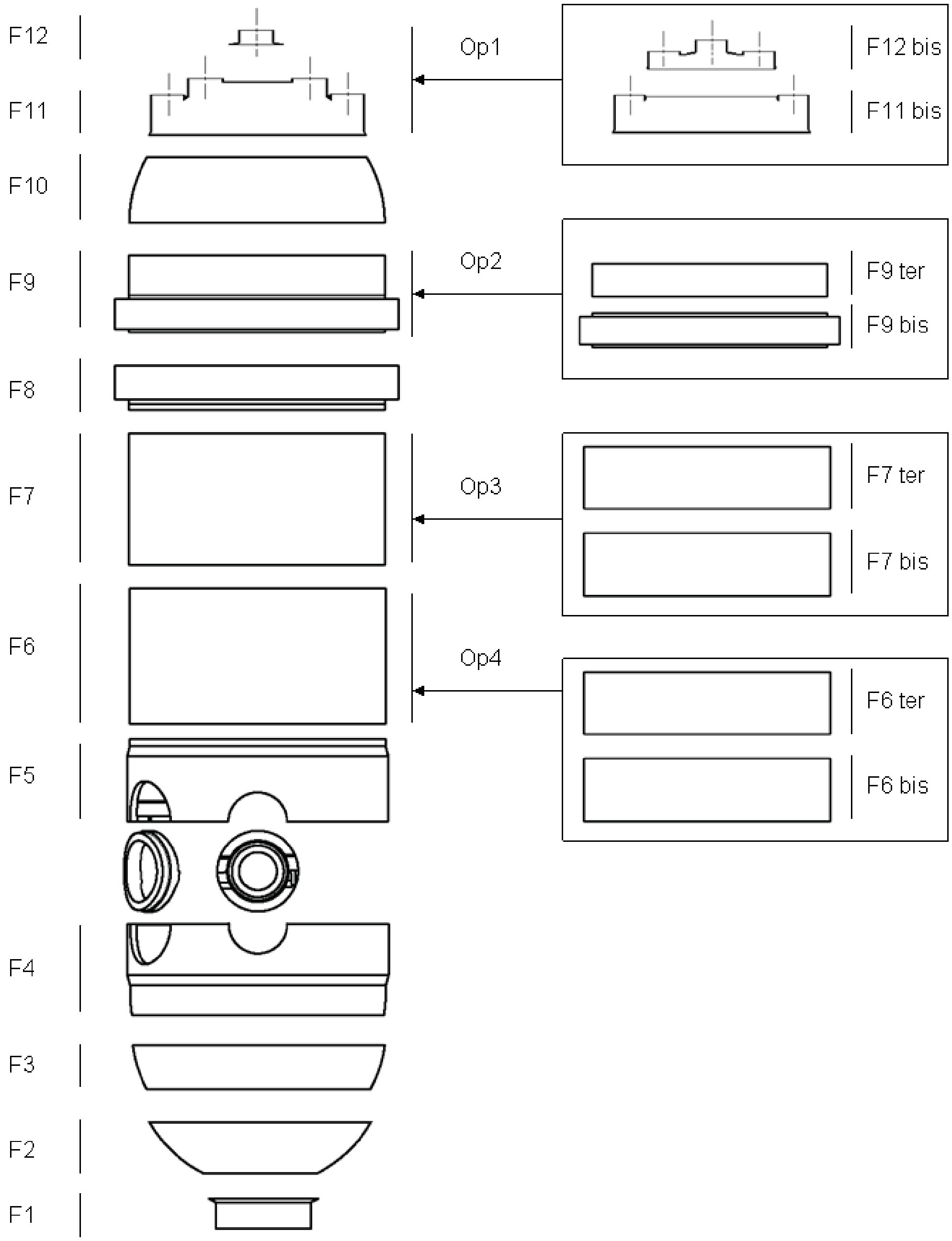


Figure 6-5: Alternative Forging breakdown for NGNP Reactor Pressure Vessel

Estimated weights have been prepared for discussion purposes with suppliers. Table 6-6 will identify finished weights associated with the reference breakdown and discuss alternatives.

The forgings will be ordered per ASME Section III NB or NH and will require impact testing. In addition the beltline region (forgings 4 and 5) will require extra material for regulatory agency required surveillance testing. The estimated excess material required for testing will be discussed later.

Table 6-6: Reference weights in metric ton (MT)of the reference breakdown

	SA-508	SA-336
Forging 1	99	104
Forging 2 (Note 1)	120	133
Forging 3 (Note 1)	109	120
Forging 4	97	106
Forging 5	94	103
Forging 6	102.3	106
Forging 7	111	116
Forging 8A (Note 2)	59	63
Forging 8B (Note 3)	237	233
Cross vessel nozzle	15.3	15.5

Note 1: Forgings 2 & 3 are full thickness for the length shown with no nozzle cut out.
Note 2: Forging 8 top line is for the spherical segment only.
Note 3: Forging 8 second line is for a solid block of height 1712 mm. The inside spherical radius has been deducted from the weight.

6.4.1.5 Allowance for impact and tensile specimens

Excess material is required on the forgings for mechanical testing. It is permitted to forge separate pieces but this is normally not done due to justification required that the separate piece has under gone the equivalent forging process. Assuming the test material will be integral with the actual forged piece it estimated that the forgings will be affected as follows. For forgings which are classified as thick and complex the minimum test piece is 155 mm by 55 mm by 1400mm.

Also SA-508 requires for forgings longer than 2032 mm that test pieces be taken from each end. Currently both nozzle belt sections (forgings 2 and 3) and the head flange (forging 7) are shown to be greater than 2032 mm. It is envisioned that forgings 2 and 3 can accommodate the test specimens by having a ID protrusion at one end and use the area for the crossover nozzle cutout at the opposite end.

The head flange forging can accommodate the test specimens by an ID protrusion and shortening the upper extension. In other words add a portion of the straight shell to the spherical portion of forging 8. The SA-336 specification does not currently have the length requirement for the testing at both ends but for the sake of this discussion the SA-336 forgings will be assumed to be modified the same as the SA-508 forgings. Table 6-7 will illustrate the modified configuration.

Forgings which are of essentially of a uniform thickness require ¼ T testing. In the case of the SA-508 forgings the test specimens must be ¼T from one quench surface and T from the second quench surface. For the case of the SA-336 normalized forgings the test specimens must be ¼ T by ¼ T from the heat treat surfaces. The cross section of the material for the test specimens is estimated to be 120 mm axial by 40 mm radial by 1400 mm circumferential. Therefore for the SA-508 the prolongation is a minimum 120 mm + T and for SA-336 the prolongation is 120 mm + ¼ T. This applies to forgings 1, 8A and cross vessel nozzle. The requirement would also apply to the forgings 4 and 5 but there is a more stringent requirement in the next paragraph.

The two main shell courses forgings 4 and 5 have an additional requirement for surveillance material for future testing to determine the affect of radiation. The required axial length for the surveillance material test block is 155 mm by T thick. Therefore for the main shell courses the prolongation is 155 mm + T and 155 + ¼ T for the SA-508 and SA-336 respectively. In addition samples will have to be taken from each end for the SA-508 and only one end for the SA-336.

The assumed mono block for forging 8 is not realistic. If this piece is to be made as a one piece forging there will have to be rough forged nozzles incorporated into the forging. The rough forged nozzle would also include the top nozzle which has a finished ID of 800 mm. The test specimens for an assumed thick and complex forging would need to be removed from at least two locations and preferably from two different axial locations. For the case of this discussion it is assumed the size of forging 8B in Table 6-6 would not be any bigger for test specimen allowance.

Table 6-7: Forging weights (MT) adjusted for test specimen allowance

	SA-508	SA-336
Forging 1	104	108
Forging 2 (Note 1)	126	137
Forging 3 (Note 1)	114	124
Forging 4	111	116
Forging 5	108	114
Forging 6	101	111
Forging 7 (Note 2)	94.2	98.5
Forging 8A (Note 2)	89	92.2
Forging 8B (Note 3)	237	233
Cross vessel nozzle	20.3	17.9

Note 1: Forgings 2 & 3 are full thickness for the length shown with no nozzle cut out.
Note 2: Forging 7 has been shortened to 1300 mm. The remaining 928 mm of straight shell has been added to the spherical segment (Forging 8A).
Note 3: Forging 8B is for a solid block of height 1712 mm. The inside spherical radius has been deducted from the weight.

6.4.2 Manufacturing

6.4.2.1 Welding process

A preliminary manufacturing scenario is necessary to separate workshop and site manufacturing operations. It must also evaluate welding processes applicable and determine the qualification and R&D needed in terms of welding processes and welding positions. This shall include also the furnace(s) to perform post weld heat treatments.

6.4.2.1.1 Mod 9Cr1Mo

The development of welding parameters and the qualification of welding procedures to the requirements of ASME Section IX need to be performed. The preheating temperature is a part of the welding process. The different AWS standards for the classification of welding consumable) indicate the following temperatures:

- 250°C ± 15 °C for SAW
- 150°C – 260 °C for GTAW
- 232°C – 288 °C for SMAW

Preheating in the 150 - 250°C range appears to be a common practice with inter-pass temperature around 200°C

A post heating is recommended after completion of the welds of the relatively thick joints (30 mm, 100 mm, 150 mm, 200 mm) before cooling down to the room temperature. The post heating is in the 300°C - 350°C temperature range with a minimum duration of 2 hours. It is also recommended to keep the welded parts in dry environment up to a post weld heat treatment.

Post-weld heat treatments have to be defined and introduced in the manufacturing scenario as intermediate and final post weld heat treatments.

The PWHT temperature is significantly higher than in the case of SA 508 grade 3 : for the fabrication of welded joints of thickness up to 200 mm, different temperatures have been chosen between 740°C and 760 C according to the target for mechanical properties, the range 750 – 760°C producing better toughness of the weld. As for the heat treatment for mechanical properties, a strict control of PWHT temperature in the whole weld is required.

Radiographic and ultrasonic non destructive examinations have to be adapted to mod 9Cr1Mo, but no particular features for NDT are expected for this material.

For mod 9 Cr1Mo, the recommended R&D actions and the associated schedule are given by the following.

SAW				
Action 1	Filler material development	1 year		
Action 2	Technological development and joints qualification		1 year	
Action 3	Full diameter ring welding demonstration			6 months

GTAW				
Action 1	Filler material development	1 year		
Action 2	Technological development and joints qualification		1 year	
Action 3	Full diameter ring welding demonstration			6 months

SMAW			
Action 1	Covered electrodes development	1 year	
Action 2	Deposit on mold and electrodes qualification		1 year

6.4.2.1.2 SA 508 grade 3 class 1

Welding processes for SA 508 grade 3 class 1 are well known thanks to the PWR experience feedback but they are to be adjusted to NGNP RPV size and geometry.

This experience includes intermediate and final post weld heat treatments (PWHT) at temperatures close to 620°C.

Automatic welding processes with increased productivity (such as MIG process) could be evaluated for the HTR vessel.

It is also recommended for the welding of SA 508 grade 3 to perform a demonstration test with full diameter rings. The purpose of this test is to check the post welding dimensional stability of the assembly (risk of out of roundness) and to develop clamping tools. 6 months are expected for such a test.

6.4.2.2 Nozzles welding

The welding solution for the nozzles (set-in or set-on) should be chosen considering:

- Procurements and manufacturing feasibility.

As pointed out in 6.4.1, the set on solution for nozzles requires heavier forged parts than the set-in solution. Possible manufacturers are less numerous and forging difficulties are growing as the parts are heavier. Due to the fact that the mass of parts evaluated in 6.4.1 are exceeding the manufacturer limits, the choice of set-in solution should be recommended.

Besides, based on the experience feedback of EPR, set-on nozzle orbital welding process need to be qualified but this kind of welding is not applicable to wire flux. In this case (as for EPR) is necessary to weld in flat position and to rotate the nozzle ring. Considering the size and the mass of the shell, this kind of operation is not recommended and is expected to be difficult to achieve.

- Codes and standards requirements.

From the view point of codes and standards requirements, both set in and set on solutions are acceptable provided that the welds can be fully inspected by radiography and by ultrasonic methods. The length of welds to be inspected is likely to be longer in the case of set-in nozzles. The access to the welds for in service inspections is perhaps easier in the case of set-on nozzles

6.4.2.3 Forming

It can be considered that rolled shells are available up to 100-150mm. Over this range of thickness, the plates should be curved by forging or obtained from a combination of forging and rolling.

Forming facilities and feasibility of the head and bottom sphere parts has to be evaluated regarding suppliers capabilities. For these particular components, the choice should be made between forged parts and heavy section rolled plates.

Curved plates could also be used for the vessel head and bottom manufacturing. But, for the cover head, the superior cap shall be performed without welds so as to avoid interferences with the control rods housing welds.

6.4.2.4 Machining

For machining, no particular features are expected for mod 9Cr1Mo at least after post weld heat treatment.

6.4.2.5 Cleaning, finishing of the surfaces.

No particular features are expected for NGNP RPV as compared with PWR RPV, except if emissivity measurements would show adverse effect of shot peening.

6.4.3 On site manufacturing issues

No particular issue is expected for the on site manufacturing expect the onsite welding between vessels and cross vessels.

It will be necessary to procure specials machines (machining centers) adapted to the size of the reactor vessel, that is to say with capacity to machine a diameter of about 8-9m, but this is not considered as an issue.

The connection between the vessels (RPV and IHX vessel) will be performed with the cross vessel. This connection will be done in situ, that is to say that the vessels are located in there respective pit.

For the welding of the cross vessel (for the two candidate materials), the two following options could be envisaged:

- If the preheating and post weld heat treatment are feasible in situ (need to be demonstrated considering the size and the geometry), the cross vessel and the vessels could be welded directly,

- If the heating in situ is not possible, it could be envisioned to overweld the weld preparations on the nozzles (buttering technique) with a high nickel filler metal and perform any required post weld heat treatment and weld joint preparation. The final welds would be made on the high nickel surface on the prepared weld joint with the high nickel weld wire. The feasibility of such a technique would need to be demonstrated.

6.4.4 Technological developments

R&D and tests have to be performed to evaluate the tightness level achievable for the sealing device of the closure head (metallic seal).

Even if this kind of seal is well know in PWR environment, the main differences for the HTR are:

- Diameter of the seal that is more important (about 7.5m diameter to compare with 4m diameter for PWR),
- The helium is more leaking that water (helium is used to test seals and welds).

6.5 Identification of potential suppliers

6.5.1 Forging providers

Large nuclear pressure vessels, such as the reactor vessel and heat exchangers are built utilizing ring and head forgings, or from welded plates that are formed into shell rings and heads. Forgings are preferred due less welding and subsequent non destructive examinations.

The NGNP gas-cooled reactor is much larger in diameter and height than conventional commercial light-water reactors (LWR). Their size presents a challenge to find, either in the U.S. or worldwide, forging vendors with the size (diameter and weight) and capacity to produce the required forgings.

This study is to provide a list of credible forging suppliers and some of their capabilities/attributes. Suppliers were gathered via internet searches, previous AREVA internal searches and from reports by the Nuclear Energy Institute and Idaho National Laboratory NGNP reports. Several U.S. and foreign forgers were contacted with respect to their capacities and potential for expanding their operations to provide heavier forged products.

Large forgings for the NGNP reactor vessel closure flange, head and nozzle belt are not available presently in the United States. LWR suppliers are obtaining these forgings from foreign suppliers for the US market. Japan Steel Works appears to be the most credible source for these components. Other potential sources are Doosan and Taewoong in South Korea and SFAR Steel (Creusot) in France. Forging suppliers in China and Russia were not researched due to potential quality issues.

There are a number of smaller forging suppliers in the U.S. that could potentially provide the larger forging if new equipment is installed. They can provide forgings for the nozzles. Several examples are Lehigh Heavy Forge, Ladish, Scot Forge, and in Europe are Bruck, Saarschmeide, Fomas and ZKM. The accompanying table lists a number of forgers and some of their capabilities.

Table 6-8 provides a summary of forging facilities. Sections 6.5.1.1, 6.5.1.2 and 6.5.13 provide more detailed information based on meetings or discussions with Japan Steel Works, SFARSTEEL and Dembiermont.

Table 6-8: Forging facilities

Manufacturer	***Press Capacity, Tons***	***Ingot Capacity, Pounds***	***Forging Capacity, Inches***	***Plate Forging Capacity, Inches***	***Materials***	***Comments/Remarks & etc.***
Ladish Co., Inc. Cudahy, WI, USA	17,000	40,000 to 60,000 Crane capacity to handle 230,000	300 to 330 OD	None	Carbon Steel, Steel Alloys, Titanium, Aluminum	Not in the nuclear market, mainly aerospace/jet engine parts. Would have to have a strong business case to do nuclear, but can supply components for commercial upgrading to NQA-1.
Lehigh Forge Bethlehem, PA, USA	10,000	Wt. Lbs. = 600,000 [272 metric ton] Dimensions = 130 in [3.3 m] Diameter	Shell, 75 OD x 46 ID x 206; could go to 200 OD	None	Ferrous and nonferrous materials; ingot lead times are >2 years + processing times; SST & alloy ingot availability is very limited.	Future Expansion Plans: Would consider expanding if the business demands. 10,000 ton hydraulic press 450,000 lbs shipping wt.s Boring/honing to 75 ft. x 30 in. deep. Isothermal press 12,500 tons. Forgings – 28 ft. Diameter, 10 ft. in face height and more than 2 tons: up to 56,000 Lbs.
Scot Forge Spring Grove, IL, USA	5,500 ton press is largest press, 7 other presses, 4 hammers all of various sizes from 4000 lb hammer to 4000 ton press	~80,000 [36 metric ton]	240 OD depending on other variables (id, wall thickness, length) wt. lbs = finished rough forging ~70,000 (depending on part configuration)	Can produce plate forging, but size dependent on shape	Numerous carbon, alloy, stainless, and non-ferrous alloys in inventory.	Future Expansion Plans: Have expanded in adding forging/machining/material handling/heat treatment capabilities and facilities. Current and future expansion continuing. Scot Forge is interested in supporting U.S commercial nuclear power industry if the future bears out. Forge furnace 1M lbs capacity. Quench Tanks to 42 ft. long or 20 ft. in diameter. 75 ton Ovrhd crane.

Manufacturer	***Press Capacity, Tons***	***Ingot Capacity, Pounds***	***Forging Capacity, Inches***	***Plate Forging Capacity, Inches***	***Materials***	***Comments/Remarks & etc.***
Japan Steel works, LTD, Muroran, Japan	14,000	1,000,000 [500 metric ton]	8565 mm OD x 7360 mm ID x 1075 mm high		Carbon Steel and Steel Alloys	100 ton electro-slag remelting furnace. 120 ton basic elec. arc furnace. 300 ton deep boring mach. 250 ton hi-spd trepan mach. 350 ton vert. lathe. 12,000 ton pipe-forming press. Serves nuclear ind.
Doosan Heavy Industries & Construction Seoul, S.Korea	13,000					100 ton elect. furnace. 155 ton Vac. Refin. Furnace. 450 ton heat furnace. 300 ton HT furnace. Horz Bor Mach 8000 mm L x 4000 mm H.
Taewoog Co. LTD Busan, S. Korea	8,000					125 MT manipulator. Ring Rolling Mill – 9000 mm OD x 2800 mm H x 60 ton max. HT furnaces. Quench Tanks.
SFAR Steel Creusot, France (acquired by AREVA)	11,300	550,000 [250 metric ton]	Flanges – 6.5 m OD Shells – 6.9 m OD Discs – 6.5 m OD			Boring Mach. 400 ton. Heat and HT furnaces. Svcs nuclear ind.
Bruck GmbH Saarbruken/Ensheim, Germany						
Fomas Italy						
ZKM Forging Stalowa, Poland (acquired by Ladish)						

Manufacturer	***Press Capacity, Tons***	***Ingot Capacity, Pounds***	***Forging Capacity, Inches***	***Plate Forging Capacity, Inches***	***Materials***	***Comments/Remarks & etc.***
OMZ (UralmashIzhora Group) Poland						
Smaller Domestic & Foreign Forgers						
Ellwood City Forge Ellwood City, PA	4,500	80,000	50 inches Dia and up to 50 foot lengths		CS, SST, Steel Alloys	Builds for the nuclear navy and power industries; Max length 600 in ; discs diameter 110 in; saddle rings OD 87 in
Jorgensen Forge Corp Seattle, WA	5,000	280,000			Steel Alloys, Ti	Open die forging on 660T, 1250T, 2500T and 5000T presses; 2400 ton ring expander capable of stretching rings to a maximum 225 inch diameter. max length 900 in ; discs diameter 95 in; saddle rings OD 150 in
Patriot Forge Branford, ONT. Canada						Max length 480 in ; discs max diameter 84 in; saddle max OD 84
Kropp Forge Cicero, IL	2,000	20,000			Steel Alloys, Ti	Hydraulic Forging Presses 750 tons, 1000 tons, 1500 tons and 2000 tons; Trim presses up to 1750 tons. Shafts max length 300 in. Shafts (high temp alloys- max length 240 in; titanium 240 in)
Canada Forgings, Inc Welland, ONT. Canada		40,000				Shaft max length 480; disc max dia 76 in; saddle rings max OD 80in
Sorel Forge Company Sorel,Quebec. Canada (Acquired by A. Finkl Forge)	5,000	84,000	63 inches Dia.	SST, Steel Alloys		5000-ton open-die hydraulic press equipped with its wide dies; produce forgings up to 50 feet long; 2000 ton press offers more flexibility. This system runs computer assisted, fully synchronized with the 40 meter-ton

Manufacturer	***Press Capacity, Tons***	***Ingot Capacity, Pounds***	***Forging Capacity, Inches***	***Plate Forging Capacity, Inches***	***Materials***	***Comments/Remarks & etc.***
						rail-bound manipulator. Stainless max shaft length - 160 in; disc max diameter 70; saddle rings max OD 70 in. Carbon Max. Shafts length - 460 in, Discs max dia - 100 in , Saddle rings max OD 115.
Dayton Forging & Heat Treating Dayton, OH		18,000		SST, Steel Alloys		Stainless max shaft length 240. Shaft max length 300 in. Disc max dia 60 in. Saddle rings max OD 60 in
A. Finkl and Sons Chicago, IL (Acquired by SCHMOLZ + BICKENBACH AG)		125,000				Max length 600 in. Discs max diameter 120 in. Saddle max OD 120 in
Forge Products Corp Cleveland, OH						
Nova Forge Trenton, NS, Canada		100,000				Carbon Max Shafts length 600 in. Discs max dia 132 in. Saddle rings Max OD 164 in.
Viking Forge Streetsboro, OH						
Wyman-Gordon Houston, TX Livingston, Scotland	 35,000 30,000				CS, steel alloys, SST, Duplex, Ni-based alloys, Ti, pwdr metals	Presses (vertical extrusion process) used to produce pipe: 9” ID to 42” ID with wall thkns 0.625” through 7.00” Services the industries of aerospace, power generation, process, oil, gas, marine and nuclear

Manufacturer	***Press Capacity, Tons***	***Ingot Capacity, Pounds***	***Forging Capacity, Inches***	***Plate Forging Capacity, Inches***	***Materials***	***Comments/Remarks & etc.***
Liberty Forge Liberty, TX						2 - 400 kW American Induction Heating Billet Furnaces 1 - 1250kW American Induction Heating Billet Furnace 1 - 2000 kW AEG Elotherm Billet Furnace 1 - 1300 Ton Ajax Forging Press 1 - 1300 Ton National Forging Press 1 - 2500 Ton National Forging Press 1 - 4000 Ton Erie Forging Press 2 - 1600 Ton National Forging Presses (Not in Production, 1 Currently Being Rebuilt) 2 - 125 Ton Minister Trim Presses 1 - 200 Ton Minister Trim Press 1 - 350 Ton Massey Trim Press 1 - 600 Ton Erie Trim Press

6.5.1.1 Japan Steel Works

Japan Steel Works is the world leader in terms of large and ultra large forgings. Compared to its competitors which are limited to forgings between press columns, JSW can carry on the forging operation in vertical axis and is therefore capable of forging pieces with a bigger outer diameter (see Figure 6-6 where the blue part of the graph is related to the between column forging and the yellow part is related to the vertical axis final forging).

A meeting was held with JSW on April 15, 2008 in order to discuss feasibility issues of procurement of forgings for the NGNP. The emphasis of the meeting was on both mod 9Cr1Mo and SA 508 options, but 2.25 Cr1Mo grades were also discussed. Discussions were supported by drawings and specifications prepared in the context of the current work.

The conclusions in terms of feasibility are the following:

- Mod 9 Cr1Mo:
    - JSW is limited to 100 ton ingot due to the need to use Electro-Slag Remelting (ESR) process to reduce segregations and carbides and nitrides precipitates. With conventional melting process, they would be limited to 60 ton ingot. Significant investment and a confirmation program would be required to extend this capability.
    - A second limitation is linked to quenching capability. For forgings with thickness lower than 240 mm, a normalization could be performed but oil quenching would be required beyond that limit.
    - Based on the above limitations, it has been identified that none of the forgings required for the NGNP could be provided by JSW. Even the small forgings required for the cross vessel nozzles would require oil quenching but the size of the nozzles would not be consistent with their quench tanks. Alternatively, it may be envisioned to perform a double normalization first on the rough forging and second after final machining. Specific investigations should be carried out within JSW to confirm that required mechanical properties would be met with such a double heat treatment. The procurement of the flanges for the RPV and the IHX vessel would still remain an issue.

      (The ability of other forging suppliers to provide mod 9Cr1Mo nozzle forgings has not been assessed)
- SA 508 grade 3 class 1:
    - JSW's current capability is up to 600 ton ingot for this material. JSW could provide forgings up to 10 meters diameter but are limited by the size of the furnace for heat treatment which can handle pieces no more than 8.5 m in diameter and 4.3 in height.
    - Most of the forgings required for the NGNP could be provided by JSW. The only part which is considered as a potential problem is the top dome of the upper head if penetrations would have to be integral to the head. It has been assessed that an ingot greater that 600 ton would be required in that case and it should be therefore envisioned to design CRDM penetrations with set-in nozzles to reduce the required ingot mass (or alternatively use two forged rings if this solution was shown to be feasible).

  - The main issue with SA 508 would be the schedule. JSW is currently full booked up to 2011 and all slots for the period 2011 to 2015 are currently under negotiation. It means that except if AREVA would be willing to change the priorities of its slots to accommodate NGNP needs, forgings would not be available before 2015 which means start-up around 2020 instead of current target of 2018. It is therefore considered that, for start-up by 2018, a design based on a combination of plates and limited forgings is required.

  - A design purely based on forgings can be envisioned for start-up by 2021 subjected that necessary actions are carried out on a timely basis to secure slots for the NGNP.

- 2.25 Cr1Mo grades

  - JSW has experience with grade 22 up to 350 ton ingots and grade 22V up to 250 ton ingots. Water quenching is currently used for petrochemical applications and there would be no problem of availability of quenching tank.

  - JSW provided the forgings for the HTTR and have therefore an experience of providing grade 22 for a nuclear application. JSW have assessed the feasibility of providing grade 22 forgings for the NGNP. Their assessment is that the closure head flange of the RPV and the cross vessel nozzles could be provided. The lower flange could not be provided due to ingot limitation and this flange would need therefore to be redesigned. It is however to be mentioned that, according to JSW experience, pro-eutectoid ferrite will appear at mid-thickness of the closure head flange and this would likely reduce mechanical properties. Another point raised by JSW concerns the requirements for drop weight tests. TNDT of -20°C was obtained for HTTR but with smaller forgings. JSW could have -20°C as a target but could commit to -5 or -10°C only.

  - It is expected that some other forgings could also be provided (for instance those of the core beltline) but this would require further detailed discussions with JSW. The availability of slots before 2015 would still be an issue.

Based on discussions with JSW, it appears clearly that SA 508 is JSW's preferred candidate from a technical and business standpoint. 2.25 Cr1Mo (grade 22) may be envisioned but there are still some technical issues to resolve for the heavy section parts. Mod 9Cr1Mo can not be supplied by JSW in the dimensions required for the NGNP and JSW do not see a strong demand to push them to upgrade their facilities. It is also to be mentioned that JSW's attention is fully consumed by the nuclear renaissance and they have therefore limited time to spend on material development.

6.5.1.2 SFARSTEEL

SFARSTEEL (Creusot Forge) ingot capacity is limited to 250 metric tons and up to 6.9 m outer diameter for shells (6.5 m outer diameter for flanges and disks) (see figure 6-6 for comparison with JSW capability).

The melting capability of SFARSTEEL (supplied by Industeel) concerning SA 508 material would be sufficient for the forgings of Figure 6-5 (except for F9) but they would be limited by their dimensional capability (see Figure 6-8).

SFARSTEEL have no experience of forging mod 9Cr1Mo but have experience with 2.25 Cr1Mo grade 22.

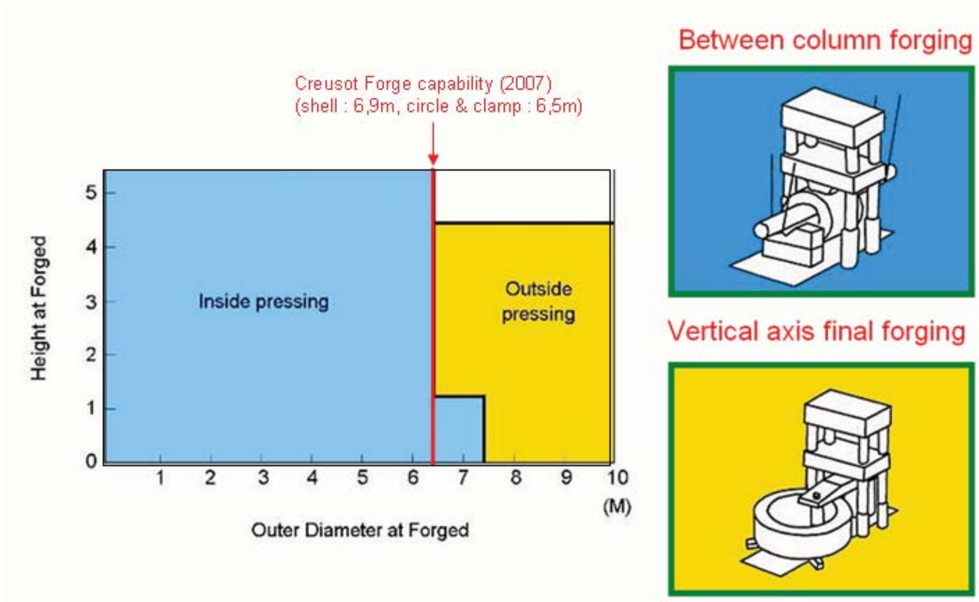
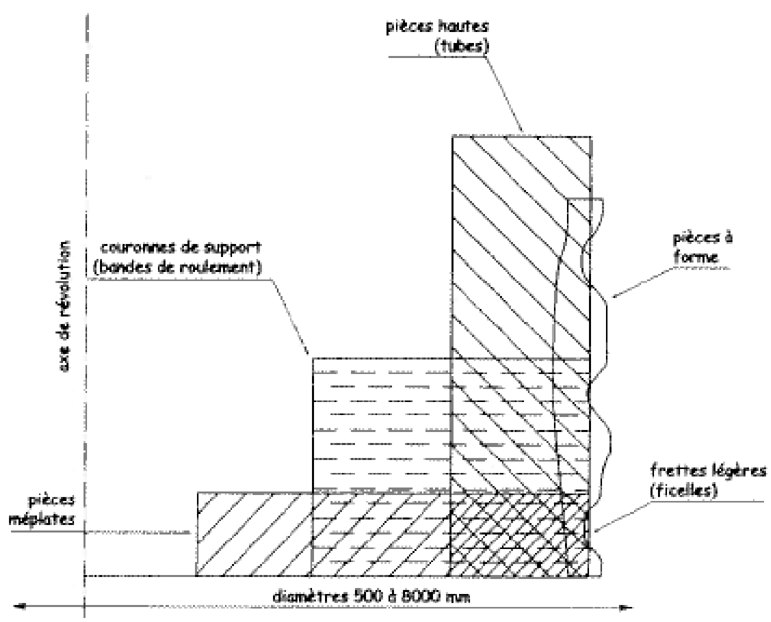


Figure 6-6: Forging capability of SFARSTEEL in comparison with JSW

6.5.1.3 Dembiermont

Figure 6-7 shows the range of dimensions of parts manufactured by Dembiermont specialized in rolled rings (FORGITAL Group).

Dembiermont is not able to manufacture the vessel parts of NGNP because of its limitations in maximal ingot weight (40t) and in dimensional capability (Figure 6-8).



Diameter	0.5 to 8.0 m
Height	1.7 m max
weight	0.1 to 40 t

Figure 6-7: Rolled rings capability of Dembiermont

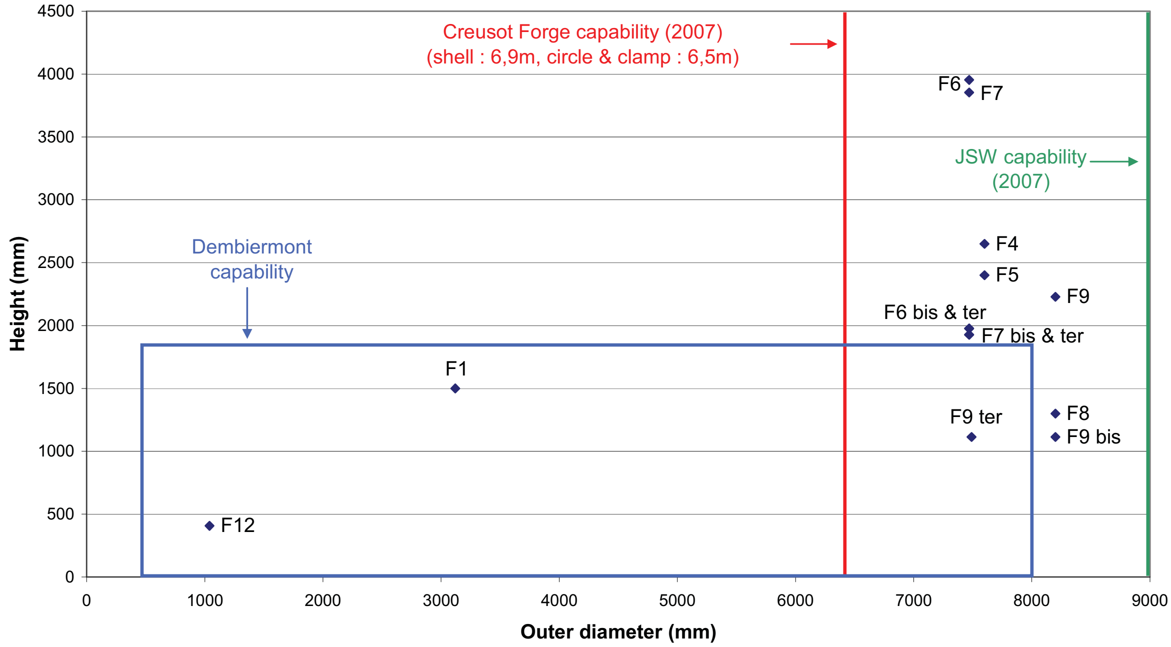


Figure 6-8: Distribution of NGNP Forgings (based on figure 6-5) in comparison with the capability of Dembiermont, Creusot Forge and JSW

6.5.2 Plate manufacturers

Industeel (subsidiary of ArcelorMittal Group) provide plates for the nuclear and non-nuclear industry. Their facilities are located in France and Belgium, with extra-heavy plates fabricated in St Chamond (France).

Industeel could provide plates for both mod 9 Cr1Mo and SA 533 material. It is however to be noticed that the commercial grade currently available for the latter is Class 2 instead of Type B Class 1. Grade SA 533 is commonly used for pressure vessels for processing equipments (separators, scrubbers etc) in the boiler and gas industry. Industeel have no experience of fabricating plates for the French nuclear industry as all French PWRs are made out of forgings. They are however currently fabricating plates for the PBMR. Industeel's current capability enables them fabricating plates up to a thickness of 400 mm.

As far as mod 9Cr1Mo is concerned, Industeel's commercial data sheet of SA387 grade 91 gives 90 mm as maximal thickness. Several tests made by the CEA on a 140 mm SA387 plate supplied by Industeel show a good homogeneity of microstructure at acceptance state and good fracture appearance transition temperatures. Characterizations show some segregation zones (Cr, Ni, Mo, Si, Nb etc.) located at the middle thickness of the plate but the chemical composition is in the normal range of variation. Industeel have also manufactured a 200 mm thick plate out of mod 9Cr1Mo.

It is also to be noted that Industeel is providing mod 9Cr1Mo plates for the Steam Generators of the Prototype Fast Breeder Reactor (PFBR) in India.

Table 6-9 gives an idea of Industeel capability concerning plate manufacturing of SA 533 and mod 9Cr1Mo.

For 2.25 Cr material, Industeel could provide plates out of grade 22 or grade 22V. They would be limited for both grades to 300 mm thick products. Beyond this limit, it seems that there is quench problems resulting in a reduction of mechanical properties.

For all materials, curved plates (either for shells or heads) could be provided up to 4300 mm width, based on a combination of rolling and pre-forging. The length would be a function of the thickness and taking into account a maximum ingot of 90 tons. Lead time would be in the order of 14 to 16 months for curved plates

Thick pressure vessel plate material is a scarce commodity with very limited availability in the U.S. Currently, only one supplier, ArcelorMittal at Burns Harbor located in northwestern Indiana near Chicago, Illinois, produces plate in thickness' of 5 inches [130 mm] to 9 inches [230 mm]. Allocation of plate from this mill would be an issue at this time. Nucor Steel Tuscaloosa, Inc. provides pressure vessel forge plate material, but not at the thickness proposed for the NGNP vessels. Steel Service Centers in the U.S. are currently procuring some plate for Asia and Europe and commercially upgrading or dedicating the material to ASTM and ASME material standards.

Table 6-9: Mechanical properties of plates grades

Designation (according to ASME)	Chemical analysis (weight %)		Thickness (mm)	Rp0,2 (MPa)	Rm (MPa)	A%	Kv (J)		
							-20°C	0°C	+20°C
SA 533C Cl. 2	C	≤0.130	t≤150	500	600–720	18	50 J guaranteed down to -50°C		
	Mn	≤1.600							
	Si	≤0.400							
	P	≤0.010	150<t≤250	480	580-720	18	50 J guaranteed down to -20°C		
	S	≤0.004							
	Ni	≤1.000							
	Mo	≤0.500							
SA 387 gr. 91 cl. 2	C	0.100	t<60	445	580-760	18	27	34	40
	S	0.002							
	P	0.018							
	Si	0.300							
	Mn	0.400							
	Cr	9.000	60<t<90	435	550-730	18			
	Mo	1.000							
	V	0.200							
	Nb	0.080							
	N	0.050							
	Al	0.020							

6.6 In-service inspection requirements

6.6.1 Introduction

This section is aimed at providing the general HTR strategy concerning the In Service Inspection, repair and maintenance. This study is based on previous experience on HTR projects. Emphasis is placed on key requirements from Safety analysis and investment protection. Identification of significant impact on ISI&R provisions is also part of the work.

In Service Inspection and Repair provisions for continuous monitoring, periodic examinations, contingency inspections, repair and maintenance must be included in the design of the nuclear island at the earlier stage of the project, in order to meet the ISI&R requirements.

These ISI & R requirements and constraints for the HTR are established based on code and standards and accordingly to the stage of the project. The target for the capacity factor is fixed to 90% as an objective. The basic input data are the NGNP cycle, transients and associated prevention measure as regards accidental situations.

6.6.2 ISI&R codes and standard

The ISI&R proposals for the HTR has been preliminary issued based on the risk informed approach (reference 7) and are mainly elaborated on the basis of the ASME code Section XI division 2 rules for Inspection and Testing of Components of Gas Cooled Plant (IGA and IWA 9000 articles) (reference 8).

6.6.3 In service Inspection philosophy and objectives

The general approach for ISI & R is recalled in ASME code section XI division 2 edition 2007:

"Methods and actions required for assuring the structural and pressure – retaining integrity of safety related components" The corresponding rules are defined in the codes in articles IG (1992 edition and addenda 1993) for the Gas Cooled Plants.

This definition is close to the IAEA definition (reference 9):

″The In Service Inspection consists in examination during the reactor operating life to detect possible deterioration of systems and components and to determine if safe operating of the plant remains possible or if counter measures are necessary ″

These definitions give the objective for the surveillance activities during the operation of a nuclear plant that is by establishing a link between the safety functions and the surveillance levels of the systems and the components ensuring these functions (Safety Related Components).

On the bases of the seen-above principles the objective of the In Service Inspection Repair and Maintenance (ISI-R & M), may be divided into "Safety assessment objectives" and "Reliability assessment objectives":

Safety assessment objectives

- To verify that the operating conditions and the loading are in accordance with the design values and that there are not abnormal evolutions, or unpredicted phenomena.
- To confirm the availability of systems (or integrity of structures and components) to ensure safety functions and to detect troubles before they become significant as regard to the function, including

the assessment of the interval of time between the examinations (based on a risk informed analysis for example).

Reliability assessment objectives

- To organize the collected data background, to analyze and to use the examinations results, in order to optimize the maintenance and inspection programs.
- To protect plant investment,
- To ensure high plant availability.

General ISI&R requirements concern the scheduled outages; but also contingency interventions lead to conceive the Safety Related Elements with accesses capabilities and to provide examinations techniques adapted to the ambient conditions in such a way the ISI program can be implemented easily.

For contingency inspections and repair, the design has to meet reasonable timescale and delay for direct examinations inside the reactor vessel must be limited.

Definition of ISI provisions must be tightly linked to the possible failure mode and the ISI techniques and criteria have to be adapted to the surveyed phenomena. The criteria must be fixed in order to detect the possible trouble before the structures or system fail.

Continuous monitoring for Elements Important for Safety will aim at providing diversified and redundant systems.

6.6.4 General requirements for HTR

The ISI&R and maintenance main guide lines are the following:

6.6.4.1 Plant design requirements - Service Life

The power units shall be designed for an operating life of 60 calendar years from authorization to operate.

The plant shall be designed to permit replacement of life-limited, and/or failed components over its lifetime. The time required to effect such replacements shall be consistent with the requirements for the capacity factor (~ 90%).

The plant designer shall develop a design life classification system and listing which categorizes items (i.e., components and subsystems) according to design life capability and shall develop the strategy to be employed to support the overall plant design life requirement of 60 years. This design life classification shall be incorporated in the planning of the preventive maintenance and inspection programs.

The plant designer shall recommend, by the end of final design, a comprehensive program for obtaining data for evaluating the actual remaining life capability of long life components, based upon their actual operating history and measurement of their life limiting characteristics.

6.6.4.2 Safety and licensing

The plant shall be designed so that the aggregate occupational radiation dose respects American regulations, the criterion will be defined later.

6.6.4.3 Reliability and availability

The target for the design capacity factor for electrical generation averaged over the plant's lifetime is 90 %. This includes unplanned shutdown (frequency and average duration will be defined later).

To the extent possible, the design shall allow all planned inspection and maintenance activities that must be accomplished with the reactor shut down to be accomplished within the period required for refueling. Exceptions shall be justified by the designer.

6.6.4.4 Maintenance and ISI

6.6.4.4.1 Plant Maintenance requirements

The design shall include provisions for monitoring power unit and equipment status, configuration, and performance and for detecting and diagnosing malfunctions as a basis for predictive maintenance plans and decision making.

To facilitate the movement of personnel within the plant, buildings and equipment shall be arranged with as few vital security areas as possible, consistent with plant safety requirements.

Building design and equipment layout shall include features (e.g., cranes, hoists, monorails) to facilitate removal and replacement of major equipment items.

Headroom, pull space, lay down areas, and work space for component and equipment maintenance shall be provided. Permanently installed lifting devices and beams, rails, etc. for temporary attachment of lifting devices shall be included where required in the design. To facilitate access for in-place maintenance or component replacement, obstructing building structural members shall be removable without cutting. Consideration shall be given to space requirements (e.g., layout of aisles, sizing of doorways, elevator(s), etc.) for moving equipment and components from their permanent location to shop facilities.

Equipment and components shall be accessible from normally provided floors and platforms to the greatest degree practicable. Where components are not accessible from floors or platforms, special access, such as permanently installed ladders and local platforms, shall be provided.

The need for access to individual components during normal plant operation and under accident conditions shall be considered in developing building and equipment arrangements. An assessment of individual component accessibility for maintenance shall be developed and maintained for normal plant operation and accident conditions, based on plant physical or computer models.

Systems and components shall be designed to facilitate hands-on maintenance, subject to the requirements. Remote maintenance techniques shall be considered in the design where reduced radiation exposure, or improved capacity factor may be economically achieved.

The design of special maintenance tools shall be provided by the equipment vendor.

The design and arrangement of plant systems, equipment, and components shall facilitate on-line maintenance.

The Nuclear Heat Source unit shall be designed to allow all components within the reactor coolant pressure boundary to be removed and reinstalled to make possible inspection, repair and replacement. A trade study to determine the method of removal and replacement of components within the primary pressure boundary, based on the degree of difficulty, time and cost and the projected probability of occurrence shall be completed and documented by completion of preliminary design.

To reduce the cost of spare parts inventory, the design shall specify components in a manner that limits the number of different types, sizes, and ratings (e.g., temperature, pressure) of parts.

The design shall incorporate standard, proven, off-the-shelf components and materials from reliable suppliers when such components and materials meet applicable codes, requirements and specifications.

A preventive maintenance plan shall be developed and documented based on the final plant design. A first draft shall be issued on completion of preliminary design. The plan shall identify the preventive maintenance requirements, tasks, methods, tools, personnel skills, and estimated worker-hours (including those for health physics personnel) on a system basis for the categories of mechanical, electrical and control and instrumentation maintenance.

A spare parts list and recommended spare parts inventory shall be provided that is consistent with the preventive maintenance plan, anticipated unscheduled maintenance, and plant capacity factor requirements.

A plan for conducting planned outages shall be developed and maintained throughout the design process. The plan shall identify the work scope, duration of major activities, and the schedule critical path. It shall also include a description of the tasks, methods, tools, personnel skills, and an estimate of worker-hours to achieve estimated activity durations.

Anticipated tasks, methods, tools, personnel skills, and worker-hour requirements to accomplish unplanned maintenance shall be documented for each plant system. The analysis of maintenance activities shall be based on industrial experience (mean-time-between-failure and mean-time-to-repair data) for like type systems and components. Estimated worker-hours, including those of health physics personnel, shall include the time required to isolate systems and equipment, prepare for and conduct maintenance activities, and return to service.

Overhead cranes shall be designed to lift the heaviest equipment or component-part to be handled during planned operations, maintenance, and inspection activities.

Separate mechanical/machine, welding, electrical, instrument, and electronic maintenance shop facilities shall be provided.

The need for on-site mechanical maintenance/machine shop facilities suitable for handling radiologically contaminated or activated components shall be determined by the end of preliminary design. Such special process facilities should include decontamination areas with appropriate water supply, drainage and storage/purification facilities.

The plant design shall enhance maintainability by including human factors considerations. Lighting levels, heating, ventilation and air conditioning and plant services, such as communications and compressed air supply, shall be provided consistent with anticipated operation and maintenance activities and in compliance with applicable regulations, code, and general industrial practice.

The design of the plant shall consider both reduction and attenuation of noise sources to reduce noise exposure during operation and maintenance activities to levels consistent with regulation and standards.

6.6.4.4.2 In Service Inspection

The design shall provide access to the reactor coolant pressure boundary to permit in-service inspection as required.

Where cost effective, the design of systems and components shall incorporate those features required to implement on-line in-service inspection. If the unit or major component must be removed from service, design features shall be included to accomplish the inspection during the power unit allotted planned outage time.

Plant piping design shall minimize the need for snubbers and restraints and shall ensure inspectability.

Design documentation shall include plans and procedures for conducting in-service inspection and shall identify equipment necessary to conduct the inspection. The equipment vendor shall furnish the design of special ISI equipment not commercially available.

An in-service inspection program shall be developed and maintained throughout the design process. The program shall include anticipated durations and worker-hours, including health physics, for isolating the equipment/system, preparing for and performing the inspections, and returning the equipment/ system to service. Physical and/or computer models shall be used to assess inspectability.

6.6.5 ISI items

The above general objectives are achieved by performing a lot of individual activities in different areas so called ISI&R items: These activities are classed in relation with the different situations:

- Normal and power conditions
  (In service -continuous- monitoring, periodic test during operation),
- At cold shut down
  (periodic examination, periodic tests, surveillance, maintenance, repair),
- After component removal (periodic examination, surveillance, tests, maintenance, repairs).

In service monitoring covers all the operations by which the Operator is continuously ensured that component remains within the limits set by the technical documents (for functional aspects).

This monitoring incorporates the review of the design conditions, the main variables control and the functional tests.

The principles of this also called continuous monitoring are based on the following elements:

- *Checking of the design conditions*

This phase aims to check the assumptions made during the dimensioning studies. It takes place mainly during the commissioning tests.

The commissioning tests cover the most of the configurations which can occur in operation in order to get a maximum of confirmations. The analysis allows confirming the assumptions made concerning the thermal hydraulic behavior of the primary circuit and thus the thermal mechanical loading of the structures.

These acquisition and checking phase are taken into account in order to constitute the reference conditions (baseline) and the observations made are used to elaborate or to specify the parameters and the criteria to be monitored (for example vibration behavior of internals). Some measurements are made before the core loading with specific heaters for prototype.

- *Main parameters follow-up*

The data of the operating variables are processed for the main working parameters of the plant (i.e. helium, structures and water, temperatures, pressures, pumps speed) and the reproducing of the values recorded is verified periodically. This follow-up completed by some specific analysis procedures allows detecting anomalies, drifts or unpredicted changes.

– *In service monitoring*

As part of the operations of the plant and within respect of safety criteria, a lot of variables are continuously recorded, compared to specified thresholds and combined with automatic actuators.

The main elements are:

- the radioactivity measurements,
- the leakage's detection,
- the pressure and flow rate measurements (hydraulic characteristics in primary circuit),
- the in-core instrumentation

- *Transients book keeping*

It consists in checking that the operating parameters (evolution during transients) are within the limits of the corresponding design transients and to count these transients in relation with those of the reference file.

The transient book keeping is applied on the main loaded structures (fatigue damage for example on the reactor vessel loaded part, the core support plate and shell function…)

- *Periodic functional tests*

It is the set of tests aiming to verify the functional operability of equipment (e g mechanical effort measurements for valve closure, safety devices translation).

6.6.6 Safety related components

6.6.6.1 Reactivity control

The components and systems involved in the shutdown function are mainly:

- The Reactor Protection System,
- The control rods, mechanism, driving system, instrumentation and control of shutdown system.
- The Reserve Shutdown System,
- The vessel head adapter and reactor cover,
- The graphite fuel blocks, reflectors and core barrel,
- The core support, so the reactor vessel, the graphite core column.

6.6.6.2 Reactivity insertion: core support failure

- Reactivity insertion is theoretically possible through the core support failure. The main structures involve in this function are:

- The reactor vessel,
- The core support structure
- The graphite columns.

6.6.6.3 Core reactivity monitoring

The core reactivity monitoring which is made sure by:

- Nuclear measurement (reactivity meter for the fast changes),
- Continuous monitoring of the shutdown function availability by means of actuators tests (one out of three).

6.6.6.4 Core cooling and decay heat removal

In nominal operation, the core is cooled by helium circulation and the removed heat is provided to the Turbine and Steam Generator through the Intermediate Heat Exchangers located in the IHX pressure vessel(s). The Primary Heat Transfer System insures this function.

Core cooling in power condition

During nominal operation the primary circulator insures the core cooling (272kg/s). The main concern is the graphite fuel block plugging. Detection would be possible through the fission product release due to the overpass of the limit temperature (1600°C). This point is to be demonstrated as well as the temperature limitation through the helium circulation in the neighboring channels.

Helium circulation through the core is insured by:

- Circulator operation,
- Piping connection, Primary Hot Gas Duct (PHGD) integrity,
- Core support (graphite support block) integrity,
- Graphite fuel block,
- Intermediate Heat eXchangers (IHX hydraulic characteristics) integrity,
- Core barrel (cold flow monitoring) integrity.

6.6.6.5 Decay heat removal

The HTR concept is favorable as regards the decay heat removal. However, the system involved in the overall function must be surveyed and available as far as possible. These systems are:

The Shutdown Cooling System (SCS) used for decay heat removal in shutdown condition (shutdown transient and cold state) when the SDHRS is not available, The Secondary Decay Heat Removal System (SDHRS) installed on the secondary loops (heat exchangers between the gas mixing (he/N_2) and air,

The reactor Cavity Cooling System (RCCS) used in PCC and DCC situations,

Even if the extreme situations are satisfactory managed prevention against the initiators is required, as they lead to severe loading on structures important for reactivity control such as the core support (reactor vessel) and graphite block support.

6.6.7 Radioactive products containment

The concerned components are those making barrier against radioactive releases:

- The fuel envelope (first barrier)
- The primary containment
- The secondary containment

-First barrier

It is constituted by fuel envelope. The monitoring is made in operation from activity measurements in the primary circuit by the in helium detection system

-Second barrier (intermediate containment)

- Reactor Pressure Vessel + safety valves,
- Penetrations (adapter and components supported by the cover vessel)
- The Helium Service System (HSS),
- Secondary separation valves at IHX outlet (if applicable)

The integrity of the second barrier is permanently monitored in -operation from the activity measurements and gas analyses in:

- Helium detection around the primary circuit,
- In the secondary circuit.

This barrier is submitted to the rules concerning the nuclear equipment operating under pressure.

-Third barrier (secondary containment)

It is constituted by the reactor building (concrete containment), its penetrations, and airlocks. The main associated circuit is the ventilation allowing maintaining a monitored negative relative pressure in operation as regard to the external one.

-Secondary pressure barrier

This barrier consists in the helium heat transport loop between the IHX and the Steam Generator.

6.6.8 ISI Techniques

ASME section XI articles IGA 2200, defines the examination methods and criteria but some complement are specified for Pressurized Water Reactors in division 1 articles IWA 2200.

Main ISI examination methods are:

- Visual examination
- Surface examination
    - Liquid penetrant testing.
    - Magnetic particle testing.
- Radiographic examination of welds
- Ultrasonic volume examination
- Geometrical control

The following would be applicable under HTR conditions.

- Visual examination

Concerning the material, no significant differences are expected between mod 9Cr1Mo and SA 508 material and for the welds (multi-pass in narrow gap), because they are both ferritic steel of similar thickness and absorption, coefficient of US beam would be quite the same.

The main differences are the coupling implementation because it is not possible to fill-up the HTR reactor vessel with water (during inspection), and the absence of the austenitic layer in case of the HTR. Due to this, the defects generated by the stainless steel welding process are not to be considered and NDT examination of RV from outside would be preferred.

The internal structures must be conceived, as far as possible to allow the sensible zones to be controlled using Ultrasonic testing (NDT) from outside. The requirement for NDT inspection of the primary circuit from outside is an important issue and the conceptual design must take it into account.

- Radiographic examination

For the HTR, remote systems seem possible for film installation inside the reactor vessel, the source being installed outside. For the selected zones, the decay irradiation level, protection and access possibilities would be studied before giving definitive assessment for radiographic method applicability.

- US NDT inspection

Outside and inside visual examination are possible for the HTR. Intervention conditions (dust deposit) must be defined but helium transparency is a favorable aspect. Inside examination would be limited or would require core unloading, internals removal and remote techniques and intervention under inert gas would probably be required (irradiation level is to be determined). Direct local leakage controls during pneumatic test are possible using helium detection but remote technique may be also required. In case of inside examination, the removed internals structures could be examined in the dedicated storage zones with appropriated tools (irradiation level is also to be determined).

The IHX in situ inspection is not feasible for the compact option, but sampling examination is conceivable in case of tubes IHX using remote techniques. For the compact option indirect techniques (leakage detection) would be acceptable but the IHX replacement possibility will be probably required.

6.6.9 Inspection intervals

The planned program for ISI and system pressure tests is defined in IGA 2410 article of the ASME (program A) for progressive intervals and IGA 2420 (program B) for fixed (10 years) intervals.

Unforeseen inspection is not part of the ISI program but some specific studies will be achieved to determine the conditions and corresponding required means and delay of typical interventions.

6.6.10 Inspection of the primary boundary

6.6.10.1 Reactor Pressure Vessel

The periodic inspection will consist in 100% NDT (UT) of welding and attachments on the reactor vessel. This periodic examination will be performed from outside following inspection intervals (see section 6.6.9). The content of visit will be defined in relation with the periodic examination performed at each refueling stage to limit as far as possible the time outage.

Specific program will concern the Primary Hot Gas Duct and surveillance of the thermal barrier efficiency (thermal instrumentation). The efficiency of the reactor cooling circuit (thermal instrumentation) will be also surveyed (reactor vessel and cover).

Applicability of Leak Before Break (LBB) principle is to be validated particularly if helium detection is foreseen around the primary circuit as continuous monitoring measure. A preliminary study has been performed in that direction, but the approach is to be qualified for mod 9Cr1Mo and the critical size of the defect remains to be determined.

Pneumatic tests will be achieved using rules of ASME code which are the following:

ASME section XI div 2 - table IGB 5222-1 (gas cooled reactor HTR)	37.8°C	1,25 P_0 *
	93.3°C	1,20 P_0
	148.9°C	1,15 P_0
	204.4°C	1,10 P_0
	260°C	1,05 P_0

P_0 nominal pressure (100 % Pn)

In case of inspection performed from outside the NDT-US technique would be adapted to allow comparable performances for detection and characterization of the defects.

It is recommended to operate the reactor vessel in the negligible creep regime in order to avoid the implementation of a specific surveillance program covering the effect of creep. If the creep effects are in the residual range the acceptance criteria retained for PWR would be applicable for the HTR reactor vessel.

It must also be noticed that ageing effect would not have to be considered if the vessel temperature is lower than 480°C.

6.6.10.2 IHX pressure vessel

The IHX pressure vessel is part of the primary barrier and thus the ISI provisions will be the same as for the reactor vessel.

Note that this vessel is insulated and then the thermal insulation will have to be removed from the outer surface for the inspection.

6.6.10.3 Tubular IHX,

The tubular IHX is also part of the primary barrier. The ISI&R proposals for the tubular IHX consist mainly in:

- Volumetric (UT) and surface (MT/UT) examination of the pressure boundary significant welding,
- Volumetric (ET/UT) inspections for Tubing and plugs,
- Visual (VT) examination for internals and interior attachments,
- Surface (MT/PT) inspection for outside welded attachments.

This approach is based on the access possibility for NDT periodic inspection of the tubing.

6.6.11 Conclusion

Provisions for continuous monitoring, periodic examinations, contingency inspections, repair and maintenance must be included in the design of the nuclear island at the earlier stage of the project, in order to meet the ISI&R requirements.

The ISI & R requirements for the HTR are established based upon preliminary Safety analyses, taking into account the investment protection and based for ISI provision definition on ISI & R examination methods and intervals issued from the ASME code (section XI division 2) devoted to gas cooled reactors.

Specific features of the HTR plant for ISI&R provisions definition are the following:

- Significant pressure level (~5MPa)
- High temperature of core outlet and of core support and pressure vessel Pressurized gas as core coolant

The main differences from ISI&R point of view, compared to classical reactors (PWR or Sodium Fast Reactors) are as follows:

- The pneumatic test and associated leakage detection technique is to be studied more in details taking into account the personel protection,
- US NDT from outside is proposed for pressure vessel reactor welding and will not lead to difficulties,

- Inside inspection seems to be easily achievable using endoscope techniques but is limited to samples examination,
- Inside inspection using US NDT (or other US volume techniques) seems to be usable but must be validated taking into account the cleaning, coupling and accesses possibilities.
- Exceptional intervention for unpredicted event (ISI extension or contingency repair) is achievable but core unloading and radioactive conditions are to be specified.
- Specific issues associated to mod 9Cr1Mo would need to be addressed.

6.7 Conclusions on alternative materials

The main issue associated to material candidates is linked to procurement. Whatever material is selected, the design of the Reactor Pressure Vessel will have to be made out of plates to be consistent with 2018 schedule. The few remaining forgings could be provided by JSW in time for start-up by 2018, subjected that the corresponding forgings could be switched with slots currently under negotiation at the time of the present report. Otherwise, a two years delay for start-up should be anticipated. Other forging suppliers could also be envisioned but should be limited to small size forgings of the RPV (such as nozzles) or at most some of the larger forgings of the IHX vessel.

In terms of material candidates, SA 508 / SA 533 material is a viable option and no major technical issue has been identified. The design could be based on forgings only if the schedule permits, or a combination of plates and forgings could be envisioned.

No forging supplier has been identified for Mod 9Cr1Mo in the dimensions required for the NGNP. JSW is limited in terms of ingot size (100 ton) and quenching capability. Even the nozzle of the cross vessel could not be quenched and a double normalization may be envisioned but would require specific investigations within JSW before committing to specific requirements. In any case, this would not solve the problem of availability of the flanges of the IHX vessel. Further investigations are therefore required to identify if other suppliers could provide forgings made of mod 9Cr1Mo. In any case, the RPV would have to be designed without flange as no supplier but JSW is capable of providing such large forgings. Plates could be procured from Industeel up to 200 mm which would be sufficient for the core beltline, but further discussions would be required to understand if the nozzle ring could also be made out of plates.

An alternative to mod 9Cr1Mo material is 2.25Cr1Mo steel (grade 22). The drawback of this material is reduced tensile properties compared to other candidates which results in increased thickness by about 150%. Based on the current design, JSW could provide the closure head flange of the RPV but the lower flange would need to be redesigned. It is however to be noted that JSW is expecting the mechanical properties at mid-thickness to be reduced and the acceptability of such a reduction should be investigated. For forgings with smaller dimensions, other suppliers than JSW could be envisioned as 2.25Cr1Mo is a material commonly used in the petrochemical industry. Plates could be procured from Industeel up to 300 mm which would be sufficient for the core beltline but the availability of plates for the nozzle ring might be an issue due to expected reduced mechanical properties beyond 300 mm. Detailed discussions with Industeel are again required to understand if 2.25Cr1Mo is a viable candidate.

As a conclusion, SA 508 / SA 533 material is a good alternative if it can be envisioned to operate at lower core inlet temperature. Other alternatives to use this material under more severe operating conditions are discussed in section 7.

The study of the viability of the “hot” vessel option should require pursuing investigations as follow:

- Identify alternative forging suppliers for mod 9Cr1Mo
- Identify limitations for fabricating plates out of mod 9Cr1Mo
- Assess as to whether the expected reduction of mechanical properties of heavy section products made of 2.25Cr1Mo could be likely to rule out this candidate.

Alternatively, it may be envisioned to limit mod 9Cr1Mo to the RPV if the fabrication of the nozzle ring was shown to be easier than that with 2.25Cr1Mo and to design the IHX vessel with 2.25Cr1Mo to take benefit of increased number of forging suppliers for the size required for the IHX vessel.

7.0 USE OF SA 508/533 MATERIAL

7.1 Scope of work

This task element is aimed at identifying operating condition changes and/or design features that would be required to permit utilization of SA508/533 material for the vessels in the prismatic design reactor. This task covers the following:

- The maximum power level and temperatures that can be achieved using SA508/533 material
- Alternatives for cooling, thermal protection, or other design features for the RPV as an alternative to revising power level and temperature to permit use of SA508/533 material for the RPV.

It is intended in this context to identify and assess alternative concepts of the AREVA prismatic design based on SA 508 material (with and without active cooling). This will be based on system engineering design and the task will provide sketches or process flow diagrams when appropriate.

The evaluation of the maximum power level and required temperature for which SA 508 could be selected without using active cooling will be based on conduction cool-down calculations and will take account of uncertainties.

7.2 Introduction

The main objective of any design option that will permit use of SA-508/SA-533 steel is to keep the reactor vessel or intermediate heat exchanger vessel wall temperatures with an acceptable temperature range as permitted by the ASME Section III Code. SA508/SA533 steels are ASME Code approved for Class 1 nuclear components and Subsection NB rules are applicable up to 371°C for normal operation. Limited high temperature excursions under off-normal and conduction cool-down conditions are permitted under Code Case N 499-2.

Selection of a vessel material for a modular HTR must meet two temperature criteria. First the vessel temperature during normal operation must be acceptable for the material. In addition, the vessel temperature transient during conduction cooldown (and other transients) must be within the specified limits for the material for the class of event being considered. In the current design for the NGNP based on 500°C core inlet temperature, SA-508/SA-533 steel is unacceptable because the calculated temperatures during normal operation exceed 371°C. Conduction cooldown temperatures could also challenge the SA 501/SA 533 limits for the reactor sizes anticipated.

In order for SA-508/SA-533 steel to be used, a number of passive and active design change options can be pursued for lowering the steady-state operating temperature for these vessels. The successful option must be able to accomplish this under the following key constraints. First, the option must limit the maximum vessel wall to about 350°C or less in all places during normal operation with core inlet and outlet temperatures as high as 500°C and 950°C, respectively. This results in a minimum operating margin of 21°C. Second, the successful option must not adversely impact the ability to passively cool the core following the design basis accidents of pressurized conduction cooldown (PCC) and depressurized conduction cooldown (DCC). This means that the maximum fuel temperatures should not significantly exceed the 1600°C guideline. It also means that the vessel wall temperature excursion remains acceptable as defined by ASME Code Case 499-2; namely, the peak vessel temperature remains below the ASME code limit of 538°C for SA-533/SA-508 steel during transient and that the time at metal temperatures above 371°C remains below the code limits (3000 hours between 371 and 427°C and 1000 hours between 427 and 538°C, with no more than 3 events where the temperatures exceeds 427°C).

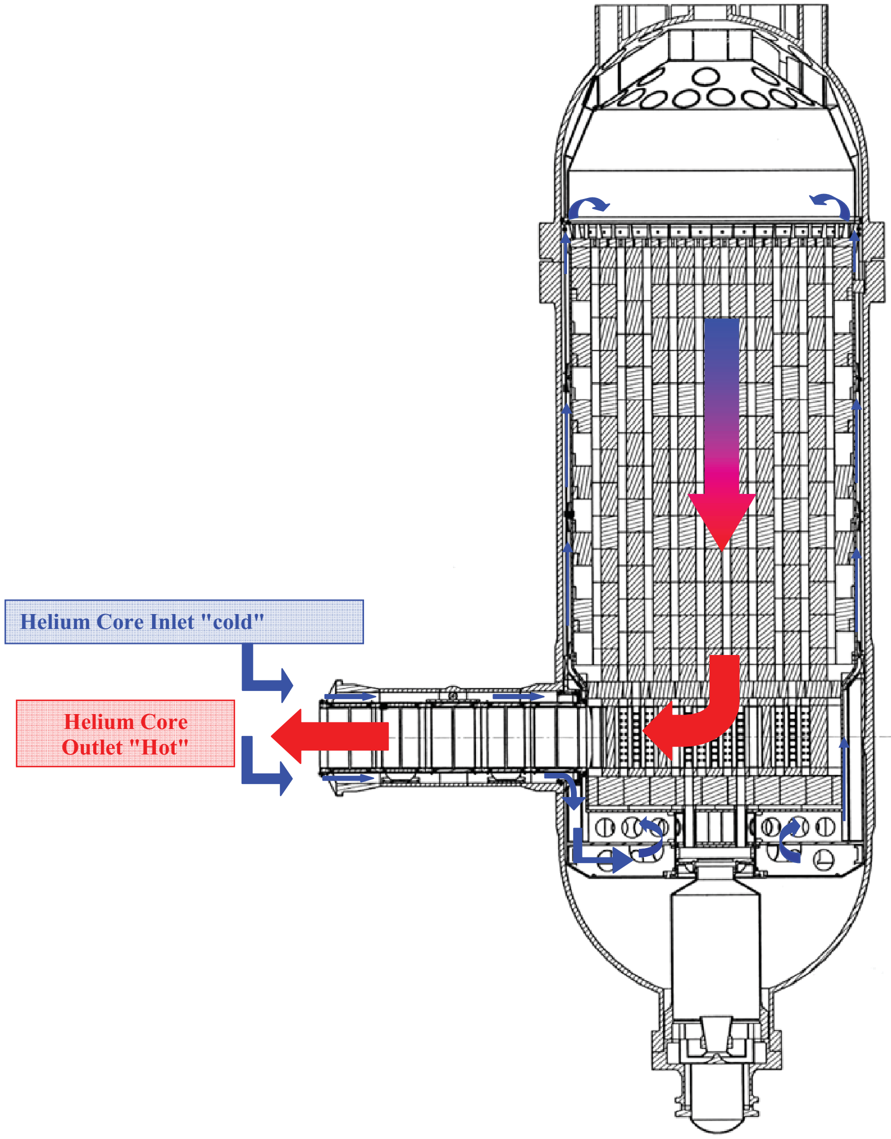
The evaluation of design options that might allow the use of SA-508/SA-533 is divided into two main parts. The first part identifies possible options and provides an initial qualitative evaluation in order to discern which options warrant more detailed analysis. Section 7.3 identifies design alternatives to address Reactor Vessel temperatures and Section 7.4 identifies design alternatives for the IHX vessel. The second part of the evaluation describes detailed analysis of the promising options. This detailed evaluation is provided in Section 7.5. Final conclusions are summarized in Section 7.6.

7.3 Identification of Options to Allow SA-508/SA533 Reactor Vessel

Figure 7-1 shows the major features of the reactor vessel-core internals assembly with the major flow paths highlighted. It is important to note that the inlet flow first enters the inlet plenum located below the core. From their, the flow is directed up along the annular space between the inner and outer core barrel. Then, it enters the area just above the core and then passes down through the core. In the current design, the reactor vessel wall temperature operates at approximately 460°C for an inlet temperature of 500°C (taking account of uncertainties), well above the 371°C limit. Hence, 9-Cr 1-Mo steel is required.

Six options, including both passive and active, are explored below as potential solutions for keeping the reactor pressure vessel temperatures within acceptable SA 508/533 limits. Which of these options will best accomplish the above objective depends on how well they work within the constraints applied, their feasibility, and their cost.

Figure 7-1: Reactor vessel



7.3.1 Passive Options for Reactor Vessel

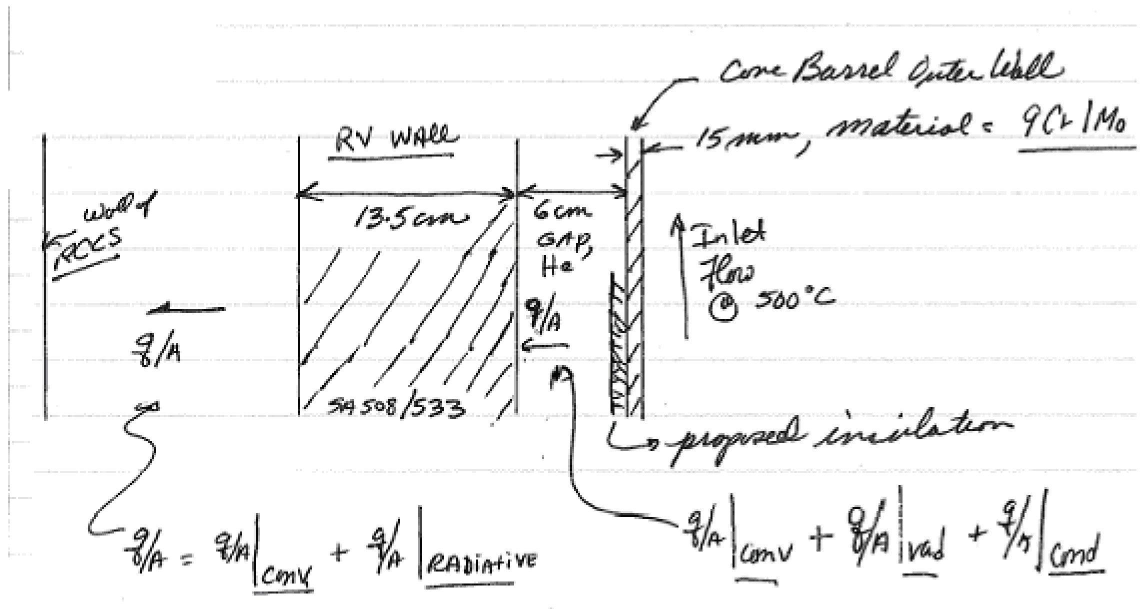
The reactor vessel and internal geometry is very conducive to a passive solution involving the application of a thin layer of insulation or a heat shield. A less attractive option involves relocating the main inlet flow path radially inward making it further away from the reactor vessel wall. These options are discussed below.

7.3.1.1 Insulate Outer Surface of the Core Barrel

7.3.1.1.1 Description

In this option, it is proposed that a layer of insulation be applied to the outer surface of the core barrel. A sketch of the applicable geometry is shown in Figure 7-2 below.

Figure 7-2: Core barrel outer wall insulation



A thin layer of calcium-silicate insulating material or other suitable insulating material is applied to the outer surface of the core barrel. Calcium-Silicate is chosen since it has an insulating value (i.e., thermal conductivity) of 0.1 W/M-K at 500°C and it has a maximum temperature capability of 1050°C. However, other materials may be suitable as well.

Without the insulation, the heat transfer in the gap is governed by a combination of radiation, convection, and to a lesser extent, conduction. Radiant heat transfer is the dominant mode of heat transfer. The balance of heat transferred is primarily driven by convection. Even though the gap region is designed to be stagnant, minor leakage flows are to be expected which will result in a convection heat transfer component. The conduction component of the heat transferred across the gap is expected to be relatively small.

7.3.1.1.2 Effectiveness

The addition of this insulating material will shield the reactor vessel wall in the core region from the high temperature of the inlet flow and reduce the radiative heat transfer component. This will reduce the operating temperature of the reactor vessel wall.

The steady-state temperatures in the other regions of the reactor vessel also need to be considered; namely, the upper and lower hemispherical segments and in the region of the cross-vessel piping. Detailed calculations are required to assess the effectiveness in these regions. However, it may be possible to consider insulating critical boundaries in these regions as well. For instance, the lower surface of the core inlet plenum and in the upper head region; however, insulation may not be as effective there because of convective flows due to bypass and parasitic leakage paths.

The vessel region in the vicinity of the cross-vessel will be susceptible to hot spots as well and these will have to be identified and managed.

7.3.1.1.3 Impact on steady state operations

Except for reducing the reactor wall temperature, the addition of insulation on the inside surface of the core barrel has no impact on steady state operation. The annular space between the inner reactor vessel wall and the outer core barrel wall is essentially a low flow or stagnant flow region. The insulation does not impact the inlet flow characteristics at all because of the way the inlet flow is routed through the double-walled core barrel.

The presence of this insulation favorably impacts plant parameters such as cycle efficiency due to reduced reactor vessel heat loss. Moreover, it does not impact core refueling. It may impede the ability to perform ISI inspections on the outer core barrel. However, the outer core barrel primarily functions as a flow channel boundary and has a limited structural role. (The outer core barrel is only 15 mm thick versus the more robust core barrel which is 50 mm thick.) The option does exist to perform this inspection from within the annular space.

7.3.1.1.4 Impact on PCC and DCC events

The presence of an insulating layer on the outer core barrel will serve to increase the maximum fuel temperatures resulting from PCC and DCC events. The thickness and type of insulation need to be optimized to ensure that fuel temperatures remain acceptable.

Conversely, the presence of the insulating layer on the outer core barrel will serve to decrease the maximum reactor vessel wall temperature during conduction cooldown over that experienced in the un-insulated case.

There is also a temporal impact due to the presence of an insulating layer on the outer core barrel. The heat transfer rate will be retarded sufficiently to potentially prolong the duration of core heat-up. It will take longer for the heat lost from the reactor vessel to match core decay heat; hence, the fuel temperature heatup and cooldown will be extended significantly. This may also impact the reactor vessel time at temperature limits. Peak vessel temperatures are expected to be reduced, but the duration of the temperature vessel transient might be extended.

In any case, sensitivity analyses will confirm the acceptability of this approach with respect to PCC and DCC events.

7.3.1.1.5 Feasibility Considerations

The addition of an insulating layer on the outer surface of the core barrel is very feasible. The core barrel is shop-fabricated in pieces and assembled on site. The insulation system can be designed to be installed along with shop fabricated pieces or installed on site. The key point is that the installation system for the core barrel can be installed outside of the reactor pressure vessel and attached to it circumferentially using established techniques for HTR thermal protection. Susceptibility to acoustic excitation would be considered in the design of the insulation system.

7.3.1.2 Insulate Reactor Vessel interior surface

7.3.1.2.1 Description

This option is similar to that for insulating the outer core barrel except that the insulation is located on the inner reactor vessel wall. At a minimum, the axial extent of the insulation must cover the reactor vessel inner wall from the top of the core to the bottom of the core. Conceptually, the remaining inner reactor vessel surfaces could be considered as well.

The heat transfer phenomena are essentially the same as described previously for the core barrel. The main difference is that the temperature of the outer core barrel will be approximately at the core inlet temperature and the gas in the annular space between the reactor vessel and the core barrel will be at slightly lower temperature.

7.3.1.2.2 Effectiveness

Similar to the core barrel option, the addition of insulating material will shield the reactor vessel wall in the core region from the high temperature of the inlet flow and reduce the operating temperature of the reactor vessel wall.

The steady-state temperatures in the other regions of the reactor vessel also need to be considered; namely, the upper and lower hemispherical segments and in the region of the cross-vessel piping. Insulation applied to these surfaces will result in lower vessel temperatures as well.

One advantage of insulating the reactor vessel inner surfaces is that any deleterious effects from the presence of hot helium gas due to bypass and parasitic leakage paths can be effectively eliminated.

As with the previous option, application of insulation in the vicinity of the cross-vessel will be difficult and this region could be susceptible to hot spots. These will have to be identified and managed.

7.3.1.2.3 Impact on steady state operations

Except for reducing the reactor wall temperature, the addition of insulation on the inside surface of the reactor vessel has no impact on steady state operation. The annular space between the inner reactor vessel wall and the outer core barrel wall is essentially a low flow or stagnant flow region. The insulation does not impact the inlet flow characteristics at all because of the way the inlet flow is routed through the double-walled core barrel.

The presence of this insulation favorably impacts plant parameters such as cycle efficiency due to reduced reactor vessel heat loss. Moreover, it does not impact core refueling.

One key drawback is that it impedes the ability to perform ISI inspections on the inner wall of the reactor vessel. The option may exist to perform ISI through the insulation; however, this will require develop of suitable methods and may require a limit on the insulation type and thickness.

7.3.1.2.4 Impact on PCC and DCC events

The presence of an insulating layer on the inner surface of the reactor vessel will serve to increase the maximum fuel temperatures resulting from PCC and DCC events. The thickness and type of insulation need to be optimized to ensure that fuel temperatures remain acceptable.

Conversely, the presence of the insulating layer on the inner surface of the reactor vessel will serve to decrease the maximum reactor vessel wall temperature over that experienced in the un-insulated case.

There is also a temporal impact due to the presence of an insulating layer on the inner surface of the reactor vessel. The heat transfer rate will be retarded sufficiently to potentially prolong the duration of core heat-up. It will take longer for the heat lost from the reactor vessel to match core decay heat; hence, the fuel heatup and cooldown will be extended significantly. This may also impact the reactor vessel time at temperature limits.

In any case, sensitivity analyses will confirm the acceptability of this approach with respect to PCC and DCC events.

7.3.1.2.5 Feasibility Considerations

The addition of an insulating layer on the inner surface of the reactor is feasible but probably more difficult than insulating the core barrel.

The NGNP reactor vessel is made from shop fabricated segments and, due to its size, assembled on site. Due to the amount of field welding involved, it will be unlikely to shop install any vessel insulation. Hence, the vessel insulation system must be designed to be installed on site.

The insulation system for the reactor vessel can be installed after the reactor vessel is in place. Conventional attachment using weldments on the inner vessel surface might be used, but the acceptability of welding to the inner vessel surface must be considered carefully.

7.3.1.3 Annular Radiative Heat Shield

7.3.1.3.1 Description

In this option, a thin metallic barrier with reasonably low emissivity is placed in the annular space between the core barrel outer wall and the inner reactor vessel wall. The main function of the shield is to reduce the amount of radiative heat flux incident upon the inner vessel wall; hence, resulting in lower vessel wall temperatures.

7.3.1.3.2 Effectiveness

The insertion of a radiative heat shield can reduce the radiative heat flux incident upon the reactor vessel inner surface by approximately a factor of two. Since radiative heat transfer is the dominant heat transfer mode between the outer core barrel and inner reactor vessel wall surfaces, the net effect will be a significant reduction in the vessel wall temperature.

7.3.1.3.3 Impact on Steady State Operations

Except for reducing the reactor vessel wall temperature, the addition of a radiative heat shield in the vessel-core barrel annular space has no impact on steady state operation. The annular space is essentially a low flow stagnant region. The radiative heat shield does not impact the inlet flow characteristics at all because of the way the inlet flow is routed through the double-walled core barrel.

The presence of this radiation shield favorably impacts plant parameters such as cycle efficiency due to reduced reactor vessel heat loss. Moreover, it does not impact core refueling.

7.3.1.3.4 Impact on PCC and DCC Events

The presence of a radiative shield on the inner surface of the reactor vessel will serve to increase the maximum fuel temperatures resulting from PCC and DCC events. The properties of radiative shield need to be optimized to ensure that fuel temperatures remain acceptable. However, due to the radiative heat transfer considerations, the degree of adjustment with a radiative shield is more limited than for insulation.

Conversely, the presence of the radiative shield will serve to decrease the maximum reactor vessel wall temperature over that experienced in the reference case without supplemental thermal protection.

There is also a temporal impact due to the presence of the radiative shield. The heat transfer rate will be retarded sufficiently to potentially prolong the duration of core heat-up. It will take longer for the heat lost from the reactor vessel to match core decay heat; hence, the fuel temperature heatup and cool-down will be extended significantly. This may also impact the reactor vessel time at temperature limits.

In any case, sensitivity analyses will confirm the acceptability of this approach with respect to PCC and DCC events.

7.3.1.3.5 Feasibility Considerations

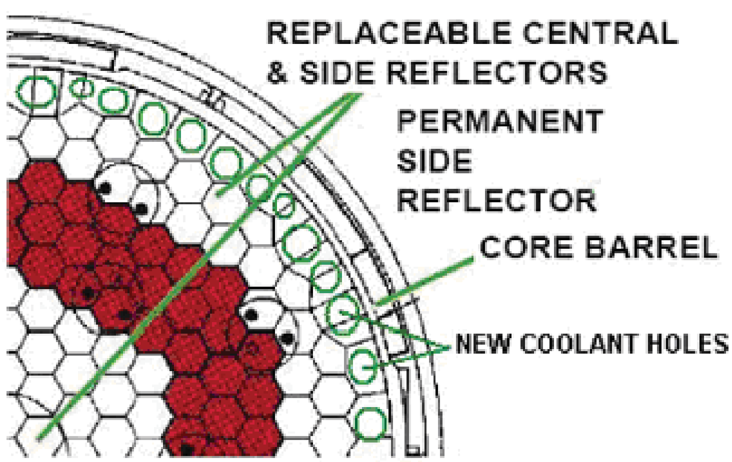
The installation of a radiative shield is considered feasible. The gap between the reactor vessel inner wall and the outer core barrel is more than sufficient to accommodate a thin radiative shield.

7.3.1.4 Relocation of Core Inlet Flow Path

7.3.1.4.1 Description

Relocation of the inlet flow path from the core barrel annulus to inside the core region could reduce the reactor vessel temperature during normal operation. General Atomics (GA) has proposed such an approach for the H2-MHR very high temperature reactor concept. In GA’s H2-MHR (ref. 10), it was reported that creating new inlet flow channels in the permanent side reflectors (PSR) could substantially reduce reactor vessel temperature. The proposed pathway is shown in Figure 7-3.

Figure 7-3: Relocation of core inlet flow path in the PSR



7.3.1.4.2 Effectiveness

GA considered two routing configurations: one routed the inlet flow through holes in the inner reflector while the other routed the inlet flow through holes in the PSR. Both configurations increase the thermal resistance between the inlet flow path and the vessel. GA evaluations showed that both configurations had nearly the same effect in terms of reducing vessel temperatures and parasitic heat losses to the RCCS. However, routing the inlet flow through the inner reflector resulted in a greater loss of heat capacity (from removal of graphite to provide the flow paths), which caused peak fuel temperatures to increase by about 40°C during a conduction cooldown.

GA reported a 170°C differential between the average vessel wall temperature (420°C) and the core inlet temperature 590°C. Hence, a somewhat smaller differential would be anticipated for a coolant inlet temperature of 500°C; nevertheless, the resulting vessel temperature should be reduced sufficiently to be within an acceptable temperature range.

7.3.1.4.3 Impact on Steady State Operations

Rerouting the core inlet flow path via the internal graphite structures does have a greater effect on normal operating vessel temperatures than the thermal insulation or radiative shield discussed above. However, in addition to lowering reactor vessel and core barrel temperatures, the revised flow path would also have several other effects on reactor components.

Local graphite temperatures at the periphery of the core near the core barrel would be reduced. This can affect the accumulated radiation damage in the PSR. It is anticipated that this impact would not have a significant effect on component lifetime, but evaluation would be required.

More importantly, creating this new flow path does make it more likely for increased bypass leakage flows. In any inlet flow path passing upward through the graphite reflectors, the coolant channel would pass through several intersections between blocks. The many resulting gaps in the channel wall would offer alternate flow paths to short circuit part or all of the core. Undoubtedly, designers would do their best to minimize these bypass paths. However, expected dimensional changes in the graphite as well as manufacturing and installation tolerances would undoubtedly result in a significant impact on the bypass flow. This will reduce the coolant flow in the active core flow passages, resulting in a potentially significant increase in local operating fuel temperatures.

This bypass flow problem is further compounded by the fact that it will be very difficult to monitor the flow field in the resulting multidimensional flow network formed by the reflector inlet passages, the active core coolant channels, and the many horizontal and vertical inter-block gaps. If very conservative analysis assumptions must be used, this could be penalizing on plant operation.

Additionally, removal of graphite material will increase the fast fluence experienced by the reactor vessel. The total fluence on the vessel would be expected to increase. More importantly, the vessel spectrum would harden significantly. An important function of the outer reflector is to attenuate the neutron flux and thermalize the spectrum. Since the spectrum at the outer edge of the reflector will be harder, the effectiveness of any boron at that location will also be minimized, further exacerbating the problem. The impact of this additional, hardened fluence will need to be carefully evaluated.

Removal of graphite from the side reflectors could also reduce the worth of the control rods. Since many neutrons are thermalized in the side reflectors, they amplify the worth of the control rods located at the boundary between the active core and the reflectors. Effectively shifting moderation from the outer reflector to the active core will reduce the worth of the rods on the boundary.

Rerouting the core inlet flow path as described favorably impacts plant parameters such as cycle efficiency due to reduced reactor vessel heat loss. Moreover, placing coolant channels in the PSR does not impact core refueling.

7.3.1.4.4 Impact on PCC and DCC Events

Removal of material from the reflector blocks to create the inlet flow path will also affect fuel temperatures during PCC and DCC events.

First of all, the removal of material from the reflectors reduces the heat capacity of the reactor structures. The overall core heat capacity plays a key role in managing the imbalance between decay heat production rates and passive decay heat removal rates. Reducing the heat capacity will generally allow fuel temperatures to rise more rapidly, and it will generally increase the ultimate peak fuel temperature.

In addition, removing graphite from the PSR blocks will alter the effective conductivity between the core and the core barrel. The precise effect depends on the specific event and the previous service life of the reflectors, since the effective conductivity represents a combination of conduction and radiation heat transfer. However, in general, the effective conductivity would be reduced by the graphite removal. The heat transfer rate will be retarded sufficiently to potentially prolong the duration of core heatup and cooldown transient. It will take longer for the heat lost from the reactor vessel to match core decay heat; hence, the maximum fuel temperatures may occur earlier and cool-down will be extended. This could impact the reactor vessel time at temperature limits. In some cases higher local conductivities due to increased radiation heat transfer could actually increase vessel temperatures.

Preliminary analyses of conduction cooldown for the H2-MHR concept have been performed by other analysts. However, more detailed analyses would be required for all potential scenarios to confirm the potential acceptability of this approach with respect to PCC and DCC events.

7.3.1.4.5 Feasibility Considerations

Relocation of the core inlet flow path is not considered to be very feasible. It might be possible to implement it successfully, if significant compromises are made in other reactor design areas and overall plant operating conditions. However, it introduces or exaggerates several other design concerns which will be difficult to assess. As discussed in the preceding subsections, major concerns include

- Reduced core heat capacity (higher conduction cooldown temperatures)
- Altered reflector effective conductivity (higher core and/or vessel conduction cooldown temperatures depending on the specific event)
- Higher bypass flows (higher operating fuel temperatures)
- Higher vessel fast fluence during normal operation
- Reduced control rod worths

Proponents of this strategy have analyzed some of these considerations, but not all of them. A significant amount of core redesign and assessment would likely be necessary.

It also appears to be costly option to implement. The number of permanent side reflectors that need to be bored out is significant. There are over 1000 PSR blocks that rest against the inner core barrel in a ring that is 12 PSR blocks high. GA proposed to have 72 inlet flow channels drilled in the PSRs. This equates to having to special machine 864 PSR blocks. In addition, design optimization to try to control bypass flows could be complex.

7.3.2 Active Options

7.3.2.1 Dedicated Vessel Cooling System

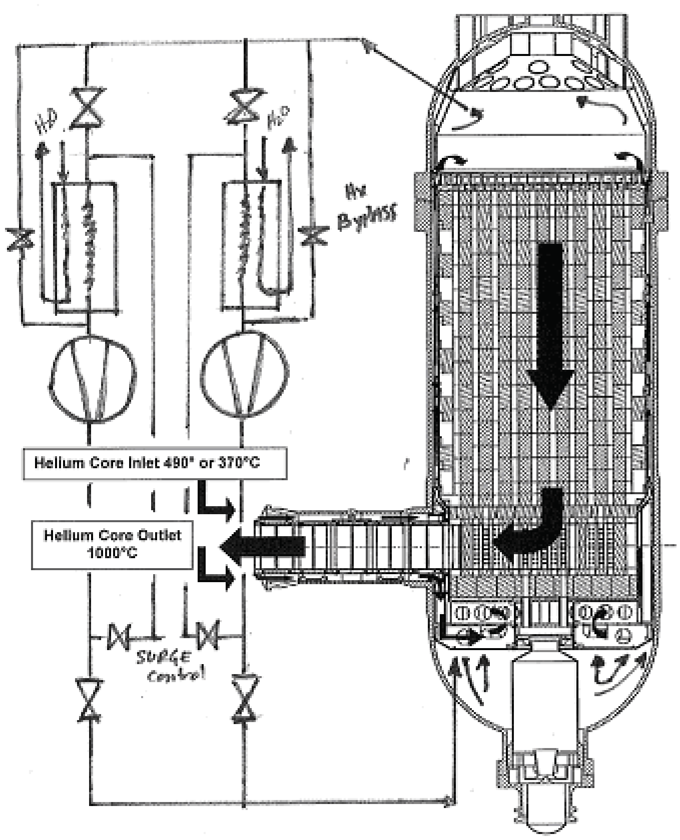
7.3.2.1.1 Description

In this option, as shown in Figure 7-4, “hot” helium is taken from the upper head region between the reactor vessel wall and the upper core inlet plenum shroud and routed through a helium-to-water heat exchanger where it is cooled down to approximately 250ºC (to be confirmed). The “cooled helium” is returned to the lower head region where it then flows up the annular space between the inner reactor vessel wall and the outer wall of the core barrel. It then reaches the upper head region, completing the circuit.

This system ensures a constant flow of cooled helium is available to “bathe” the entire inner surface of the reactor vessel with relatively cool helium, keeping the reactor vessel temperature at an optimum temperature (i.e., less than 350ºC) for its service conditions.

In the current design, the lower part of the core support structure communicates openly with the lower plenum region. Modifications will be required to partition the lower plenum area to provide a separate pathway that will convey vessel cooling flow to the reactor vessel annulus region.

Figure 7-4: Direct vessel cooling



7.3.2.1.2 Effectiveness

A dedicated active cooling system, when properly designed, will be highly effective in maintaining the reactor vessel at a temperature less than 350°C. This not only includes the portion of the reactor vessel adjacent to the core but also the lower and upper hemispherical head regions.

7.3.2.1.3 Impact on Steady State Operations

Except for reducing the reactor wall temperature to less than 350ºC, the use of an active vessel cooling system should not significantly impact steady state operation. The annular space between the inner reactor vessel wall and the outer core barrel wall provides the major flow path for the vessel cooling system. The system does not impact the inlet flow characteristics at all because of the way the inlet flow is routed through the double-walled core barrel.

The presence of an active vessel cooling system may have a slight negative impact on plant parameters such as cycle efficiency due to increased heat loss from the reactor vessel and to the power required to run its associated pumps and blowers. However, it does not impact core refueling. A key advantage of the system, however, is that it does not in any way impede the ability to perform ISI inspections on the reactor vessel or the outer core barrel.

An active vessel cooling system will also have an impact on plant availability, because it will be required for power operations. Should it not be operational, a plant shutdown could be required.

7.3.2.1.4 Impact on PCC and DCC Events

An active vessel cooling system will not adversely impact PCC and DCC results. Moreover, it has a slight beneficial effect in that the average reactor vessel temperature will begin these events at a lower temperature than without the system.

However, while insulation or radiative shield between the core barrel and the vessel may provide some reduction in peak conduction cooldown vessel temperatures, the active cooling system probably would not provide a significant benefit in this area.

If the cooling system remains in operation during these events, it can assist in the removal of decay heat.

7.3.2.1.5 Feasibility Considerations

The addition of an active vessel cooling system is a highly feasible option. It consists of standard power plant equipment: blowers, pumps, valves, heat exchangers etc. It simply remains an engineering task to design, fabricate, procure, construct and operate.

It should be noted that the current version of the PBMR uses this concept of vessel cooling to permit use of SA-508/SA-533 material.

7.3.2.2 Integral Cooling System

7.3.2.2.1 Description

This option is similar to that proposed previously in Section 7.3.2.1, Dedicated Vessel Cooling System, except that one of the existing auxiliary systems is used to provide helium cooling flow to the reactor vessel. The most likely candidate system is the helium purification system.

As with the dedicated vessel cooling option, (Figure 7-4), “hot” helium is taken from the upper head region above the core inlet plenum and routed to the helium purification system. There it is cooled and processed. The “cooled clean helium” is then returned to the reactor vessel lower head region where it is directed up the annular space between the inner reactor vessel wall and the outer wall of the core barrel. It then reaches the upper head region, completing the circuit.

In the current design, the lower plenum communicates directly with lower core support structure and any helium return flow in that region would mix with the reactor inlet flow. Modifications in this region would therefore be required to direct the cool helium flow to the vessel annulus.

Note: Another candidate system could be the shutdown cooling system; however, the SCS system is not a good match for this application. First, its capacity is much too large. And, second, it takes its source from the core exit which is far from being an optimum choice. Furthermore, it would be very difficult to adapt the SCS to provide a vessel cooling role.

7.3.2.2.2 Effectiveness

As with the dedicated vessel cooling option, this option could be effective at maintaining the reactor vessel temperature within acceptable limits. However, the main disadvantage with this system is one of capacity. The normal helium purification flow rate is 5% of system helium mass per hour or 150 kg/hour. For handling any water ingress, the system can operate at 3300 kg/hour to remove moisture (this is equivalent to a clean-up constant of 100% per hour). Even at this higher rate, equivalent to about 1 kg/sec), the flow rate is probably too low for cooling flow purposes.

7.3.2.2.3 Impact on Steady State Operations

Except for reducing the reactor wall temperature to less than 350ºC, the use of an active vessel cooling system should not significantly impact steady state operation. The annular space between the inner reactor vessel wall and the outer core barrel wall provides the major flow path for the vessel cooling system. The system does not impact the inlet flow characteristics at all because of the way the inlet flow is routed through the double-walled core barrel.

The presence of an active vessel cooling system may have a slight negative impact on plant parameters such as cycle efficiency due to increased heat loss from the reactor vessel and to the power required to run its associated pumps and blowers. However, it does not impact core refueling. Another advantage of the system, is that it does not impede the ability to perform ISI inspections on the reactor vessel or the outer core barrel.

Again, the active vessel cooling system will also have an impact on plant availability, because it will be required for power operations.

7.3.2.2.4 Impact on PCC and DCC Events

As with the dedicated vessel cooling system, using an installed system to perform vessel cooling will not adversely impact PCC and DCC results. Moreover, it has a beneficial effect in that the average reactor vessel temperature will begin these events at a lower temperature than without the system.

However, as for the dedicated active system, the use of an existing system for active vessel cooling does not provide a significant benefit for peak conduction cooldown vessel temperatures unless the system operates through the transient. Thus the licensing analyses will not take benefit of this system unless it is a safety related system with the required redundancy, etc.

7.3.2.2.5 Feasibility Considerations

It is feasible to use an auxiliary system such as helium purification. However, the practicality of adapting this system for vessel cooling duty is questionable. Using a dedicated system as discussed in the previous section is a more attractive option.

7.3.3 Initial Screening of Reactor Vessel Options

The following reactor vessel options were presented above:

- Passive Options-Reactor Vessel
    - Insulating the Core Barrel Outer Surface
    - Insulating the reactor vessel inner wall
    - Installing a radiative heat shield
    - Relocating the core inlet flow path
- Active Options – Reactor Vessel
    - Dedicated vessel cooling system
    - Integral vessel cooling system

A qualitative comparison of these options is presented in Table 7-1.

For each option, the impact of the option on following attributes is qualitatively rated:

System effectiveness
Achievable Power Level
Plant availability
Plant efficiency
Vessel heat loss
Reactor Design
Core hydraulics – bypass flows, others
Plant operation due to loss of function
Vessel properties to due neutron damage
Source of cooling
Required equipment
Maintenance
Reactor Vessel ISI capability
Refueling
PRA
Nuclear safety
PCC accident
DCC accident
Fabrication costs
Construction cost
Cost of operation
Disadvantages

Based on the initial evaluation of the candidate options, some preliminary conclusions are noted below.

Of the passive options for the reactor vessel, insulating the core barrel outer surface seems promising relative to the other passive options for the following reasons. First, it is very effective in reducing the radiative heat flux that would emanate for a bare core barrel surface. Second, it should be low cost. In conjunction with insulating the core barrel, insulation of the upper and lower head regions may also be required In order to achieve a uniform temperature distribution throughout the vessel. The key to this option's success will be how well the type and amount of insulation can be optimized to achieve a balance between steady state reactor vessel thermal performance and fuel and reactor vessel thermal performance during PCC and DCC events. If this balance is shown difficult to obtained, the thermal shielding option could be a good alternative.

The dedicated cooling system is the most promising of the active options for the reactor vessel. There is no uncertainty in its effectiveness – it will work. It also does not in any way interfere with the effectiveness of passive heat transfer following PCC and DCC events. It is also attractive with respect to the synergy it offers in being coupled to the cooling needs of the IHX and cross-vessel piping.

By far, the least promising option for the reactor vessel is the relocation of the core inlet flow path. It will have a significant impact on core design and cost. More importantly, it will adversely affect several other normal operating characteristics including vessel fluence, control rod worths, core bypass flow and operating fuel temperatures. It would also affect the behavior of conduction cooldown events.

Next, insulating the reactor vessel inner wall suffers from the main drawbacks of interfering with the ability to perform ISI and installation method. Finally, matching the needs of vessel cooling to an existing system such as Helium Purification will only work if there is similarity in size and needs. This is not the case since vessel cooling loads will be much larger than the capacity of a small auxiliary system like He purification.

Based on these observations, more detailed analysis of the the following options was pursued as described in Section 7.5:

- Insulation on outside of core barrel
- Radiative shield between core barrel and reactor vessel
- Dedicated active vessel cooling system

Table 7-1: Comparison of vessel options

	Active In-Vessel Cooling	
Option -->	Dedicated Vessel Cooling System	Integral Cooling System
Description	A vessel cooling pathway is created with dedicated cooling system (i.e., similar to CBCS in PBMR). Cooling flow is provided such that all regions of the reactor vessel are maintained acceptable temperatures.	Same as dedicated vessel cooling system but cooling flow is taken from an exisiting source - e.g., helilum purification.
ATTRIBUTES:		
> System effectiveness	**HIGH - Need to confirm by calculation.**	**HIGH - Need to confirm by calculation.**
>Impact on Achievable Power Level	**NONE**	**NONE**
>Impact on plant availability	**LOW**	**LOW**
>Impact on plant efficiency	**LOW**-heat gained by cooling system is discharged as waste heat	**LOW**-heat gained by cooling system is discharged as waste heat
>Impact on reactor vessel heat loss	**LOW.** Will reduce vessel heat loss.	**LOW.** Will reduce vessel heat loss.
>Impact on reactor design	**MEDIUM** - requires designing a separate cooling system and modifications to core internals to develop cooling flow paths.	**MEDIUM** - requires designing a separate cooling system and modifications to core internals to develop cooling flow paths.
>Impact on core hydraulics – bypass flows, others	**LOW**	**LOW**
>Impact on plant operation due to loss of function (e.g., cooling)	**HIGH** - loss of system will require reduction of core inlet temperature and/or core power	**HIGH** - loss of system will require reduction of core inlet temperature and/or core power
>Impact on vessel properties to due neutron damage	**LOW** - system will need to be designed to maintain optimum irradiation temperature	**LOW** - system will need to be designed to maintain optimum irradiation temperature
>Impact on source of cooling	**MEDIUM** - a reliable source of cooling water will be required	**HIGH** - Using an existing source like He purification places an additional burden on that system which couples it's availability directly to plant availability.
>Impact on required equipment	**MEDIUM** - a dedicated system will be required along with reliable source of back-up power	**LOW** - use of existing cooling source lessens need for dedicated equipment.
>Impact on maintenance	**LOW**- Another system to maintain	**LOW**- Another system to maintain
>Impact on reactor vessel ISI capability	**NONE**	**NONE**
>Impact on refueling	**NONE**	**NONE**
>Impact on PRA	**MEDIUM.** The addition of cooling system will result in consideration of new scenarios (e.g., water ingress via this system).	**MEDIUM.** The addition of cooling system will result in consideration of new scenarios (e.g., water ingress via this system).
>Impact on nuclear safety	**Low.** Water leakage from system could affect reactivity.	**Low.** Water leakage from system could affect reactivity.
>Impact on PCC accident	**LOW or NONE**	**LOW or NONE**
>Impact on DCC accident	**LOW or NONE**	**LOW or NONE**
>Impact on fabrication costs	**LOW**	**LOW**
>Impact on construction cost	**LOW**	**LOW**
>Impact of cost to operate system	**LOW**	**LOW**
DISADVANTAGES	Requires a dedicated system and source of cooling water.	Places undue burden on an auxiliary system by linking the availability of the system to overall plant availability.

Table 7-1: Comparison of vessel options (cont'd)

	Passive Options			
Option -->	**Relocate Inlet Flow Path (H2-MHR option)**	**Heat Shield**	**Core Barrel Outer Wall Insulation**	**Reactor Vessel Inner Wall Insulation**
Description	The inlet flow path is relocated inside the outer reflector region via flow holes in the permanent side reflector.	A thin metal annular shield is placed between the reactor vessel inner wall and the core barrel to create a thermal radiation shield.	A layer of insulation is applied to the outer core barrel surface.	A layer of insulation is applied to the inner surface of the reactor vessel wall.
ATTRIBUTES:				
> System effectiveness	Need calculation results	Need calculation results	Need calculation results	Need calculation results
>Impact on Achievable Power Level	Need calculation results	Need calculation results	Need calculation results	Need calculation results
>Impact on plant availability	LOW	LOW	LOW	LOW
>Impact on plant efficiency	LOW	LOW	LOW	LOW
>Impact on reactor vessel heat loss	**LOW.** Will reduce vessel heat loss.	**LOW.** Will reduce vessel heat loss.	**LOW.** Will reduce vessel heat loss.	**LOW.** Will reduce vessel heat loss.
>Impact on reactor design	**HIGH** - requires re-routing inlet flow through reflector or permanent side reflectors.	**Low-to-Medium.** Requires design and installation of the shield but does not require major inlet flow re-routing.	NONE	NONE
>Impact on core hydraulics – bypass flows, others	**HIGH** - inlet flow re-routing will increase amount of core bypass flow. Need calculation results.	NONE	NONE	NONE
>Impact on plant operation due to loss of function (e.g., cooling)	**N/A** - complete loss of inlet flow function is not credible.	**High.** - The loss of function for the heat shield is unlikely but the impact is high if it is lost.	**HIGH** - loss of insulating function will result in higher reactor vessel wall temperatures which may necessitate lower temperature/lower power level operations or plant shutdown	**HIGH** - loss of insulating function will result in higher reactor vessel wall temperatures which may necessitate lower temperature/lower power level operations or plant shutdown
>Impact on vessel properties to due neutron damage	**MEDIUM-HIGH** - Removing reflector material to create a flow path will increase the fast fluence on the reactor vessel. Need calculation results to confirm extent of increase and acceptability.	NONE	NONE	NONE
>Impact on source of cooling	NONE.	NONE	NONE	NONE
>Impact on required equipment	NONE.	NONE	NONE	NONE
>Impact on maintenance	NONE.	**LOW** - will require periodic inspection	**LOW** - will require periodic inspection	**LOW** - will require periodic inspection
>Impact on reactor vessel ISI capability	NONE.	**HIGH** - Addition of a heat shield barrier will impede ability to perform vessel inspections. Making a removable shield will be difficult.	**LOW.** The addition of an insulation layer will decrease amount of annular space (6 cm) between core barrel outer wall and reactor vessel inner wall. This may impede ISI capability.	**High.** Insulating the vessel wall will impede ability to perform vessel inspections. However, a thin layer may not pose a significant problem for ECT.
>Impact on refueling	NONE.	NONE	NONE	NONE
>Impact on PRA	NONE.	NONE.	NONE.	NONE.
>Impact on nuclear safety	NONE.	NONE.	NONE.	NONE.
>Impact on PCC accident	Need calculation results	Need calculation results	Need calculation results	Need calculation results
>Impact on DCC accident	Need calculation results	Need calculation results	Need calculation results	Need calculation results
>Impact on fabrication costs	**HIGH** - Hundreds of PSR or reflector blocks need to be drilled with flow holes, other modifications to inlet piping etc. will also be required.	LOW-MEDIUM	LOW-MEDIUM	LOW-MEDIUM
>Impact on construction cost	LOW	LOW	LOW	LOW
>Impact of cost to operate system	NONE	NONE	NONE	NONE
DISADVANTAGES	This option requires a major re-design of the core internals and flow path arrangment.	By themselves, these options are **NOT** an integral solution to the problem of thermal protection for the reactor vessel, IHX and interconnecting piping. These options only focus on achieving acceptable temperatures in the RV core beltline region. Other regions of the vessel will need to be evaluated for hot spots etc. Any of these options could be combined with other options to arrive at an integral solution.		

7.4 Identification of Options to Allow SA-508/SA-533 for the IHX Vessel

The constraints for using SA-508/SA-533 material for the IHX vessel are the same as described in Section 7.3 for SA-508/SA-533 material use in the reactor vessel.

7.4.1 Passive Options

7.4.1.1 Insulate IHX Inner/Outer Shell

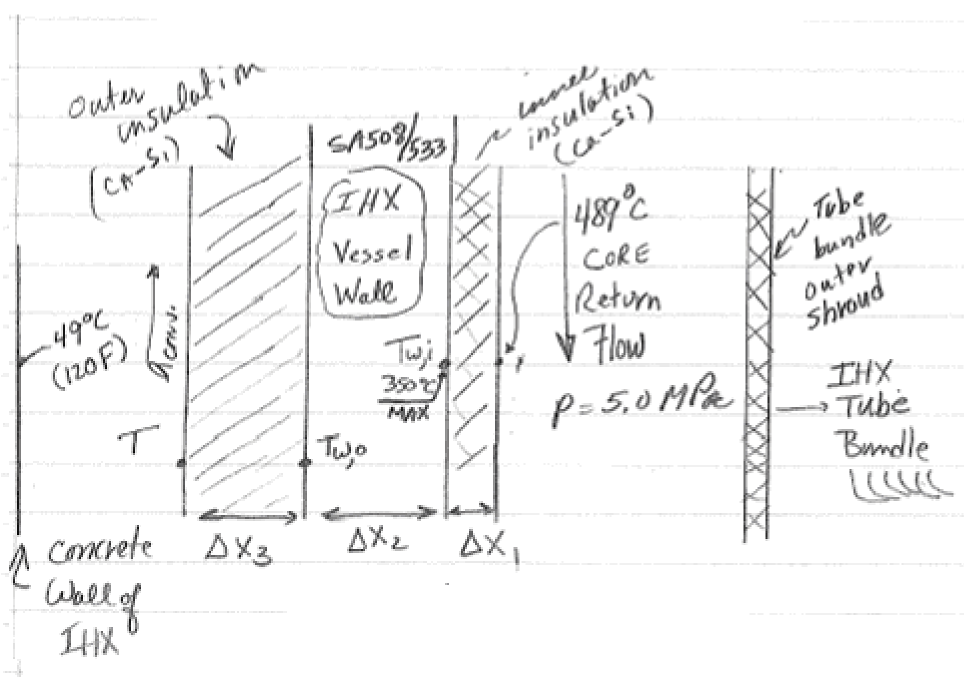
7.4.1.1.1 Description

In the tubular IHX concept, the core return He (~500ºC) flows along an annular space between the tube-bundle package shroud and the IHX inner pressure vessel wall for essentially the entire length of the IHX. In this option a layer of insulating material is applied to both the inner and outer surfaces of the IHX vessel wall. See Figure 7-5 below.

Insulating both interior and exterior surfaces is required so that the IHX vessel wall temperature operates in an acceptable temperature range below 350ºC. Over-insulating the interior surface will result in too cold of a wall temperature range. Hence, both insulation layers need to be optimized.

Over-insulating the interior surface of the IHX vessel will significantly reduce the heat loss form the IHX; however, the resulting wall temperatures may be too low and would make the IHX susceptible to thermal shock should the insulation fail.

Figure 7-5: IHX vessel insulation



7.4.1.1.2 Effectiveness

Applying insulation to the IHX as described above will effectively reduce the IHX vessel wall temperatures.

7.4.1.1.3 Impact on Steady State Operations

There is no impact on steady state operations except for the reduction in heat loss from the IHX. This improves plant efficiency slightly.

7.4.1.1.4 Feasibility

The addition of an insulating layer on the inner and outer surfaces of the IHX is feasible. It may be more difficult to insulate the interior surface that the exterior surface.

The IHX vessel is shop fabricated and the installation of insulation can be performed under shop conditions.

7.4.2 Active Options

7.4.2.1 Dedicated Cooling System

7.4.2.1.1 Description

In this option a dedicated cooling system will provide helium cooling flow to the IHX vessel. Helium cooling flow passes through the annular space formed by the inner IHX vessel surface and a flow baffle. The sole purpose of the baffle is to form the pathway for vessel cooling. Additionally, the exterior surface of the IHX vessel is fully insulated to minimize heat loss.

7.4.2.1.2 Effectiveness

A dedicated active cooling system, when properly designed, will be highly effective in maintaining the IHX vessel at a temperature less than 350ºC. This not only includes the portion of the IHX vessel wall along its length but also the lower and upper hemispherical head regions.

7.4.2.1.3 Impact on Steady State Operations

Except for reducing the IHX wall temperature to less than 350ºC, the use of an active vessel cooling system should not significantly impact steady state operation. The system may impact the IHX exit flow characteristics due to the reduction in flow area due to the flow baffle; however, this effect can be minimized or completely eliminated by re-design.

The presence of an active IHX vessel cooling system may have a slight negative impact on plant parameters such as cycle efficiency due to increased heat loss from the IHX vessel and due to the power required to run the associated pumps and blowers. However, it does not impact core refueling. A disadvantage of the system, however, is that it may impede the ability to perform ISI inspections on the inner IHX vessel wall.

An active vessel cooling system will also have an impact on plant availability because it will be required for power operations. Should it not be operational, a plant power reduction or shutdown will be required.

7.4.2.1.4 Feasibility

The addition of an active IHX vessel cooling system is a highly feasible option. It consists of standard power plant equipment: blowers, pumps, valves, heat exchangers etc. It simply remains an engineering task to design, fabricate, procure, construct and operate.

7.4.3 Initial Screening of IHX Vessel Options

The following IHX vessel options were presented above:

- Passive Options-IHX Vessel
    - Insulating the IHX vessel inner wall
- Active Options – IHX vessel
    - Dedicated cooling system

A qualitative comparison of these options is presented in Table 7-2.

For each option, the impact of the option on following attributes is qualitatively rated:

System effectiveness	Required equipment
Achievable Power Level	Maintenance
Plant availability	IHX Vessel ISI capability
Plant efficiency	PRA
IHX Vessel heat loss	Nuclear safety
IHX Design	Fabrication costs
IHX hydraulics	Construction cost
Plant operation due to loss of function	Cost of operation
Source of cooling	Disadvantages

Based on the initial evaluation of the candidate options, some preliminary conclusions are noted below.

The active cooling system appears to be the most promising for essentially the same reason stated for the reactor vessel dedicated cooling system – there is no uncertainty in its effectiveness. It also permits the IHX vessel to be fully insulated from the outside while being maintained at an acceptable operating temperature. Furthermore, the IHX cooling needs can be coupled to that of the reactor vessel and be served by a single system.

For the IHX, the option of insulating the inside surface of the vessel can be envisioned but could lead to difficulty selecting and installing suitable insulation.

Based on these observations, more detailed analysis of the the following options was pursued as described in Section 7.5:

- Insulation inner and outer surface of IHX vessel
- Active IHX vessel cooling

Table 7-2: Comparison of IHX vessel options

	Active In-Vessel Cooling	Passive Cooling
Option -->	**Dedicated Vessel Cooling System**	**Reactor Vessel Inner Wall Insulation**
Description	An IHX vessel cooling pathway is created with dedicated cooling system (i.e., similar to CBCS in PBMR). Cooling flow is provided such that all regions of the IHX vessel are maintained acceptable temperatures.	A layer of insulation is applied to the inner surface of the reactor vessel wall.
ATTRIBUTES:		
> System effectiveness	**HIGH - Need to confirm by calculation.**	**MEDIUM - Need calculation results**
>Impact on Achievable Power Level	**NONE**	**NONE**
>Impact on plant availability	**LOW**	**LOW**
>Impact on plant efficiency	**LOW**-heat gained by cooling system is discharged as waste heat	**LOW**
>Impact on IHX vessel heat loss	**LOW.** Will reduce vessel heat loss.	**LOW.** Will reduce vessel heat loss.
>Impact on IHX design	**MEDIUM** - requires designing a separate cooling system and modifications to IHX internals to develop cooling flow paths.	**LOW**
>Impact on IHX hydraulics – bypass flows, others	**LOW**	**LOW**
>Impact on plant operation due to loss of function (e.g., cooling)	**HIGH** - loss of system will require reduction of core inlet temperature and/or core power	**HIGH** - loss of insulating function will result in higher IHX vessel wall temperatures which may necessitate lower temperature/lower power level operations or plant shutdown
>Impact on source of cooling	**MEDIUM** - a reliable source of cooling water will be required	**NONE**
>Impact on required equipment	**MEDIUM** - a dedicated system will be required along with reliable source of back-up power	**NONE**
>Impact on maintenance	**LOW**- Another system to maintain	**LOW** - will require periodic inspection
>Impact on IHX vessel ISI capability	**HIGH -** development of cooling channels will make inspection of inner IHX vessel wall difficult.	**High.** Insulating the vessel wall will impede ability to perform vessel inspections. However, a thin layer may not pose a significant problem for ECT.
>Impact on refueling	**NONE**	**NONE**
>Impact on PRA	**MEDIUM.** The addition of cooling system will result in consideration of new scenarios (e.g., water ingress via this system).	**LOW**
>Impact on nuclear safety	**Low.** Water leakage from system could affect reactivity.	**NONE.**
>Impact on PCC accident	**NONE**	**NONE**
>Impact on DCC accident	**NONE**	**NONE**
>Impact on fabrication costs	**MEDIUM**	**LOW-MEDIUM**
>Impact on construction cost	**LOW**	**LOW**
>Impact of cost to operate system	**LOW**	**NONE**
DISADVANTAGES	Requires a dedicated system and source of cooling water.	

7.5 Analysis of Design Options for SA-508/SA-533

Based on the previous evaluation, the following options are further studied:

- Reduced normal operating temperature and/or power level
- Passive Options-Reactor Vessel
    - Insulating the Core Barrel Outer Surface
    - Installing a radiative heat shield
- Active Options – Reactor Vessel
    - Dedicated vessel cooling system
- Passive Options-IHX Vessel
    - Insulating the IHX vessel inner wall (in addition to the outer wall)
- Active Options – IHX vessel
    - Dedicated cooling system

The evaluations are based on simplified heat transfer thermal calculations of nominal conditions as well as CFD calculations of both nominal conditions and conduction cool-down accidents. These evaluations are performed based on NGNP recommended parameters, namely 565 MWth and core inlet and outlet temperatures of respectively 500 and 900°C.

A first evaluation is however performed to evaluate under which operating conditions the current design could be used with SA-508/SA-533. This case is intended to probe the limits of SA-508/SA-533 achievable without adding any special design features to control vessel temperature.

7.5.1 Reactor Vessel – Required operating conditions changes without design modification

To evaluate the operating condition changes, calculations were performed using STAR-CD, a general-purpose finite-volume heat-transfer and computational fluid dynamics (CFD) code. This code is capable of modeling heat transfer by conduction, convection, and radiation in arbitrary geometries. The system that is modeled here consists of physical phenomena that occur on a wide variety of temporal and spatial scales, more than are typically modeled by modern CFD software and a traditional CFD approach. Thus, the STAR-CD code was enhanced by a set of additional subroutines to model the hydraulic resistance and heat transfer in the coolant channels, the heat transfer across the reactor cavity to the RCCS, and the thermal output of the reactor core. These subroutines also provide the temperature-specific properties of the component materials.

An existing three-dimensional model representation of the reactor vessel and its internal components was used for these analyses. Only a 30° section of the reactor is explicitly modeled, and the 12-fold symmetry of the core in the circumferential direction is used to represent the rest of the core through symmetric boundary conditions. Obviously, features that cannot be represented by this symmetry, such as the exit to the cross-duct, are necessarily excluded from being modeled explicitly.

The graphite fuel elements and reflector blocks are normally assumed for these calculations to be new and hence unaffected by irradiation from the core. However, irradiated graphite properties are used when confirming the effect of the individual options on fuel temperatures during conduction cooldown . The reactor vessel has the physical properties of SA508/SA533 steel.

The conduction cool-down scenario was numerically simulated by first determining the steady-state solution that describes the conditions during normal operation. Then, the thermal field of this solution was used as the initial condition for the conduction cool-down calculation, which was performed for 500 hours of simulated time.

The results of three cases are reported in this section. The reference case is based on 600 MWth and 400 and 800°C core inlet and outlet temperatures. In one of the cases the parameters were adjusted to be penalizing for the vessel temperatures; in another case, the parameters were adjusted to be penalizing for the fuel. The parameters used in all three sets of cases are presented in Table 7-3 for comparison.

The results of these calculations are given in Table 7-4. This table presents the peak vessel temperature during normal operation and the peak fuel and vessel temperatures during the conduction cool-down event. It also gives the time at metal temperature of the vessel for the temperature range between 371 and 427°C and between 427 and 538°C (although in none of the results presented here did the peak vessel temperature exceeds 538°C).

This table shows that, even for the conservative case for the vessel, temperatures during normal conditions are acceptable for SA 508 material (below 371°C) and temperatures reached during DCC satisfy values prescribed by Code Case N 499-2 (no more than 3000 h between 371 and 427°C and no more than 1000 h between 427 and 538°C) with margins to allow for several occurrences. The peak fuel temperature for the conservative case for the fuel is also considered as acceptable. It can be therefore concluded that the current RPV design can be considered acceptable without design modifications up to a power level of 600 MWth and a core inlet temperature of 400°C.

Table 7-3: Parameters Used for the DCC Analysis (Case without design modification)

Parameter	Reference case (R)	Conservative for the fuel (F)	Conservative for the vessel (V)
Power level (MWth)	600	616	624
Inlet temperature (°C)	400	400	428
Outlet temperature (°C)	800	800	816
Graphite conductivity	irradiated	irradiated & reduced	non-irradiated
Residual power	−	+6.6%	+10%
Steel emissivity	0.7	0.633	0.9
Outer RPV emissivity	0.8	0.8	0.7

Table 7-4: Results of Changing Operating Conditions

Power MWth	Inlet/outlet temp., °C	Case parameters	Max. fuel temp. DCC, °C	Max. RPV temp. Normal, °C	Max. RPV temp. DCC, °C	RPV, hours at 371−427°C	RPV, hours above 427°C
600	400/800	R	1400	332	437	207	81
624	428/816	V	1497	359	495	217	245
616	400/800	F	1645	333	440	265	107

7.5.2 Reactor Vessel − Insulating the Core Barrel Outer Surface

A calculation was performed to determine the effect of insulating the outer surface of the core barrel. The reference case is this time based on 565 MWth and 500 and 900°C core inlet and outlet temperature. This was modeled by adding a thermal resistance along this surface that is equivalent to placing a 3 mm thick layer of insulation with conductivity of 0.1 W/m K. This insulating material was assumed to be located along the entire outer core barrel surface, from the top of the core to the lower core support structures.

The results of this calculation are compared to the reference case in Table 7-5. The insulation results in lower peak vessel temperatures, particularly during normal operation. During conduction cooldown, the presence of the insulation keeps the vessel temperatures below 427°C. The detailed analysis shows that it reduces the amount of time that the vessel is above 371°C from 273 hours to 212 hours.

Although the reduced rate of heat flow out of the core barrel is beneficial to the vessel, it results in higher fuel temperatures during conduction cooldown, raising the peak temperature in the active core by 35°C. The core barrel would reach in such a case a temperature of 856°C and this temperature would be even larger if a conservative set of parameters was used to account for uncertainties. This temperature is considered as unacceptable even for a material like alloy 800H and it is considered at this stage that optimization of the insulation of the core barrel outer surface would be difficult to achieve.

Table 7-5: The Effect of Adding Insulation

Insulation	Power MWth	Inlet/outlet temp., °C	Max. fuel temp. DCC, °C	Max. RPV temp. Normal, °C	Max. RPV temp. DCC, °C	RPV, hours at 371−427°C	RPV, hours above 427°C
None	565	500/900	1399	404	439	183	90
Insulation, 3 mm	565	500/900	1434	315	399	212	0

7.5.3 Reactor Vessel - Installing a radiative heat shield

The option of installing a radiative heat shield was examined by introducing a set of two-dimensional baffle elements to the numerical model. These elements affect neither the thermal mass nor the heat transfer by conduction in the model. (Since they are located close to the outer surface of the core barrel, it is assumed that their effect on natural convection in the space between the barrel and the vessel is negligible.) Their only purpose is to shield the surfaces of the core barrel and the vessel from each other in STAR-CD's discrete-beam radiation heat transfer model.

The effect of the radiative heat shield on the peak temperatures is given in Table 7-7, with and without uncertainties. The set of parameters considered is given in Table 7-6. The effect is rather similar to the effect of the insulation on the outside of the core barrel. In both cases, the peak vessel temperatures decrease and the peak fuel temperature during conduction cooldown increases. The effect of the heat shield is less than the effect of the 3 mm insulation, both on the vessel and the fuel. For example, in these calculations, the radiation shield does not prevent the peak vessel temperature from exceeding 427°C; although the number of hours that the vessel material is above this temperature is greatly reduced.

Table 7-7 shows that, for the conservative case for the vessel, temperatures during normal conditions are slightly above the 371°C limit of SA 508 material. Temperatures reached during DCC satisfy values prescribed by Code

Case N 499-2 with margins to allow for several occurrences. The peak fuel temperature for the conservative case for the fuel is considered as acceptable and the temperature reached by the core barrel (782°C) is also considered as acceptable if a material like alloy 800H is selected.

This option with radiative shield is therefore considered as promising but would require a slight adjustment of the core inlet temperature or would necessitate revisiting the assumptions for the conservative case for the vessel.

Table 7-6: Parameters Used for the DCC Analysis (Radiative Shield Case)

Parameter	Reference case (R)	Conservative for the fuel (F)	Conservative for the vessel (V)
Power level (MWth)	565	580	588
Inlet temperature (°C)	500	500	535
Outlet temperature (°C)	900	900	918
Graphite conductivity	irradiated	irradiated & reduced	non-irradiated
Residual power	−	+6.6%	+10%
Steel emissivity	0.7	0.633	0.9
Outer RPV emissivity	0.8	0.8	0.7

Table 7-7: The Effect of Adding a Radiative Shield

Radiation shield	Power MWth	Inlet/outlet temp., °C	Case parameters	Max. fuel temp. DCC, °C	Max. RPV temp. Normal, °C	Max. RPV temp. DCC, °C	RPV, hours at 371−427°C	RPV, hours above 427°C
None	565	500/900	R	1399	404	439	183	90
Present	565	500/900	R	1410	347	428	247	9
Present	588	535/918	V	1511	382	482	202	216
Present	580	500/900	F	1635	345	429	294	33

7.5.4 Reactor Vessel - Dedicated vessel cooling system

7.5.4.1 Required Vessel cooling system

In this option, hot helium is taken from the upper head region above the upper plenum and routed through a helium-to-water heat exchanger where it is cooled down to approximately 250 ºC (482°F). The cooled helium is returned to the lower head region where it then flows up the annular space between the reactor vessel wall and the outer wall of the core barrel. It then reaches the upper head region, completing the circuit. This system ensures a constant flow of cooled helium is available to cool the entire inner surface of the reactor vessel, keeping the reactor vessel temperature at an optimum for its service conditions.

The reactor vessel is modeled as shown in figure 7-6.. The inside of the cylinder is exposed to 500°C (932°F) helium at 5.0 MPa, flowing at 600 lbm/s (272 kg/s). This forced convection flow and radiation are the inner boundary condition. The helium coolant flows in the annular space where two boundary conditions apply to each wall: forced convection to the flowing helium and radiation between the walls. Natural convection and radiation in air to a 65°C (149°F) wall are the outer boundary condition for the vessel wall.

The results of a steady state analysis for the reactor vessel cooling cases are shown in Table 7-8.

Table 7-8: Results of Reactor Vessel Cooling Analysis

Flowrate lbm/s	Flowrate kg/s	forced flow velocity m/s	heat transfer coefficient W/m^2 ºC	heat into flow MW	heat lost to ambient MW	max. core barrel T ºC	max. vessel wall T ºC	He flow out T ºC
0	0.0	0.00	21.9	0.00	2.07	494.	380.	N/A
15	6.8	0.33	23.8	1.74	1.46	491.	328.	298.8
20	9.1	0.44	30.0	2.16	1.41	490.	323.	295.4
55	25.0	1.20	67.0	4.39	1.24	484.	303.	283.6
200	90.7	4.34	186.1	9.82	1.04	467.	277.	270.7
600	272.0	13.02	448.1	17.14	0.95	442.	262.	262.0

These calculations indicate that a vessel wall temperature less than 350°C could be easily obtained with a limited loss of efficiency (1.13 MW for the case with 6.8 kg/s, taking into account that the heat transferred to the Reactor Cavity Cooling System would be reduced due to the presence of the cooling system).

7.5.4.2 Behavior of the vessel during conduction cooldown accident

The effect of vessel cooling was modeled by beginning the conduction cool-down simulation with the reactor vessel uniformly at 350°C. The results of this calculation and the reference case (both run with a core power of 600 MWth) are given in Table 7-9.

Because of the relatively small thermal mass of the vessel compared to the other components in the reactor, such as the graphite blocks, active cooling of the vessel has very little effect on the temperatures during conduction cool-down. Thus, the primary benefits of a dedicated vessel cooling system are limited to situations such as normal operation, when the active cooling is available.

Further calculations have however shown that, based on a reference power of 565 MWth, vessel temperatures reached during DCC would satisfy Code Case N499-2 limits (even when uncertainties are taken into account).

Table 7-9: The Effect of Cooling the Vessel on Conduction Cool-down

Vessel cooling	Max. fuel temp. DCC, °C	Max. RPV temp. Normal, °C	Max. RPV temp. DCC, °C	RPV, hours at 371−427°C	RPV, hours above 427°C
not cooled	1440	405	451	186	123
cooled	1440	350	450	185	123

7.5.5 IHX Vessel - Insulating the IHX vessel inner wall

To reduce the temperature of the IHX pressure vessel, insulation is to be placed as shown on its inner and outer surfaces (Figure 6-7). Analysis is necessary to determine the right balance of insulation on the inside and the outside surfaces. The goals for the design are 1) keep the temperature of the IHX vessel wall less than 350°C (662°F), and 2) keep the thermal losses from the IHX to ambient at less than 0.05 MW for each of the two tubular IHX vessels.

The IHX vessel wall is modeled as shown in figure 7-7. The inside of the cylinder is exposed to 489°C (912°F) helium at 5.0 MPa. This forced convection flow and radiation are the inner boundary condition. Various amounts of insulation are applied to the inner and outer walls of the IHX (with a conductivity of about 0.1 W/m°C in the range of temperature of interest). Natural convection and radiation in air to a 49°C (120°F) wall are the outer boundary condition.

The results for the insulated IHX cases are shown in Table 7-10. The case that meets the goal is shown in bold. The results indicate that significant external insulation is required to achieve the desired vessel heat loss, and that the inner and outer insulation must be properly balanced in order to achieve an acceptable vessel temperature.

Table 7-10: Results of IHX Insulation Analysis

inner insulation thick.	outer insulation thick.	metal T	heat loss to ambient
inches	inches	°C	MW
0.5	0.	201.	0.806
0.5	1.	397	0.331
0.5	10.	476.	0.055
1.	0.	150.	0.468
1.	1.	343.	0.256
1.	10	464.	0.052
3.	5.	374.	0.068
3.	6.	388.	0.061
3.	10.	419.	0.043
5.	4.	306.	0.059
5.	5.	343.	0.054
5.	**6.**	**343.**	**0.049**

The choice of the insulating material and the practicality of implementing the insulation should be further studied.

7.5.6 IHX Vessel - Dedicated cooling system

This option is similar to the IHX insulation option except that insulation is applied only to the outside of the IHX vessel wall (Figure 7-8). The inside is cooled actively, with 250°C, similarly to the vessel cooling option. A cylinder to separate the helium coolant at 250°C, from the core return flow at 489°C, would have to be added to the IHX. This configuration is analyzed similarly to the vessel cooling option.

The results for the IHX cooling cases are shown in Table 7-11.

Table 7-11: Results of IHX vessel Cooling Analysis

Flowrate lbm/s	Flowrate kg/s	Insulation thickness inches	forced flow V m/s	heat transfer coefficient $W/m^2\,°C$	heat into flow MW	heat lost to ambient MW	max. core passage T ºC	max. IHX wall T ºC	He flow out T ºC
20.	9.1	0.0	0.84	47.9	3.64	1.76	469.	322.	326.5
20.	9.1	1.0	0.84	48.3	4.55	0.27	472.	386.	345.5
20.	9.1	2.0	0.80	48.3	4.62	0.15	473.	391.	347.0
55.	25.0	1.0	2.30	106.8	8.28	0.21	456.	337.	313.2
55.	25.0	2.0	2.30	106.8	8.36	0.12	456.	339.	313.9
200.	90.7	1.0	8.35	295.5	15.49	0.17	419.	289.	282.5
200.	90.7	2.0	8.35	295.6	15.56	0.09	419.	290.	282.8

This analysis indicates that the required power to cool one IHX vessel would be between 5 to 8 MW which is considered as too significant. The alternative with insulation on both inside and outside the IHX vessel would be therefore preferable.

7.6 Conclusions Regarding SA-508/SA-533 Alternatives

Several options have been identified and investigated to enable the use of SA 508 material. The conclusions can be summarized as follows:

- The current RPV design can be considered acceptable using SA-508/SA-533 without design modifications up to a power level of 600 MWth and a core inlet temperature of 400°C.
- The implementation of a thermal insulation at the outer surface of the core barrel seems difficult to optimize and results in an unacceptable temperature for the core barrel.
- The alternative with a thermal shield provides promising results, even though further refinement would still be required.
- The implementation of active cooling for the RPV could be achieved with a limited impact in terms of overall plant efficiency. Such a cooling system would have no effect on temperatures reached during DCC situations, but vessel temperatures would be acceptable.

For the IHX vessel,

- The implementation of an active cooling of the IHX vessels would have a large impact on the efficiency.
- The option based on insulation on both inside and outside the IHX vessel would be preferable.

Thus, for systems with operating temperatures of 400°C (core inlet) and 800°C (core outlet), an SA-508/SA-533 vessel is a clear option.

For higher temperature operation, feasible alternatives appear to be available to allow the use of an SA-508/SA-533 vessel. However, whether these options are preferable to a vessel made of a higher temperature alloy remains to be determined. This question depends foremost on the availability of such a vessel. If a high temperature vessel such as modified 9Cr-1Mo is available, that would be a simpler option which would avoid the added complexity of the alternatives explored in this section. On the other hand, if such a vessel is not available, then these solutions may represent the only option.

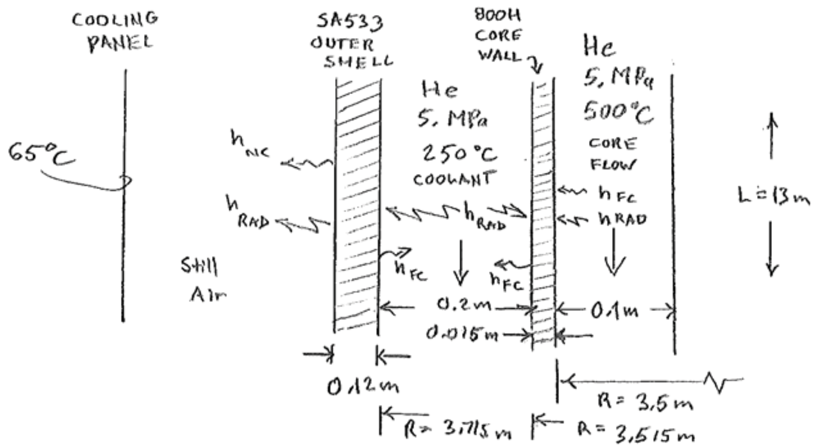


Figure 7-6: HTR Vessel Cooling Option

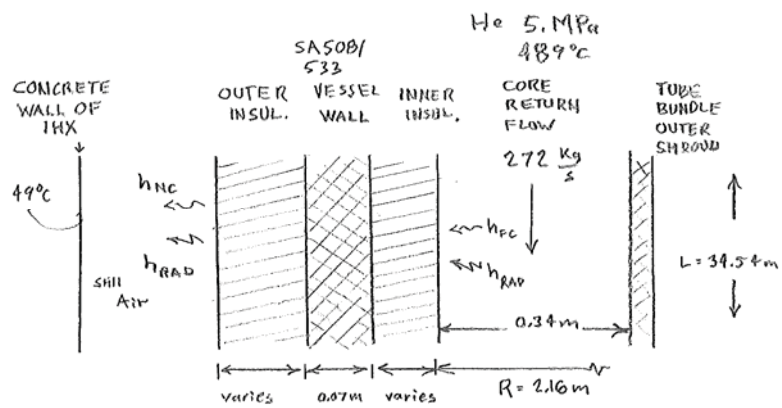


Figure 7-7: IHX vessel Insulation Option

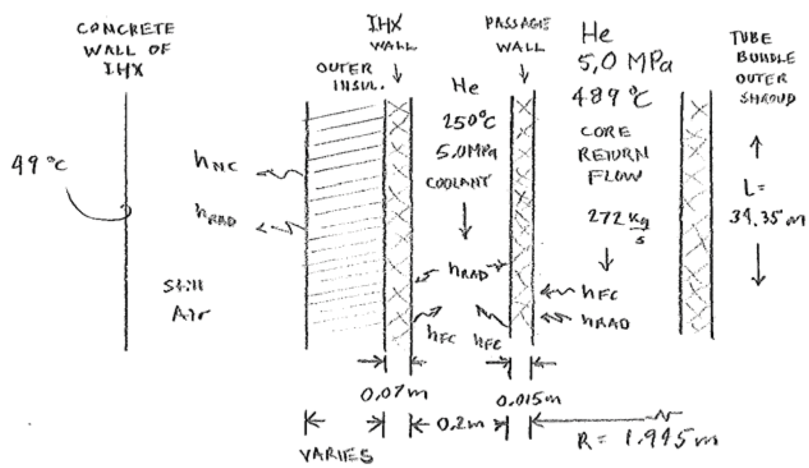


Figure 7-8: IHX vessel Insulation and Cooling Option

8.0 PROPOSED FUTURE STUDIES

The following lists items identified in the context of this work which would need to be further studied.

- Identify alternative forging suppliers for mod 9Cr1Mo
- Identify limitations for fabricating plates out of mod 9Cr1Mo
- Assess as to whether the expected reduction of mechanical properties of heavy section products made of 2.25Cr1Mo could be likely to rule out this candidate.
- Perform a detailed assessment of the RPV fabrication issues and a coolant pressure trade study considering circulator feasibility, circulator power, RPV feasibility, and RPV cost.
- Perform detailed transient analyses and Finite Element calculations for the RPV, cross vessel and IHX vessel to confirm preliminary assessments
- Evaluate the practicality of implementing thermal insulation or thermal shielding to enable the use of SA508/533 material.

9.0 CONCLUSIONS

This study has evaluated alternatives for the Reactor Pressure Vessel (RPV) materials and design, the cross vessel, and IHX pressure vessel materials considering the range of potential design and initial operating conditions for NGNP.

The main issue associated to material candidates is linked to procurement. Whatever material is selected, the design of the Reactor Pressure Vessel will have to be made out of plates to be consistent with 2018 schedule. The few remaining forgings could be provided by JSW in time for start-up by 2018, subjected that the corresponding forgings could be switched with slots currently under negotiation at the time of the present report. Otherwise, a minimum two years delay for start-up should be anticipated, subjected to taking a decision in the very near term to reserve forging slots.

Procurement issues have been identified with mod 9Cr1Mo and it is recommended to pursue investigations to clarify, if this option is still viable. 2.25Cr 1Mo annealed (grade 22) could also be envisioned for the “hot” vessel option, but this material requires to increase the thickness by about 150% compared to other candidates. It must be clarified if mechanical properties expected for the thicker parts (flanges and nozzle ring) would still be acceptable.

2.25Cr 1Mo V is also considered as a good candidate for such an application, with expected reduced feasibility issues for welding compared to mod 9Cr1Mo. However, the time required to qualify it for the NGNP is not expected to be consistent with NGNP schedule.

No procurement issue has been identified with the PWR grade (SA 508 / SA 533 grades) and this material could be procured in the required dimensions for the NGNP.

Design alternatives have been identified and potential suppliers listed. Japan Steel Works (JSW) is confirmed to be the only supplier capable of providing the large forgings necessary for the RPV, but its present capabilities do not permit JSW to fabricate forgings made out of mod 9Cr1Mo for the dimensions required for the NGNP.

This study also identifies other fabricability issues, required Codes and Standards modifications and discusses In-Service Inspection requirements.

Preliminary stress analyses have been performed and indicate that a refined assessment of the IHX vessel would be required to confirm the current sizing.

This study finally identifies and evaluates the conditions under which the PWR grade can be used. It is shown that:

- The current RPV design can be considered acceptable using SA-508/SA-533 without design modifications up to a power level of 600 MWth and a core inlet temperature of 400°C.

- For higher temperature operation, feasible alternatives (active cooling or implementation of a thermal shielding) appear to be available to allow the use of an SA-508/SA-533 vessel. However, whether these options are preferable to a vessel made of a higher temperature alloy remains to be determined.

10.0 REFERENCES

1- Doc. 12-9051191-001 NGNP with Hydrogen Production Pre-conceptual Design Studies Report

2- Doc. 12-9076325-001 NGNP - IHX and Secondary Heat Transport Loop Alternatives

3- Doc. 12-9040130-001 Improvement of ASME NH for grade 91 (negligible creep)

4- Doc. 12-9076324-001 Proposed Test Program to assess negligible creep conditions of modified 9Cr1Mo grade

5- Doc. 12-9045964-001 Improvement of ASME NH for grade 91 (creep-fatigue)

6- Doc. 12-9061054-000 Proposed test program to validate creep-fatigue procedures for modified 9Cr1Mo

7- NRC -AERB Nuclear Safety Projects Meeting 2004, August 30 - September 3 NRC prospective on risk-informed In Service Inspection

8- ASME 1992 Boiler and pressure vessel code. Rules for In Service Inspection Section XI (division1 – articles IW) and rules for inspection and testing of components for gas cooled plants (division 2 – Articles IG)

9- Safety guide IAEA n^r 50 - 56 - 02 (1980) In Service Inspection of Nuclear plant.

10- GA-A25401 - H2-MHR Pre-Conceptual Design - Report: SI-Based Plant - April 2006

APENDIX A: DRAFT FORGING SPECIFICATION FOR NGNP VESSEL MODIFIED 9CR1MO SA-336 GRADE F91

Contents

1. TECHNICAL PROCUREMENT CONDITIONS

The manufacturing, tests and inspections of the NGNP forgings in SA-336 Grade F91 Grade 3 alloy steel shall be completely in accordance with all the requirements defined in this specification.

NOTE: Any conflict between this specification and the applicable Codes and Standards must be brought to the attention of AREVA NP for clarification prior to start of manufacturing.

1.1 APPLICABLE CODE

The forgings shall be in accordance with the following:

ASME Boiler and Pressure Vessel Code, 2001 Edition with Addenda through 2003:
Section III division 1 subsection NB article NB-2000.
SA-336 specification of section II part A including the following supplementary requirements:

1) Simulated Postweld Heat Treatment of test coupons.

2) Ultrasonic Testing-reference Block Calibration
3) Charpy- Notch Impact Transition Curve
4) Restrictive Chemistry (Modified)

1.2 ASTM STANDARDS

The standards shall be used at the latest applicable edition unless otherwise specified in the article NB-2000 of the ASME Code Section III division I subsection NB for which it shall be applied the applicable edition required by this article.

2. MELTING PROCESS

The steel shall be made in a basic electric furnace, vacuum degassed and fully killed.
The melting process to be used by the Forging Supplier shall be indicated in the Technical Manufacturing Program.

3. CHEMICAL REQUIREMENTS

The heat and product analyses shall comply with the following requirements. The other elements constitutive of the steel not listed in the table must be considered as residuals.

Product analysis shall be performed on each individual forging at locations to be indicated in the Technical Manufacturing Program, one being made on a sample taken from the top and the other on a sample taken from the bottom of the ingot. These analyses may be made on mechanical test specimen discards.

The Forging Supplier must ensure an hydrogen content lower than or equal to a proposed limit after completion of the forging and before the first cooling down to room temperature. The proposed and guaranteed hydrogen content limit shall be demonstrated and documented by the Forging Supplier.

Table 1 : Specified Chemical Analyses

Chemical Requirements	Heat Analysis		Product Analysis	
	Min %	**Max %**	**Min %**	**Max %**
Carbon	0.08	0.12	0.08	0.12
Manganese	0.30	0.60	0.30	0.60
Phosphorus		0.012		0.015
Sulfur		0.005		0.005
Silicon	0.20	0.50	0.20	0.50
Nickel		0.40		0.40
Chromium	8.00	9.50	8.00	9.50
Molybdenum	0.85	1.05	0.85	1.05
Vanadium	0.18	0.25	0.18	0.25
Columbium	0.06	0.10	0.06	0.10
Nitrogen	0.03	0.07	0.03	0.07
Aluminum		0.040		0.040
Copper		0.1		0.1
Cobalt, Tin, Arsenic		Info		Info

4. MANUFACTURING

Prior to manufacturing, the Forging Supplier shall submit a Technical Manufacturing Program including at least the following data to the AREVA NP approval:
> Steel making and ingot pouring process,
> Ingot size, weight and discards,
> Sketch of the forging after each forging operation with the achieved dimensions and the indication of the main working direction,
> Heat treatment cycles applied to the forging, with the dimensions at these stages,

>The locations of the thermocouples on the forging during Heat Treatment for Mechanical Properties,
> Dimensions, configuration and metallurgical condition of forging material presented for ultrasonic testing (preliminary and final)
> Sketch showing the test coupons on the forging,
> Sketch showing the test specimens on the test coupons.

After the final Quenching and Tempering or Normalization and Tempering, only machining and grinding method are authorized on the forging, the thermal processes are forbidden to remove extra-material, or it shall be used thermal processes by air-arc and/or oxygen cutting and a minimum of 1/16” of the cut surface shall be removed by a mechanical method.

During heat treatments, if fuel is used, it shall not contain more than 0.45% sulfur by weight for oil or 15 grains per 100 ft^3 for gas.

5. HEAT TREATMENTS

Table 2 : Heat treatment conditions

TREATMENT	STAGE	CHARACTERISTICS
HEAT TREATMENT FOR MECHANICAL PROPERTIES (HTMP)	After preliminary heat treatment and in any case prior to test coupons removal	- Austenitizing at a temperature in the range of 1900 to 1960°F, producing an austenitic structure with a sufficient holding time to have an homogeneous temperature throughout the part. - Water quenching by immersion. or Normalization - Tempering to a temperature greater than or equal to 1390°F held for at least 1 hour per inch of maximum thickness to be heat treated.
SIMULATED POSTWELD HEAT TREATMENT (SPWHT)	Performed on the test coupons after they are taken from the part see § 7 of this specification.	Test coupons shall undergo the following: - Heating rate above 800°F: ≤ 100°F/hr - Holding temperature 1375°F to 1400°F for 20 hours - Cooling rate down to 800°F ≤ 100°F/hr - Still air cooling under 800°F

Heat treatment procedures, as minimum, shall specify holding time, temperature, heating and cooling rates, heating method, temperature distribution, and location of the thermocouples.

6. STRUCTURE AND GRAIN SIZE

A micrographic examination with photographs shall be performed on the part in each of the test coupons. The austenitic grain size index shall be determined for information purposes.

7. MECHANICAL PROPERTIES

Table 3 : Specified Mechanical Properties

TYPE OF TEST	TEMPERATURE	PROPERTIES	Min	Max	Unit
DROP WEIGHT (1)	/	TNDT		-15	°F
CHARPY V IMPACT	TNDT+ 60°F	Energy - Single value	50		ft-lbs
CHARPY V IMPACT	TNDT+ 60°F	Lateral Expansion	35		Mils
RTNDT DETERMINATION (1)	/	RTNDT		-15	°F
CHARPY V IMPACT	-5°F on axial and circumferential specimens	Average energy value (3 specimens) Energy-Single value	30 21		ft-lbs
CHARPY V IMPACT	30°F on axial and circumferential specimens	Average energy value (3 specimens) Energy-Single value	45 30		ft-lbs
CHARPY V IMPACT	on the upper shelf (100% shear) on axial and circumferential specimens	Energy-Single value	75		ft-lbs
TENSILE TEST	ROOM	Tensile strength (S_u)	85	110	Ksi
TENSILE TEST	ROOM	Yield strength (S_y)	60		Ksi
TENSILE TEST	ROOM	Elongation on 2 inches	20		%
TENSILE TEST	ROOM	Reduction of area	40		%
TENSILE TEST	800°F	Tensile strength (S_u)	66		Ksi
TENSILE TEST	800°F	Yield strength (S_y)	52		Ksi
TENSILE TEST	800°F	Elongation on 2 inches		Info	%
TENSILE TEST	800°F	Reduction of area		Info	%

(1) The actual value of TNDT and RTNDT shall be determined.

8. TEST COUPONS AND SPECIMENS REMOVAL

The test coupons shall be removed as per the subparagraph NB-2223. of the ASME Code section III subsection NB :

- In the RPV beltline forgings, the specimen axis position should be t/4 from the heat treated surface, t being the wall thickness of the heat treated forging.

- In large and complex forgings (RPV nozzles, RPV flanges), specimen axis should be at least at 19 and 38 mm from the two nearest heat treated surfaces.

Test specimens for tension tests, for full Charpy V impact curves and Drop weight tests shall be machined in two test coupons removed from each of 2 locations at the end of the forging corresponding to the bottom of the ingot and 180^0 apart. They are identified as X and Y in § 9 and table 4 of this specification. If the forging height exceeds 3.7 m. X and Y coupons shall be taken at each end of the forging. Specimens in the test coupons will be oriented in axial or circumferential directions as required in § 9 and table 4 of this specification.

Sketches showing the exact locations shall be approved by the Purchaser as part of the Technical Manufacturing Program.

X and Y test coupons shall be able to provide all specimens necessary to perform the series of tests indicated in § 9 of this specification.

9. NUMBER OF MECHANICAL TESTS

9.1 Tests on X and Y coupons in as received condition

The series of tests in as received condition (quenched and tempered or normalized and tempered) which shall be performed on X and Y coupons comprise:

- Circumferential tensile test at room temperature

- Circumferential tensile test at elevated temperature (800°F)

- Axial Charpy V impact test at -5°F

- Axial Charpy V impact test at 30°F

- Axial Charpy V impact test on the upper shelf (100% shear).

9.2 Tests on X coupon in as received condition and after simulated post weld heat treatment

X coupon is the test coupon where lower impact values at 30°F will be found. The following complementary test will be performed on X test coupon:

- in as received conditions

  - Circumferential Charpy V impact test at -5°F

  - Circumferential Charpy V impact test at 30°F

- Circumferential Charpy V impact test on the upper shelf (100% shear).

- after simulated post weld heat treatment

  - Circumferential tensile test at room temperature
  - Circumferential tensile test at elevated temperature (800°F)
  - Circumferential Charpy V impact test at 30°F
  - Axial Charpy V impact test at 30°F
  - Charpy V transition curve (axial orientation)
  - Drop weight TNDT (axial orientation)
  - RTNDT determination (axial orientation)
  - Drop weight TNDT (circumferential orientation)
  - RTNDT determination (circumferential orientation)

10. TESTS AND INSPECTIONS

Table 4 : Specified Tests and Inspections

TYPE	STAGE	METHOD	CRITERION
HEAT ANALYSIS	During steel melting and ingot pouring	As per § 7.1 of SA-336	As per § 3 of this specification
PRODUCT ANALYSES	During mechanical testing, on bottom and head of ingot	As per § 7.2 of SA-336	As per § 3 of this specification
GRAIN SIZE	During mechanical testing on X and Y coupons	As per § 6 of this specification	For information
ROOM TENSILE TEST	On as received X and Y test coupons and as per § 7 and 9.1 of this specification	As per § 8.1.1 of SA-336	As per § 7 of this specification
ROOM TENSILE TEST	After SPWHT of X test coupons and as per § 7 and 9.2 of this specification	As per § 8.1.1 of SA-336	As per § 7 of this specification

800°F TENSILE TEST	On as received X and Y test coupons and as per § 7 and 9.1 of this specification	As per § 8.1.1 of SA-336	As per § 7 of this specification
800°F TENSILE TEST	After SPWHT of X test coupons and as per § 7 and 9.2 of this specification	As per § 8.1.1 of SA-336	As per § 7 of this specification
-5°F CHARPY V	On as received X and Y test coupons and as per § 7 and 9.1 of this specification (axial orientation)	As per § 8.3 of SA-336	As per § 7 of this specification
30°F CHARPY V	On as received X and Y test coupons and as per § 7 and 9.1 of this specification (axial orientation)	As per § 8.3 of SA-336	As per § 7 of this specification
UPPER SHELF CHARPY V	On as received X and Y test coupons and as per § 7 and 9.1 of this specification (axial orientation)	As per § 8.3 of SA-336	As per § 7 of this specification
-5°F CHARPY V	On as received X test coupons and as per § 7 and 9.2 of this specification (circumferential orientation)	As per § 8.3 of SA-336	As per § 7 of this specification
30°F CHARPY V	On as received X test coupons and as per § 7 and 9.2 of this specification (circumferential orientation)	As per § 8.3 of SA-336	As per § 7 of this specification

UPPER SHELF CHARPY V	On as received X test coupons and as per § 7 and 9.2 of this specification	As per § 8.3 of SA-336	As per § 7 of this specification
30°F CHARPY V	After SPWHT of X test coupons and as per § 7 and 9.2 of this specification (circumferential orientation)	As per § 8.3 of SA-336	As per § 7 of this specification
30°F CHARPY V	After SPWHT of X test coupons and as per § 7 and 9.2 of this specification (axial orientation)	As per § 8.3 of SA-336	As per § 7 of this specification
CHARPY V IMPACT TRANSITION CURVE	After SPWHT of X test coupon and in axial orientation	NB-2300 and § 7 of this specification	The transition curve shall be plotted from six sets of specimens. One set shall be on the lower shelf (10% shear), one set on the upper shelf (100% shear), one set at T_{NDT} + 60°F and the three remaining sets as necessary to develop the optimum transition curve.
RTNDT DETERMINATION	After SPWHT of X test coupon, per § 8 of this specification and in circumferential orientation	As per NB-2300	As per § 7 of this specification
RTNDT DETERMINATION	After SPWHT of X test coupon, per § 8 of this specification and in axial orientation	As per NB-2300	As per § 7 of this specification
MAGNETIC PARTICLE EXAMINATION	After machining	As per NB-2545 (only yoke method shall be used)	As per NB-2545

ULTRASONIC EXAMINATION	After final machining or at I stage as advanced as possible for the areas which cannot be examined at this stage.	As per Attachment A of this specification	As per Attachment A of this specification
DIMENSIONAL AND VISUAL INSPECTION	After final machining	/	As per procurement drawing given in the purchase order

11. DEFECTIVE AREA REMOVAL

The surface defects shall be removed in accordance with the paragraph NB-2538 of the ASME Code section Ill division 1 Article NB-2000.

Minor surface defects may be removed by grinding without informing AREVA NP to produce a smooth transition, provided the dimensions thereby reduced are not under the minimum specified dimensions.

No repair by welding shall be allowed at the Forging Supplier's shop.

After grinding a magnetic particle examination shall be performed according to the paragraph 'TESTS AND INSPECTIONS" of this specification.

12. ARCHIVE MATERIALS

The Supplier shall deliver, at the same time as the forgings:

- Archive material for each forgings which will be the balance of the material not used for mechanical testing. These archive materials shall be delivered in the Heat Treatment condition as the forgings and shall be NDE examined in the same manner. The archive material shall be marked in accordance with paragraph MARKING of this specification and in the same condition as the forgings.

- All prolongation of the NGNP forgings not machined into test specimens shall have received the same heat treatment as the as received forgings (SPWHT shall not be performed).

Archive material and prolongation will be cut by mechanical means. They shall be marked in such a way that the orientation in the forging before cutting can be identified.

13. MARKING

Marking shall be performed with a low stress “blunt-nosed continuous” or “blunt-nosed interrupted dot die stamp” as per material specification SA-336 and NB-2150 of ASME Code section III.

The marking shall be performed on the locations required on the procurement drawing given in the order and shall include at least the following information:

- The supplier name or symbol
- The Heat Number
- Grade
- AREVA NP Purchase Order N°
- Item codification of the part given in the Purchase Order
- Equipment number given in the Purchase Order

14. CLEANLINESS - PACKING - TRANSPORTATION

Mercury or mercury compound-containing instruments shall not be used for any purposes during fabrication, assembly, testing or packaging.

Every effort shall be made to prevent lead, sulfur, zinc, mercury and other low melting point metals or halogens, or materials containing theses elements, coming in contact with the closure head forgings. Such contaminants which are unavoidably present shall be removed prior to heat treatments and hot forming. A list of expendable materials to be used by the Supplier shall be submitted to AREVA NP approval prior to use.

The final cleaned surfaces of the forgings shall be free of the aforementioned o compounds.

The forgings shall be adequately packaged protection from damage and environmental exposure during shipping and handling.

The Supplier shall provide for AREVA NP to approve the details of preservation and packing methods prior to shipping.

15. REPORTS

The Supplier shall deliver a Certified Material Test Report for each forging. This CMTR shall be written as per NB-2130 of ASME Code section III.

The CMTR shall include all the following test reports:

- heat and product analyses,
- heat treatment records (showing the complete time-temperature cycle) and analysis of the heat treatment diagrams (for preliminary heat treatment, heat treatment for mechanical properties and simulated post-weld heat treatment),
- Certified copy of actual heat treatment charts
- destructive tests
- non destructive examinations
- dimensionnel checks

The Supplier shall provide high resolution digital photographs of the forgings at the following stages of production: Forging, Normalizing, Rough machining, Quenching, Final Machining, Packing and Transportation.

ATTACHMENT A: ULTRASONIC EXAMINATION

The part shall be examined by UT inspection as per:
> The paragraph NB-2542 of the ASME Code Section III division 1 Article NB-2000.
> The section 7.3 and supplementary requirement S2 of the SA-508 specification of the ASME Code Section II Part A.
and the following additional requirements:

A-PROCEDURE

The Supplier shall submit the UT procedure to the AREVA NP approval.

B - STAGE OF EXAMINATION

The UT shall be performed on the part after machining at the delivery configuration.
However for areas which cannot be examined in the final configuration UT may be performed at one stage as advanced as possible. In this case these particular areas and the relevant stage shall be clearly identified in the UT procedure.

C - TRANSDUCER FREQUENCIES

- Straight beam: 4 MHz

- 45° angle beam: 2MHz

However, if the grain structure does not permit good results lower frequencies can be used with AREVA NP agreement. In no case may the frequencies be lower than:
- 2 1/4 MHZ for straight beam transducers
- 1 MHz for angle beam transducers.

D - ADDITIONAL REQUIREMENTS

The supplier shall set-up its UT equipment, for straight beam method only, in order to detect any indication with equivalent diameter of 0.12” or greater. All indications greater than or equal to (0.12”) shall be recorded.

APENDIX B: DRAFT FORGING SPECIFICATION FOR NGNP VESSEL SA- 508 GRADE 3 CLASS 1

Contents

1. TECHNICAL PROCUREMENT CONDITIONS

The manufacturing, tests and inspections of the NGNP reactor vessel forgings in SA-508 Grade 3 Class 1 (fine grain practice) low alloy steel shall be completely in accordance with all the requirements defined in this specification.

NOTE: Any conflict between this specification and the applicable Codes and Standards must be brought to the attention of AREVA NP for clarification prior to start of manufacturing.

1.1 APPLICABLE CODE

The NGNP forgings shall be in accordance with the following:

ASME Boiler and Pressure Vessel Code, 2001 Edition with Addenda through 2003:
Section III division 1 subsection NB article NB-2000.
SA-508 specification of section II part A including the following supplementary requirements:
S1 Simulated Postweld Heat Treatment of test coupons.
S2 Ultrasonic Testing-reference lock Calibration
S3 Charpy- Notch Impact Transition Curve
S4 Additional Charpy Data
S9 Restrictive Chemistry (Modified)

S10 Alternative fracture toughness test

S13 Minimum tempering temperature
S15 Product Analysis

1.2 ASTM STANDARDS

The standards shall be used at the latest applicable edition unless otherwise specified in the article NB-2000 of the ASME Code Section III division I subsection NB for which it shall be applied the applicable edition required by this article.

2. MELTING PROCESS

The steel shall be made in a basic electric furnace, vacuum degassed and fully killed.
The melting process to be used by the Forging Supplier shall be indicated in the
Technical Manufacturing Program.

3. CHEMICAL REQUIREMENTS

The heat and product analyses shall comply with the following requirements. The other elements constitutive of the steel not listed in the table must be considered as residuals.

Product analysis shall be performed on each individual forging at locations to be indicated in the Technical Manufacturing Program, one being made on a sample taken from the top and the other on a sample taken from the bottom of the ingot. These analyses may be made on mechanical test specimen discards.

The Forging Supplier must ensure an hydrogen content lower than or equal to proposed limit after completion of the forging and before the first cooling down to room temperature. The proposed and guaranteed hydrogen content limit shall be demonstrated and documented by the Forging Supplier.

Table 1 : Specified Chemical Analyses

Chemical Requirements	Heat Analysis		Product Analysis	
	Min %	Max %	Min %	Max %
Carbon		0.20		0.22
Manganese	1.20	1.50	1.15	1.60
Phosphorus		0.008		0.008
Sulfur		0.005		0.005
Silicon	0.15	0.40	0.15	0.40
Nickel	0.40	1.00	0.37	1.03
Chromium		0.25		0.25
Molybdenum	0.45	0.60	0.43	0.62
Vanadium		0.010		0.010
Copper		0.10		0.10
Aluminum		0.040		0.040
Cobalt		0.03		0.03

4. MANUFACTURING

Prior to manufacturing, the Forging Supplier shall submit a Technical Manufacturing Program including at least the following data to the AREVA NP approval:
> Steel making and ingot pouring process,
> Ingot size, weight and discards,
> Sketch of the forging after each forging operation with the achieved dimensions and the indication of the main working direction,
> Heat treatment cycles applied to the forging, with the dimensions at these stages,

>The locations of the thermocouples on the forging during Heat Treatment for Mechanical Properties,
> Dimensions, configuration and metallurgical condition of forging material presented for ultrasonic testing (preliminary and final)
> Sketch showing the test coupons on the forging,
> Sketch showing the test specimens on the test coupons.

After the final Quenching and Tempering, only machining and grinding method are authorized on the forging, the thermal processes are forbidden to remove extra-material, or it shall be used thermal processes by air-arc and/or oxygen cutting and a minimum of 1/16" of the cut surface shall be removed by a mechanical method.

During heat treatments, if fuel is used, it shall not contain more than 0.45% sulfur by weight for oil or 15 grains per 100 ft^3 for gas.

5. HEAT TREATMENTS

Table 2 : Heat treatment conditions

TREATMENT	STAGE	CHARACTERISTICS
HEAT TREATMENT FOR MECHANICAL PROPERTIES (HTMP)	After preliminary heat treatment and in any case prior to test coupons removal	- Austenitizing at a temperature in the range of 1560 to 1700°F producing an austenitic structure with a sufficient holding time to have a homogeneous temperature throughout the part. - Water quenching by immersion. - Tempering to a temperature greater than or equal to 1175°F held for at least 1/2 hour per inch of maximum thickness to be heat treated.
SIMULATED POSTWELD HEAT TREATMENT (SPWHT)	Performed on the test coupons after they are taken from the part see § 7 of this specification.	Test coupons shall undergo the following: - Heating rate above 800°F: ≤ 100°F/hr - Holding temperature:1103°F to 1148°F - Holding time: 16 hr - Cooling rate down to 800°F ≤ 100°F/hr - Still air cooling under 800°F

Heat treatment procedures, as minimum, shall specify holding time, temperature, heating and cooling rates, heating method, temperature distribution, and location of the thermocouples.

6. STRUCTURE AND GRAIN SIZE

A micrographic examination with photographs shall be performed on the part in each of the test coupons. The austenitic grain size index shall be equal or greater than 5.

7. MECHANICAL PROPERTIES

Table 3 : Specified Mechanical Properties

TYPE OF TEST	TEMPERATURE	PROPERTIES	Min	Max	Unit
DROP WEIGHT (1)	/	TNDT		-20	°F
CHARPY V IMPACT	TNDT+ 60°F	Energy - Single value	50		ft-lbs
CHARPY V IMPACT	TNDT+ 60°F	Lateral Expansion	35		Mils
RTNDT DETERMINATION (1)	/	RTNDT		-20	°F
CHARPY V IMPACT	-5°F on Axial specimens	Average energy value (3 specimens) Energy-Single value	42 30		ft-lbs
CHARPY V IMPACT	-5°F on Circumferential specimens	Average energy value (3 specimens) Energy-Single value	30 21		ft-lbs
CHARPY V IMPACT	30°F on Axial specimens	Average energy value (3 specimens) Energy-Single value	60 42		ft-lbs
CHARPY V IMPACT	30°F on Circumferential specimens	Average energy value (3 specimens.) Energy-Single value	60 42		ft-lbs
CHARPY V IMPACT	on the upper shelf (100% shear) on axial and circumferential specimens	Energy-Single value	75		ft-lbs
TENSILE TEST	ROOM	Tensile strength (S_u)	80	105	Ksi
TENSILE TEST	ROOM	Yield strength (S_y)	50		Ksi
TENSILE TEST	ROOM	Elongation on 2 inches	18		%
TENSILE TEST	ROOM	Reduction of area	38		%
TENSILE TEST	660°F	Tensile strength (S_u)	72		Ksi
TENSILE TEST	660°F	Yield strength (S_y)	44		Ksi
TENSILE TEST	660°F	Elongation on 2 inches		Info	%
TENSILE TEST	660°F	Reduction of area		Info	%

(1) The actual value of TNDT and RTNDT shall be determined.

8. TEST COUPONS AND SPECIMENS REMOVAL

The test coupons shall be removed as per the subparagraph NB-2223 of the ASME Code section III subsection NB:

- In the RPV beltline forgings, the specimen axis position should be t/4 from the heat treated surface, t being the wall thickness of the heat treated forging.

- In large and complex forgings (RPV nozzles, RPV flanges), specimen axis should be at least at 19 and 38 mm from the two nearest heat treated surfaces.

Test specimens for tension tests, full Charpy V impact curves and Drop weight tests shall be machined in two test coupons removed from each of 2 locations at each end of the forging corresponding to the bottom of the ingot and 180^0 apart. They are identified as X and Y in § 9 and table 4 of this specification. If the forging height exceeds 3.7 m. X and Y coupons shall be taken at each end of the forging. Specimens in the test coupons will be oriented in axial or circumferential directions as required in § 9 and table 4 of this specification

Sketches showing the exact locations shall be approved by the Purchaser as part of the Technical Manufacturing Program.

X and Y test coupons shall be able to provide all specimens necessary to perform the series of tests indicated in § 9 of this specification.

The specimen for grain size determination shall be taken in the prolongation of the tensile test specimen.

9. NUMBER OF MECHANICAL TESTS

9.1 Tests on X and Y coupons in as received condition

The series of tests in as received condition (quenched and tempered or normalized and tempered) which shall be performed on X and Y coupons comprise:

- Circumferential tensile test at room temperature

- Circumferential tensile test at elevated temperature (800°F)

- Axial Charpy V impact test at -5°F

- Axial Charpy V impact test at 30°F

- Axial Charpy V impact test on the upper shelf (100% shear).

9.2 Tests on X coupon in as received condition and after simulated post weld heat treatment

X coupon is the test coupon where lower impact values at 30°F will be found. The following complementary test will be performed on X test coupon:

- in as received conditions

    - Circumferential Charpy V impact test at -5°F

- Circumferential Charpy V impact test at 30°F
- Circumferential Charpy V impact test on the upper shelf (100% shear).

- after simulated post weld heat treatment
  - Circumferential tensile test at room temperature
  - Circumferential tensile test at elevated temperature (800°F)
  - Circumferential Charpy V impact test at 30°F
  - Axial Charpy V impact test at 30°F
  - Charpy V transition curve (axial orientation)
  - Drop weight TNDT (axial orientation)
  - RTNDT determination (axial orientation)
  - Drop weight TNDT (circumferential orientation)
  - RTNDT determination (circumferential orientation)

10. TESTS AND INSPECTIONS

Table 4 : Specified Tests and Inspections

TYPE	STAGE	METHOD	CRITERION
HEAT ANALYSIS	During steel melting and ingot pouring	As per § 5.1 of SA-508	As per § 3 of this specification
PRODUCT ANALYSES	During mechanical testing, on bottom and head of ingot	As per § 5.2 of SA-508	As per § 3 of this specification
GRAIN SIZE	During mechanical testing on X and Y coupons	ASTM E 112	Grain size ASTM 5 or finer
ROOM TENSILE TEST	On as received X and Y test coupons and as per § 7 and 9.1 of this specification	As per § 6.1 of SA-508	As per § 7 of this specification
ROOM TENSILE TEST	After SPWHT of X test coupons and as per § 7 and 9.2 of this specification	As per § 6.1 of SA-508	As per § 7 of this specification

660°F TENSILE TEST	On as received X and Y test coupons and as per § 7 and 9.1 of this specification	As per § 6.1 of SA-508	As per § 7 of this specification
660°F TENSILE TEST	After SPWHT of X test coupons and as per § 7 and 9.2 of this specification	As per § 6.1 of SA-508	As per § 7 of this specification
-5°F CHARPY V	On as received X and Y test coupons and as per § 7 and 9.1 of this specification (axial orientation)	As per § 6.2 of SA-508	As per § 7 of this specification
30°F CHARPY V	On as received X and Y test coupons and as per § 7 and 9.1 of this specification (axial orientation)	As per § 6.2 of SA-508	As per § 7 of this specification
UPPER SHELF CHARPY V	On as received X and Y test coupons and as per § 7 and 9.1 of this specification (axial orientation)	As per § 6.2 of SA-508	As per § 7 of this specification
-5°F CHARPY V	On as received X test coupons and as per § 7 and 9.2 of this specification (circumferential orientation)	As per § 6.2 of SA-508	As per § 7 of this specification
30°F CHARPY V	On as received X test coupons and as per § 7 and 9.2 of this specification (circumferential orientation)	As per § 6.2 of SA-508	As per § 7 of this specification
UPPER SHELF CHARPY V	On as received X test coupons and as per § 7 and 9.2 of this specification	As per § 6.2 of SA-508	As per § 7 of this specification
30°F CHARPY V	After SPWHT of X test coupons and as per § 7 and 9.2 of this specification	As per § 6.2 of SA-508	As per § 7 of this specification

	(circumferential orientation)		
30°F CHARPY V	After SPWHT of X test coupons and as per § 7 and 9.2 of this specification (axial orientation)	As per § 6.2 of SA-508	As per § 7 of this specification
CHARPY V IMPACT TRANSITION CURVE	After SPWHT of X test coupon and in axial orientation	As per supplementary requirement S3 of SA-508 and NB-2300 and § 7 of this specification	The transition curve shall be plotted from six sets of specimens. One set shall be on the lower shelf (10% shear), one set on the upper shelf (100% shear), one set at T_{NDT} + 60°F and the three remaining sets as necessary to develop the optimum transition curve.
RTNDT DETERMINATION	After SPWHT of X test coupon, per § 8 of this specification and in circumferential orientation	As per NB-2300	As per § 7 of this specification
RTNDT DETERMINATION	After SPWHT of X test coupon, per § 8 of this specification and in axial orientation	As per NB-2300	As per § 7 of this specification
MAGNETIC PARTICLE EXAMINATION	After machining	As per NB-2545 (only yoke method shall be used)	As per NB-2545
ULTRASONIC EXAMINATION	After final machining or at I stage as advanced as possible for the areas which cannot be examined at this stage.	As per Attachment A of this specification	As per Attachment A of this specification
DIMENSIONAL AND VISUAL INSPECTION	After final machining	/	As per procurement drawing given in the purchase order

11. <u>DEFECTIVE AREA REMOVAL</u>

The surface defects shall be removed in accordance with the paragraph NB-2538 of the ASME Code section Ill division 1 Article NB-2000.

Minor surface defects may be removed by grinding without informing AREVA NP to produce a smooth transition, provided the dimensions thereby reduced are not under the minimum specified dimensions.

No repair by welding shall be allowed at the Forging Supplier’s shop.

After grinding a magnetic particle examination shall be performed according to the paragraph ‘TESTS AND INSPECTIONS” of this specification.

12. <u>ARCHIVE MATERIALS</u>

The Supplier shall deliver, at the same time as the forgings:

- Archive material for each forgings which will be the balance of the material not used for mechanical testing. These archive materials shall be delivered in the Heat Treatment condition as the forgings and shall be NDE examined in the same manner. The archive material shall be marked in accordance with paragraph MARKING of this specification and in the same condition as the forgings.

- All prolongation of the NGNP forgings not machined into test specimens shall have received the same heat treatment as the as received forgings (SPWHT shall not be performed).

Archive material and prolongation will be cut by mechanical means. They shall be marked in such a way that the orientation in the forging before cutting can be identified.

13. MARKING

Marking shall be performed with a low stress “blunt-nosed continuous” or “blunt-nosed interrupted dot die stamp” as per material specification SA-508 and NB-2150 of ASME Code section III.

The marking shall be performed on the locations required on the procurement drawing given in the order and shall include at least the following information:

- The supplier name or symbol
- The Heat Number
- Grade and Class
- AREVA NP Purchase Order N°
- Item codification of the part given in the Purchase Order
- Equipment number given in the Purchase Order

14. CLEANLINESS - PACKING - TRANSPORTATION

Mercury or mercury compound-containing instruments shall not be used for any purposes during fabrication, assembly, testing or packaging.

Every effort shall be made to prevent lead, sulfur, zinc, mercury and other low melting point metals or halogens, or materials containing theses elements, coming in contact with the closure head forgings. Such contaminants which are unavoidably present shall be removed prior to heat treatments and hot forming. A list of expendable materials to be used by the Supplier shall be submitted to AREVA NP approval prior to use.

The final cleaned surfaces of the forgings shall be free of the aforementioned o compounds.

The forgings shall be adequately packaged protection from damage and environmental exposure during shipping and handling.

The Supplier shall provide for AREVA NP to approve the details of preservation and packing methods prior to shipping.

15. REPORTS

The Supplier shall deliver a Certified Material Test Report for each forging. This CMTR shall be written as per NB-2130 of ASME Code section III.

The CMTR shall include all the following test reports:

- heat and product analyses,
- heat treatment records (showing the complete time-temperature cycle) and analysis of the heat treatment diagrams (for preliminary heat treatment, heat treatment for mechanical properties and simulated post-weld heat treatment),
- Certified copy of actual heat treatment charts
- destructive tests
- non destructive examinations
- dimensional checks

The Supplier shall provide high resolution digital photographs of the forgings at the following stages of production: Forging, Normalizing, Rough machining, Quenching, Final Machining, Packing and Transportation.

ATTACHMENT A: ULTRASONIC EXAMINATION

The part shall be examined by UT inspection as per:
> The paragraph NB-2542 of the ASME Code Section III division 1 Article NB-2000.
> The section 7.3 and supplementary requirement S2 of the SA-508 specification of the ASME Code Section II Part A.
and the following additional requirements:

A-PROCEDURE

The Supplier shall submit the UT procedure to the AREVA NP approval.

B - STAGE OF EXAMINATION

The UT shall be performed on the part after machining at the delivery configuration.
However for areas which cannot be examined in the final configuration UT may be performed at one stage as advanced as possible. In this case these particular areas and the relevant stage shall be clearly identified in the UT procedure.

C - TRANSDUCER FREQUENCIES

- Straight beam: 4 MHz

- 45° angle beam: 2MHz

However, if the grain structure does not permit good results lower frequencies can be used with AREVA NP agreement. In no case may the frequencies be lower than:
- 2 1/4 MHZ for straight beam transducers
- 1 MHz for angle beam transducers.

D - ADDITIONAL REQUIREMENTS

The supplier shall set—up its UT equipment, for straight beam method only, in order to detect any indication with equivalent diameter of 0.12” or greater. All indications greater than or equal to (0.12”) shall be recorded.

APENDIX C: COMMENTS ON DRAFT FORGING SPECIFICATIONS

1. INTRODUCTION

Draft forging specifications provided in Appendices A and B were presented to JSW on April 2008. This appendix is aimed at summarizing the comments made by JSW during this meeting. The preparation of component specifications is usually an iterative process which involves negotiations between the parties involved. The specification should reflect what suppliers would agree to provide for a given cost, taking into account their experience and practices. Further discussions would need to take place to finalize the forging specifications.

2. COMMENTS ON MODIFIED 9CR1MO SPECIFICATION

JSW comments on modified 9Cr1Mo SA -336 grade F91 specification were the following:

- The maximum value of phosphorus content is said to be difficult to achieve and JSW would rather have a value of 0.015% for both heat and products analyses.
- Austenitizing at a temperature in the range of 1900 to 1960°F [1038 to 1071 °C] is considered as a too narrow window. The ASME upper limit is 2000 °F and they would prefer to keep this requirement.
- JSW consider that water quenching is not appropriate for this material and that either oil quenching or normalization should be performed depending on the thickness of the product to be heat treated.
- JSW consider that tempering to a temperature greater than or equal to 1390°F [755°C] is too high and that tempering for half and hour per inch is sufficient.
- Simulated Post Weld Heat Treatment temperature is judge to be too high by JSW and they would recommend 30°C difference between the SPWHT and the tempering temperature.
- A value of TNDT of -30°C can be achieved but JSW would not have enough data to guaranty such a value.
- JSW do not have enough information to guaranty high temperature tensile properties and those values should be specified for information only.
- Drop weight tests should be specified in one direction only.

2. COMMENTS ON SA 508 SPECIFICATION

The only comment made by JSW on SA 508 grade 3 class 1 specification is that they could meet a TNDT value of -30°C but this could be very costly due to the need to use high cost scrap.