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Revision 0

Next Generation Nuclear Plant Reactor Pressure Vessel Acquisition Strategy

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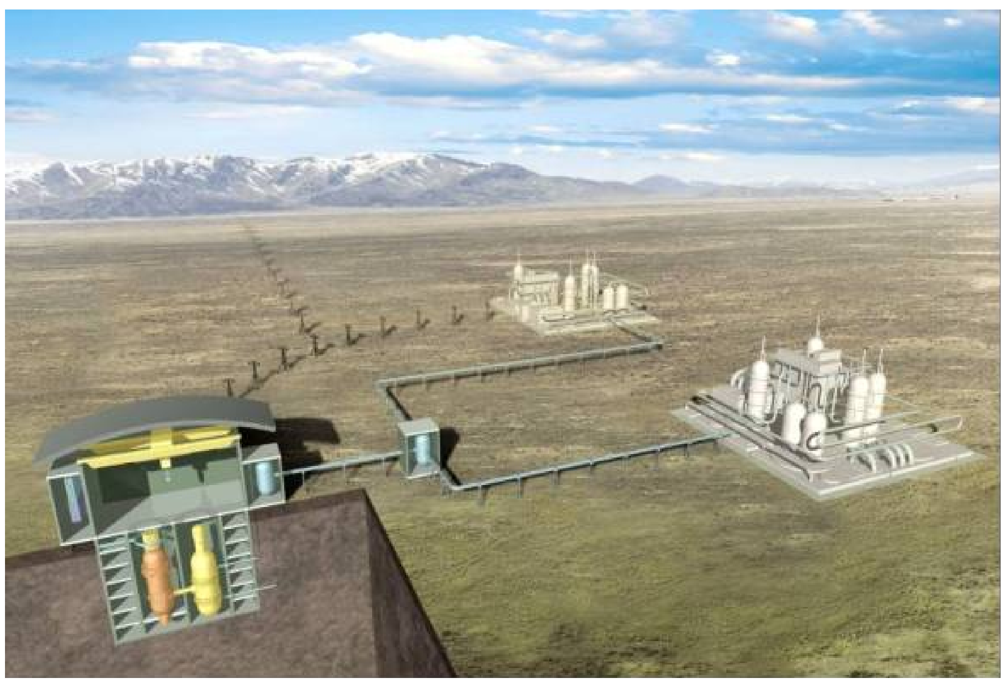
April 2008



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U.S. Department of Energy
Assistant Secretary for the Office of Nuclear Energy
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EXECUTIVE SUMMARY

The Department of Energy has selected the High Temperature Gas-cooled Reactor design for the Next Generation Nuclear Plant (NGNP) Project. The NGNP will demonstrate the use of nuclear power for electricity and hydrogen production. It will have an outlet gas temperature in the range of 850 to 950°C and a plant design service life of 60 years. The reactor design will be a graphite moderated, helium-cooled, prismatic or pebble-bed reactor and use low-enriched uranium, TRISO-coated fuel. The plant size, reactor thermal power, and core configuration will ensure passive decay heat removal without fuel damage or radioactive material releases during accidents.

Selection of the technology and design configuration for the NGNP must consider both the cost and risk profiles to ensure that the demonstration plant establishes a sound foundation for future commercial deployments. The NGNP challenge is to achieve a significant advancement in nuclear technology while at the same time setting the stage for an economically viable deployment of the new technology in the commercial sector soon after 2020.

The purpose of this report is to address the acquisition strategy for the NGNP Reactor Pressure Vessel (RPV). Because of it's size the RPV will most likely be fabricated at the Idaho National Laboratory (INL) site from multiple subcomponent pieces. The pressure vessel steel can either be a conventional material already used in the nuclear industry such as listed within ASME A 508/A533 specifications or it will be fabricated from newer pressure vessel materials never before used for a nuclear reactor in the US. Each of these characteristics will present a procurement and fabrication challenge to the producer of the RPV.

Studies of potential RPV steels have been carried out as part of the pre-conceptual design studies. These design studies generally focus on ASME Code status of the steels, temperature limits and allowable stresses. Three realistic candidate materials have been identified by this process: conventional light water reactor RPV steels A 508/533, 2¼Cr-1Mo in the annealed condition and modified Grade 91 (vanadium modified 9Cr-1Mo ferritic martenistic steel). Based on superior strength and higher temperature limits the Grade 91 has been identified by the majority of design engineers as the preferred choice for the RPV. All of the vendors have concluded, however, that with adequate engineered cooling of the vessel the A 508/533 steels are also acceptable. The 2¼Cr-1Mo steel has insufficient strength at elevated temperature to remain a serious candidate for application in the NGNP RPV.

Discussions with the very few vendors that have the potential to produce large forgings for nuclear pressure vessels indicate a strong preference for conventional A508 steel. This preference is based in part on long experience with forging these steels for nuclear components. It is also based on the inability to cast large ingots of the higher alloy steel due to segregation during ingot solidification. Smaller ingots restrict the possible mass of forger components; Grade 91 steel is restricted to 120 ton ingots compared to 480 tons for A 508. Smaller forgings increase the amount of welding required for completion of the RPV. The higher alloy steels are also prone to weld cracking and must be post-weld heat treated to ensure adequate high temperature strength. There are also questions about the ability to produce, and very importantly, verify the through thickness properties of thick sections of the Grade 91 steel material.

The availability of large components, ease of fabrication and nuclear service experience with the A508/533 steels strongly favor their use in the RPV for the NGNP. This material selection reduces the need for further research and development and the associated technical risk to the project. Availability of with experience fabricating nuclear components with these steels minimizes the schedule risk to the project as well.

There is a growing worldwide demand for heavy section pressure vessel materials in the petrochemical, fossil energy and nuclear areas. The lack of orders for nuclear power plant construction in the last twenty years has resulted in fewer suppliers with the required ASME certifications. Given the lack of worldwide capacity it is imperative that the design of the RPV be finalized and long lead time orders be placed at least ten years before anticipated construction.

The following recommendations are made to further define the acquisition strategy and define the risk for obtaining the properly designed and fabricated RPV and IHX pressure vessels to meet the 2021 NGNP start up date:

1. Complete the overall NGNP reactor system design
2. Choose the appropriate RPV and IHX vessel materials
3. Complete the detailed design of the RPV and IHX pressure vessels
4. Work with material suppliers and vessel fabricators to ascertain the delivery schedule for the heavy section materials and the completed components to the INL site.
5. Work with the construction contractor and/or vessel fabricator to assure correct assembly of these vessels as regarding welding and heat treatment procedures.

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ACRONYMS

AGR	Advanced Gas-Cooled Reactor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
B&PV	Boiler and Pressure Vessel
CCGT	Combined Cycle Gas Turbine concept
CRBRP	Clinch River Breeder Reactor Plant
CT	X-ray Tomography
DCC	Depressurized Conduction Cool-down
DHI	Doosan Heavy Industries, South Korea
ETD	elevated temperature design
FSV	Fort St. Vrain
GA	General Atomics
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum
GT-MHR	Gas Turbine-Modular Helium Reactor
HFIR	High-Flux Isotope Reactor
HTDM	high-temperature design methodology
HTGR	High-Temperature Gas Reactor
HTV	High Temperature Vessel
IHX	Intermediate Heat Exchanger
INL	Idaho National Laboratory
ISI	In-Service Inspection
JSW	Japan Steel Works
KAERI	Korean Atomic Energy Research Institute
LWR	Light-Water Reactor
M&C	materials and components
MOC	Materials of Construction
MRC	INL Materials Review Committee
MTR	Material Test Reactor
NDE	nondestructive examination
NE	Department of Energy Office of Nuclear Energy
NERAC	Nuclear Energy Research Advisory Committee
NERI	Nuclear Energy Research Initiative
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Helium Reactor
PCC	Pressurized Conduction Cool-down
PCHE	Printed Circuit Heat Exchangers
PBR	Pebble Bed Reactor
PCDR	Preconceptual Design Report

PCU	Power Conversion Unit
PMB	GIF HTGR Materials and Components Project Management Board
PAR	Preliminary Safety Assessment Report
PWR	Pressurized Water Reactor
PWHT	Post Weld Heat Treatment
RCS	Reactivity Control System
R&D	Research and Development
RPV	Reactor Pressure Vessel
SS	Stainless Steel
TRISO	Tri-isotopic (fuel)
UT	Ultrasonic Testing
VCS	Vessel Cooling System

Next Generation Nuclear Plant Reactor Pressure Vessel Acquisition Strategy

1. INTRODUCTION AND PURPOSE

The Department of Energy has selected the High Temperature Gas-cooled Reactor (HTGR) design for the Next Generation Nuclear Plant (NGNP) Project. The NGNP will demonstrate the use of nuclear power for electricity and hydrogen production. The reactor design will be a graphite moderated, helium-cooled, prismatic or pebble-bed, thermal neutron spectrum reactor. The NGNP will use very high burn-up, low-enriched uranium, Tri-Isotopic (TRISO)-coated fuel and have a projected plant design service life of 60 years. The HTGR concept is considered to be the nearest-term reactor design that has the capability to efficiently produce hydrogen. The plant size, reactor thermal power, and core configuration will ensure passive decay heat removal without fuel damage or radioactive material releases during accidents.

The basic technology for the NGNP was established in former high-temperature gas-cooled reactor plants (e.g., DRAGON, Peach Bottom, Albeitsgemeinschaft Versuchsreaktor, Thorium Hochtemperatur Reaktor, and Fort St. Vrain). These reactor designs represent two design categories: the Pebble Bed Reactor and the Prismatic Modular Reactor. Commercial examples of potential NGNP candidates are the Gas Turbine-Modular Helium Reactor (GT-MHR) from General Atomics (GA), the High Temperature Reactor concept (ANTARES) from AREVA, and the Pebble Bed Modular Reactor (PBMR) from the PBMR consortium. Furthermore, the Japanese High-Temperature Engineering Test Reactor and Chinese High-temperature Reactor are demonstrating the feasibility of the reactor components and materials needed for NGNP. (The High-Temperature Engineering Test Reactor reached a maximum coolant outlet temperature of 950°C in April 2004.) Therefore, the NGNP is focused on building a demonstration plant, rather than simply confirming the basic feasibility of the concept.

The operating conditions for the NGNP represent a major departure from existing water-cooled reactor technologies. Few choices exist for metallic alloys for use at NGNP conditions and the design lifetime considerations for the metallic components may restrict the maximum operating temperature. Qualification of materials for successful and long-life application at the high-temperature conditions planned for the NGNP is a large portion of the effort in the NGNP Materials Research and Development (R&D) Program.

Selection of the technology and design configuration for the NGNP must consider both the cost and risk profiles to ensure that the demonstration plant establishes a sound foundation for future commercial deployments. The NGNP challenge is to achieve a significant advancement in nuclear technology while at the same time setting the stage for an economically viable deployment of the new technology in the commercial sector soon after 2020.

A major component of the NGNP is the Reactor Pressure Vessel (RPV). The purpose of this report is to address the acquisition strategy for the RPV. This vessel will be larger than any nuclear reactor pressure vessel presently in service in the United States (taller, larger in diameter, thicker walled, and heavier) which will mean it must be fabricated through welding and heat treatment at the Idaho National Laboratory (INL) site from of multiple subcomponent pieces. The pressure vessel steel can either be a conventional material already used in the nuclear industry such as A 508/533 or it will be fabricated from newer pressure vessel materials never before used for a nuclear reactor in the US. The worldwide capability to produce very large forgings for these subcomponent pieces is very limited and direct experience with forgings of the size required for NGNP is restricted to conventional pressure vessel steels. Each of these characteristics will present a procurement and fabrication challenge to the producer of the RPV.

1.1 Mission Statement

The objective of the NGNP Materials R&D Program is to provide the essential materials R&D needed to support the design and licensing of the reactor and balance of plant, excluding the hydrogen plant. The materials R&D program was initiated prior to the design effort to ensure that materials R&D activities are initiated early enough to support the design process. The thermal, environmental, and service life conditions of the NGNP will make selection and qualification of the high-temperature materials a significant challenge; thus, new materials and approaches may be required. The mission of the NGNP Materials Program must support the objectives associated with the NGNP in the Energy Policy Act of 2005 and provide any materials related support required during the development of the NGNP. Preparation of this report was done as part of the NGNP Materials Program.

1.2 Assumptions

The following assumptions were used in preparing this strategy:

- The NGNP will be a full-sized reactor plant capable of electricity generation with a hydrogen demonstration unit of appropriate size.
- The reactor design will be helium-cooled, graphite moderated core design fueled with TRISO-design fuel particles in carbon-based compacts or pebbles.
- The NGNP must demonstrate the capability to obtain a Nuclear Regulatory Commission (NRC) operating license. The design, materials, and construction will need to meet appropriate Quality Assurance methods and criteria and other nationally recognized codes and standards.
- The demonstration plant will be designed to operate for a nominal 60 years.
- The NGNP Program including the materials program will continue to be directed by INL based on the guidelines given in the Energy Policy Act of 2005. The scope of work will be adjusted to reflect the level of congressional appropriations.
- Application for an NRC operating license and fabrication of the NGNP will occur with direct interaction and involvement of one or more commercial organizations.

1.3 Issues

The last HTGR design reactor built in the U.S. was the Fort St. Vrain (FSV) reactor which was constructed in the early 1970's, generated the first power sent to the grid in 1976, and was taken out of service in 1989. Along with the long gap in construction of Light Water Reactors in the US this puts the NGNP in the situation where there is a lack of current industry technical information and experience with regard to the materials of construction and fabrication practices associated with the NGNP pre-conceptual designs under consideration. This includes the use of A508 and vanadium-modified 9Cr-1Mo (referred to as Grade 91 steel in this report) pressure vessel steels in recent pressure vessel applications.

There needs to be new information developed regarding the primary metals producers who can produce the required alloys (in a suitable large ingot) to the requirements of the various specifications and provide this material to a supplier who can convert the steel to the required forging or plate sizes. There is a lot of steel capacity in the world, but little capacity for these heavy section forgings or plates. A lot of attention is being given to a single supplier (Japan Steel Works [JSW)]), which could cause scheduling difficulties for meeting the 2021 NGNP start up date. Additional suppliers and their capabilities must be identified as there is worldwide competition for heavy section forgings for new petrochemical and gas treatment facilities.

Another issue will be the identification of vessel fabrication vendors with the appropriate ASME certifications to perform nuclear work. The number of these firms has declined over the last 20 years and the NGNP will be competing for these services with resurgent orders for light water reactors (LWRs) and chemical process facility components in a world market. There is significant competition for these services just in the nuclear industry as shown In Table 1-1.[1]

Table 1-1. New nuclear plant status.

NEI NUCLEAR ENERGY INSTITUTE

New Nuclear Plant Status

Company	Site(s)	Design, # of Units	Early Site Permit (ESP)	Construction/ Operating License Submittal
Alternate Energy Holdings/UniStar	Owyhee County, ID	EPR	-	FY 2009
Amarillo Power/UniStar	Vicinity of Amarillo, TX	EPR	-	FY 2009
AmerenUE/ UniStar	Callaway County, MO (Callaway)	EPR	-	FY 2008
Constellation/ UniStar	Calvert County, MD (Calvert Cliffs)	EPR	-	Partial - Under Review, Full – FY 2008
Constellation/ UniStar	Oswego County, NY (Nine Mile Point)	EPR	-	FY 2009
Detroit Edison	Fermi, MI (Fermi)	Not yet determined	Not yet determined	FY 2008
Dominion	Louisa County, VA (North Anna)	ESBWR (1)	Approved November 2007	November 2007
Duke	Cherokee County, SC (William States Lee)	AP1000 (2)	-	December 2007
Duke	Davie County, NC	Not yet determined	Under consideration	Not yet determined
Duke	Oconee County, SC (Oconee)	Not yet determined	Under consideration	Not yet determined
Entergy	West Felciana Parish, LA (River Bend)	ESBWR (1)	-	FY 2008
Entergy (NuStart)	Claiborne County, MS (Grand Gulf)	ESBWR (1)	Approved April 2007	FY 2008
Exelon	Clinton, IL (Clinton)	Not yet determined	Approved March 2007	Not yet determined
Exelon	Victoria County, TX	ESBWR (2)	-	FY 2008
Florida Power & Light	Miami-Dade County, FL (Turkey Point)	Not yet determined (2)	Not yet determined	FY 2009
Luminant	Glen Rose, TX (Comanche Peak)	APWR (2)	-	FY 2008
NRG Energy/STPNOC	Matagorda County, TX (South Texas Project)	ABWR (2)	-	September 2007

Table 1-1. (continued).

New Nuclear Plant Status

Company	Site(s)	Design, # of Units	Early Site Permit (ESP)	Construction/ Operating License Submittal
PPL Corp./UniStar	Luzerne County, PA (Susquehanna)	EPR	-	FY 2009
Progress Energy	Wake County, NC (Harris)	AP1000 (2)	-	February 2008
Progress Energy	Levy County, FL	AP1000 (2)	-	FY 2008
South Carolina Electric & Gas	Fairfield County, SC (V.C. Summer)	AP1000 (2)	-	FY 2008
Southern Company	Burke County, GA (Vogtle)	AP1000 (2)	Under review, Approval expected early 2009	FY 2008
TVA (NuStart)	Jackson County, AL (Bellefonte)	AP1000 (2)	-	October 2007

FY - Federal Fiscal Year, CY - Calendar Year

Updated: 2/08

To meet the NGNP startup date of 2021, the RPV must be delivered much earlier. The needed delivery date must be identified and a schedule for material acquisition and fabrication must be developed. Once a delivery date is specified a schedule for the following steps needs to be completed:

- Place material order with primary metal producer to obtain position in the melting schedule to secure material for fabrication
- Finalize material shapes and sizes (forgings, plate) and choose the appropriate specifications for the intermediate product mill
- Secure fabrication vendor services and ship material to his facility
- Completion date for fabrication
- Shipment to Idaho
- Completion of field fabrication and heat treatment of sub-assemblies (assumes that field fabrication facility if ready)
- Installation of the RPV, intermediate heat exchanger (IHX), and other major equipment to meet start up schedule.

1.4 NGNP Reactor Vendors (Pre-Conceptual Design Phase)

HTGR is an inherently safe nuclear reactor concept with a straightforward safety basis that has the potential to substantially reduce emergency planning requirements and improved siting flexibility compared to current and advanced light water reactors. The viability of a graphite core planned for the NGNP has previously been demonstrated in former high-temperature gas-cooled reactor plants (e.g., DRAGON, Peach Bottom, Albeitsgemeinschaft Versuchsreaktor, Thorium Hochtemperatur Reaktor, and Fort St. Vrain).

This section describes the current pre-conceptual designs in summary form that are described in detail in the NGNP Pre-Conceptual Design Report.[2] In FY-07 pre-conceptual design work was initiated by the NGNP Project at the INL. This work was completed by three contractor teams with extensive experience in HTGR technology, nuclear power applications and hydrogen production. Each contractor developed a recommended design for NGNP and a commercial version of the HTGR. R&D, data needs, and future studies required to achieve operation of the NGNP were identified as part of the work. In addition, a number of special studies were requested from all three or two of the three teams. The special studies include Reactor Type Trade Study,[3] Pre-conceptual Heat Transfer and Transport Studies,[4] Primary and Secondary Cycle Trade Study,[5] and Power Conversion System Trade Study.[6] The three designs developed are as follows:

1. The **GT-MHR** concept; team led by **General Atomics** teamed with; Washington Group International; Rolls-Royce (United Kingdom); Toshiba Corporation and Fuji Electric Systems (Japan); Korean Atomic Energy Research Institute and OKB Mechanical Design (Russia).
2. The **ANTARES** concept; team led by AREVA NP, Inc. teamed with; Burns & Roe; Washington Group International, BWXT, Dominion Engineering, Air Products, Hamilton-Sundstrand-Rocketdyne, Mitsubishi Heavy Industries, Nova Tech, and Entergy.
3. **The PBMR** concept; team led by **Westinghouse Electric Company**, LLC teamed with; Pebble Bed Modular Reactor (Pty) Ltd. and M-Tech Industrial (Pty) Ltd. (South Africa); The Shaw Group; Technology Insights; Air Products and Chemicals, Inc.; Nuclear Fuel Services; and Kadak Associates.

All three designs utilize TRISO fuel, graphite moderation and high temperature helium coolant in the primary system in the 800^0C -950^0C temperature range. All of the concepts feature various passive neutronic design features which result in a core with relatively low power density and a negative temperature coefficient of neutron reactivity. The shut-down cooling system, the secondary reactivity shut-down system, and the control rod design are all similar among the three designs. All of the reactor concepts could be used as a basis for the NGNP HTGR. Although the designs will not be presented in detail here, the features that relate to RPV material selection and challenges will be discussed. The key operating parameters and design features for all three designs are listed in Table 1-2 along with information for the Fort St. Vrain high-temperature gas reactor, the largest and most recent gas-cooled reactor to operate in the U.S.

Table 1-2. Key operating parameters for the NGNP designs and the Fort St. Vrain HTGR.

Condition or Feature	Fort St. Vrain HTGR	General Atomics GT-MHR	AREVA ANTARES	Westinghouse PBMR
Power Output [MW(t)]	842	550-600	565	500
Average power density (w/cm^3)	6.3	6.5		4.8
Moderator	Graphite	Graphite	Graphite	Graphite
Core Geometry	Cylindrical	Annular	Annular	Annular
Reactor type	Prismatic	Prismatic	Prismatic	Pebble Bed
Safety Design Philosophy	Active	Passive	Passive	Passive
Plant Design Life (Years)	30	60	60	60
Fuel – Coated Particle	HEU-Th/ ^{235}U (93% enriched)	TRISO UCO (startup UO_2)	TRISO UCO (backup UO_2)	TRISO UO_2
Fuel Max Temp – Normal Operation (°C)	1260	1250	1300	1057
Fuel Max Temp – Emergency Conditions (°C)	NA - Active Safety System cools fuel.	1600	1600	1600
Power Conversion Configuration	Direct	Direct	Indirect	Indirect
PCS Cycle Type	Reheat Steam	Brayton	Steam Rankine	Rankine
IHX Design Power			Shell & Tube	
Process	NA	PCHE	PCHE or Fin-Plate	PCHE
Core outlet temperature (°C)	785	Up to 950	900	950
Core inlet temperature (°C)	406	590	500	400
Coolant Pressure (MPa)	4.8	7	5	9
Coolant Flow Rate (kg/s)	428	320	240	193
Secondary outlet temperature (°C)	538	925	850/875 PCS/H_2	900
Secondary inlet temperature (°C)	NA	565	450/475 PCS/H_2	NA
Secondary Fluid	Steam	He	He	He-N
RPV Material	Pre-stressed concrete	2-¼Cr-1Mo	Grade 91	A 508/533
RPV Outside Diameter (m)		8.2	7.5	6.8
RPV Height (m)		31	25	30
RPV Thickness (mm)		281	150	>200

1.4.1 General Atomics – GT-MHR Concept

General Atomics recommended a prismatic reactor design. The core consists of graphite blocks with an annular-fueled region of 1020 prismatic fuel blocks arranged in three columns. They argue that a prismatic reactor inherently allows higher reactor power density levels, resulting in better plant economics, involves fewer uncertainties (and therefore less risk) and allows more flexibility with respect to the use of alternate fuel cycles, such as those fabricated from surplus weapons grade plutonium or transuranics separated from spent LWR fuel.[2,3] The temperature rise of the coolant in the various flow paths through the core varies over a wide range. Good mixing of the outlet coolant is needed to avoid excessive thermal stresses in the downstream components resulting from large temperature gradients and fluctuations, and to assure that the gas entering the turbine has a uniform mixed mean temperature.

General Atomics recommends the use of a direct Brayton Cycle vertical power conversion system (PCS) for electricity generation and an indirect heat transport loop to transport thermal energy to the hydrogen production plant. The primary loop and the hydrogen heat transport loop would both use helium at 7 MPa as a heat transport medium.

1.4.1.1 Reactor Pressure Vessel

The reactor vessel operates at a maximum through-wall average temperature of 440°C during normal operation and reaches about 550 °C during a conduction cooldown event. The core's fuel elements and graphite reflectors, and the shutdown ball channels are all non-metal, capable of withstanding the prescribed maximum core temperatures (~1600°C) or higher in the design-limiting loss-of-coolant accident.

The RPV of the GT-MHR is approximately 31 m high, 8.2 m in diameter, and 281 mm thick. The reference RPV material is 2-¼Cr-1Mo, but this particular material has low strength at the temperatures of interest, which will require very thick sections. A 2¼Cr-1Mo-V steel has better strength at the temperatures of interest (similar to Grade 91 steel), but is not in Section III (nuclear section) of the ASME code. General Atomics is also considering a design alternative to incorporate cooling of the reactor vessel, which could potentially lower reactor vessel temperatures to a level that would allow use of proven light water reactor vessel materials (e.g., A 508/A533 steel).

1.4.2 AREVA – ANTARES Concept

AREVA recommended that the NGNP be a 565 MWt prismatic reactor, citing greater economic potential, higher power level and passive safety, more useable power, greater design flexibility, higher degree of license-ability (concept previously licensed for Fort St. Vrain), higher degree of predictability in core performance, forced outages and scheduled outages than a pebble bed reactor design. They suggest a gas outlet temperature of 900°C as the best compromise between energy efficiency and the ability to produce hydrogen, and the durability of equipment. AREVA proposes using He/N_2 mixture in the power conversion unit (PCU); 900°C is the maximum temperature they advise for nitrogen bearing gas because of nitriding concerns.[6] Use of the high nitrogen gas on the secondary side was specified because it simplifies technology development for the power turbine.

The ANTARES design[7,8] is also based in part on the GT-MHR concept, with 1020 fuel blocks arranged in three columns to form the annular core between inner and outer graphite reflectors. The primary loop pressure is limited to 5.5 MPa which is substantially less than the 7 to 9 MPa specified by the other contractors.

AREVA provided two plant configurations – a plant configuration with a Brayton Cycle to generate electrical power, and a plant configuration with steam to generate electricity by using a Rankine Cycle. The Brayton Cycle configuration is based upon the original ANTARES design. AREVA has recently concluded that the Rankine Cycle is more mature and may be more adaptable to NGNP requirements, and therefore preferable.

1.4.2.1 Reactor Pressure Vessel

The RPV is approximately 25 meters high, 7.5 meters in diameter, and 150 mm thick and made from Grade 91 steel. This steel is preferred because of its superior high temperature properties compared to A 508/533 LWR steel. AREVA has concluded that the properties of this material minimize risk and uncertainty in the design process and maximizes operational margin. This is a developmental material for this application. An ASME code extension is needed, but AREVA does not believe qualification for this application will not be substantially more difficult than qualification for LWR steel, and the resulting margin during a transient situation would be greater. Issues with availability, fabricability, through thickness properties and post-weld heat treatment need to be resolved.

1.4.3 PBMR Concept

This reactor is being developed in South Africa by PBMR (Pty) Ltd. through a world-wide development effort.[9,10,11,12] The program includes testing of mechanical systems and components, a comprehensive fuel development effort and a testing and verification program to support the licensing process. A full-sized demonstration PBMR reactor will be built at the Koeberg nuclear reactor site (owned by Eskom, the South African national utility) near Capetown, South Africa. Westinghouse recommended a pebble-bed reactor over a prismatic reactor design based on the fuel and fueling system demonstrated in Germany (Albeitsgemeinschaft Versuchsreaktor and Thorium Hochtemperatur Reaktor), minimal development costs and reduced risks because of progress in South Africa, higher capacity leading to higher performance capability, lower fuel temperatures, and a strong vendor/supplier infrastructure.

The PBMR utilizes 450,000 graphite-based spherical fuel elements, called pebbles, which are approximately 6 centimeters in diameter. The design of these pebbles, based on the German High-Temperature Reactor, is located in an annular cavity in the reactor vessel. Pebbles proceed vertically downward until they are removed at the bottom. On removal they are checked, and if they are intact and not past the burnup limit, they are circulated to the input queue again. Otherwise, they are replaced with fresh pebbles. This on-line refueling feature makes refueling shutdowns unnecessary, and it also allows the reactor to operate with almost no excess reactivity, which confers advantages in safety, economy, and resistance to nuclear weapons proliferation.

The building design for a single PBMR module consists of a reinforced concrete confinement structure, called the citadel, which houses the PCU. The function of the citadel is as a confinement structure to protect the nuclear components of the power conversion unit from external missiles and to retain the vast majority of fission products that might be released in the event of a reactor accident. The limited total core power allows the reactor to be designed for passive heat conduction from the core, thermal radiation and convection from the vessel and conduction to the confinement structure, keeping temperatures low enough to prevent core or fuel damage.

The present design of the PBMR allows the use of readily available materials that have ASME design allowables. PBMR has concluded that these materials will not need any additional development or data base generation for use at the NGNP system design conditions.

Westinghouse recommends the use of an indirect power conversion cycle and an indirect hydrogen heat transport loop arranged in a serial fashion. The intermediate heat exchanger for the hydrogen heat transport loop would be placed first in the series in order to obtain the highest temperature gas from the nuclear reactor. After the IHX extracts 50 MW, the cooled primary loop gas would then go to the PCU. The pressure of the primary loop is 9 MPa, and the secondary loop between 8.1 and 8.5 MPa. The power conversion cycle uses steam generators and a traditional Rankine Cycle to generate electricity, and would be designed to receive the full power of the reactor.

1.4.3.1 Reactor Pressure Vessel

The vessel design consists of a welded cylindrical shell welded to the bottom head. The top head, containing numerous penetrations for fuel handling and reactor control systems, will be bolted to the cylindrical section. The dimensions are 6.8 m in diameter × 30 m high.

The RPV design configuration is such that its normal operating temperature range is from 300-350°C. The selection is A 508 forgings, A 533 plates and A 504 Grade 24B Class 3 (bolts) steels. A separate stream of helium actively cools the RPV, however during postulated severe accident conditions calculations indicate that the proposed RPV material temperature may be in the creep range. However these LWR pressure vessel steels provide the following benefits:

1. There is manufacturing experience in forging large diameter, thick ring sections thus ensuring predictable through-thickness material properties.
2. There is welding experience with these materials.
3. The A508 and A533 steels are ASME qualified material for nuclear pressure vessel design.
4. ASME design rules, in the form of a nuclear code case, for limited use of these materials in the temperature range 371°C to 538°C are available.
5. There is an extensive irradiation response database at the normal operating temperatures incorporated in the NRC licensing guidelines (NRC Regulatory Guide 1.99) and other international standards (ASTM E 900).
6. The IHX vessels, made from the same steel as the RPV, connect the primary system to the secondary heat transport loop, and contain the heat exchangers.

1.5 NGNP Reactor Vendors (FY 2008 Studies)

The NGNP Project opened contracts with AREVA and General Atomics in FY 2008 to perform a further review of the RPV/ IHX vessels and hot ducts.[13] The study description is as follows:

"This study will evaluate alternatives for the Reactor Pressure Vessel (RPV) materials and design, the primary system hot ducts, 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 options to provide cooling or other design features where desirable to permit use of traditional materials (e.g., A508 used in similar applications for light water reactors) for these components that may reduce cost and schedule risk to the NGNP Project.

In all of the following tasks the evaluations shall begin with the recommendations of the three contractor teams that performed the pre-conceptual engineering work on operating conditions, (e.g., 500 to 600 Mwt, 900-950°C primary system gas outlet temperature, 350 – 500°C primary gas inlet temperature, 7 to 9 MPa primary system pressure) and materials, (e.g., A 508, Grade 91, 2¼Cr-1Mo steels). Where a material is precluded from use because it is not acceptable for these conditions, the conditions for which it would be acceptable should be identified and the design features or changes in operating conditions necessary to achieve this condition should be developed. The range of alternative operating conditions considered in this case should be consistent with the results of the study WBS NHS.S11, which is assessing the appropriate design and initial operating conditions for NGNP."

1.6 AREVA Study

1.6.1 AREVA Background Information[14]

The Pre-conceptual Design Studies Report[2] 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. One important assumption in carrying out this study is the objective of beginning initial operation of NGNP in 2021.

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 and the Steam Generator (see Figure 1-1). The study is performed assuming direct production of Helium at very high temperature (900-950°C) to feed a H_2 production facility.

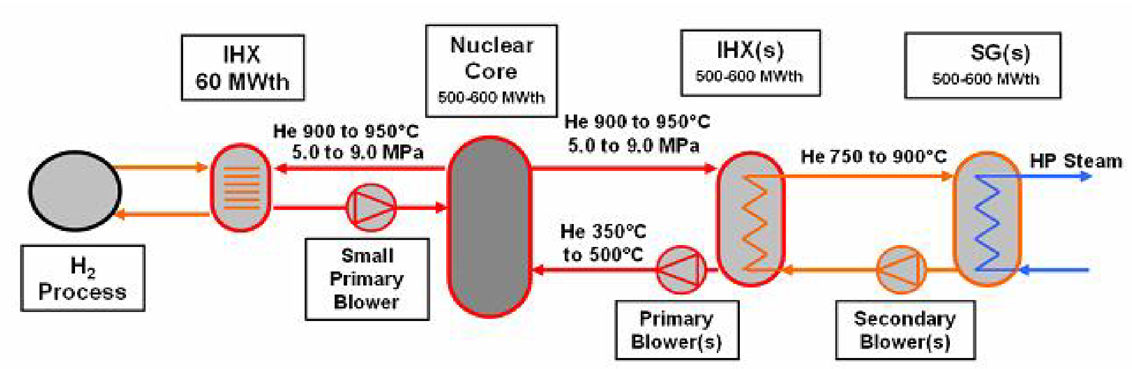


Figure 1-1. NGNP configuration considered in this study.

The configuration recommended by AREVA for the indirect steam concept is defined in Reference 15.[15] 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 1-2. IHX vessels are themselves connected to Steam Generator vessels whose design and specific feasibility issues will not be discussed in the present document.

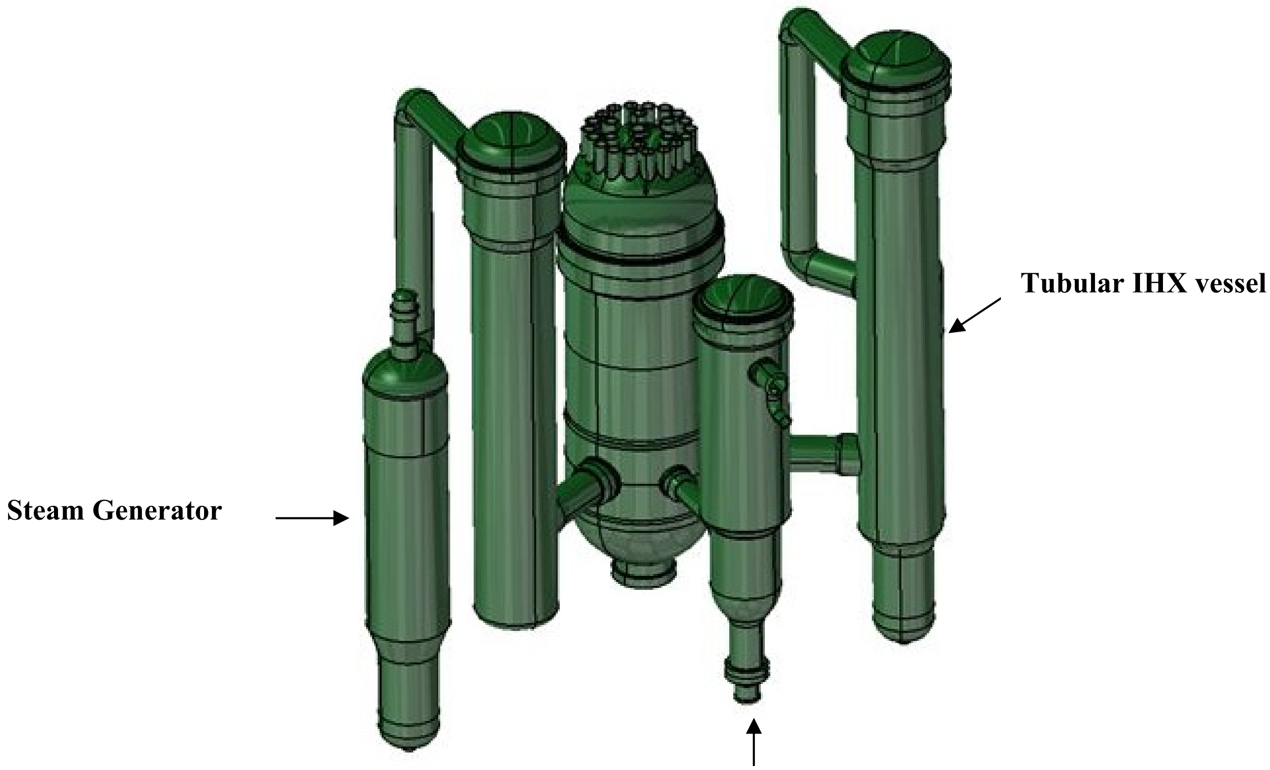


Figure 1-2. Vessel arrangement.

The values of the normal operating parameters recommended as a result of Ref. 2 are indicated in Table 1-3.

Table 1-3. 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

1.6.2 AREVA Pressure Vessel Description[14]

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 is Grade 91 steel.

1.6.2.1 Reactor Vessel

The Reactor Vessel (see Figures 1-3 and 1-4) is approximately 25 meters high, 7.5 meters in diameter and 150 mm thick in the core belt line region.

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 pressurized-water reactor (PWR) 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 Figure 1-5). The on-site welding could be performed in a dedicated on-site workshop including the corresponding heat treatment, final machining, non-destructive examination, hydrotest and cleaning facilities.

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

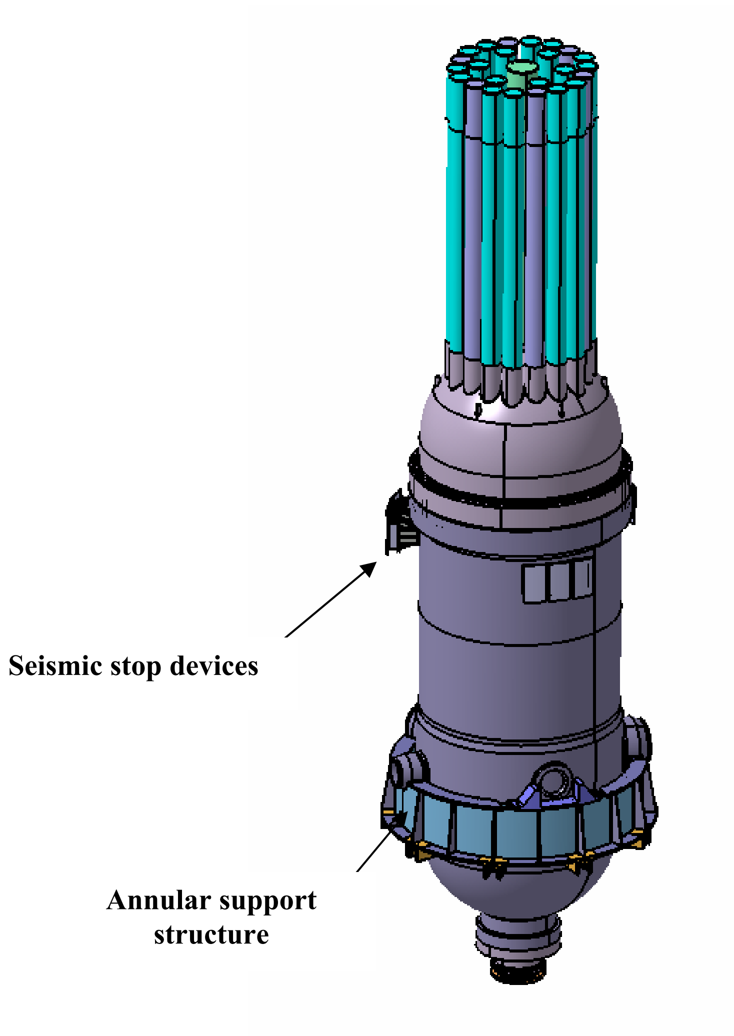


Figure 1-3. Reactor vessel and support system.

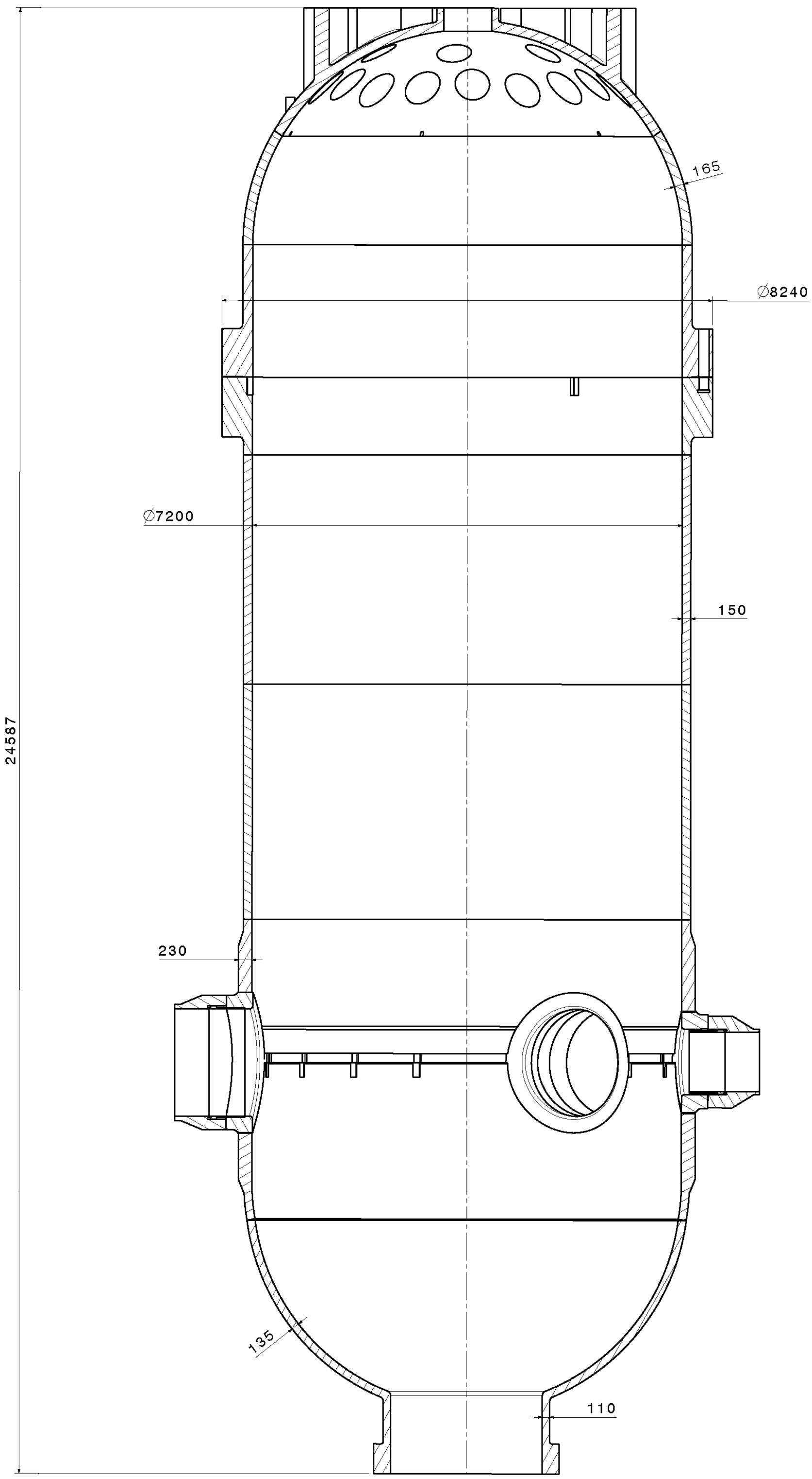


Figure 1-4. Reactor pressure vessel cross section (Grade 91 steel option). Dimensions are in mm.

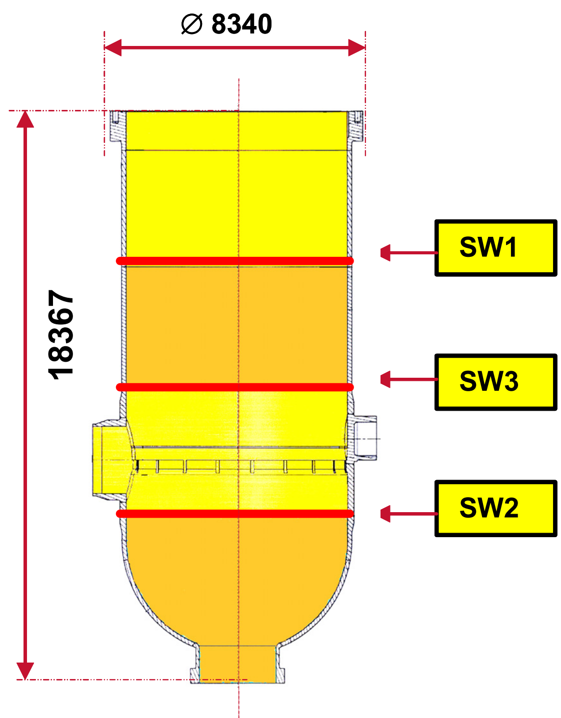


Figure 1-5. Reactor vessel on site weld locations. Dimensions are in mm.

1.6.2.2 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.

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

1.6.2.3 IHX Vessels

Figures 1-6 and 1-7 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).

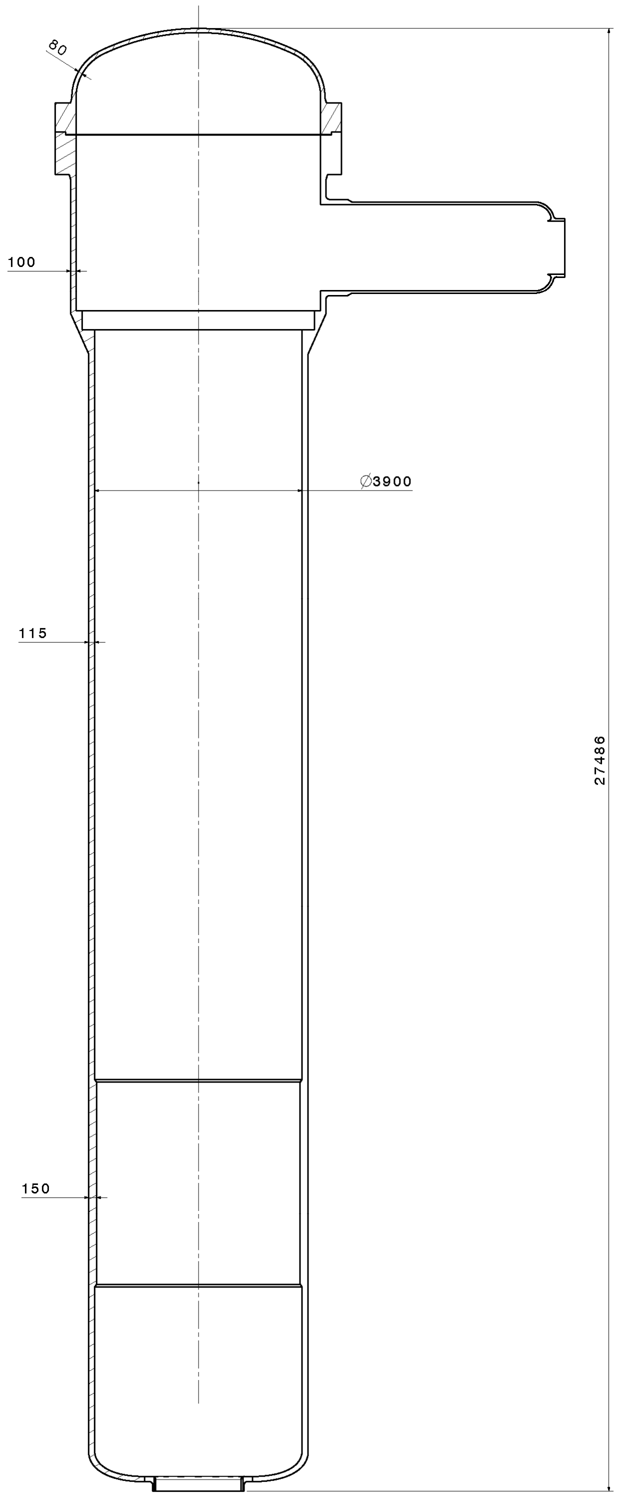


Figure 1-6. Tubular IHX vessel (Grade 91 steel option). Dimensions are in mm.

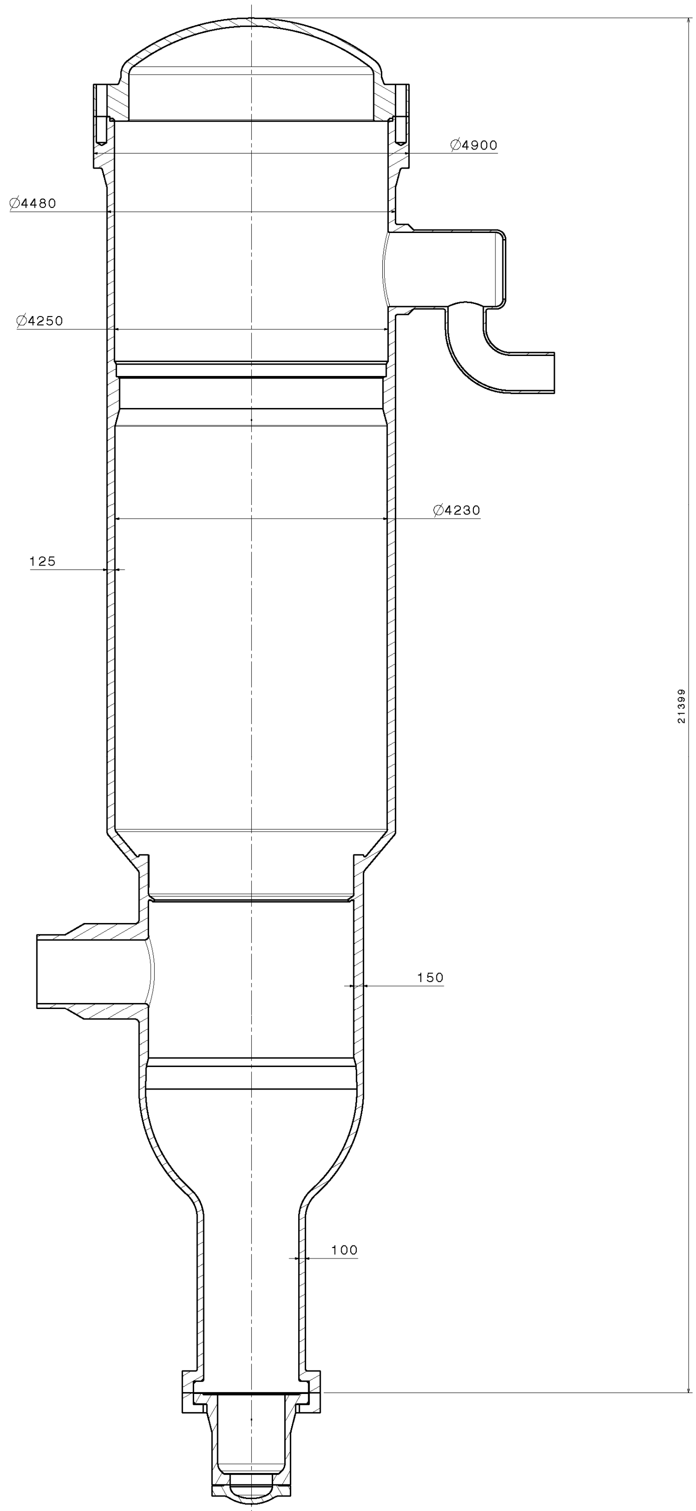


Figure 1-7. Compact IHX vessel (Grade 91 steel option). Dimensions are in mm.

1.6.3 AREVA Material Alternatives

The reference material selected for the vessel system is Grade 91 steel. This material is 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 Grade 91 steel vessels which remain to be solved and Table 1-4 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 does not appear to be consistent with NGNP schedule.

Materials like grade 92, 12 or 122 steels 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 Grade 91 steel 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 1-4 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:

- A 508/533
- Grade 91
- 2-¼Cr-1Mo annealed
- 2-¼Cr-1Mo quenched / normalized and tempered
- 2-¼Cr-1Mo with very high tensile strength (A 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 2021.

Table 1-4. AREVA 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)	A 508 Grade 3 Class 1 A 533 Grade B Class 1	371	NP (1)	371	427	371
Grade 91 (Mod 9Cr-1Mo)	A 336 grade F91 A 182 grade F91 A 387 grade P91	NP 371 371	NP 650 650	NP 371 ($t \leq 3$ in) 371 ($t \leq 3$ in)	650 650 650	NP 482 482
2-¼Cr-1Mo annealed	A 336 grade F22 A-387 Grade 22 Class 1	371	650	371	650	482
2-¼Cr-1Mo quenched/ normalized and tempered	A-336 grade 22 class 3 A-387 grade 22	371	NP	371	650	482
2-¼Cr-1Mo with high tensile strength	A 541 grade 22 class 3	NP	NP	NP	454	454
2-¼Cr-1Mo with very high tensile strength	A 541 grade 22 class 4	371	NP	NP	NP	NP
2-¼Cr-1Mo-V	A-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 1-8 provides a comparison of allowable stresses for the different material candidates. Notice that PWR steel and Grade 91 steel have similar allowable stress around 370°C; 2¼Cr-1Mo grades with high allowables show a significant drop in properties beyond 430°C. A 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 2¼Cr-1Mo grades. It is also expected that fracture toughness properties would be low for this material. The annealed 2¼Cr-1Mo material would require a significant increase of thickness compared to other candidates to compensate for the reduced tensile properties. The 2¼Cr-1Mo-V steel has similar allowables as PWR steel and Grade 91 steel at low temperature and keeps its strength at higher temperatures with allowable even slightly above that of Grade 91 steel. 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.

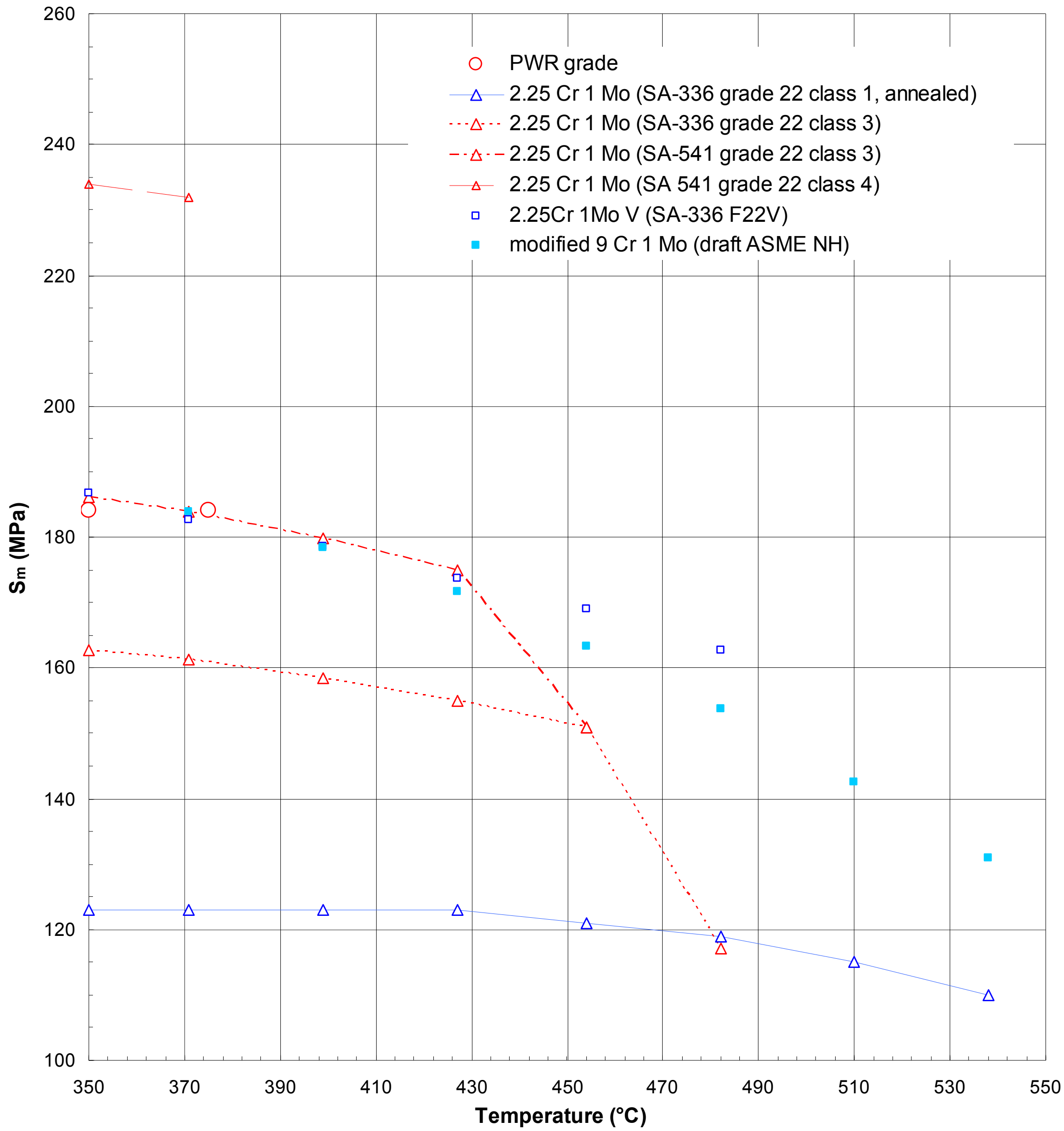


Figure 1-8. Comparison of allowable stress for candidate RPV materials.

1.6.4 Survey of Materials Used in Japan for Nuclear and Non-Nuclear Pressure Vessel Applications

A survey has been performed of material used in Japan for pressure vessel applications because of their more recent construction experience. Table 1-5 provides a summary of materials used in the nuclear industry and Table 1-6 a summary of material used in the non nuclear industry for medium and high temperature applications.

Table 1-6 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 1-5. Material uses in Japan's nuclear industry.

Material Name (JIS Symbol)	ASME Equivalent	Operating Conditions (Temperature, Pressure)	Geometry (Diameter, Height Thickness)	Adopted Component
Mn-Mo alloy steel (SQV2A, SFVQ2A)	A 533B, A508	343ºC, 17.2 MPa	~5.2m, ~13.6 m, ~255 mm	RPV, SG etc. for LWR
2¼Cr-1Mo alloy steel (SCMV4, SFVAF22B)	A 387 Gr22, A 336 F22	395ºC, 3.9 MPa (1)	RPV : 5.5m, 13.2m, 122~160 mm	RPV&Heat Exchanger Vessels of High-Temperature Engineering Test Reactor
Note: (1) Design temperature and pressure of 440ºC and 4.8 MPa Table 1-5, Material Used in the Non-nuclear Industry				

Table 1-6. Material used in the non-nuclear industry.

Material Name (JIS Symbol)	ASME Equivalent	Operating Conditions (Temperature, Pressure)	Thickness	Adopted Component
Grade 91 steel (KA-SCMV28, KA-SFVAF28)	A 387 Gr. 91, A 182 Gr. F91, A 335 Gr. P91	~600ºC		Main steam piping for Supercritical Pressure Boiler
Carbon steel for boiler and other pressure vessels (SB410,450,480)	A 285/285M Gr. A,B,C	~450ºC	~ 200 mm (thickness)	Boiler and pressure vessel in medium and high temperature
Mo alloy steel (SB450M, SB480M)	A 204/204 M Gr. A, B, C	~500ºC	~ 150 mm (thickness)	Boiler and pressure vessel in medium and high temperature
Mn-Mo, Mn-Mo-Ni alloy steel (SBV1A,1B,2,3)	A 302/302M Gr. A, B, C, D	~500ºC	~ 150 mm (thickness)	Boiler and pressure vessel in medium and high temperature

Table 1-6. (continued).

Material Name (JIS Symbol)	ASME Equivalent	Operating Conditions (Temperature, Pressure)	Thickness	Adopted Component
Mn-Mo, Mn-Mo-Ni alloy steel (SQV1A,1B,2A,2B,3A,3B)	A 533 Gr. A, B, C, D	~400ºC	~ 150 mm	Boiler and pressure vessel in medium and high temperature
Cr-Mo alloy steel (SCMV1,2, 3,4,5,6)	A 387/387M Gr. 2, 12, 11, 22, 22L,21, 21L, 5	~400ºC	SCMV1,2,3 : ~ 150 mm , SCMV4,5,6 : ~ 300 mm , (thickness)	Boiler and pressure vessel in medium and high temperature

1.6.5 AREVA Comparison of Material Candidates

Table 1-7 provides a comparison of material candidates on key selection criteria. The following materials are considered as credible candidates for start-up by 2021:

- A 508/533
- Grade 91
- 2-¼Cr-1Mo annealed.

The allowables of 2-¼Cr-1Mo annealed are however probably too low and would require thicknesses which would make this option not economical. Fabricability issues would have also to be clarified.

Table 1-7. Comparison of material candidates.

Material	ASME III Acceptance	Allowables	Negligible Creep Conditions	Procurement	Fabricability
A 508/533	Permitted up to 371°C for normal operation and up to 538°C under specific conditions as per Code Case N499-2		<371°C	No issue	No issue
Grade 91	Permitted up to 650°C but would required the acceptance of A 336 grade F91 specification		To be defined but expected between 400 and 450°C	Availability of heavy section forgings to be clarified	Welding qualification to be completed. Practicality of performing post-weld heat treatment (PWHT) on site to be studied

Table 1-7. (continued).

Material	ASME III Acceptance	Allowables	Negligible Creep Conditions	Procurement	Fabricability
2¼Cr-1Mo annealed	Permitted up to 650°C	Lower than PWR grade and Grade 91 which would require an increase of thickness by 150%	To be defined but expected around 400°C	Availability of heavy section forgings to be clarified	Should be less a concern than for Grade 91 but welding qualification will be required
2¼Cr-1Mo quenched / normalized and tempered	Permitted for Class 1 components but not above 371°C	Lower than those for PWR grade and Grade 91 which would require an increase of thickness by 120%	To be defined but expected around 400°C	Availability of heavy section forgings to be clarified	Should be less a concern than for Grade 91 but welding qualification will be required
2¼Cr-1Mo with high tensile strength (A 541 grade 22 class 3)	Not permitted	Similar to those for PWR grade and Grade 91 but significant drop in properties beyond 430°C	To be defined but expected around 400°C	Availability of heavy section forgings to be clarified	Should be less a concern than for Grade 91 but welding qualification will be required
2¼Cr-1Mo with very high tensile strength (A 541 grade 22 class 4)	Permitted for Class 1 components but not above 371°C	Higher than those for PWR grade and Grade 91 but drop in properties expected as for A 541 grade 22 class 3	To be defined but expected around 400°C	Availability of heavy section forgings to be clarified	Should be less a concern than for Grade 91 but welding qualification will be required
2¼Cr-1Mo-V	Not permitted	Similar to those for PWR grade and Grade 91	To be defined but expected around 400°C	Availability of heavy section forgings to be clarified	Should be less a concern than for Grade 91 but welding qualification will be required

1.6.6 AREVA Analysis of Forged Components Procurement[14] (Section 6.3)

1.6.6.1 ASTM-ASME Standards for Grade 91 Steel

The starting requirements are those of ASME-ASTM A 336 for Grade 91 steel, which covers forged parts without limitation of mass. In A 182, devoted to pipe flanges, forged fittings and valves, the mass is limited to 4,250 kg. It is to be noted that subsection NH of ASME III concerning components in elevated temperature service allows grade F91 according A 182 forgings but not according A 336 forged parts.

The specified chemical analyses are quoted in Table 1-8. They are very similar in A 336 and in A 182. Differences are found for maximum P and S contents only. The required mechanical properties are quoted in Table 1-9. They are similar in A 336 and A 182. The unique difference is a maximum for ultimate tensile strength S_u of 760 MPa in A 336; no maximum limit for S_u in A 182.

Both A 336 and A 182 grades F91 are normalized and tempered with the Ame temperature requirements (Table 1-9).

Table 1-8. Specified chemical compositions (Grade 91).

	Heat analysis		Product analysis	
%	A 336 grade F91	A 182 grade F91	A 336 grade F91	A 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 1-9. Mechanical properties at room temperature and specification for heat treatment for Grade 91.

	A 336 grade F91	A 182 grade F91	A 387 grade P91
S_y	≥ 415 MPa	≥ 415 MPa	≥ 415 MPa
S_u	585–760 MPa	≥ 585 MPa	585–760 MPa
A %	≥ 20	≥ 20	≥ 20
Normalize	1040–1095°C	1040–1095°C	1040–1080°C
Temper	≥ 730°C	≥ 730°C	730-800°C

1.6.6.2 Filler Metals for Grade 91 Steel

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

During preliminary work for HTGR, 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 industrial welding energy compatible with industrial operations
- Toughness of weld metals similar to base metal
- Cross - weld creep strength of the welded joints.

The first point is met by using filler metals with severe limit in sulfur content (S ≤ 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. 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 needs 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 AW, grade ER 905-B9 of SFA 5.28 for rods for gas shielded procedure including gas tungsten arc welding and grades E9015-B9, E9016-B9 or E9018-B9 of SFA 5.5 for covered electrode for shielded metal arc welding.

The technological development for the welding operations of the NGNP RPV will then needs similar studies as in the case of A 508 base metal.

1.6.6.3 ASTM-ASME Standards for A 508

Use of this grade for the RPV, is subject to the assumption that the normal service temperature of the RPV does not exceed 371°C 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.

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

The specified chemical analyses are quoted in Table 1-10. Particular limitations are found for maximum Al, Ca B and Ti.

The required mechanical properties for A 508 Grade 3 Class 1 are quoted in Table 1-11.

A 508 grade 3 Class 1 is quenched by immersion or by spraying and tempered with the temperature requirement indicated in Table 1-11.

Table 1-10. Specified chemical composition (A 508 Grade 3).

	Heat Analysis		Product Analysis
%	A 508 Grade 3 Class 1	A 533 type B Class 1	A 508 Grade 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 1-11. Specified mechanical properties at room temperature with heat treatment for A 508 Grade 3 Class 1.

	A 508 Grade 3 Class 1	A 533 Type B Class 1
S_y	≥ 345 MPa	≥ 345 MPa
S_u	550 – 725 MPa	550 – 690 MPa
A %	≥ 18	≥ 18
Austenitizing before quenching		845 – 980°C
Temper	≥ 635°C	≥ 595°C

1.6.7 AREVA Procurement Discussion[14]

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 Grade 91 steel and A 508 materials are considered.

1.6.7.1 Nozzle Ring - Grade 91 Steel

The following table provides the estimated required vessel thickness in the core belt line region for Grade 91 steel for a design temperature of 460°C and different values of primary pressure.

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

The thickness has to be reinforced due to the presence of nozzles (see Figure 1-9) 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

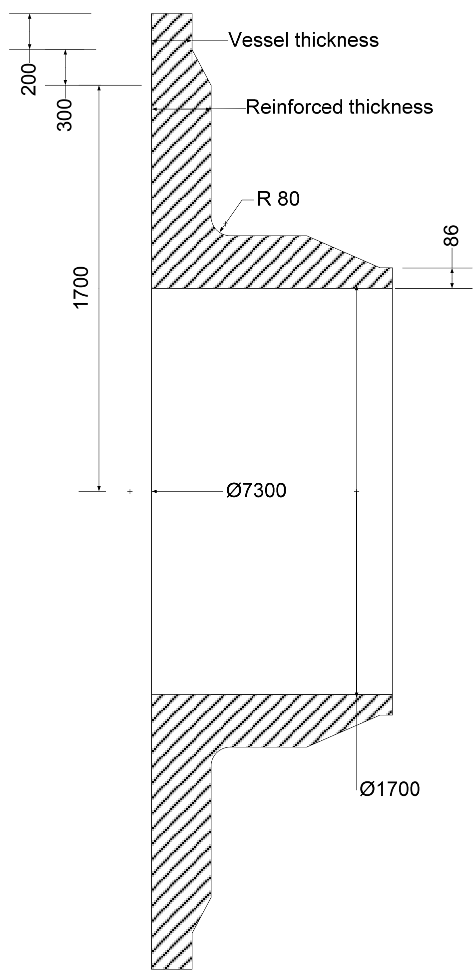


Figure 1-9. Nozzle ring. Dimensions are in mm.

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 1-10).

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

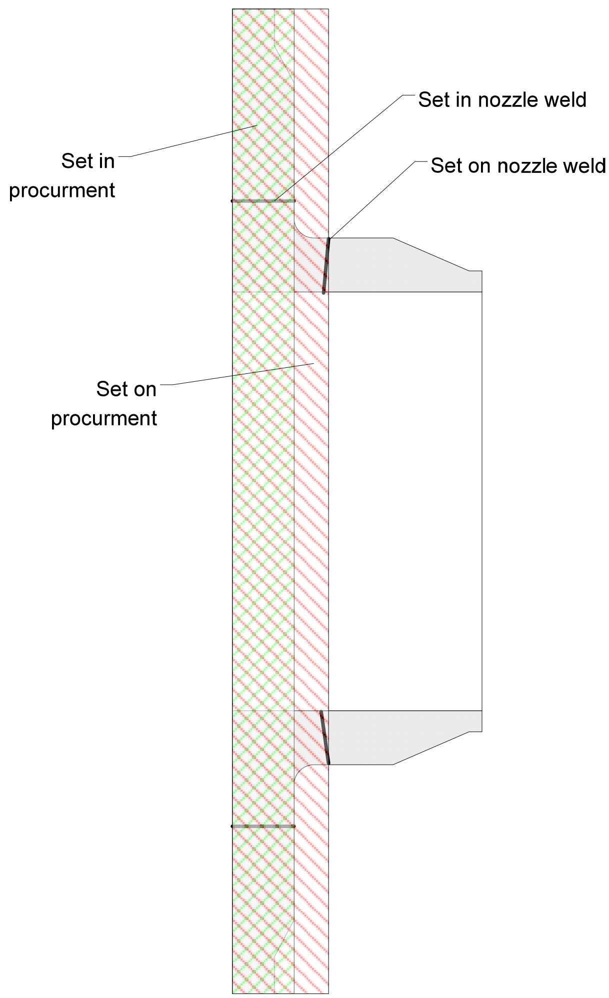


Figure 1-10. Nozzle ring procurement needs with set-in or set-on nozzle.

Consequently, the ingot procurement masses for these rings are roughly estimated:

Pressure (MPa)	Ingot mass with set-in design	Ingot mass with set-on design
4	320 tons	460 tons
5	400 tons	530 tons
6	460 tons	600 tons

The set-in nozzle procurement could be expected similar to the sketch in Figure 1-11. 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.

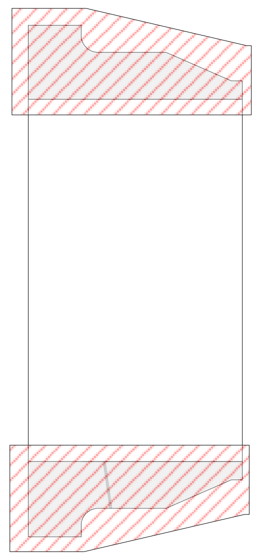


Figure 1-11. Set-in nozzle procurement need.

1.6.7.2 Nozzle Ring – A 508 Steel

The following table provides the estimated required vessel's thickness in the core belt line region for A 508 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 1-9) 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 1-12).

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:

Pressure (MPa)	Ingot Mass with Set-in Design	Ingot Mass with Set-on Design
4	290 tons	430 tons
5	340 tons	480 tons
6	420 tons	560 tons

The set-in nozzles procurements are expected to be similar to those for Grade 91.

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

Thus the mass of the shell used to machine the flange is about 130 tons and the ingot mass about 260 tons.

1.6.7.3 Flange Procurement

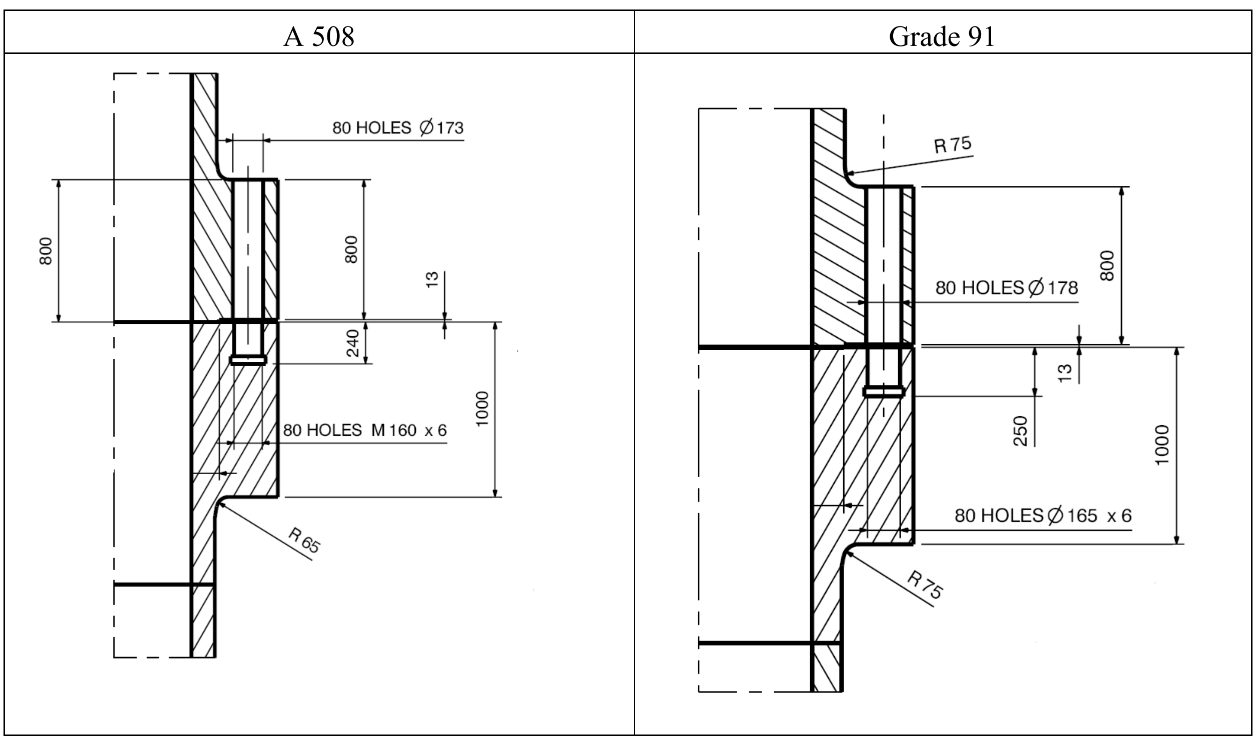


Figure 1-12. RPV flange design. Dimensions are in mm.

1.6.8 AREVA Final Procurement Analysis[14]

For Grade 91 steel, it is expected that the larger ingots available would be in the order of 270 tons.

The following table summarizes Grade 91 steel design feasibility for the nozzle ring.

Pressure (MPa)	Ingot Mass with Set-in Design	Comments	Ingot Mass with Set-on Design	Comments
4	320 tons	Beyond supplier capacity	460 tons	Beyond supplier capacity
5	400 tons	Beyond supplier capacity	530 tons	Beyond supplier capacity
6	460 tons	Beyond supplier capacity	600 tons	Beyond supplier capacity

For A 508, it is expected that ingots up to about 450 tons could be fabricated by JSW. The following table summarizes A 508 design feasibility.

Pressure (MPa)	Ingot Mass with Set-in Design	Comments	Ingot Mass with Set-on Design	Comments
4	290 tons	Expected feasible	430 tons	Expected feasible
5	340 tons	Expected feasible	480 tons	Feasibility not insured but design optimizations could make it possible
6	420 tons	Expected feasible	560 tons	Beyond supplier capacity

Based on this preliminary analysis, it seems that the set-in nozzle option should be selected to minimize feasibility issues. With this option, the nozzle ring could be fabricated in one piece for A 508 material. For Grade 91 steel, primary pressure should be far lower than 4 MPa to increase chances of procurement of the nozzle ring in one piece. This pressure value would have however consequences on the circulator design and total required house load and it is therefore recommended for the time being to retain a value of 5 MPa but split the nozzle ring in two pieces.

1.6.9 Estimated Weight for NGNP Forgings

As a result of this analysis, forging breakdowns have been defined as a basis for discussions with forging suppliers. Figure 1-13 provides the reference breakdown proposed for discussions with Japan Steel Works. This figure is generic for A-508 and Grade 91 steel. The thickness dimensions are somewhat larger for the Grade 91 steel material primarily due to the higher temperature assumptions for the design. The Grade 91 steel material is also slightly lower strength even at the same temperature. As discussed previously, it has however to be noted that, for procurement of A 508 from JSW, it could be envisioned to merge forgings 2 and 3 in one piece. Figure 1-14 provides another breakdown defined for discussions with suppliers with more limited capabilities.

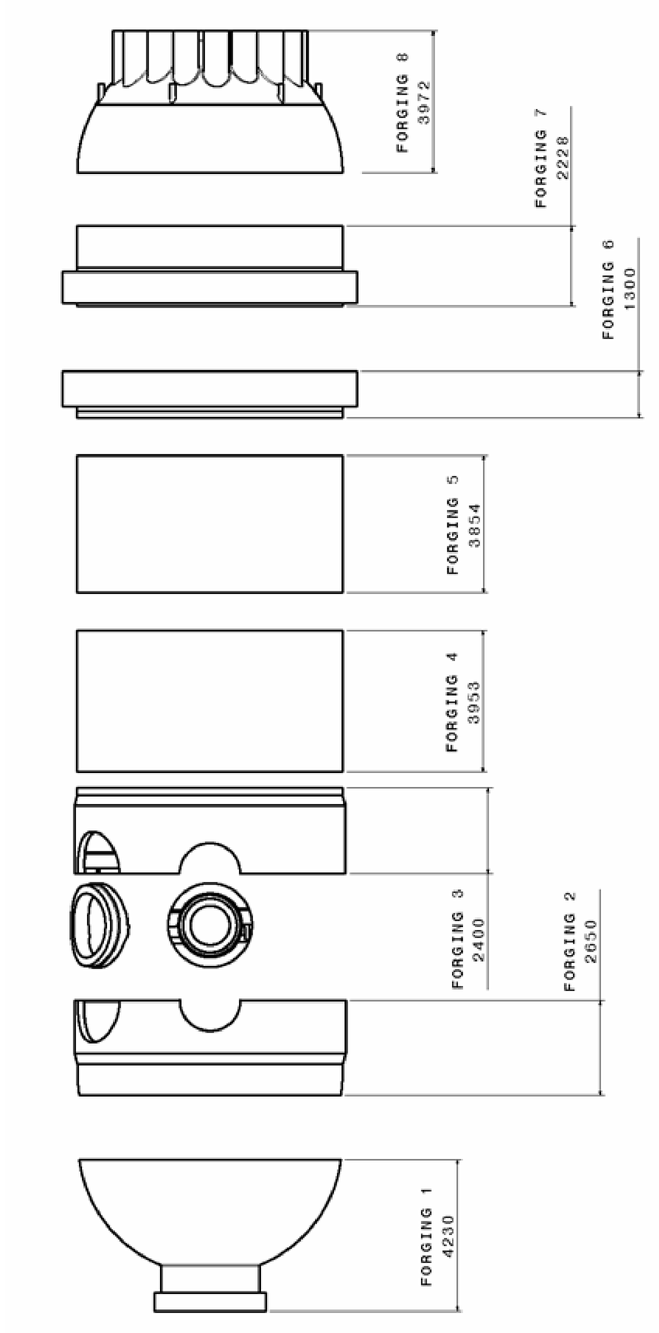


Figure 1-13. Reference forging breakdown for NGNP RPV.

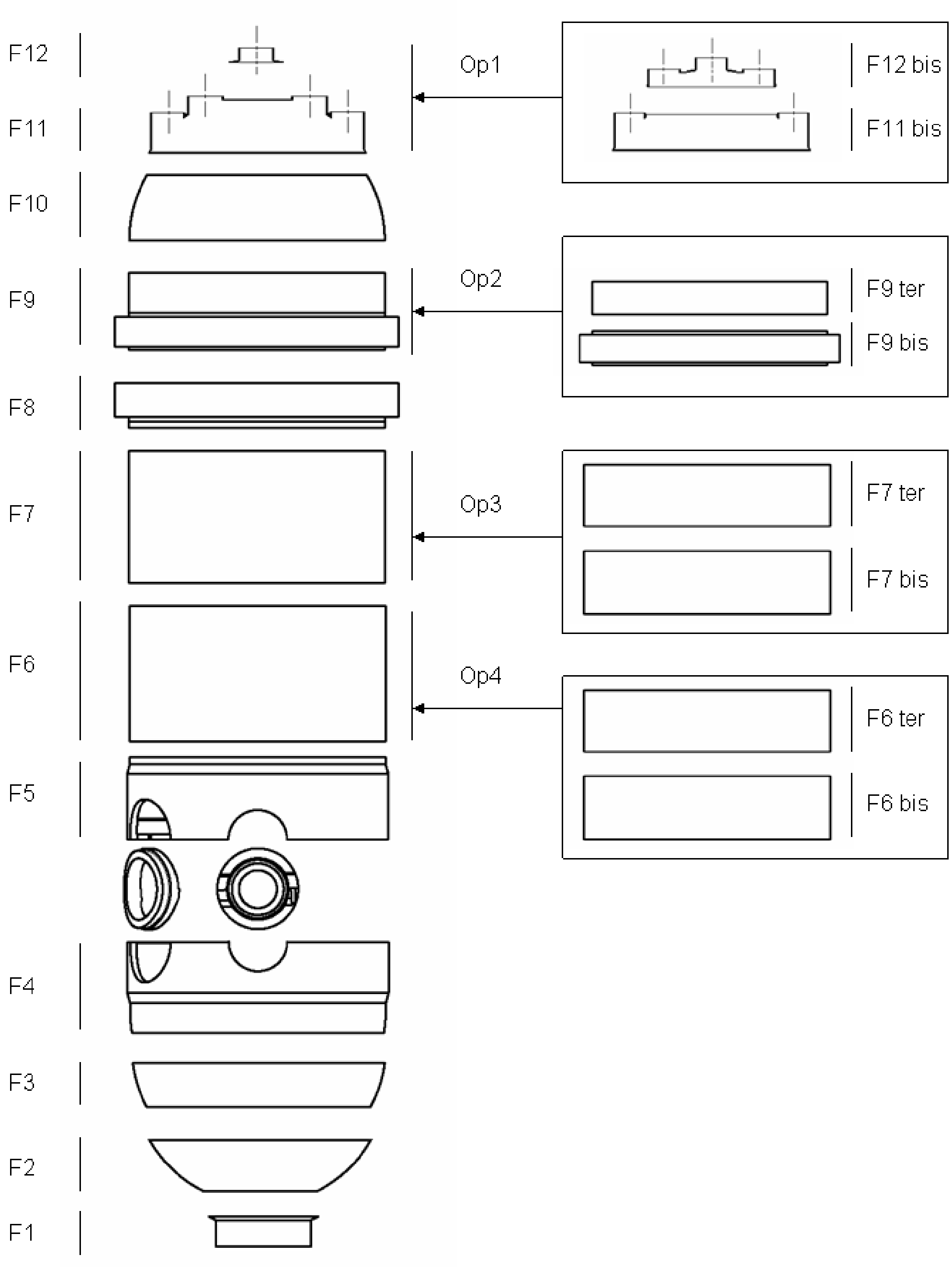


Figure 1-14. Alternative forging breakdown for NGNP RPV.

Estimated weights have been prepared for discussion purposes with suppliers. Table 1-12 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 1-12. Reference weights in metric ton (MT) of the reference breakdown.

	A-508	Grade 91
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.

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 1400 mm.

Also A 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 an 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 Grade 91 specification does not currently have the length requirement for the testing at both ends but for the sake of this discussion the Grade 91 forgings will be assumed to be modified the same as the A 508 forgings. Table 1-13 illustrates the modified configuration.

Forgings which are of essentially of a uniform thickness require ¼ T (where T is the thickness of the forging) testing. In the case of the A 508 forgings the test specimens must be ¼T from one quench surface and T from the second quench surface. For the case of the Grade 91 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 A 508 the prolongation is a minimum 120 mm + T and for Grade 91 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.

Table 1-13. Forging weights (MT) adjusted for test specimen allowance.

	A-508	Grade 91
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.

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 A 508 and A-336 respectively. In addition samples will have to be taken from each end for the A 508 and only one end for the Grade 91.

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 1-13 would not be any bigger for test specimen allowance.

1.6.10 AREVA Assessment of Cooled Vessel Concept

1.6.10.1 AREVA Scope

This task is aimed at identifying operating condition changes and/or design features that would be required to permit utilization of A 508/533 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 A 508/533 material
- Identifying 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 A 508/533 material for the RPV.

It is intended to identify and assess alternative concepts of the AREVA prismatic design based on A 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 A 508 could be selected without using active cooling will be based on conduction cool-down calculations and will take account of uncertainties.

1.6.10.2 AREVA Background Discussion on Cooled Vessel Concept

The main objective of any design option that will permit use of A 508/533 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. A 508/533 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 high-temperature reactor 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, A 508/533 steel is unacceptable because the calculated temperatures during normal operation exceed 371°C. Conduction cooldown temperatures could also challenge the A 501/533 limits for the reactor sizes anticipated.

In order for A 508/533 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 A 508/533 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).

Six options, including both passive and active, were explored by AREVA as potential solutions for keeping the reactor pressure vessel temperatures within acceptable A 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.

1.6.10.3 AREVA Conclusions Regarding A 508/533 Alternatives

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

- The current RPV design can be considered acceptable using A 508/533 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 A 508/533 vessel is a clear option.

For higher temperature operation, feasible alternatives appear to be available to allow the use of an A 508/533 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 Grade 91 steel 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.

1.7 General Atomics Assessment[16]

GA assessed the procurement of the NGNP RPV and IHX vessels in a special study.[16] The report focused on evaluating design options for these two vessels, taking into account the anticipated operating conditions for NGNP, the available materials and their associated metallurgical and physical properties, and acquisition, fabricability, and reliability factors that could impact NGNP startup by 2021. This report includes the following assessments:

- An assessment performed by URS Washington Division of RPV design criteria based on Nuclear Regulatory Commission (NRC) guidance.
- Detailed thermal and structural analyses performed by Korea Atomic Energy Research Institute (KAERI) of prismatic-block Modular Helium Reactor with RPVs manufactured from both industry-proven A 508/533 steel and more developmental Grade 91 steel, in order to assess requirements for an active Vessel Cooling System and to estimate structural design margins.
- Parametric, accident-condition analyses performed by Fuji Electric Systems to estimate the sensitivity of peak fuel and RPV temperatures to key design parameters, in order to better establish priorities for technology development.
- An assessment performed by URS Washington Division of Grade 91 steel and other high-alloy steels for potential use as the RPV material of construction, and recommendations provided by Toshiba Corporation for the IHX material of construction based on their IHX designs for various NGNP configurations that are currently under consideration.
- Information provided by JSW on their current and future capabilities for manufacturing large forgings from A 508/533 steel and high-alloy steels.
- Information provided by KAERI and a Korean supplier on their RPV fabrication capabilities and issues associated with transportation and on-site assembly of an RPV.

1.7.1 KAERI Assessment of Cooled Vessel Concept

KAERI performed an initial structural analysis of the NGNP Grade 91 RPV under the normal operating condition and the transient conditions high pressure conduction cooldown and low pressure conduction cooldown are performed, and the structural integrity of the vessel is confirmed per the ASME B&PV Code, Section III, Subsections NB and NH. The design criteria for the Level A&B Service Conditions are applied in the evaluation of the structural integrity of the vessel instead of those for the Level C&D Service Conditions conservatively. The vessel material is Grade 91 steel and it is assumed that during the normal operation the core inlet and outlet gas temperatures are 490°C and 950°C, respectively, and the reactor internal pressure is constant 7 MPa in all conditions considered, conservatively.

The reactor vessel temperature is maintained below 371°C during the normal operation and the structural integrity of the vessel is confirmed with proper margin per the design criteria of the subsection NB. All the check items for the stress intensities given by the structural analysis and post-process are below the certain allowable limits required in the subsection NB. Even though the reactor vessel temperature exceeds 371°C during the transients of the high pressure conduction cooldown and low pressure conduction cooldown for time, the structural integrity of the vessel is confirmed with proper margin per the design criteria in the subsection NH.

All the check items for the stress intensities, the creep strains and the creep-fatigue damage given by the structural analysis and post-process are below the corresponding allowable limits required in the subsection NH.

In summary, the structural integrity and design adequacy of the NGNP Grade 91 RPV was confirmed through this preliminary evaluation study. However, it should be noted that the evaluation and discussion so far is preliminary and is based on the simplified vessel configuration in which the important loadings such as nozzle loads, or support loads, seismic loads, and flange effects are not considered.

1.7.2 URS-Washington Division Assessment of RPV Materials

This section addresses the acceptability of candidate materials of construction (MOC) for the RPV under current ASME Code definitions, allowable properties, and design stresses. Applicable, relevant regulatory and industry guidelines concerning potentially suitable candidate MOC were reviewed to meet the RPV operating conditions of this very high-temperature reactor.

The various candidate MOC for evaluation included the LWR steels (ASME A 508/A 533), 2¼Cr-1Mo, 2¼Cr-1Mo-V, Grade 91 steel, and other potential candidates as identified by GA. An assessment was made of the time and effort required to extend or develop new ASME Code cases as necessary for candidate MOC not qualified for use under the current ASME Code Section III, Division 1, Subsection NH for Class 1 Components in Elevated Temperature Service (649°C for the NGNP). LWR primary coolant system MOC must only meet Sect. III, Div. 1, Subsection NB temperature limits up to 391C.

The candidate MOC being considered for the primary coolant pressure boundary system must meet the required design criteria for this NGNP RPV based upon current ASME Code and NRC regulations for LWR licensing per 10 CFR 50.55a. For the GA GT-MHR RPV, candidate MOC should meet the rules and requirements of ASME Code Section III, Division 1, Subsection NH, Class 1 Components in Elevated Temperature Service (to 649°C). Sect. III, Div. 1, Subsection NB covers Class 1 Components up to 371°C , too low for the 490°C (or 590°C) inlet gas coolant design temperature of the RPV.

The baseline properties of MOC candidates include tensile strength, yield strength, elongation, reduction in area, creep-rupture strength, low-cycle fatigue, creep-fatigue, and fracture toughness (impact strength). Other key MOC characteristics include availability, fabricability, weldability, and good high temperature corrosion resistance to (compatibility with) the He environment. Potential RPV MOC must not only have good room temperature properties, but more importantly, possess high strength and stress intensities (allowable stresses) at elevated temperatures – i.e., good hot strength for extended operation, under neutron irradiation. In addition, the MOC must meet the applicable ASME Code and NRC rules and regulations for Section III, Division 1, Class 1 components per Subsection NH (to 649°C) to qualify for long-term RPV service.

From a corrosion/erosion standpoint, the following factors and effects on MOC must be addressed:

1. Effect of He coolant chemistry on MOC degradation – i.e., contaminants/impurities in He gas.
2. Corrosion effects on mechanical properties on candidate MOC.
3. Fission product release and its effect on MOC candidates.
4. Corrosion/erosion due to particulate-laden He gas flow velocities.

From a welding and heat treatment standpoint, the following factors should be adequately addressed:

1. Effect of welding processes and heat input on mechanical properties and microstructures.
2. Effect of post-weld heat treatment (PWHT) on high temperature creep strength. PWHT needs to be optimized to maintain high temperature creep properties.
3. The mechanical properties of thick sections (> 15 cm).
4. Post-forming heat treatment (PFHT). The higher the amount of cold work performed, the lower the high temperature creep properties without a PFHT.

The LWR steels A 508 and A 533 are approved in ASME Code Sect. III, Subsection NB, for Class 1 components only up to 371°C for normal operation, well below the inlet core coolant temperature of 490°C. Therefore, this class of LWR steels was not further reviewed because of inadequate hot strength (371°C Code limit). Unless it is economically feasible to cool the inlet He gas temperature to <371°C, which would reduce the thermal efficiency of the Very High-temperature Reactor, these steels are not considered suitable for the above RPV conditions. The following ferritic alloy steels appear to be potentially suitable as RPV forging and plate candidates for MOC consideration and investigation and are shown in Table 1-14.

a. Grade 22(Fe-2.25Cr-1Mo), UNS K21590, ASME A-182 and Grade 91, F22

The well-established Grade 22, used in both fossil and nuclear power plants, is approved in ASME Section III for use up to 593°C. However, its lower allowable stress values at NGNP RPV temperatures (490°C) would require greater RPV wall thicknesses to meet the above design conditions. Thus, while applicable, this proven low alloy steel is not considered to be an optimum RPV candidate due to excessive wall thickness and attendant loads. However, a vanadium (V) modified version of this steel has significantly higher stress intensities at elevated temperatures than Grade F22 as follows:

b. Grade 22V(Fe-2.25Cr-1Mo-0.25V), UNS K31835, ASME A-182, A-336, A-541, F22V

Grade 22V is approved for use under ASME Code Section VIII but not under Section III. While there are adequate tensile strength data at 500°C, it is only approved up to 482°C. Limited high temperature creep and thermal aging data are available. There is thick section fabrication and welding experience derived from the oil and gas industry. As with other MOC, more data are needed on compatibility with impure He gas. This steel has good hot strength properties but requires an ASME code case for Sect. III, Div. 1, Class 1 applications up to at least 490°C, preferably per Subsection NH, which entails additional property testing and a series of quarterly ASME Code committee meetings to prepare and approve a code case.

c. Grade 91 steel, UNS K90901, ASME A-182, Grade 91 steel (forgings); ASME A-387, Grade 91 steel (plates)

Grade 91 (ferritic/martensitic) steel has the best mechanical properties and is the most industrially mature of the high strength steels. Its superior hot strength properties result from the addition of alloying elements such as V, Nb and N and optimum heat treatment. Grade 91 steel is designated as a creep strength enhanced ferritic steel. It is widely used in cogeneration power plants and supercritical fossil fuel power units up to the maximum temperatures and pressures. Grade 91 steel is ASME Code-approved for up to 649°C per Section III, Class 1, Subsection NH component applications. This alloy is much more resistant to thermal fatigue than austenitic stainless steels because of its lower thermal expansion coefficient and higher thermal conductivity. Grade 91 steel provides excellent mechanical properties at elevated temperatures when produced and heat treated to form the optimum tempered martensitic microstructure.

Proper PWHT and welding practices are essential for the successful use of Grade 91 steel as a durable RPV MOC. There are adequate data on long-term thermal aging with conservative creep-fatigue limits. As with all other potential MOC, additional data on hot He compatibility must be obtained. Also, more data are needed on the Grade 91 steel properties in the thick sections required for the RPV. Grade 91 steel is thus considered the best available high strength ferritic alloy steel regarding high temperature properties provided it is heat treated and welded properly. Sound welding and PWHT are crucial to the successful use of Grade 91 steel. Hardness testing is one quality control/nondestructive examination method of measuring and monitoring the proper hardness of the base metal, weld metal and weld heat affected zones of Grade 91 steel forgings and plate to assure proper PWHT of this alloy. The Energy Production Research Institute has established hardness testing and other inspection programs to assess and confirm the Grade 91 steel properties. Certified Material Test Reports of Grade 91 steel and all RPV grades are essential to confirm the chemical and mechanical properties including hardness at both the mill and the field.

Some other candidate MOC for the RPV also offer superior high temperature properties including:

d. Grade 23(2.25Cr-1.6W-V-Cb), UNS K41650, ASME A-182(Forgings) and A-387(Plate)

Grade 23 ferritic alloy steel is another modification of Grade 22 in which W, V, and Nb are used as alloying elements to obtain superior elevated temperature properties. Its high temperature tensile strength and allowable stresses are significantly better than Grade 22 up to 649°C. In fact, the stress intensities of Grade 23 are only slightly less than Grade 91 steel up to 649°C and are essentially equal to Grade 91 steel in the 482-510°C temperature range, covering the 490°C inlet He gas coolant temperature. An ASME Code Case 2199-3 on Grade 23 was approved on 04/18/06 allowing the use of Grade 23 for Section I construction, which lists its allowable stresses up to 649°C as forgings (A-182), plate (A-387), pipe (A-335), and tube (A-213). The next step would be a Code case for use in Section III, Class 1 components, Subsection NH. Grade 23 has the necessary hot strength if heat treated properly as specified by this code case. It appears to be the economically best option in the 490-580°C core inlet temperature range. Grade 23 would permit thinner wall components than Grade 22 and is more fabrication-friendly than Grade 91 steel. However, tight controls of fabrication are required to prevent possible reheat cracking. Still, a Code case for Sect. III, Div. 1, Subsection NH must be developed and approved for Grade 23.

e. Grade 24(2.25Cr-1Mo-0.25V-Ti-B), ASME A-182(Forgings)

Grade 24 ferritic alloy steel is economically comparable to Grade 23 and, unlike Grade 23, is not susceptible to reheat cracking. It is also expected to be more fabrication friendly than Grade 23. However, Grade 24 is not yet approved for use by ASME Code Section I, much less Section III. It has very good high temperature properties but lacks the necessary ASME Code approvals.

f. Grade 92(9Cr-2W), ASME A-182(Forgings, seamless)

Grade 92 has the best elevated temperature tensile properties of the above ferritic alloy steels at both 490°C core inlet temperature and at 649°C. For example, here are its approximate stress intensities:

482°C : 137MPa; 510°C: 131MPa; 593°C: 83MPa; and 649°C: 39MPa. These are outstanding allowable design stresses over the entire temperature range from ambient (almost 180MPa) to 649°C. It would enable the RPV designer to use lighter wall forgings and components than with the other ferritic steel grades. ASME Code Case 2179-6 was approved on 08/04/06 for the use of seamless Grade 92 tubes, pipes, and forgings in both Sections I and VIII, Division 1 construction. As with all the other ferritic alloy steels, proper heat treatment and welding are critical for its successful applications. Again, Grade 92 needs ASME Section III, Div. 1, Subsection NH approval, but it has some of the best high temperature properties available as an RPV MOC for the NGNP.

Other potential ferritic alloy steels that are being researched and tested for elevated temperature power plant service conditions include Grade E911 (9Cr-1Mo-0.2V-Nb-N). Such grades as E911 have shown significantly improved high temperature strength versus Grade 91, but need ASME code cases for Section III, Class 1, Subsection NH to justify serious consideration as a viable RPV pressure boundary MOC. Grade E911 is covered in ASME A-182 specification for forged alloy (and steam system) piping components for use in pressurized systems.

1.7.2.1 URS-WG Recommendation

Grade 91 steel is our current recommended RPV MOC because it is approved for ASME Code Section III, Division 1, Subsection NH for Class 1 Components in Elevated Temperature Service(to 649°C) in addition to its excellent high temperature properties and extensive power plant service duty. Grade 91 was approved for Subsection NH is both the 2004 and 2007 ASME Codes. The two other alloys with superior elevated temperature properties are Grades 23 and 92, both comparable in hot strength to Grade 91 steel. Their ASME Code status is as follows: (1) Grade 23 was approved for Code Section I construction by Code Case 2199-3 with allowable stresses up to 649°C and offers good MOC economics with its lower alloy content; and (2) Grade 92 was approved for both Code Section I and VIII Div. 1 construction by Code Case 2179-6 with superior high temperature strength properties. Grades 23 and 92 appear to warrant the development of code cases for Sect. III, Div. 1, Subsection NH use to provide alternate RPV MOC to Grade 91 steel. With several URS/Washington Division Engineers actively engaged in ASME Code committees which meet quarterly, if we proactively promoted and supported code cases for Grades 23 and 92 in forgings and plate for Subsection NH use, it should be possible to prepare and issue such code cases within about one year which includes four Code committee meetings. With nobody championing these code cases, the approval process would probably take approx. two years. Additional high temperature property testing must also be conducted on these two grades to qualify them for Sect. III, Subsection NH use. Thus, Grade 91 steel needs the least additional testing and code work to certify its use as an RPV MOC. Grades 23 and 92 merit the necessary code case time and efforts for Sect. III, Div. 1, Sub. NH usage.

If a new alloy chemistry is proposed as a MOC that cannot be included into an existing ASME Section II approved grade or as a modification of an existing grade, then the time frame for approval would at least another year beyond the time required for ASME Code approval – i.e., two + years. ASME requires that the proposed MOC already be accepted by a major materials society such as ASTM before being considered for inclusion as an approved ASME material. Table 1 below summarizes the comparison of major RPV MOC Candidates.

Table 1-14. URS-Washington division assessment of RPV materials.

Material	Chemistry	Current ASME Code Status	ASME Action Needed	Advantages	Disadvantages (Note 2)	Time for ASME NH Approval
Grade 91	9 Cr-1Mo-V	Section III, NH approved to 649°C	None	Excellent high temperature properties Widely Used ASME Section III, NH approved	Sensitive to weld and PWHT variations	Already approved
Grade 92	9 Cr-2 W	Section I & VIII approved	Section III, NH approval – Code Case	Best elevated temperature properties Allows use of the thinnest sections	Sensitive to weld and PWHT variations Not ASME, NH approved	1 year
Grade 23	2.25 Cr-1.6 W-Nb	Section I approved	Section III, NH approval – Code Case	Not as sensitive to welding and PWHT variations More economical that Grade 91 steel	Susceptible to reheat cracking Less stress intensity factors than Grade 91 steel Not ASME, NH approved	1 year
Grade 24	2.25 Cr-1 Mo-2.5-V-Ti-B	No Approvals	Section III, NH approval – Code Case	Good high temperature properties Not as susceptible to reheat cracking Not as sensitive to weld and PWHT variations	No ASME approvals Need greater thicknesses to meet design conditions	1 year
Grade 22V	2.25 Cr-1 Mo-0.25 V	Section VIII approved	Section III, NH approval – Code Case	Good high temperature properties Widely used	Limited high temperature property data available Only approved to 482C Need greater thicknesses to meet design conditions	1 year
New	TBD	No approvals	Section III, NH approval – Code Case	Have ability to choose alloy with potentially superior high temperature properties	Significant time to produce Significant costs to produce No track record of performance	2 – 3 years

Notes:

1. Given the temperatures in the NGNP it is assumed that all material must be ASME Section III, Subsection NH approved in order to qualify for use.
2. All above grades need data on thick section performance and compatibility with hot He.
3. The time frame for ASME approval column includes approximate time frame for inclusion into ASME Section III, Subsection NH. There is no guarantee, however, that approval will be given for use in temperatures equal to Grade 91 steel. The only exception to this is Grade 92. It is very likely that Grade 92 would be approved to the same temperatures of Grade 91 steel or beyond.

1.7.3 KAERI Acquisition and Fabricability Assessment

KAERI in association with a local Korean supplier has provided a preliminary report on the acquisition and fabricability of the following RPV candidate alloys: A 508/533, 2-¼Cr-1Mo, and Grade 91 steel. Their assessment states that material manufacturability and weldability is judged as being poor for the Grade 91 steel materials. They also assessed the transportation and field erection issues associated with building the NGNP at the INL.

1.7.3.1 Melt/Forging Sequence (Korean Supplier Facilities)

Scrap iron is charged and melted in the electric arc furnace and alloy materials (Cr, Ni etc.) are added in the refining ladle furnace to meet the chemical requirements in the steel foundry shop. This Al-killed and vacuum degassed steel is poured into the mold to make the ingot. Ingots ranging from 2 tons to 500 tons have been manufactured. These ingots are shaped into shells, heads, nozzles, blocks and bars by using forging presses of 10,000 tons, 4,200 tons and 1,600 tons.

Heat treatment steps, such as normalizing, quenching and tempering are performed for stress relieving and mechanical property improvement. A test coupon is removed from the forged material, and then simulated PWHT is carried out prior to performing mechanical and metallurgical tests.

When all the requirements are met (Table 1-15), the forged material is finally machined for required dimensions. These forged materials are delivered to manufacturing shop for fabrication with the Certified Material Test Report. Figure 1-15 shows the typical melting/forging process.

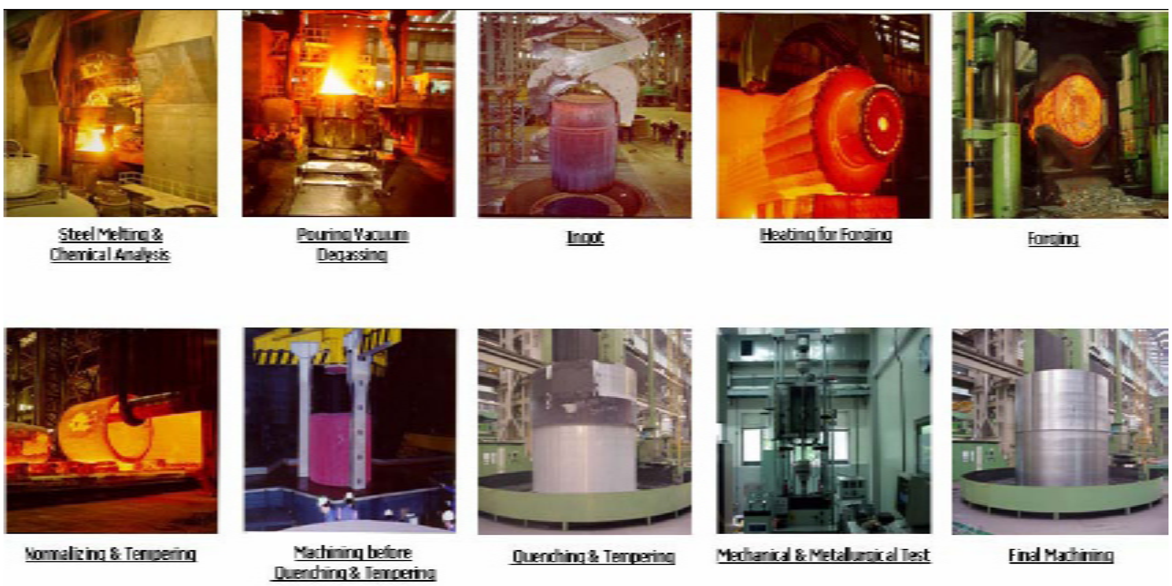


Figure 1-15. Korean supplier melting/forging fabrication sequence.

Table 1-15. Material chemistry and mechanical property requirements

Description		A508 Gr.3 Cl.1	A533, Type B, Cl.1	2.25Cr-1Mo (A387, Gr.22)	Grade 91 Steel (A387, Gr.91)
Chemical Composition Req't	Carbon	0.25	0.25	0.05 ~ 0.15	0.08 ~ 0.12
	Mg	1.20 ~ 1.50	1.15 ~ 1.50	0.30 ~ 0.60	0.30 ~ 0.60
	Copper (Max.)	0.07	0.15	-	-
	Phosphorus (Max.)	0.012	0.015	0.035	0.020
	Sulfur (Max.)	0.010	0.005	0.035	0.010
	Vanadium	0.03	-	-	0.18 ~ 0.25
	Aluminum (Max.)	0.04	-	-	0.04
	Nickel	0.40~1.00	0.40~0.70	-	0.40
	Chromium	0.25	-	2.00 ~ 2.50	8.00 ~ 9.50
	Si	0.15 ~ 0.40	0.15 ~ 0.40	0.50	0.20 ~ 0.50
Mechanical Req't	Tensile strength (MPa)	480-650	550-690	410-585	585-760
	Yield strength, min. 0.2% offset (MPa)	250	345	207	415
	Elongation 50 mm min (%)	20	18	18	18
	Reduction of area, min (%)	38	-	45	-
* Refer to attached Certified Material Test Report for detail information.					

1.7.3.2 *Welding/Fabrication Analysis*

The material manufacturability and weldability of plain carbon steel is satisfactory, and it has been qualified in the overall areas of industries. The Korean supplier has the accumulated experience and knowledge about carbon steel.

The higher alloyed Cr-Mo steels are not as easy to fabricate and weld. The preheating and PWHT are required to prevent cracking in the welding process and special care should be taken when selecting the weld filler material. Therefore, a longer time for component design is required and there will be more manufacturing cost associated with the Cr-Mo vessels. In addition, a test mock-up for Cr steel is required to verify the manufacturability and weldability before final fabrication.

Table 1-16. Welding/fabrication characteristics.

Description	A508 Gr.3 Cl.1 /A 533 B 1	2¼Cr-1Mo (A387 Grade 22)	Grade 91 steel (A387 Gr.91)
Classification	Carbon Steel	Heat Resisting Steel (Boiler equip't)	Heat Resisting Steel (LNG Tank Ship)
Material Manufacturability	Good	Poor (Due to segregation)	Poor (Due to segregation)
Welding Rod	EA-3 Type (Mn-Mo-Ni Alloy) ASME Sec.II Part C SFA 5.23 (Annex A7.1.1)	EB-3 Type ASME Sec.II Part C SFA 5.23 (Annex A7.1.2)	EB-9 Type ASME Sec.II Part C SFA 5.23 (Annex A7.1.2.1)
Weldability	Satisfactory	Poor (Weld cracking)	Poor (Weld cracking)

1.7.3.3 Korean Supplier RPV Assembly and Transportation and Assembly Options

Subassemblies by Shop Manufacturing

- **Option 1**: Head and 1 piece body are manufactured in the shop, and assembled on the site.
- **Option 2**: Head and 2 piece body (Upper Body and Lower Body) are manufactured in the shop, and assembled on the site.

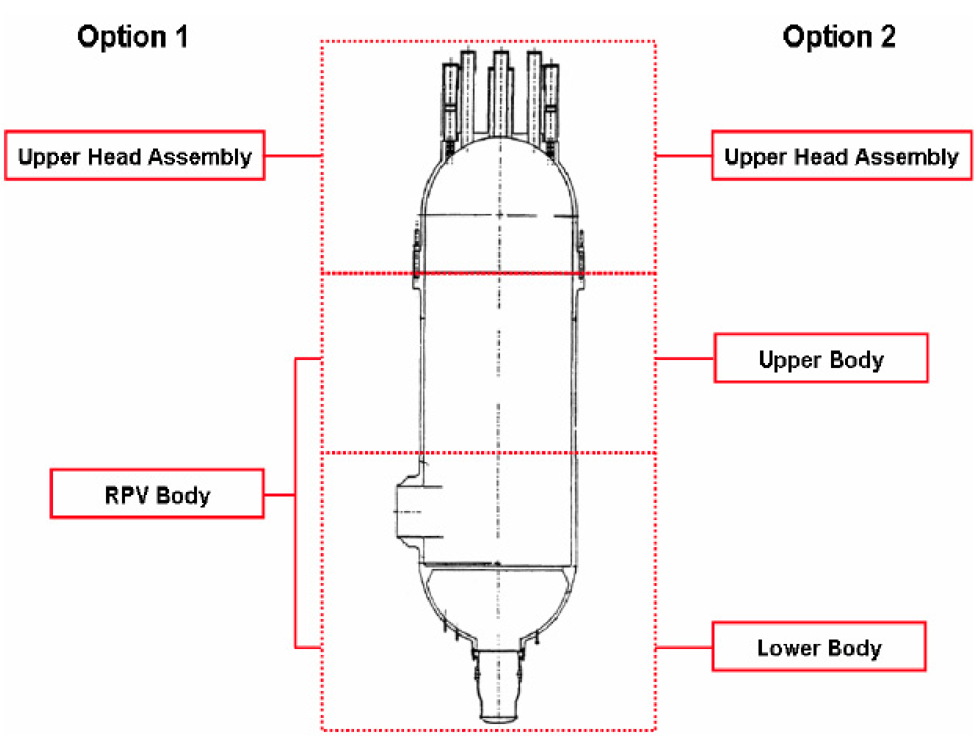


Figure 1-16. Subassembly method for different configurations.

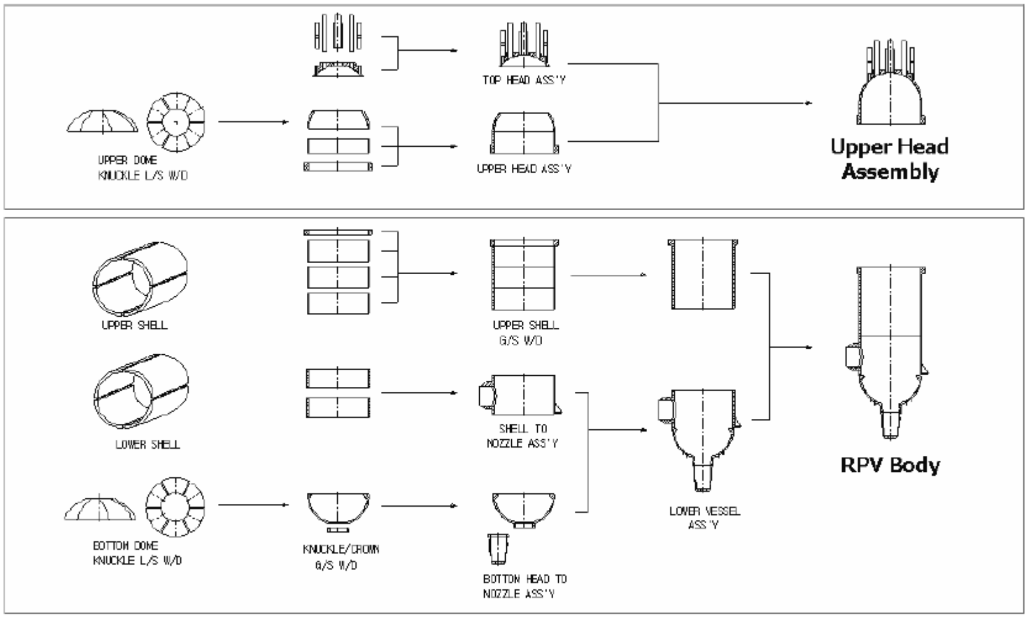


Figure 1-17. Option 1(1 piece body).

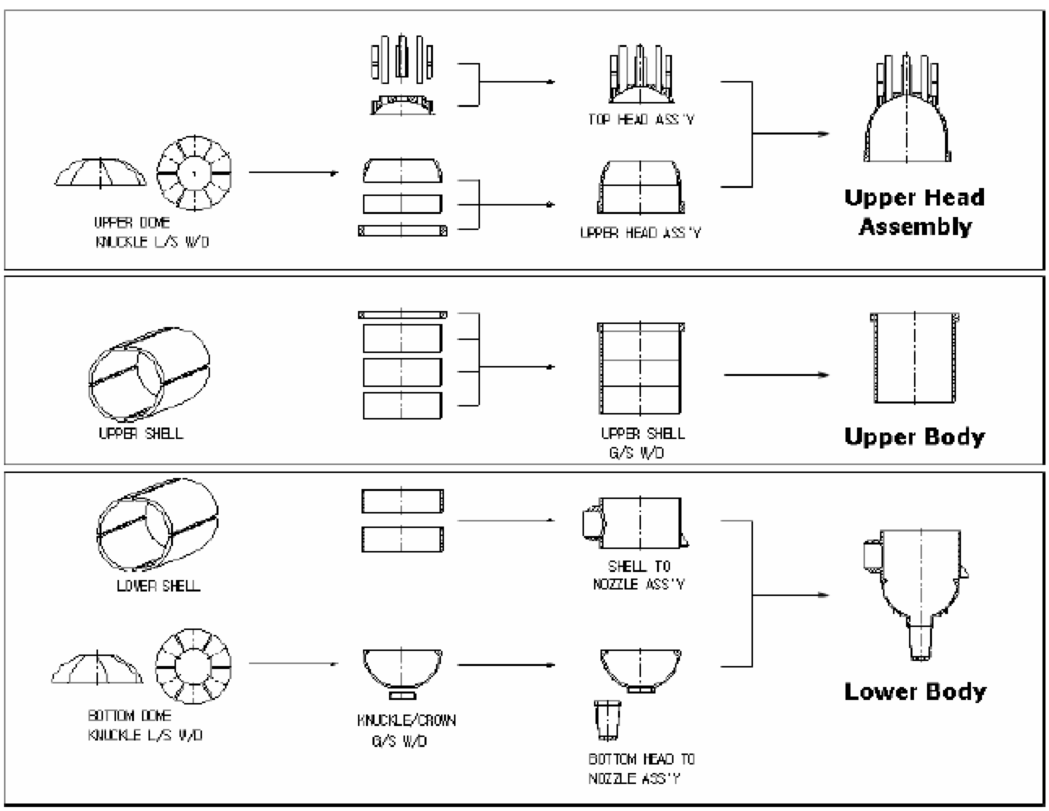


Figure 1-18. Option 2 (2 piece body).

S/N	Plate Thickness (mm)	Plate Length (mm)		Arrangement	Configuration
		Single	Total		
1	220	2,800	2,800	60°	
2	225	1,660	1,660	90°	
3	600	997	997	90°	
4	600	718	718	90°	
5	225	2,624	7,872	90°	
6	270	2,370	4,740	90°	
7	170	4,620	4,620	60°	
8	50	1,796	1,796	180°	
9	50	508	508	180°	
10	50	712	712	180°	

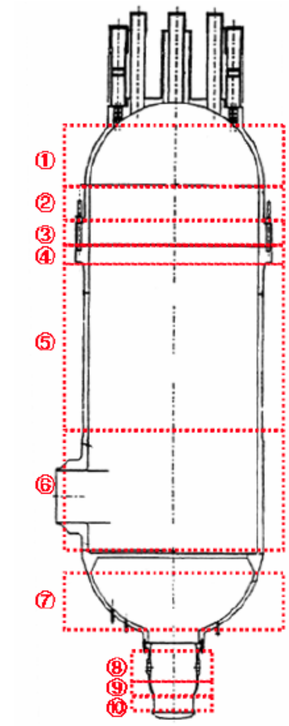


Figure 1-19. Shell/head fabrication by plate bending with bill of material.

Table 1-17. Forging bill of material.

SN	OD (mm)	ID(mm)	Depth (mm)	Qty	Remark
1	640.0	–	–	31	Nozzle
2	5,718.0	5,163.0	318.7	1	–
3	7,632.7	7,226.3	203.2	2	–
4	7,658.1	7,226.3	215.9	4	–
5	3,046.4	–	260.4	1	Nozzle
6	7,747.0	7,226.3	260.4	2	–
7	7,504.4	6,807.8	347.4	1	–
8	127.0	–	200.0	45	Nozzle
9	3,060.0	–	181.4	1	Nozzle
10	2,032.0	1,932.0	50.0	2	–
11	533.0	–	50.0	2	Nozzle
12	2,032.0	1,932.0	50.0	1	–

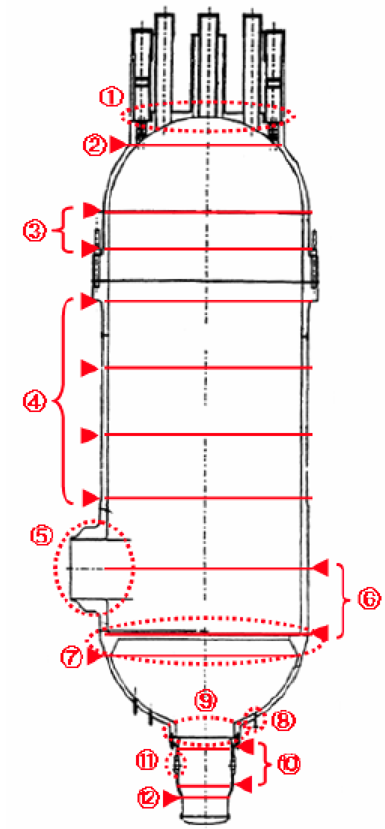


Figure 1-20. Shell/head fabrication by forging.

Table 1-18. Weight and dimensions of each subassembly, one piece body.

Component		OD (mm)	ID (mm)	Height (mm)	Weight (ton)
RV	Head	8,420.1	7,226.3	10,207.5	435*
	RPV	8,420.1	7,226.3	21,008.3	924*
	Total	8,420.1	7,226.3	31,215.8	1,364
* The weight of stud and gasket is excluded. (But it is included in total weigh.)					

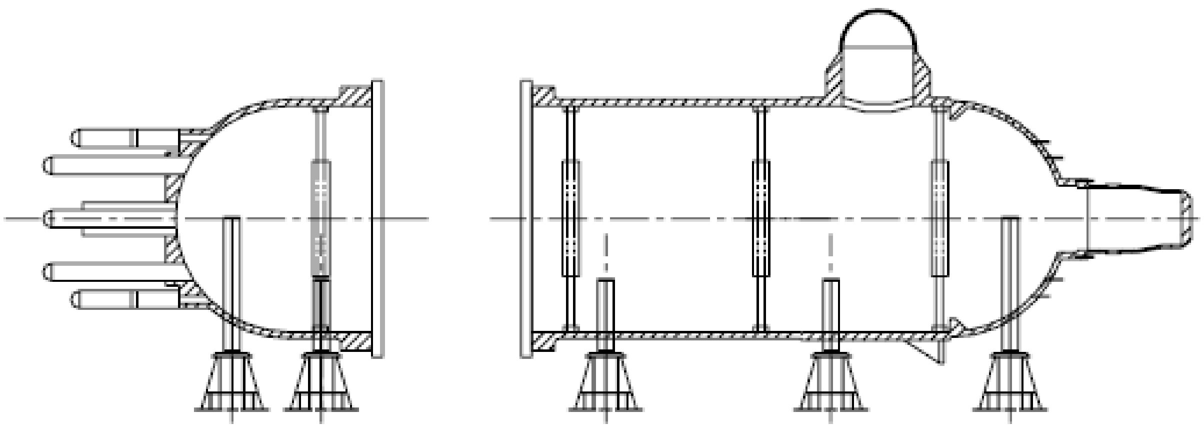


Figure 1-21. One piece body.

Table 1-19. Weight and dimensions of each subassembly, two piece body.

Component		OD (mm)	ID (mm)	Height (mm)	Weight (ton)
RV	Head	8,420.1	7,226.3	10,207.5	435*
	Upper Body	8,420.1	7,226.3	21,008.3	924*
	Lower Body	8,420.1	7,226.3	12419.8	599*
	Total	8,420.1	7,226.3	31,215.8	1,364
* The weight of stud and gasket is excluded. (But it is included in total weigh.)					

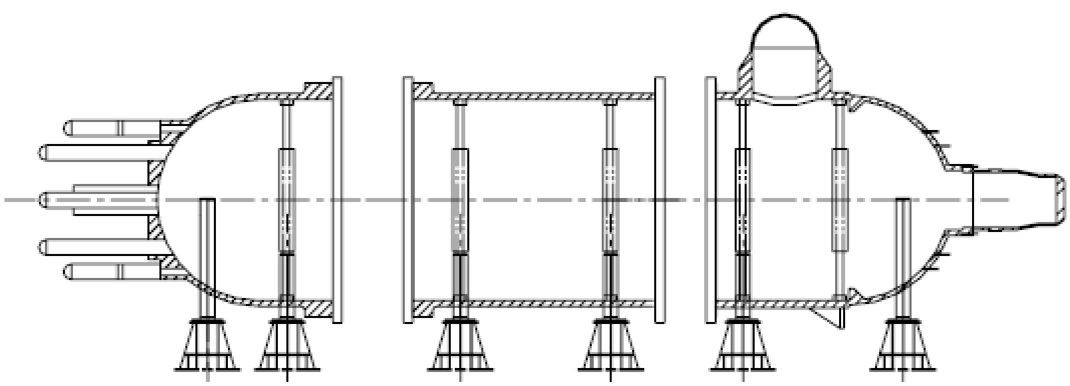


Figure 1-22. Two piece body.

1.7.3.4 Transportation/Site Assembly

Considerations for transportation and on-site fabrication are:

- If the transportation and fabrication conditions are same, there is no difference in the vessel transportation among the candidate materials. Both A 508/533 and Grade 91 steel have painting and package requirements, and the internal support to prevent deformation during transportation is required for all materials.

- When the Cr-Mo vessel is assembled on the site, special control is required in the selection of weld material, preheating, and PHWT. Because the Cr-Mo steel has poor weldability compared to carbon steel.
- Considerations for the Ground (Marine) Transportation – Route Survey
  a. Bridges, tunnels, roundabout ways in all transportation routes should be investigated.
  b. Packing, painting and nitrogen fill-up should be performed to prevent corrosion of the internal/external component.
  c. The barge arrival date considering the flux and reflux of the tides should be determined, and the preparation of the pull-up process is required.
  d. Cranes, lifting/handling equipment, pedestals and multi-loaders should be prepared for both the ground and marine transportation.
  e. The estimated problems should be considered for the approval of the transportation.
- Considerations for the Option 1 (1 Piece Body) Manufacture
  a. In case of long-term storage on the site, continuous control is required to prevent corrosion.
  b. Saddle, Up/Down Ending method, Tie-Down requirement and the method of using Lift Lugs should be checked.
  c. Extra-large crane (1000 ton) for the overweight RPV body is required.
  d. When assembling the head and body, a special tensioner equipment is required for the stud/nut assembly.

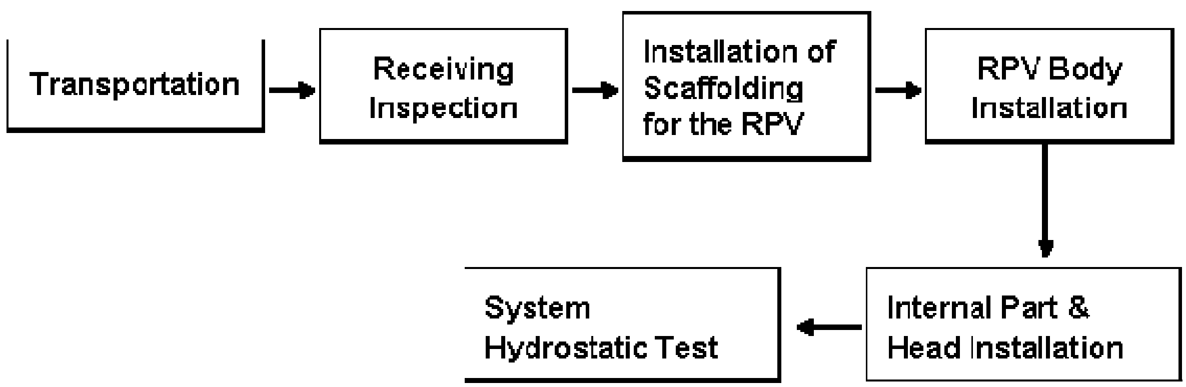


Figure 1-23. Option 1 (1 piece body) transportation and assembly sequence.

- Considerations for the Option 2 (2 Piece Body) Manufacture
  a. In case of long-term storage on the site, continuous control is required to prevent corrosion.
  b. Girth seam welding for the bodies is required on the site. Welding is to be performed in the horizontal direction after the installation of the Lower Body.

     Therefore, the following items should be developed.
     - Automatic welding equipment considering the weldability
     - Weld preparation
     - Requirement for the weld misalignment
     - Mock-up for welding
     - Welding process
  c. Saddle, Up/Down Ending method, Tie-Down requirement and the method of using Lift Lugs should be checked.

d. Extra-large Crane (1000 ton) for the overweight RPV body is required.

e. When assembling the head and body, a special tensioner equipment is required for the stud/nut assembly.

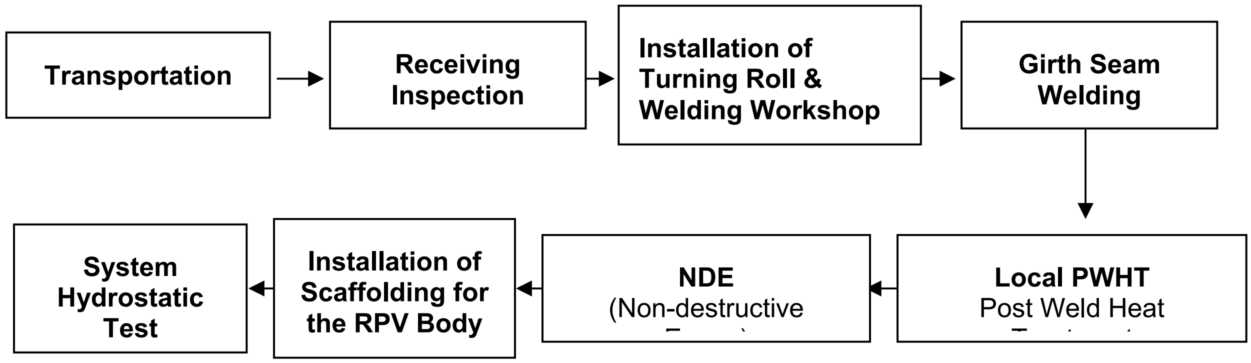


Figure 1-24. Option 2 (2 piece body) transportation and assembly sequence.

1.7.4 Toshiba Assessment of JSW Forging Capabilities

Toshiba Corporation assessed the capabilities of JSW to fabricate forgings for the RPV for a 600-MW(t) prismatic block NGNP from the various candidate materials currently under consideration (A 508, 2¼Cr-1Mo, and Grade 91 steel). Toshiba Corporation also assessed the current backlog for forging construction at JSW to determine the approximate date by which forgings for the RPV would have to be ordered to obtain delivery of the RPV in time for a 2021 NGNP startup.

Toshiba Corporation met with JSW to discuss the current capabilities of JSW. In addition, JSW provided answers to specific questions posed by GA (see Table 1-20). As indicated in Table 1-20, JSW is starting to develop capability to supply Grade 91 steel forgings to support the Fast Breeder Reactor (FBR) program in Japan,[a] but this program is still in the very early stages, and it is highly unlikely JSW would be able to supply forgings of this material in time to meet a 2021 NGNP startup. For this reason, JSW strongly recommends use of A 508 steel for the NGNP RPV. Estimates for the RPV thickness using A 508 steel are given below:

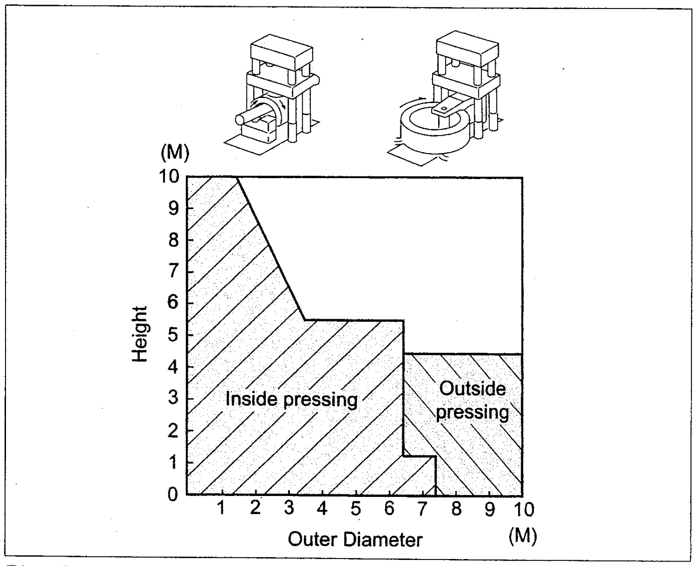
Cylindrical Shell:	152 mm
Hemispherical Domes:	102 mm – 127 mm
Vessel Support Interfaces:	203 mm

Figure 1-25 shows the dimensional capabilities of the JSW forging facilities. Ring forgings are limited to heights of 10 m and outside diameters of 10 m. Further limitations are imposed by the round furnace and quench tank, both of which can accommodate ring forgings with diameters of 9 m and heights of 6 m. However, because the quench tank has water-circulation nozzles installed on the tank wall, ring forging diameters are further limited to 8.2 m, unless the height of the forging is below the height of the nozzles. JSW did previously manufacture a ring forging for the Monju FBR with dimensions 8.760 m OD × 7.780 m ID × 0.783 m H.

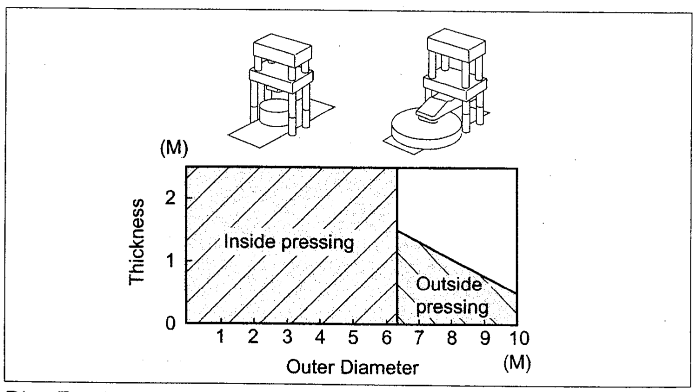
a. Japan intends to re-start the Monju prototype FBR in the near term and intends to start deployment advanced FBRs around 2030[16]

Table 1-20. JSW answers to questions from GA.

Question from General Atomics	Answer from Japan Steel Works
For what materials does JSW have existing nuclear pressure vessel component forging manufacture capability (forging process equipment, process technology and ASME B&PV code qualification)?	The manufacturing facilities are not specific material and are used for LWR RPV material, turbine rotor material (high-Cr steel) and other materials.
What are the basic steps and process conditions used by JSW for manufacture of nuclear pressure vessel forgings?	The basic steps for manufacturing of the nuclear reactor pressure vessel can be obtained from the home page of JSW: http://www.jsw.co.jp/en/guide/facilities.html Basically, it involves, melting, refining, pouring, ingot-making, heat treatment, machining, and products.
What size limitations does the JSW nuclear vessel component forging manufacturing process have (finished weight, finished ring forging ID & OD, and finished ring forging length)?	The product size and weight are strongly affected by the maximum possible ingot weight. The maximum possible ingot weights are 600 t for A-508, 250 t for 2¼Cr-1Mo, and 120 t for Grade 91 steel. The product weights are typically results less than about 30 % of the ingot weights (~10 % in case of complex shapes).
In previous discussions, the maximum JSW finished ring forging OD has been indicated to be about 8.2m and that the maximum OD was limited by the size of the existing quench tank. Could the quench tank size be increased, or could an alternative quench process be used (e.g., water spray)?	At present, there are no plans to increase the size of the quench tank, and current operations at JSW do not provide any schedule leeway to remodel any major facilities. The 8.2 m diameter limitation for RPVs results from water-circulation nozzles on the inside of the tank wall, which has an ID of 9 m. If the product height is sufficiently small (below the height of the tank nozzles), larger diameter forgings can be put into the quench tank. An example is a ring forging for the Monju FBR (8.760 m OD × 7.780 m ID × 0.783 m H).
JSW has also previously indicated to GA that they will only supply A 508 nuclear vessel forgings based on their currently developed forging process. If this is still the case, could JSW develop the necessary forging process capability, including ASME code qualification, for the alternative NGNP nuclear vessel materials under consideration (2¼Cr-1Mo and Grade 91 steel)? If so, what would be the order of magnitude for both the cost and schedule of the process development work for each of these alternative NGNP vessel materials? If the development work is done, what would be JSW's projection of the finished forging size limitations for the alternative materials?	JSW still recommends using A 508. Technically, 2¼Cr-1Mo is possible. But the product height is relatively small and the number of the welding lines increases because of the relatively small ingot weight of 250t. Also, further study is needed for the cross vessel joint region to maintain the required height. For Grade 91 steel the ingot size is limited because of segregation. Because of the current emphasis on FBR development in Japan, work has started on developing large-sized forged products of Grade 91 steel appeared for FBR in Japan, but this work is still in the design study phase. Because of the segregation problem, experimental work is also needed. Also, ingot manufacturing facilities require reservations of 5 to 6 years in advance. Hence, the cost, schedule, and forging size limitations cannot be determined at this time.
Does JSW have nuclear vessel welding process capability for joining forged components (process equipment, process technology and ASME B&PV code qualification)? If so, for what materials and what are the size limitations for finished welded assemblies?	JSW performs prefabrication of nuclear vessel components. However, welding and fabrication of these components into final products are conducted by other companies, typically heavy industries companies. JSW does not perform ASME certification of nuclear components. These certifications are performed on the final product by other companies.



Ring Forging Process and Capacity



Disc Forging Process and Capacity

FACILITIES

14,000 Ton Hydraulic press with facility for outside pressing
8,000-Ton Hydraulic Press with facility for cross die forging
3,000-Ton Oil Hydraulic Press with numerical control

Figure 1-25. Dimensional capabilities of JSW forging facilities.

Manufacturing large-sized forged products requires large-sized ingots. JSW has used ingot sizes of 600 T for A 508. For 2¼Cr-1Mo, the largest ingot size used by JSW is 250 T, but this size was determined by product requirements. It may be possible to use larger-sized ingots for 2.25Cr-1Mo, but quality requirements need to be confirmed. For Grade 91 steel, the ingot weight is currently limited to 120 T, because segregation causes difficulty with making homogeneous ingots. As discussed above, R&D efforts have been initiated for the purpose of making large-sized ingots with Grade 91 steel, in order to manufacture large-size forgings for the Japan's advanced FBR concept. Regardless of the original ingot weight, product weights are typically 30% or less of the ingot weight, and as little as 10% of the ingot weight for complex shapes.

According to JSW, ingot manufacturing facilities require reservations of 5 to 6 years in advance. An additional 3 to 4 years is required to produce the final RPV product. Hence, even if the recommendation by JSW to use A 508 steel is adopted, the RPV could not be procured in time to support a 2021 NGNP startup, unless an effort is made to assign a high priority to production of the NGNP RPV. This would likely require some sort of government-to-government cooperation between the US and Japan on NGNP. However, development of the FBR and deployment of additional LWRs currently have a much higher priorities in Japan.

2. FORGING SUPPLIERS

AREVA generated a list of potential worldwide forging suppliers (See Table 1-21) as part of their study.[14] The first seven companies listed would be of greater interest as suppliers of RPV forgings or forged plates because of their capability to produce heavier pieces.

Table 2-1. Worldwide forging capability.

Manufacturer	Press Capacity, Tons	Ingot Capacity, Tons	Forging Capacity, mm	Plate Forging Capacity, mm	Materials	Comments/Remarks & etc.
Ladish Co., Inc. Cudahy, WI, USA	17,000	18 to 27 Crane capacity to handle 100	7,500 to 8,250 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	272 ton Dimension 3,300 mm Diameter	Shell, 1,875 OD × 1,150 ID × 5,150; could go to 5,000 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 Isothermal press 12,500 tons. Forgings – 8,400 Diameter, 3,000. in face height and more than 2 tons
Scot Forge Spring Grove, IL, USA	5,500 ton is largest press, 7 other presses, 4 hammers all of various sizes from 2 hammer to 4000 ton press	36	6,000 OD depending on other variables (id, wall thickness, length) wt. lbs = finished rough forging ~35 ton (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 450 ton capacity. Quench Tanks to 12,600 long or 6,000 mm in diameter. 75 ton Ovrhd crane.
Japan Steel works, LTD, Muroran, Japan	14,000	500	8565 mm OD × 7360 mm ID × 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.

Table 2-1. (continued).

Manufacturer	Press Capacity, Tons	Ingot Capacity, Tons	Forging Capacity, mm	Plate Forging Capacity, mm	Materials	Comments/Remarks & etc.
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 × 4000 mm H.
Taewoog Co. LTD BuAn, S. Korea	8,000					125 MT manipulator. Ring Rolling Mill – 9000 mm OD × 2800 mm H × 60 ton max. HT furnaces. Quench Tanks.
SFAR Steel Creusot, France (acquired by AREVA)	11,300	250	Flanges – 6,500 OD Shells – 6,900 mm OD Discs – 6,500 mm OD			Boring Mach. 400 ton. Heat and HT furnaces. Svcs nuclear ind.
Smaller Domestic & Foreign Forgers						
Ellwood City Forge Ellwood City, PA	4,500	35	1,250 Dia and up to 15,000 lengths		CS, SST, Steel Alloys	Builds for the nuclear navy and power industries; Max length 15,000 mm ; discs diameter 2,750 mm; Addle rings OD 2,175
Jorgensen Forge Corp Seattle, WA	5,000	125			Steel Alloys, Ti	Open die forging on 660T, 1250T, 2500T and 5000T presses; 2400 ton ring expander capable of stretching rings to a maximum 5,650 mm diameter. Max length 22,500 mm ; discs diameter 2,375 mm; Addle rings OD 3,750 mm
Patriot Forge Branford, ONT. Canada						Max length 12,000 mm ; discs max diameter 12,000 mm; Saddle max OD 2,000 mm

Table 2-1. (continued).

Manufacturer	Press Capacity, Tons	Ingot Capacity, Tons	Forging Capacity, mm	Plate Forging Capacity, mm	Materials	Comments/Remarks & etc.
Kropp Forge Cicero, IL	2,000	9			Steel Alloys, Ti	Hydraulic Forging Presses 750 tons, 1000 tons, 1500 tons and 2000 tons; Trim presses up to 1750 tons. Shafts max length 7,500 mm Shafts: (high temp alloys- max length 6,000 mm
Canada Forgings, Inc Welland, ONT. Canada		20				Shaft max length 120,000 mm; disc max dia 1,900 mm; saddle rings max OD 2,000 mm
Sorel Forge Company Sorel, Quebec. Canada (Acquired by A. Finkl Forge)	5,000	38	1,575 Dia.	1,575 Dia.		5000-ton open-die hydraulic press equipped with its wide dies; produce forgings up to 15,000 mm long; 2000 ton press offers more flexibility. This system runs computer assisted, fully synchronized with the 40 meter-ton rail-bound manipulator. Stainless max shaft length - 4,000 mm; disc max diameter 70; Saddle rings max OD 1,750 mm. Carbon Max. Shafts length - 11,500 mm, Discs max dia – 2,500 mm , Addle rings max OD 2,875 mm.
Dayton Forging & Heat Treating Dayton, OH		8		SST, Steel Alloys		Stainless max shaft length 6,000 mm. Shaft max length 7,500 mm in. Disc max dia 1,500 mm. Addle rings max OD 1,500 mm
A. Finkl and Sons Chicago, IL (Acquired by SCHMOLZ + BICKENBACH AG)		57				Max length 15,000 mm Discs max diameter 3,000 mm Saddle max OD 3,000 mm

Table 2-1. (continued).

Manufacturer	Press Capacity, Tons	Ingot Capacity, Tons	Forging Capacity, mm	Plate Forging Capacity, mm	Materials	Comments/Remarks & etc.
Nova Forge Trenton, NS, Canada		45				Carbon Max Shafts length 15,000 mm. Discs max dia 3,300 mm. Saddle rings Max OD 4,100 mm.
Wyman-Gordon Houston, TX Livingston, Scotland	35,000 30,000				CS, steel alloys, SST, Duplex, Nibased alloys, Ti, pwdr metals	Presses (vertical extrusion process) used to produce pipe: 225 mm ID to 1,000 mm ID with wall thkns 15 through 175 mm Services the industries of aerospace, power generation, process, oil, gas, marine and nuclear
Liberty Forge Liberty, TX						2 - 400 kW American Induction Heating Billet Furnaces 1 – 1250 kW 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

3. POTENTIAL NUCLEAR PRESSURE VESSEL MANUFACTURERS

An issue for the fabrication of the RPV and the IHX pressure vessel the identification of vessel fabrication vendors with the appropriate ASME certifications to perform nuclear work. The number of these firms has declined over the last 20 years and the NGNP will be competing for these services with resurgent orders for LWR's and chemical process facility components in a world market. Table 1-22 lists the prospective vendors for this work.

Table 3-1. Qualified nuclear vendors.

Vendor	Capability	Quality Assurance	Nuclear Experience
Precision Custom Components (PCC), York, PA.	The PCC facility size exceeds 25,000 square meters under one roof and is conveniently located to major transportation routes including rail, truck, and deep water access in Baltimore, MD and Philadelphia, PA. PCC's core competency is in fabrication of heavy vessels (to 600 tons) involving special materials with challenging welding and machining requirements, tight tolerances, and robust Quality Assurance procedures, including NQA-1. PCC has fabricated large pressure vessels and other vessels and equipment for the commercial nuclear and process industries including Westinghouse, GE, AREVA, ExxonMobil, Dow, DuPont, and others. Facilities include large horizontal and vertical boring mills, gantry mills, complete automated and manual welding capability and weld development laboratory, heat treatment facilities, 150 ton overhead cranes, deep assembly and test pits, and complete NDT capability in house including a 4MeV radiographic inspection facility.	The PCC Quality Assurance program is routinely audited to the requirements of ASME NQA-1, 10 CFR 50, Appendix B, 10 CFR 71, Subpart H and 10 CFR 72, Subpart G by various nuclear equipment designers and electric utilities. PCC was also audited to the ASME Code, Section III, Division 1, Subsection NCA, Article 4000 and Section III, Division 3, Article WA-4000. As a result of this audit, PCC has received an "N", "NPT" and "NS" Certificate of Authorizations and a "NTP" Certificate of Authorization. Since 1991 PCC's quality system has been audited twenty-two times by utilities and equipment designers and five times by the Nuclear Regulatory Commission and has been found to be in compliance. PCC is one of only nine firms worldwide who maintain ASME Section VIII Division 3 certification for the design and manufacture of ultra high pressure vessels.	PCC has supplied major nuclear reactor primary system components (e.g., reactor heads, closure heads, and steam generators), reactor service equipment, fuel cycle, and related components to the U.S. Navy, NSSS providers, EPC's, and electric utilities and has supplied major equipment to the Department of Energy's National Laboratories including Lawrence Livermore Lab, Sandia Lab, Los Alamos Lab, Brookhaven Lab, and Jefferson Lab.
ENA (Equipos Nucleares, S.A.), Catabria, Spain		ASME "N" and other stamps since 1978 AD - HPO ÜBERPRÜFUNG (TÜV) ISO 9001	PBMR RPV and other components
GE-Hitachi Nuclear Energy, Custom	ASME Sections III and VIII Pressure vessel work has been a core business	GEH has established a Quality Assurance Program to assure that all fabrication and	US Nuclear Navy Nuclear Propulsion system components.

Table 3-1. (continued).

Vendor	Capability	Quality Assurance	Nuclear Experience
Fabrications, Canonsburg, PA	since its inception. The business has maintained its U certificate of authorization continuously since the 1960's. Additionally, the business has maintained its N type certificates until 1986; these were reestablished in 2000 and maintained since. The facility is equipped with all the necessary welding and machining capabilities and serves as a state-of-the-art heavy fabrication facility. Major facility highlights are as follows: 30,300 m^2 of Manufacturing Space Under Roof (with room for expansion) In-Plant Rail Spur 250 Ton Overhead Lifting Capacity 3 Shift Non-Union Facility CNC Five Axis Waterjet Cutting Machine Vertical Milling Parts Up To 3,000 mm In Diameter and 2,750 mm High Horizontal Milling Parts Up To 5,000 mm Long and 2,500 mm Wide Lathes with 750 mm Swing and 12,900 mm Between Centers 6,000 X 6,000 X 5,000 mm Assembly Pit Blasting and Painting Facilities 4,500 X 7,500 mm State-Of-The-Art X-Ray Facility Rolling and Bending Capabilities Full Range of Welding and Inspection Equipment	construction of items and supply of material are in compliance with the latest requirements of the ASME Boiler and Pressure Vessel Code Section III, Division 1 and Division 3, 10 CFR 50, Appendix B, 10 CFR 71 Sub-Part H, 10 CFR 72 Sub-Part G, 10 CFR 21, ASME NQA-1 (Quality Assurance Requirements for Nuclear Facility Applications) and contractual requirements. All ASME B&PV Code work performed will be as defined in the scope of our Certificates of Authorization. GEH Custom Fabrication has been qualified and audited by such distinguished groups as the US Nuclear Regulatory Commission (NRC), the American Society of Mechanical Engineers (ASME), the United States Department of Defense (DOD), and numerous commercial enterprises (Westinghouse, Bechtel, NAC International, Transnuclear, Packaging Technology and Fluor Hanford, etc.).	LWR replacement parts such as pressure vessels, control rod drive mechanisms, strainers, steam dryers, etc. Nuclear fuel management systems such as spent fuel canisters, transfer casks, wet storage racks, etc.
Doosan Heavy Industries & Construction,	Steel foundry with electric arc furnaces, vacuum ladle refining	ASME N, NA,NPT, U, u2, etc.	Korean nuclear power plant components including:

Table 3-1. (continued).

Vendor	Capability	Quality Assurance	Nuclear Experience
Seoul, S. Korea	vacuum steam degassing. Forge shop with 10,000/13,000, 4,200 and 1,600 ton presses Nuclear fabrication shop Heavy machine shop		Nuclear Steam Supply System at Yonggwang 1&2, Ulchin 1&2 Design and construction of Ulchin 3&4 Yongggwang 5&6
FRAMATOME ANP, Chalon/St. Marcel, France	Shipping on Saone River Three bays, 35,800 m^2 of shops 1000T lifting capacity Fully automated large welding gantries Vertical lathes, boring and milling machines Heat treatment to 600T Steam generator fabrication capability	ASME Section III, N, NPT ISO 9001 ISO 14001 RCC-M per AFCEN (French Assoc. For Design and Construction of Nuclear Power Plant Materials) French Regulations on Pressurized Water Reactor (PWR), French Nuclear Steam Supply System Control Office	600 heavy components Manufactured all heavy components for French PWR's, (over 500) Worldwide nuclear industry heavy components (nearly 100)

4. CONCLUSIONS

4.1 AREVA Conclusions

The AREVA 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.

As far as material selection is concerned, the following materials are considered as credible candidates for start-up by 2021:

- A 508/533
- Grade 91 steel
- 2 ¼Cr-1Mo annealed.

The allowables of 2¼ Cr-1Mo annealed are however probably too low and would require thicknesses which would make this option not economical. Fabricability issues would have also to be clarified.

2¼Cr 1Mo V is also considered as a good candidate for such an application, with expected reduced feasibility issues for welding compared to Grade 91 steel but the time required to qualify it for the NGNP is not expected to be consistent with NGNP schedule.

Design alternatives have been identified and potential suppliers listed. Japan Steel Works (JSW) is confirmed to be the only supplier capable of providing the forgings necessary for the RPV. JSW's ability to fabricate large forgings made of Grade 91 steel will be discussed in a revision to this report, following a meeting with them on mid April 2008. Draft component specifications have been prepared in preparation to this meeting.

This study identifies also 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 A 508/533 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 A 508/533 vessel. However, whether these options are preferable to a vessel made of a higher temperature alloy remains to be determined.

4.2 GA Conclusions

1. Information provided by both JSW and KAERI confirm that use of A 508/533 for the RPV material of construction is essentially required in order to support a 2021 NGNP startup date. Use of Grade 91 steel (or other high-alloy steels) is probably a better material in terms of design optimization, but it is highly unlikely an RPV manufactured from this material could be procured in time to support a 2021 startup date, even with a dedicated, international effort. A key issue associated with Grade 91 steel is uniformity of properties in the ingot sizes required to manufacture the large forgings needed for an RPV. Developmental work should continue on Grade 91 steel (and possible some other high-alloy

steels), since higher temperature capability could be a significant advantage for follow-on commercial plants. Opportunities for collaboration with Japan on development of Grade 91 steel should be investigated, since Japan has started developmental work on this material to support deployment of advanced FBRs in the 2030 time frame.

2. Assuming A 508/533 steel is used as the material of construction for the RPV, operation with a coolant inlet temperature of 590°C will require use of an active VCS to ensure compliance with ASME code limits. Thermal analyses performed by KAERI and Fuji Electric Systems indicate a VCS should not be required if it is possible to operate the NGNP with a higher core ΔT (and lower coolant mass flow) by lowering the inlet temperature to 490°C. Design measures to optimize power and coolant flow distributions should result in acceptable fuel temperatures during normal operation with coolant inlet/outlet temperatures of 490°C/950°C. Some of these design measures (e.g., restraint mechanisms and sealing keys to reduce bypass flow) will require additional design work and technology development to demonstrate their feasibility and effectiveness.
3. Stress analyses using the ANSYS code have confirmed the structural integrity with respect to ASME code guidelines of RPVs manufactured from either A 508/533 or Grade 91 steel.
4. Toshiba Corporation has recommended A 508/533 steel as the material of construction for IHX vessels and has included Kaowool insulation as part of the design to protect the vessels from creep damage.

4.3 INL Recommendations

Two steels are being seriously considered for application in the NGNP reactor pressure vessel; conventional light water reactor pressure vessel steels A 508/533 and the higher alloy Grade 91 ferritic/martensitic steel. A 508 has the advantage of being fully incorporated into the Nuclear Section of the ASME Code. Due to extensive use of this steel in LWRs there is an extensive experience base with the material and sufficient irradiation testing so that no further irradiation experiments will be required. The major drawback for application of A 508/533 for the NGNP is that the material is only suitable up to 371°C. Assuming reasonable operating margins, this would require that the RPV be cooled to a temperature probably below 350°C. A 508/533 are low alloy steels that do not require a quench and temper heat treatment to fully develop their properties.

Grade 91 steel is also incorporated in the ASME Code and its use would allow operation of the RPV to a temperature above 400°C. There is no significant experience with this material for nuclear reactor vessels and irradiation testing would be required as part of the licensing. In order to obtain optimal creep properties this steel must be normalized at 1080°C, quenched into a fully martensitic condition and tempered at about 750°C. Insufficiently rapid quenching from the normalizing temperature will result in a microstructure that contains ferrite and coarse carbides which seriously degrades the creep properties. It is not clear for the heavy sections required in the NGNP RPV design that through-thickness properties can be obtained for this steel.

The NGNP RPV will be very large diameter, likely greater than 7 m, and very heavy section, greater than 150 mm in thickness. Restrictions on the diameter of a vessel that can be transported by road or rail suggest that the vessel will have to be fabricated on the reactor site. On-site fabrication will require the ability to do field welding and post-weld heat treatment. Welding methods are well established for A 508/533 and no significant technical issues are anticipated for on-site fabrication of an RPV from these steels. Grade 91 requires sophisticated pre-weld heating, careful temperature control during the welding process, and post-weld heat treatment. The usual post-weld heat treatment is tempering the martensitic weld metal. Restricting the PWHT to only tempering the weld metal can leave the heat affected zone susceptible to premature creep cracking. It would be necessary to normalize and quench the fabricated vessel to obtain optimum creep properties. It is not clear that such a heat treatment is possible for on-site fabrication.

There is currently a world-wide shortage of fabrication capability for nuclear qualified components. Potential vendors for components of sufficient size for the NGNP RPV have considerable experience with A 508/533 steel. There is no experience with forging Grade 91 steel in heavy sections for nuclear applications. No plans appear to be in place currently to develop capability to forge large Grade 91 components. The maximum forging size that can be obtained is closely related to the maximum ingot that can be cast by the forge shop. Ingots up to 450 T are possible for A 508, while issues associated with segregation during solidification limit Grade 91 ingots to about 120 T.

The combination of technical maturity, availability, and fabricability strongly suggest that an A 508/533 RPV presents the minimum technical and schedule risk for the NGNP project. While this steel is not the preferred candidate of all of the vendors that have completed pre-conceptual design studies, all of these studies have indicated that this is an acceptable choice of materials.

Given the shortage of capacity to fabricate heavy sections for nuclear components and the anticipated demand for LWR vessels, the current delivery time is estimated to be ten years. It is imperative that the NGNP design be finalized and an order placed for the vessel in the next several years to support construction in 2021. It is also recommended that the NGNP project continue to work with potential vendors that are considering adding capacity to fabricate large nuclear components to ensure that large components are available.

The following recommendations/comments are made to further define the RPV and IHX vessel acquisition strategy and define the risk for obtaining the properly designed and fabricated RPV and IHX pressure vessels to meet the 2021 NGNP start up date:

1. Complete the overall NGNP reactor system design
2. Choose the appropriate RPV and IHX vessel materials
3. Complete the detailed design of the RPV and IHX pressure vessels
4. Work with material suppliers and vessel fabricators to ascertain the delivery schedule for the heavy section materials and the completed components to the INL site.
5. Work with the construction contractor and/or vessel fabricator to assure correct assembly of these vessels as regarding welding and heat treatment procedures.

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