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Report on the Survey of the Design Review of New Reactor Applications

Volume 4: Reactor Coolant and Associated Systems

Working Group on the Regulation of New Reactors

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BETTER POLICIES FOR BETTER LIVES

NEA
NUCLEAR ENERGY AGENCY

Organisation de Coopération et de Développement Économiques
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NUCLEAR ENERGY AGENCY
COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 35 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Latvia, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, Korea, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 31 countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, Korea, Russia, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission and the International Atomic Energy Agency also take part in the work of the Agency.

The mission of the NEA is:

– to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally sound and economical use of nuclear energy for peaceful purposes;

– to provide authoritative assessments and to forge common understandings on key issues as input to government decisions on nuclear energy policy and to broader OECD analyses in areas such as energy and the sustainable development of low-carbon economies.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES

The Committee on Nuclear Regulatory Activities (CNRA) shall be responsible for the programme of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. The Committee shall constitute a forum for the effective exchange of safety-relevant information and experience among regulatory organisations. To the extent appropriate, the Committee shall review developments that could affect regulatory requirements with the objective of providing members with an understanding of the motivation for new regulatory requirements under consideration and an opportunity to offer suggestions that might improve them and assist in the development of a common understanding among member countries. In particular, it shall review current management strategies and safety management practices and operating experiences at nuclear facilities with a view to disseminating lessons learnt. In accordance with the *NEA Strategic Plan for 2017-2022* and the *Joint CSNI/CNRA Strategic Plan and Mandates for 2011-2016*, the Committee shall promote co-operation among member countries to use the feedback from experience to develop measures to ensure high standards of safety, to further enhance efficiency and effectiveness in the regulatory process and to maintain adequate infrastructure and competence in the nuclear safety field.

The Committee shall promote transparency of nuclear safety work and open public communication. The Committee shall maintain an oversight of all NEA work that may impinge on the development of effective and efficient regulation.

The Committee shall focus primarily on the regulatory aspects of existing power reactors, other nuclear installations and the construction of new power reactors; it may also consider the regulatory implications of new designs of power reactors and other types of nuclear installations. Furthermore it shall examine any other matters referred to it by the Steering Committee. The Committee shall collaborate with, and assist, as appropriate, other international organisations for co-operation among regulators and consider, upon request, issues raised by these organisations. The Committee shall organise its own activities. It may sponsor specialist meetings and working groups to further its objectives.

In implementing its programme, the Committee shall establish co-operative mechanisms with the Committee on the Safety of Nuclear Installations (CSNI) in order to work with that Committee on matters of common interest, avoiding unnecessary duplications. The Committee shall also co-operate with the Committee on Radiological Protection and Public Health (CRPPH) and the Radioactive Waste Management Committee (RWM) on matters of common interest.

FOREWORD

The Committee on Nuclear Regulatory Activities (CNRA) of the Nuclear Energy Agency (NEA) is an international committee composed primarily of senior nuclear regulators. It was set up in 1989 as a forum for the exchange of information and experience among regulatory organisations and for the review of developments which could affect regulatory requirements. The Committee is responsible for the NEA programme concerning the regulation, licensing and inspection of nuclear installations. In particular, the Committee reviews current practices and operating experience.

The CNRA created the Working Group on the Regulation of New Reactors (WGRNR) at the Bureau meeting of December 2007. Its mandate is to "be responsible for the programme of work in the CNRA dealing with regulatory activities in the primary programme areas of siting, licensing and oversight for new commercial nuclear power reactors (Generation III+ and Generation IV)."

At its second meeting in 2008, the Working Group agreed on the development of a report based on recent regulatory experiences describing; 1) the licensing structures, 2) the number of regulatory personnel and the skill sets needed to perform reviews, assessments and construction oversight, and 3) types of training needed for these activities. The Working Group also agreed on the development of a comparison report on the licensing processes for each member country. Following a discussion at its third meeting in March 2009, the Working Group agreed on combining the reports into one, and developing a survey where each member would provide his/her input for the completion of the report.

During the fourth meeting of the WGRNR in September 2009, the Working Group discussed a draft survey containing an extensive variety of questions related to the member countries' licensing processes, design reviews and regulatory structures. At that time, it was decided to divide the workload into four phases: general, siting, design and construction. The general section of the survey was sent to the group at the end of the meeting with a request to the member countries to provide their response by the next meeting. The "Report on the Survey of the Review of New Reactor Applications" NEA/CNRA/R(2011)13, which covers the members' responses to the general section of the survey, was issued in March 2012.

At the tenth meeting of the WGRNR in March 2013, the members agreed that the report on responses to the Design section of the survey should be presented as a multi-volume text. As such, each volume will focus on one of the eleven general technical categories covered in the survey. It was also agreed that only those countries with design review experience related to the technical category being reported are expected to respond to that section of the survey. Since the March 2013 meeting, the following reports have been published:

- "Report on the Survey of the Design Review of New Reactor Applications, Volume 1: Instrumentation and Control", NEA/CNRA/R(2014)7, June 2014.
- "Report on the Survey of the Design Review of New Reactor Applications, Volume 2: Civil Engineering Works and Structures", NEA/CNRA/R(2015)5, November 2015.
- "Report on the Survey of the Design Review of New Reactor Applications, Volume 3: Reactor, NEA/CNRA/R(2016)1", March 2016.

The reports on the survey of the design review of new reactor applications are to serve as guides for regulatory bodies to understand how technical design reviews are performed by member countries. It therefore follows that the audience for these reports are primarily nuclear regulatory organisations, although the information and ideas may also be of interest to other nuclear industry organisations and interested members of the public.

This report was prepared by the WGRNR under the co-ordination of Dr Steven Downey (Nuclear Regulatory Commission [NRC], United States). The following regulatory body members also contributed to the report with relevant proposals and inputs:

- Janne Nevalainen, Säteilyturvakeskus (STUK), Finland
- Philippe Joyer, Autorité de sûreté nucléaire (ASN), France
- Jaharlal Koley, Atomic Energy Regulatory Body (AERB), India
- Tomonori Kawamura, Nuclear Regulation Authority (NRA), Japan
- Yeon-Ki Chung, Korea Institute of Nuclear Safety (KINS), Korea
- Ladislav Haluska, Úrad Jadrového Dozoru (UJD), Slovak Republic
- Andreja Persic, Slovenian Nuclear Safety Administration (SNSA), Slovenia
- Craig Reierson, Office for Nuclear Regulation (ONR), United Kingdom
- John Monninger, Nuclear Regulatory Commission (NRC), United States
- Steven Downey, NRC, United States
- Mr Janne Nevalainen (STUK, Finland) chaired the meetings and Mr Young-Joon Choi (NEA Secretariat) supervised the work carried out by the group.

TABLE OF CONTENTS

TABLE OF ABBREVIATIONS AND ACRONYMS

ACPSR	Advisory Committee of Project Safety Review
AERB	Atomic Energy Regulatory Body
ANS	American Nuclear Society
ANSI	American National Standard Institute
ASME	American Society of Mechanical Engineers
ASN	Autorité de sûreté nucléaire
B&PV	Boiler and pressure vessel
BARC	Bhabha Atomic Research Centre
BP	Break preclusion
CNRA	Committee on Nuclear Regulatory Activities
COL	New reactor combined licence
DAC	Design acceptance confirmation
EDF	Électricité de France
EPR	Evolutionary power reactor
ESPN	Équipements sous pression nucléaires
GDA	Generic design assessment
IAEA	International Atomic Energy Agency
ITAAC	Inspection, tests, analyses, and acceptance criteria
KEPIC	Korea Electric Power Industry Code
KINS	Korea Institute of Nuclear Safety
LBB	Leak before break
LOCA	Loss-of-coolant accident
LTOP	Overpressure protection under low temperature
MCL	Main coolant line
MCP	Main coolant pump
NCC	Natural circulation cooling
NDE	Non-destructive examination
NEA	Nuclear Energy Agency
NPCIL	Nuclear Power Corporation of India Ltd
NPE	Nuclear pressure equipment

NPP	Nuclear power plant
NPS	Nuclear power station
NRA	Nuclear Regulation Authority
NRC	Nuclear Regulatory Commission
NSL	Nuclear site licence
NSSC	Nuclear Safety and Security Commission
ONR	Office for Nuclear Regulation
PCSR	Pre-construction safety report
PDSC	Project Design Safety Committee
PHWR	Pressurised heavy-water reactor
PRA	Probabilistic risk assessment
PSAR	Preliminary safety analysis report
PWR	Pressurised thermal shock
RAI	Request for additional information
RCIC	Reactor core isolation cooling
RCPB	Reactor coolant pressure boundary
RCS	Reactor coolant system
RHR	Residual heat removal
RPV	Reactor pressure vessel
SAP	Safety assessment principles
SAR	Safety analysis report
SER	Safety evaluation report
SG	Specialist group
SI	Structure integrity
SNSA	Slovenian Nuclear Safety Administration
SRG	Safety review guidelines
SRP	Standard review plan
SSC	Structures, systems and components
TH	Thermohydraulics
TSO	Technical Support Organisation
WENRA	Western European Nuclear Regulators Association
WG	Working group
WGRNR	Working Group on the Regulation of New Reactors
YVL	Ydinturvallisuusohjeet (STUK’s Regulatory Guides on nuclear safety)

EXECUTIVE SUMMARY

At the tenth meeting of the Committee on Nuclear Regulatory Activities (CNRA) Working Group on the Regulation of New Reactors (WGRNR) in March 2013, the Working Group agreed to present the responses to the Second Phase, or Design Phase, of the licensing process survey as a multi-volume text. As such, each report will focus on one of the eleven general technical categories covered in the survey. The general technical categories were selected to conform to the topics covered in the International Atomic Energy Agency (IAEA) Safety Guide GS-G-4.1. This report provides a discussion of the survey responses related to the Reactor Coolant and Associated Systems category.

The Reactor Coolant and Associated Systems category includes the following technical topics: overpressure protection, reactor coolant pressure boundary, reactor vessel, and design of the reactor coolant system. For each technical topic, the member countries described the information provided by the applicant, the scope and level of detail of the technical review, the technical basis for granting regulatory authorisation, the skill sets required and the level of effort needed to perform the review. Based on a comparison of the information provided by the member countries in response to the survey, the following observations were made:

- Although the description of the information provided by the applicant differs in scope and level of detail among the member countries that provided responses, there are similarities in the information that is required.
- All of the technical topics covered in the survey are reviewed in some manner by all of the regulatory authorities that provided responses.
- It is common to consider operating experience and lessons learnt from the current fleet during the review process.
- The most commonly and consistently identified technical expertise needed to perform design reviews related to this category are mechanical engineering and materials engineering.

The complete survey inputs are available in the appendices.

INTRODUCTION

During the five decades of commercial nuclear power operation, nuclear programmes in NEA countries have grown significantly. Over the years, communication among member countries has been a major reason for the steady improvements to nuclear power plant safety and performance around the world. Member countries continue to learn from each other, incorporating past experience, and lessons learnt in their regulatory programmes. They consult each other when reviewing applications and maintain bilateral agreements to keep the communication channels open. This has been vital and will continue to be extremely important to the success of the new fleet of reactors being built.

The Design Phase Survey Reports continue along these lines by providing detailed information on the design-related technical topics that are reviewed by the regulatory organisation as part of the regulatory authorisation process. This report focuses on the survey responses related to the Reactor Coolant and Associated Systems category.

SURVEY

The Second Phase, or Design Phase, of the licensing process survey conducted by the Committee on Nuclear Regulatory Activities (CNRA) Working Group on the Regulation of New Reactors (WGRNR) covers eleven general technical categories that are based on IAEA Safety Guide GS-G-4.1. Under these eleven general categories, there are a total of 69 specific technical topics to be addressed. For each topic, a member country is asked to answer seven survey questions. At the March 2013 meeting, the Working Group agreed that the report of the responses to the Design section of the survey should be presented as a multi-volume text. As such, each volume will focus on one of the eleven general technical categories covered in the survey. This report will present the results of the survey related to the Reactor Coolant and Associated Systems category.

The following pages present high-level summaries provided by the members and a discussion of the survey results. Complete survey responses are presented in the appendices.

HIGH-LEVEL SUMMARIES

Finland

The information provided is based on the construction licence application review of Evolutionary Power Reactor (EPR) type nuclear power plant, Olkiluoto 3. The review is based on Finnish Safety regulations and STUK YVL Guidance.

The strength design of the primary circuit and its most important mechanical components is described in the Preliminary Safety Analysis Report (PSAR) and the related Topical Reports. Because these components take a long time to manufacture, STUK addressed the strength design when reviewing the pre-inspection documentation related to manufacturing. Special attention in the review was paid to adequate dimensioning: the basic dimensions in respect of pressure and other mechanical loads were reviewed, and preliminary stress, fatigue and brittle fracture analyses of the most critical locations were carried out.

In this case, the plant supplier designed the plant unit using the French RCC-M standard applicable to the design of nuclear facilities. The design criteria presented in the said standard are based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, NB, Class 1 Components, Rules for Construction of Nuclear Power Plant Components (American Society of Mechanical Engineers), to which reference is also made in the Finnish YVL Guides.

The loads acting on the primary circuit in the various operational states and accident situations, and the other effects of the operating environment, were duly considered. The design of the basic version of the EPR is based on the Break Preclusion (BP) approach, which includes the application of the leak before break (LBB) principle. The Preliminary Safety Analysis commented on the application of the LBB approach.

The safety review also consisted of an evaluation of the design basis pipe ruptures, the manufacturing technologies, the limitation of ruptures, the minimisation of the risks, the in-service inspections, the provisions for secondary circuit pipe ruptures, the primary circuit over-pressurisation protection, the water chemistry, and the brittle fracture of the reactor pressure vessel.

Confirmatory analysis was performed in the assessment of the BP Concept, the loading and stress analyses, and the material data files

France

The main regulatory document on nuclear pressure equipment in France is the 12 December 2005 Order related to nuclear pressure equipment, or the so-called “ESPN Order”.

As regards manufacturing, the ESPN Order extends to nuclear pressure equipment the approach and essential requirements of the European Pressure Equipment Directive (PED – transposed in France by the 13 December 1999 Decree) while adding specific nuclear and radiation safety requirements. During this regulatory process a conformity assessment against regulatory safety essential requirements (design, materials, manufacturing, welding, non-destructive tests, etc.) is carried out by an "independent body" for each nuclear pressure equipment. For N1 nuclear pressure equipment (primary and secondary circuit of nuclear power plant) such conformity assessment is carried out by ASN with the help of a Technical Support Organisation (TSO) chosen among the notified agreed bodies. For other nuclear pressure equipment these conformity assessments are carried out directly by notified agreed bodies.

For in-service inspection, the ESPN Order sets additional provisions like periodic inspections, repairs instruction, requalification, etc., to those of the 13 December1999 Decree so that the nuclear and radiation safety requirements applying in nuclear facilities can also be taken into account for those equipment. The specific case of PWR main primary and secondary systems is covered by the 10 November 1999 Order.

The ESPN Order leaves a major part to industrial codes and standards. Nevertheless this regulation makes it necessary to check that the codes used for nuclear pressure equipment comply with the new requirements and may induce modifications of the codes.

India

Atomic Energy Regulatory Body (AERB) Safety Guide No. AERB/NPP&RR/SG/G-1, "Consenting Process for Nuclear Power Plants and Research Reactors", specifies the relevant information to be submitted by the applicant for review and assessment during various stages of consenting/licensing of an NPP. Further, the same document, along with AERB/SG/G-7, is meant to provide information on the methods of review and assessment to be carried out by AERB. The design description part of the safety analysis report should bring out the design criteria/bases and functional requirements and should describe how these are met in the detailed design of the Reactor Coolant and Associated Systems.

The staff of the AERB (1) conducts initial checks for adequacy of information submitted and conducts preliminary reviews of the information provided, and then (2) the AERB asks for additional information as necessary. (3) Detailed reviews are conducted in a specialist group (SG) or working group (WG) constituted for the purpose and (4) the SG or WG resolves technical issues with utility. (5) The unresolved issues and recommendations of SGs are then brought to the Project Design Safety Committee (PDSC) of the AERB. (6) Specific issues are referred to AERB Standing Committee on Reactor and Coolant System. The same committee also reviews the operating experience feedback of such systems as applicable. (7) The PDSC makes its recommendation to the Advisory Committee of Project Safety Review (ACPSR) for the final disposition. (8) After its review, the ACPSR makes the necessary recommendation to the board of AERB.

The scope and level of detail of the safety review is based on the guidance of applicable codes and guides of the AERB. In specific areas where AERB documents are not prepared, relevant IAEA or other codes/standards acceptable to AERB are used. During the review AERB committees also consider emerging technical and construction issues, operating experience, and lessons learnt related to this category. Confirmatory analyses are performed, if necessary, on a case-by-case basis by the technical service organisation or at the designated division of AERB. The commonly performed confirmatory analyses are to verify the adequacy of the submissions related to Failure Modes and Effects Analysis.

The review is carried out based on general design principles relevant to assuring safety as enunciated in AERB safety code AERB/NPP-PHWR/SC/D-1, AERB/NPP-PWR/SC/D and guides AERB/NPP-PHWR/SG/D-8, AERB/NPP-PHWR/SG/D-1 and AERB/NPP-PHWR/SG/D-5, AERB/NPP-PHWR/SG/D-23 and AERB/SG/QA-1.

Reviewers from the regulatory staff have undergone formal training in reactor systems in performing regulatory/safety reviews. The regulatory staff is also trained in various review areas through participation in the safety review and regulatory inspection process. The other members of the review team are from the TSO's who work in specialised areas.

Japan

The information provided is based on the new regulatory requirements for commercial nuclear power plants that went into force on 8 July 2013. In the sense of "Back-fit", the new regulations are applied to the existing nuclear power plants. After the Tokyo Electric Power Company's (TEPCO's) Fukushima Daiichi Nuclear Power Station (NPS) accident, all nuclear power plants were stopped. Only the nuclear power plants that conform to the new regulatory requirements could restart. The Nuclear Regulation Authority (NRA) that was established to improve its nuclear safety management and regulation in 2012 reviews application to restart.

The new regulatory requirements significantly enhance the design basis and strengthen the protective measures against natural phenomena which may lead to common cause failure. For example, the new regulatory requirements include strict evaluation of earthquakes, tsunamis, volcanic eruptions, tornadoes and forest fires, and countermeasures against tsunami inundation. They also enhance countermeasures against events other than natural phenomena that may trigger common cause failures. For example, the new regulatory requirements include strict and thorough measures for fire protection and, countermeasures against internal flooding.

The new regulatory requirements require preventing core damage under postulated severe accident conditions, such as establishing structures, systems, and components (SSCs), procedures, etc., which make a reactor sub-critical and maintain the integrity of the reactor coolant pressure boundary and the containment. They also require preventing containment vessel failure under postulated severe core damage. Moreover they require countermeasures against the loss of a large area of the NPP due to extreme natural hazards or terrorisms. Applicants should provide information including Probabilistic Risk Assessment (PRA) report and safety analysis reports.

The NRA has issued many requirements, standards, and guidelines on the above since its establishment. The NRA staff reviews accident progression and, reactor design, in terms of design-basis events and severe accident conditions.

Korea

The information provided in this report is based on the application review of APR1400 type nuclear power plant. Safety reviews of the licence application documents are performed by the Korea Institute of Nuclear Safety (KINS) at the request of the Nuclear Safety and Security Commission (NSSC). The review process is started only after the docket review is confirmed as satisfactory in accordance with the laws and regulations. The safety reviews are conducted twice; for the purpose of issuing a construction permit and for the purpose of issuing an operating licence purpose. The review plan is made to allow an in depth review to be conducted on the important items related to: (1) design changes compared to the previous approved plants; (2) application of the latest technical criteria; (3) first of a kind design issues, and so on. For certain aspects, the key review items are selected and their adequacy verified through a confirmatory audit analysis that is presented in the response.

The principal criteria for regulatory review related to the reactor coolant systems (RCS) design and other topical areas are provided in the "Regulation on Technical Standards for Nuclear Reactor Facilities, etc." This Regulation prescribes the specific requirements for acceptance criteria stipulated in the Articles (Standards for Construction Permits and Standards for Operating Licences) of the Nuclear Safety Act. In addition, the relevant NSSC Notices prescribe the specific requirements for the design and other topical areas of the RCS. Korea Electric Power Industry Codes (KEPIC) and Standards endorsed through NSSC Notices can be used as applicable codes and standards for the detailed design, manufacturing, testing, and so on.

The KINS also developed safety review guidelines that prescribe acceptance criteria and review procedures, and applies them during a safety review.

In order to maintain the quality of review activities, KINS ensures that the regulatory review activities are performed only by those who have more than 2 years of practical experience as per the Rules for Entrusted Regulatory Activities (Specific Rules on Safety Review for Nuclear Reactor and Related Facilities). Those in charge of review activities are required to take continuing education following a training schedule established annually, in order to enhance their technical expertise. Each technical staff member takes at least 40 hours of training a year, which helps to ensure technical competence of the staff engaged in regulatory activities.

Slovak Republic

The information provided is based on the Slovak legal framework which accommodates Western European Nuclear Regulators Association (WENRA) reference levels and IAEA standards. The fulfilment of these requirements is reported via the safety analysis report, technical documentation, and quality documentation.

The applicant has to demonstrate that the reactor coolant and associated systems are designed so that during normal operation, during abnormal operation, and during design basis accidents, the robustness, lifetime, and functional reliability of its parts and equipment are ensured with a sufficient margin of error. The applicant has to demonstrate that the normal coolant system is able to ensure that the boundary parameters of the fuel will not be exceeded. Regarding the emergency cooling system, the applicant has to demonstrate the ability of reliable core cooling during loss of coolant accidents (LOCA). These abilities are demonstrated via a description of the design basis; a detailed description of all components which are part of the systems; a demonstration of the resistance to single failure and common cause failure; compliance with project requirements arising from codes, standards and regulations; a reliability analysis; a description of interdependencies with other operating systems and structures; and requirements for testing and maintenance. The main goal of all submitted documentation is to ensure that all legislative requirements are fulfilled and that the nuclear facility will be operated safely and the public will be protected.

Review of the documentation submitted by the applicants is usually performed by regulatory body employees with the assistance of a TSO. When using support services from a TSO, there is a condition of TSO independence. This condition results from the fact that the Slovak Republic is small and there are not many organisations with relevant skills in the nuclear field. Therefore, we must prevent the situation where the same organisation provides support services to both the nuclear facility and the regulatory body.

Slovenia

The information provided is based on the review of a licensing process for reactor coolant and associated systems design approval. The fundamental purpose is for the applicant to demonstrate that the facility systems, the operating procedures, the processes to be performed and other technical requirements described in the Safety Analysis Report offer reasonable assurance that the plant will comply with the regulations and standards. The most extensive review is performed at the design certification stage. During the operation stage, in case of the systems changes for example, the licensing review is carried out in the same way, only less intensive.

The basic nuclear power plant design bases are set in Rules on Radiation and Nuclear Safety Factors (JV5). They are based on WENRA reference levels. The requirements for technical acceptance, safety functions, safety analyses, reactor trip systems, residual heat removal, protection systems and instrumentation and control are set in Rules. Technical acceptance includes criteria for primary coolant system pressure boundary protection and criteria for protection of the secondary coolant system.

The information provided by the applicant is based on a detailed system description with drawings, material properties, and the design basis. For overpressure protection, it is important to provide system reliability and testing information. Stress evaluation and studies of engineering mechanics and fracture mechanics of components are important for reactor pressure-temperature limits and information on fracture toughness. A justification of the design features and performance shall be provided to ensure that the components of the RCS, and the subsystems interfacing with the RCS, meet the safety requirements for design.

Additionally, during the licensing process the SNSA evaluates whether applicant has provided complete information to demonstrate that the design, materials, fabrication methods, and inspection techniques used, conformity to all applicable regulations, industrial codes and standards. The review of the results of testing, inspection and surveillance is also performed.

Materials engineering, mechanical engineering and nuclear engineering are the primary expertise needed to successfully perform reactor coolant and associated systems design review and assessment. In some areas, experience with codes and standards are also needed to completely review the technical topic.

United Kingdom

In the UK a generic design assessment (GDA) process has been put in place for the assessment of reactor designs proposed for construction in the UK, on a generic basis, in advance of any site-specific proposals. It covers safety, security and environmental protection, as the project is run jointly with the Environment Agency. GDA is a four-step process with increasing levels of technical assessment detail. The process allows the ONR to get involved with reactor designers at an early stage where ONR can maximise its influence. It also allows designers to understand and address regulatory concerns while the design is still in progress, which reduces the financial and regulatory risks for power station developers. The outcome of a successful GDA is the issue of a Design Acceptance Confirmation (DAC), which together with a site licence is required prior to the start of any nuclear island construction.

In parallel with GDA, a prospective licensee needs to inform ONR of its intention to apply for a nuclear site licence (NSL) to build and operate new nuclear power stations in Great Britain. ONR will then start to engage with the prospective licensee to provide constructive challenge and advice in order to inform their development of a “right first time” NSL application. Once the NSL application has been formally submitted, ONR assesses it culminating in a recommendation to the Chief Inspector on whether or not a licence should be granted.

A site-specific pre-construction safety report (PCSR) does not need to be in place when the NSL is granted; instead, ONR expects a licensee to provide a site-specific PCSR to support the start of nuclear safety-related construction. However, before a licence is granted ONR needs to be satisfied that the licence applicant’s safety documentation provides assurance that the site will be suitable for the proposed activities if the plant is adequately designed, constructed and operated (http://www.onr.org.uk/licensing-nuclear-installations.pdf).

Prior, and during the licensing period, ONR engages with the prospective licensees across a full range of technical disciplines (including all the disciplines relevant to the report on Reactor Coolant and Associated Systems, such as structural integrity, fault analysis, etc.) to ensure that the prospective licensee:

- Puts in place effective means to transfer relevant knowledge from the Responsible Designer, and an appropriate mechanism for adopting the GDA design and safety case.
- Develops an organisation capable of taking control of the design and safety case and able to develop the site-specific aspects outside or beyond the scope of the GDA safety case and design.
- Establishes an appropriate programme of submissions leading to the site-specific safety case.

Specifically with regard to the reactor coolant and associated systems, the PCSR is expected to cover all structural integrity claims including supporting arguments and evidence. This should include an identification of the highest reliability components where demonstration over and above pressure vessel code compliance will be required. Justification is needed for material selection, compositional specification and forging processes, transient definitions and loading envelopes, nuclear pressure vessel code assessment (e.g. ASME, RCC-M, etc.), avoidance of fracture demonstrations, and in-service inspection requirements. In particular, the avoidance of fracture demonstration needs to include fracture mechanics assessments, manufacturing inspection capabilities and qualification proposals, and confirmatory fracture toughness testing proposals.

These expectations are described in greater detail in the ONR Safety Assessment Principles (SAPs) within the engineering section on integrity of metal components and structures (SAPs EMC.1 to EMC.34) and ageing and degradation (SAPs EAD.1 to EAD.3). For example, EAD.2 requires that the effects of the coolant chemistry on ageing and degradation processes affecting material properties of structures, systems and components are considered. In support of EAD.3, the engineering SAP ECH.3 on control of chemistry requires suitable and sufficient systems, processes and procedures which should be provided to maintain chemistry parameters within the limits and conditions of the safety case.

United States

The information provided in response to the survey is based on the technical review of a new reactor design certification application, but is also applicable to the review of applications for new reactor design approvals and combined licences issued under 10 CFR Part 52. Typically, the most extensive review of the Reactor Coolant and Associated Systems is performed at the design certification stage. New reactor combined licence (COL) applicants typically incorporate the RCS design by reference to a certified standard plant design. As such, the staff's review of the Reactor Coolant and Associated Systems at the COL application stage would typically focus on site-specific information, operational programmes, and departures from the approved standard design.

Regardless of the type of application, the fundamental purpose is for the applicant to demonstrate that the facility and equipment, the operating procedures, the processes to be performed, and other technical requirements described in the Safety Analysis Report (SAR) offer reasonable assurance that the plant will comply with the regulations and that public health and safety will be protected. Design information provided by the applicant in this technical category should demonstrate that the RCS is adequate to accomplish its intended objective and to maintain its integrity under conditions imposed by all foreseeable reactor behaviours, including both normal and accident conditions. Special consideration should be given to the integrity of the reactor coolant pressure boundary (RCPB), which, if not maintained, could result in a significant loss of coolant, fuel damage, and subsequent fission product release to the environment.

The regulations related to this technical category require that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition. It is also required that the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical to ensure both structural and leak-tight integrity. Several generic communications and guidance documents have been developed to provide guidance to applicants and licences on acceptable approaches meeting the regulatory requirements.

Once an application has been formally accepted, the Nuclear Regulatory Commission (NRC) staff reviews the information provided for compliance with the regulatory requirements and performs confirmatory analyses, as necessary, to make a reasonable assurance finding. The scope and level of detail of the staff's safety review of Reactor Coolant and Associated Systems is based on the guidance provided in the applicable sections of The Standard Review Plan (SRP), NUREG-0800. As part of the review, the staff also considers emerging issues, operating experience, and lessons learnt from the current fleet.

Materials engineering, Mechanical engineering, and reactor systems engineering are the expertise needed to successfully perform design reviews in this area. In addition to knowledge of RCS components and design, it is important for technical reviewers in this area to have experience with codes and standards, particularly the ASME Boiler and Pressure Vessel (B&PV) Code, which is incorporated by reference into US NRC regulations.

DISCUSSION

Under the category of Reactor Coolant and Associated Systems, there were four technical topics to be addressed in the survey. These topics were selected to conform to the topics covered in International Atomic Energy Agency (IAEA) Safety Guide No. GS-G-4.1. For each of the four technical topics under this category, the member countries were asked seven questions in order to gather some insights on the level of detail needed for regulatory authorisation. In responding to these questions, each member country described the following:

- The design information provided by the applicant.
- The analysis, reviews, and/or research performed by the regulatory authority's reviewer(s) and the scope of the review.
- The types of confirmatory analyses performed (if any) by the regulatory authority.
- The technical basis (standards, codes, acceptance criteria) for regulatory authorisation.
- The skill sets required to perform the review.
- The specialised training, experience, education, and/or tools needed to perform the regulatory review.
- The level of effort needed for the regulatory authority to perform the review.

Design information provided by the applicant

Among the regulatory organisations that responded to the survey, there are similarities in the information provided by an applicant. In the area of overpressure protection, most countries responded that the applicant provides a description of the design and design basis of the overpressure protection system. Aspects of the overpressure protection system design and design bases that were identified in several responses include the materials specifications, applicable codes and standards, and the description of applicable instrumentation. It is also common for the applicant to describe the reliability of the overpressure protection system as well as perform some type of failure analysis. In addition, a description of the plans for, or results of, testing and inspections are also commonly provided by the applicant.

For the reactor coolant pressure boundary (RCPB), most countries responded that the applicant provides a description of the components comprising the RCPB. Details commonly provided to describe the RCPB include the materials (including the materials specifications and their compatibility with the reactor coolant), fabrication/manufacturing processes, applicable codes, and provisions for leakage detection and monitoring. It is also common for the applicant to describe the in-service inspection and testing of the RCPB, including how access is provided to perform the inspections. In addition, several countries responded that the applicant performs some type of analysis of the reactor coolant boundary. Commonly identified analyses were stress or strength analysis, brittle fracture analysis, and leak before break (LBB) analysis.

In the area of reactor vessel, most countries responded that the applicant provides a description of the design and design bases of the reactor vessel. Aspects of the reactor vessel design that were commonly identified in the survey responses are the materials specifications, fabrication processes, and limits on operating pressure and temperature. It is also common for the applicant to describe how the integrity of the reactor vessel is evaluated and maintained considering radiation embrittlement. In addition, it is also common for the applicant to provide information on inspection, testing, and/or surveillance of the reactor vessel.

For the design of the RCS, most countries responded that the applicant provides a description of the design, design bases, and performance requirements for each component and/or subsystem of the RCS.

Analysis, reviews and/or research performed

All of the technical topics covered in the survey are reviewed by all of the regulatory organisations that provided responses. While the responses show that most regulatory organisations have the framework in place to perform separate design reviews related to each survey topic, the responses also indicate that some survey topics are reviewed concurrently.

All countries review the information provided by the applicant to confirm compliance with the applicable regulatory requirements, guidelines, or codes and standards. Confirmatory analyses/assessment or independent evaluation/verification of information provided by the applicant are commonly mentioned as part of the design reviews related to this technical category.

Technical basis

In all cases, the technical basis for regulatory authorisation is provided by a combination of regulations and regulatory guidance. In addition to the regulations and guidance documents, member countries also make use of internationally recognised consensus standards related to the technical category. For example, the AFCEN RCC-M Code, Design and Conception Rules for Mechanical Components of PWR Nuclear Islands, was identified as part of the technical basis for granting regulatory authorisation in Finland and the United Kingdom. Finland and the United Kingdom commonly identified the RCC-M Code in relation to the RCPB and the reactor vessel. Also, the United Kingdom and the United States refer to ASME Codes as part of the technical basis for regulatory authorisation in the area of overpressure protection. Lastly, Korea and the United States both identified ASTM E185, Design of Surveillance Programmes for Light-Water Moderated Nuclear Power Reactor Vessels, as part of the technical basis for regulatory authorisation in the area of the reactor vessel.

Skill sets required to perform review

Mechanical engineering and materials engineering were the most consistently identified technical skills needed to perform the reviews related to the reactor coolant and associated systems. Other technical staff members that were identified on a less consistent basis include civil/structural engineers, chemical engineers, nuclear engineers, reactor systems engineers, and risk assessment engineers.

Specialised training

Although the specific training requirements may vary, all countries indicated that experience related to the technical review topic is important.

Level of effort

The total level of effort required for each member country to review the Reactor Coolant and Associated Systems category is provided in the table below. It is noted that in France, India and Japan, resources (hours) are not set up for each individual review area. Also, in the Slovak Republic, the level of effort allotted for the review of submitted documentation is defined by regulation and dependent upon the activity to be approved.

Country	Total level of effort for reactor	Basis for estimate
Finland	7640 hours.	Construction licence application review of Olkiluoto 3.
France	-	Resources (hours) are not set up for each individual review area.
India	-	Resources (hours) are not set up for each individual review area.
Japan	-	Resources (hours) are not set up for each individual review area.
Korea	4520 hours.	Application review of APR1400 type nuclear power plant.
Slovak Republic		Level of effort defined by regulation and dependent upon the activity to be approved.
Slovenia	2320 hours.	The level of effort was estimated from the analysis, which was prepared in order to assess the resources needed in case of construction of new nuclear power plants.
United Kingdom		Technical review of a PCSR.
United States	4040 hours.	Standard design certification review.

CONCLUSION

This report focused on the results of the design survey related to the Reactor Coolant and Associated Systems. Based on a comparison of the information provided in response to the survey, the following observations were made:

- Although the description of the information provided by the applicant differs in scope and level of detail among the member countries that provided responses, there are similarities in the information that is required.
- All of the technical topics covered in the survey are reviewed by all of the regulatory authorities that provided responses.
- It is common to consider operating experience and lessons learnt from the current fleet during the review process.
- The most commonly and consistently identified technical expertise needed to perform design reviews related to this category are mechanical engineering and materials engineering.

Additional reports will be issued by the Working Group in order to discuss the results of the Design Phase survey in other technical areas.

APPENDIX A:
OVERPRESSURE PROTECTION

Summary Table

Country	Is this area reviewed?	Are confirmatory analyses performed?	Expertise of reviewers	Level of effort
Finland	Yes	No	Knowledge of thermal hydraulics.	20 Working Days. (1600 hours).
France	Yes	Yes	Mechanical engineer, Material engineer, Risk assessment engineer.	-[1]
India	Yes	Yes	Reviewer should have sufficient review experience in the concerned field. Education requirements provided in Appendix.	–[1]
Japan	Yes	Yes	Civil, Structural and Mechanical engineers. Generally staff who has more than 10-year experience is taken on the task.	-[1]
Korea	Yes	Yes	Mechanical engineer, Material engineer, Nuclear engineer, Reactor systems engineer.	840 hours.
Slovak Republic	Yes	No	Technical engineer.	-[2]
Slovenia	Yes	No	Mechanical engineer, Nuclear engineer.	200 hours.[3]
United Kingdom	Yes	Yes	Chartered engineer.	-
United States	Yes	Yes	Mechanical engineer, reactor systems engineer.	600 hours.

Notes:

1. In France, India and Japan, resources (hours) are not set up for each individual review area.
2. In the Slovak Republic, the standard level of effort for the review of submitted documentation is defined by regulation and dependent upon the activity to be approved.
3. In Slovenia, the level of effort was estimated from the analysis, which was prepared in order to assess the resources needed in case of construction of new nuclear power plants.

Overpressure protection	Finland STUK
Design information provided by applicant	– Design bases; – Design evaluation; – Applicable codes and classification; – Material specification; – Process instrumentation.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Evaluate that the applicant has provided information required by the YVL Guides; – Ensure that the system fulfils requirements set by the YVL Guides.
What type of confirmatory analysis (if any) is performed?	
Technical basis: • **Standards** • **Codes** • **Acceptance criteria (e.g. can come from accident analysis, regulatory guidance)**	YVL Guide 2.4.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	No formal requirements**.**
Specialised training, experience and/or education needed for the review of this topic	Knowledge of thermal hydraulics.
Level of effort in each review area	20 working days**.**

Overpressure protection	France ASN
Design information provided by applicant	Design information is provided by the manufacturer to demonstrate the conformity of a nuclear pressure equipment (primary and secondary circuits and other safety systems. *“Equipment shall be designed in such a way as to minimise the risk of loss of integrity, taking account of foreseeable alterations in the materials. The design shall take account of ageing due to irradiation.”* Extract of nuclear pressure equipment (NPE) order.
Analysis, reviews and/or research performed by the reviewer and scope of review	According to NPE regulation, for N1 NPE (primary and secondary circuits) ASN performs an examination of the design, and determinates their conformity with essential safety requirements. For N2 and N3 NPE and for non-nuclear pressure equipment this examination is carried out by an independent notified agreed body.
What type of confirmatory analysis (if any) is performed?	A conformity assessment that leads to a certification.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria (e.g. can come from accident analysis, regulatory guidance)**	The technical basis of such assessment are regulatory requirements (essential safety requirements), standards harmonised, codes and general standards.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Mechanical, material, risk assessment; – Junior: Mechanical engineer; – TSO: Mechanical, material, risk assessment.
Specialised training, experience and/or education needed for the review of this topic	Knowledge of nuclear power plant design and operation, metallurgy, manufacturing process, safety risk analysis, non-destructive tests …
Level of effort in each review area	During this review, ASN is supported by a TSO (Notified agreed body).

Overpressure protection	India AERB
Design information provided by applicant	Design information with respect to over pressure protection should be submitted by applicant in SAR (Section 3.5 of Standard Format And Contents of Safety Analysis Report For Nuclear Power Plants (AERB/NPP/SG/G-9 (draft)). These SAR should contain details of pressure-relieving devices (safety and relief valves) of the following systems: – Reactor Coolant System (RCS); – Primary side of auxiliary or emergency systems connected to the RCS; – Moderator system (over pressure rupture disc), if applicable; – Any blow down or heat dissipation systems connected to the discharge of these; – pressure-relieving devices; – Secondary side of steam generators.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Reviewers evaluate that the applicant has provided adequate information required for review; – Ensure that the system fulfils requirements as required by design code; – Confirms that design basis considered for over pressure is adequate; – Calculates relieving capacity requirement and checks adequacy with given design; – Checks supplier certificate on pressure rating and relieving capacity and test results conducted at simulated condition; – Calibration provision and requirements.
What type of confirmatory analysis (if any) is performed?	– Confirms that design basis considered for over pressure is adequate; – Calculates relieving capacity requirement; – Reviews test results of the relieving capacity submitted by supplier.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	1. Design requirements for light-water reactor are given in section 6.8 of Design of Light-Water Reactor Based Nuclear Power Plants (AERB/NPP-LWR/SC/D 2014); 2. Design requirements of pressurised heavy-water reactor (PHWR) based NPPs are given in section 6.3.1.4 of Design of Pressurised Heavy-Water Reactor Based Nuclear Power Plants (AERB/NPP-PHWR/ SC/D (Rev.1)); 3. Section 4.9 of Primary Heat Transport System for PHWRs (AERB/NPP-PHWR/SG/D-8).
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	The identified reviewer should have a graduate/ master's degree in Mechanical Engineering with sufficient review experience for leading the team. The identified junior reviewer should have a graduate/ master's degree in Mechanical Engineering with knowledge of design and Thermal Hydraulics.
Specialised training, experience and/or education needed for the review of this topic	Knowledge of thermal-hydraulics code for over pressure estimation. Knowledge over process for identifying most severe case of over pressure causing event.
Level of effort in each review area	No particular limitation.

Overpressure protection	**Japan** **NRA**
Design information provided by applicant	– The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors, Article3(1)(ii)(e) ”Structure and equipment of the reactor cooling system equipment”; – The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors, Annex2 “Reactor cooling system facilities”.
Analysis, reviews and/or research performed by the reviewer and scope of review	These activities are to conform to the requirements, standards, criteria, and the like described below.
What type of confirmatory analysis (if any) is performed?	In the establishment permit application stage, adequacy of an applicant's analytic method and the analysis results are verified. Independent evaluation is also performed to comprehend the uncertainties of the analytic method, if needed.
Technical basis: • **Standards;** • **Codes;** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	The following regulatory requirements and guides are applicable to this technical area: – The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors (S53 #77); – The NRA Ordinance on Standards for the Location, Structure and Equipment of Commercial Power Reactors (H25 #5); – The Regulatory Guide of the NRA Ordinance on Standards for the Location, Structure and Equipment of Commercial Power Reactors (#1306193); – The NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities (H25 #6); – The Regulatory Guide of the NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities (#1306194); – The NRA Ordinance on Technical Standards for Nuclear Fuel Material Being Used as a Fuel in Commercial Power Reactors (H25 #7); – Guide for Evaluation of Effectiveness of Preventive Measures Against Core Damage and Containment Vessel Failure of Commercial Power Reactors (#13061915); – Guide for Establish Permit Application of Commercial Power Reactors (#13061919); – The Standard Review Plan on Technical Capability of Severe Accident Management of Commercial NPPs (#1306197); – Guide for Procedure of Construction Work Approval (#13061920).
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Manager and engineer; – Junior: Engineer; – TSO: Researcher. Generally staff who has more than 10-year experience is taken on the task.

Overpressure protection	**Japan** (Cont.) **NRA**
Specialised training, experience and/or education needed for the review of this topic	– Basic training for the examiner for nuclear safety; – Practical application training for the examiner for nuclear safety.
Level of effort in each review area	Resources (hours) are not set up for the individual review area. Regarding the standard processing duration, 2 years are set up for establishment permit of an entire plant, and 3 months per one application are set up for construction work approval. Divided application is granted for construction work approval.

Overpressure protection	Korea KINS
Design information provided by applicant	As part of the SAR, the applicant should describe or provide the following related to the overpressure protection: – Design Bases; – Design Evaluation; – Equipment and Component Description; – Mounting of Pressure Relief Devices; – Applicable codes and Classification; – Process Instrumentation; – System Reliability; – Testing and Inspection; – Overpressure protection under low temperature(LTOP); – Overpressure Protection for the nuclear steam supply system.
Analysis, reviews and/or research performed by the reviewer and scope of review	The Korea Institute of Nuclear Safety (KINS) staff reviews the information provided in the SAR and request for additional information (RAI) responses for compliance with the regulations. The scope and level of detail of the staff's safety review is based on the KINS Safety Review Guidelines (SRG) for Light-Water Reactors. The sections of the KINS SRG that are applicable to this area are as follows: – SRG 5.2.2, "Overpressure protection"; – SRG Appendix 5.2.2-1, "Overpressure Protection while Operating at Low Temperatures".
What type of confirmatory analysis (if any) is performed?	KINS staff performs the confirmative analysis by using regulatory safety analysis computer code to verify the results of LTOP analysis and Overpressure protection analysis submitted by the Construction Permit/Operating Licence applicant.
Technical basis: • **Standards;** • **Codes;** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	The applicable NSSC Regulatory Requirements include the following: 1. Regulations on Technical Standards for Nuclear Reactor Facilities, Etc., Article 21, "Reactor Coolant Pressure Boundary"; 2. Regulations on Technical Standards for Nuclear Reactor Facilities, Etc., Article 22, "Reactor Coolant System, etc."; 3. Regulations on Technical Standards for Nuclear Reactor Facilities, Etc., Article 37, "Overpressure Protection"; 4. Notice No. 2014-15, "Regulation on Safety Classification and Applicable Codes and Standards for Nuclear Reactor Facilities"; 5. Notice No. 2014-19, "Guidelines on Technical Standards of Nuclear Reactor Facilities of Korean Electric Power Industry Code" of the Nuclear Safety and Security Commission; 6. Notice No. 2014-20, "Standards for Safety Valves and Relief Valves of Nuclear Reactor Facilities"; 7. Notice No. 2014-24, "Regulation on Pre-operational Inspection of Nuclear Reactor Facilities"; 8. Notice No. 2014-29, "Regulation on In-Service Test of Safety-related Pumps and Valves". The applicable Codes and Standards related to this area are: 1. Korean Electric Power Industry Code (KEPIC) MN (Nuclear-Mechanical); 2. KEPIC MO (In-service Tests); 3. KEPIC MI (In-service Inspection); 4. KEPIC MD (Materials).

Overpressure protection	**Korea** (Cont.) **KINS**
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Mechanical Engineer; – Material Engineer; – Nuclear engineer; – Reactor systems engineer.
Specialised training, experience and/or education needed for the review of this topic	– Experience in Plant Systems Engineering; – Experience in Thermal-Hydraulics/Fluid Dynamics; – Experience in reactor core analyses; – Experience in or knowledge of reactor physics; – Knowledge of reactor design; – Knowledge of material for reactor vessel; – Knowledge of metallography, water chemistry and fracture mechanics; – Knowledge of code and standard for safety class 1 components; – Knowledge of and/or experience with welding and non-destructive examinations; – Knowledge of material degradation mechanism including radiation embrittlement; – Experience in reactor vessel integrity evaluations; – Knowledge of operational programme requirements.
Level of effort in each review area	Total : 920 hours – Material review: 80 hours; – ISI/IST review: 40 hours; – Overpressure protection system and analysis review: 800 hours.

Overpressure protection	Slovak Republic UJD
Design information provided by applicant	– Design bases; – Detail system description; – Fulfilment of the requirements arising from standards, codes and national regulator; – Material specification; – System reliability; – Single failure analysis; – Strength analysis; – Testing and inspection.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Evaluate that the applicant has provided complete information to demonstrate that the materials, fabrication methods, inspection techniques and load combinations used conform to all applicable regulations, industrial codes and standards; – Review the results of testing, inspection and surveillance.
What type of confirmatory analysis (if any) is performed?	
Technical basis: • **Standards;** • **Codes;** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: technical engineer; – Junior: technical engineer; – TSO: technical engineer.
Specialised training, experience and/or education needed for the review of this topic	– Detail knowledge of system, subsystems and supporting systems design; – Experience with methods for testing.

Overpressure protection	**Slovak Republic** (Cont.) **UJD**
Level of effort in each review area	Review of the submitted design information is a part of approval process which is performed as an administrative procedure based on administrative proceeding code. Based on this act we have 60 days for approval of the submitted documentation. In case that we need more time (for example if we need review from TSO or the other support organisation) we can ask our chairperson about extending the period for approval. In some cases, which are strictly defined in the atomic act the time period for reviewing is longer. These cases are as follows: – Four months if siting of nuclear installation, except repository is concerned; – Six months if nuclear installation commissioning or decommissioning stage is concerned; – One year if building authorisation, siting and closure of repository or repeated authorisation for operation of a nuclear installation are concerned.

Overpressure protection	**Slovenia** **SNSA**
Design information provided by applicant	– Design bases for overpressure protection system; – Design evaluation; – Piping and instrumentation diagrams; – Equipment and components description; – Mounting of pressure relief devices; – Applicable codes and classification; – Material specification; – Process instrumentation; – System reliability; – Testing and inspection information.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Evaluate that the applicant has provided complete information to demonstrate that the materials, fabrication methods, inspection techniques and load combinations used conform to all applicable regulations, industrial codes and standards; – Ensure adequate safety margins.
What type of confirmatory analysis (if any) is performed?	
Technical basis: • **Standards;** • **Codes;** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	– JV5, Rules on Radiation and Nuclear Safety Factors: Protection System; – IAEA Safety Standards.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Mechanical Engineer; – Junior: Mechanical Engineer; – TSO: Mechanical Engineer, Nuclear Engineer.
Specialised training, experience and/or education needed for the review of this topic	– Detail knowledge of system, subsystems and supporting systems design; – Knowledge in TH and fluid dynamics; – Knowledge in reactor physics; – Experience with reactor computer codes; – Experience with methods for testing and inspection of the system.
Level of effort in each review area	– Regulator review: 80 hrs; – TSO’ review time: 120 hrs.

Overpressure protection	United Kingdom ONR
Design information provided by applicant	Fault Studies: – Design basis analysis was provided in the PCSR. This was supplemented by additional analysis performed in support of diversity analysis in response to RO-41; – Description and validation of computer codes (and methodologies) used; – Systems descriptions. SI Pre-Construction Safety Report describing the Overpressure Protection systems, and how they operate.
Analysis, reviews and/or research performed by the reviewer and scope of review	The primary focus of ONR's assessment during GDA was on the scope of the fault sequences analysis, the adequacy and validation of the methods applied, and the acceptability of the predicted safety margins for comparison with ONR's fault analysis SAPs FA.1 to FA.9. In addition to the design basis faults, ATWT analysis for loss of feed water faults and analysis covering the common mode failure of the pressuriser relief valves following the closure of all 4 MSIV fault were requested through RO-41. A summary of the assessment is given in Section 4.2.3 of the Step 4 Design Basis Assessment of the EDF and AREVA UK EPR Reactor ONR Assessment Report ONR-GDA-AR-11-020a Rev 0 (www.onr.gov.uk/newreactors/step-four-technical-assessment-reports.htm#edf _) SI Specific Review: Limited to the basis for establishing suitable pressure-temperature limits for operation of the reactor. See SI Step 4 Report Section 4.5.
What type of confirmatory analysis (if any) is performed?	Coupled reactor kinetics and thermal hydraulic analysis. None
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	*[SAP] Safety Assessment Principles for Nuclear Facilities*. 2006 Edition Revision 1. HSE. January 2008. www.onr.gov.uk/newreactors/step-four-technical-assessment-reports.htm#edf _ *[T/AST/34] ND BMS Technical Assessment Guide.* T/AST/034 Issue 1. HSE. November 1999. Transient Analysis for Design Basis Accidents in Nuclear Reactors. *[T/AST/42] ND BMS Technical Assessment Guide.* T/AST/042 Issue 1. HSE. November 1999. Validation of Computer Codes and Calculational Methods. ONR Safety Assessment Principles: – Integrity of Metal Components and Structures – EMC.1 to EMC.34; – Ageing and degradation – EAD.1 to EAD.3. Nuclear Pressure Vessel Design standards:

	– RCC-M and ASME III.
Overpressure protection	**United Kingdom** (Cont.) **ONR**
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	ONR ONR training requirements for ONR Principal Inspector / Inspector of Nuclear Safety. TSO: – Experienced senior consultant; – Chartered engineer status required for the Regulator in a discipline related to the topic under consideration, with no differentiation in requirement for the Senior or Junior regulator. TSO expertise required in relation to the topic under consideration, but no specific level required
Specialised training, experience and/or education needed for the review of this topic	– Experience and knowledge of modelling techniques for performing reactor physics and thermal hydraulic transient analysis. Knowledge in nuclear power plants and systems; – Understanding of the structural integrity safety principles; – Understanding of the background to basis for setting pressure-temperature limits in the nuclear pressure vessel design codes.
Level of effort in each review area	See estimate for part 3 of assessment and verification section below. TSO support used on this aspect.

Overpressure protection	**United States NRC**
Design information provided by applicant	As part of the SAR, the applicant should describe or provide the following related to the overpressure protection system: – Design of the system, including any subsystems and supporting systems; – Material specifications for each component; – System reliability and the consequence of equipment/component failures; – Testing, analyses, and inspections; – Technical specifications; – Instrumentation and Control; – Power supply.
Analysis, reviews and/or research performed by the reviewer and scope of review	The NRC staff (1) reviews the information provided in the SAR for compliance with the regulations, (2) issues RAIs as necessary, (3) reviews RAI responses, (4) resolves technical issues with applicants or licensees, and (5) produces a safety evaluation report (SER) documenting its findings. The scope and level of detail of the staff's safety review is based on the guidance of NUREG-0800, SRP. The section of the SRP that is applicable to this area is as follows: – SRP 5.2.2, "Overpressure Protection." The above SRP section also reference additional SRP sections that interface with and supplement this review area. The staff also considers emerging technical and construction issues, operating experience, and lessons learnt related to this category.
What type of confirmatory analysis (if any) is performed?	The staff performs independent calculations to verify that the applicant's overpressure protection analyses are valid.

<table>
<tr><th>Overpressure protection</th><th>United States (Cont.)
NRC</th></tr>
<tr><td>Technical basis:
Standards
Codes
Acceptance criteria
(e.g. can come from accident analysis, regulatory guidance)</td><td>The applicable NRC Regulatory Requirements include the following:
1. 10 CFR 50, Appendix A, GDC 1, “Quality Standards and Records”;
2. 10 CFR 50, Appendix A, GDC 15, “Reactor Coolant System Design”;
3. 10 CFR 50, Appendix A, GDC 30, “Quality of RCPB”;
4. 10 CFR 50, Appendix A, GDC 31, “Fracture Prevention of the RCPB”;
5. 10 CFR 50, Appendix G, “Fracture Toughness Requirements”;
6. 10 CFR 50.34(f), “Additional TMI-related requirements”;
7. 10 CFR 50.55a, “Codes and Standards”;
8. 10 CFR 52.47, “Contents of Applications; Technical Information”;
9. 10 CFR 52.79, “Contents of Applications; Technical Information in Final Safety Analysis Report”;
10. 10 CFR 52.80(a), “Requirement for COL application to contain the proposed inspection, tests, analyses, and acceptance criteria (ITAAC)”.
The NRC guidance documents that provide an acceptable approach for satisfying the applicable regulatory requirements include the following:
1. Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants”;
2. Regulatory Guide 1.29, “Seismic Design Classification”;
3. RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III”.
Note: Guidance documents are not a substitute for regulations, and compliance with guidance documents is not required.
The applicable codes and Standards related to this area are as follows:
1. ANSI/ANS 52.1, “Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants”;
2. ASME B&PV Code Section II;
3. ASME B&PV Code Section III.</td></tr>
<tr><td>Skill sets required by (education):
Senior (regulator)
Junior (regulator)
TSO</td><td>– Reactor Systems Engineer;
– Mechanical Engineer.</td></tr>
<tr><td>Specialised training, experience and/or education needed for the review of this topic</td><td>All technical reviewers are required to complete a formal training and qualification programme prior to performing safety reviews independently.
Other specialised training, experience, and education that is needed to successfully perform reviews in this technical area include:
– Experience in Plant Systems Engineering;
– Experience in Thermal-Hydraulics/Fluid Dynamics;
– Experience in reactor core analyses;
– Experience in or knowledge of reactor physics.</td></tr>
<tr><td>Level of effort in each review area</td><td>NRC (Staff and Contractors): 600 hours</td></tr>
</table>

APPENDIX B: REACTOR COOLANT PRESSURE BOUNDARY (MATERIALS, IN-SERVICE INSPECTION, TESTING AND LEAKAGE DETECTION)

Summary Table

Country	Is this area reviewed?	Are confirmatory analyses performed?	Expertise of reviewers	Level of effort
Finland	Yes	Yes	Materials scientist/engineer.	375 working days (3000 hours).
France	Yes	Yes	Mechanical engineer, Material engineer, Risk assessment engineer.	-[1]
India	Yes	Yes	Reviewer should have sufficient review experience in the concerned field. Education requirements provided in Appendix.	-[1]
Japan	Yes	Yes	Civil, Structural and Mechanical engineers. Generally staff who has more than 10-year experience is taken on the task.	-[1]
Korea	Yes	No	Mechanical engineer, Material engineer. Nuclear engineer, Reactor systems engineer.	680 hours.
Slovak Republic	Yes	No	Mechanical engineer.	-[2]
Slovenia	Yes	No	Mechanical engineer, nuclear engineer.	600 hours[3].
United Kingdom	Yes	Yes	Chartered engineer.	-
United States	Yes	No	Chemical engineer, materials engineer, mechanical engineer, reactor systems engineer.	640 hours.

Notes:

1. In France, India, and Japan, resources (hours) are not set up for each individual review area.
2. In the Slovak Republic, the standard level of effort for the review of submitted documentation is defined by regulation and dependent upon the activity to be approved.
3. In Slovenia, the level of effort was estimated from the analysis, which was prepared in order to assess the resources needed in case of construction of new nuclear power plants.

Reactor coolant pressure boundary	**Finland STUK**
Design information provided by applicant	– PSAR and TR; – Design specification for each component; – Preliminary loading specification (pressure and temperature); – Dimensioning; – Preliminary stress analyses reports; – Preliminary brittle fracture analysis; – LBB analysis; – Material Data File; – Description of Manufacturing; – Evaluation of new manufacturing method; – Large Forging (Nozzle Shell and Main Coolant Line [MCL]); – Narrow Gap Welding; – RPV Safe End; – Inspectability of primary components; – Evaluation of the Inspection Qualification for In-service Inspection; – Sensitivity and working reliability of the Leakage Monitoring System.
Analysis, reviews and/or research performed by the reviewer and scope of review	– STUK's inspectors inspect all documents by themselves. The inspection is performed by specialist from different branch of technology (process, component, strength, manufacturing, quality, non-destructive testing, quality assurance); – Simplified analysis by STUK if needed; – More detailed analysis by TSO if needed.
What type of confirmatory analysis (if any) is performed?	– Assessment of the BP Concept; – Loading and stress analyses.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	– RCCM; – KTA.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	No formal requirements. – Senior: M.Sc./engineer; Working experience of sector. – Junior: M.Sc./engineer. – TSO: Specialist of sector. Competence of research institute shall be evaluation by audit.

Reactor coolant pressure boundary	**Finland** (Cont.) **STUK**
Specialised training, experience and/or education needed for the review of this topic	– Introduction course; – YK Basic professional training course on nuclear safety Finland; – Training for standard; – YTD/SAHA archives tools: Diary tools.
Level of effort in each review area	375 working days.

Reactor coolant pressure boundary	**France ASN**
Design information provided by applicant	The manufacturer provides an instruction sheet that describes the in-service inspection activities to be carried out on nuclear pressure equipment (NPE). The operator delivers in-service inspection programmes for pressure NPE, feedback experience analysis.
Analysis, reviews and/or research performed by the reviewer and scope of review	ASN examines the conformity of in-service inspections provided in instruction documents provided par the manufacturer. ASN reviews the in-service inspection programmes for N1 NPE provided by the operator and deliver an authorisation. ASN verify the respect of the in-service inspection programmes for NPE.
What type of confirmatory analysis (if any) is performed?	A conformity assessment that leads to a certification.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria (e.g. can come from accident analysis, regulatory guidance)**	The technical basis of such reviews are regulatory requirements, codes and general standards.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Mechanical, Material, risk assessment; – Junior: Mechanical engineer; – TSO: Mechanical, Material, risk assessment.
Specialised training, experience and/or education needed for the review of this topic	Knowledge of nuclear power plant design and operation, metallurgy, manufacturing process, safety risk analysis, non-destructive tests.
Level of effort in each review area	During this review, ASN is supported by a TSO (Notified agreed body).

Reactor coolant pressure boundary	**India AERB**
Design information provided by applicant	Design information with respect to Reactor coolant Pressure Boundary should be submitted by applicant in Safety Analysis Report (Section 3.5 of Standard Format And Contents of Safety Analysis Report For Nuclear Power Plants (AERB/NPP/SG/G-9 (draft)). Following details are required to be submitted: – Design specification for each component along with material; – Mechanical and thermal loading specification (class of service and loading); – Stress analyses results; – Ductile to Brittle Transition Temperature; – LBB analysis; – Radiation effect and brittle fracture consideration; – Process of Manufacturing; – Evaluation of critical manufacturing process related to Large Forging (Nozzle Shell and core belt region), welding in core belt region; – Inspectability of pressure boundary components and requirements; – Evaluation of the Inspection requirements for In-service Inspection; – Sensitivity and working reliability of the Leakage Monitoring System.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Reviewers of AERB review the documents about their completeness for containing required information; – Detailed review is carried out by experts of different fields like: Materials, Design adequacy, Layout, provision of inspection, manufacturing requirement, QA aspects etc.; – Independent stress analysis carried out if required; – For a new design, validation of thermal hydraulic calculation, loop test reports etc. may be required; – Materials properties meeting requirement to fulfil LBB – Adequacy of leak monitoring; – Pipe and equipment support.
What type of confirmatory analysis (if any) is performed?	– Stress analysis of selected equipment/component/piping; – Performance of leak monitoring in mock up.

Reactor coolant pressure boundary	**India** (Cont.) **AERB**
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	Technical basis and applicable codes and Guides: 1. Design of Light-Water Reactor Based Nuclear Power Plants (AERB/NPP-LWR/SC/D 2014); 2. Design of Pressurised Heavy-Water Reactor Based Nuclear Power Plants (AERB/NPP-PHWR/ SC/D (Rev.1)); 3. Safety Classification and Seismic Categorisation for Structures, Systems and Components of Pressurised Heavy-Water Reactors (AERB/NPP-PHWR/ SG/D-1); 4. Design Basis Events for Pressurised Heavy-Water Reactors (AERB/SG/D-5); 5. Primary Heat Transport System for Pressurised Heavy-Water Reactors (AERB/NPP-PHWR/SG/D-8); 6. Seismic Qualification of Structures, Systems and Components of Pressurised Heavy-Water Reactors (AERB/NPP-PHWR/ SG/D-23); 7. Quality Assurance in the Design of Nuclear Power Plants (AERB/SG/QA-1); 8. Quality Assurance in the Manufacture of Items for Nuclear Power Plants (AERB/SG/QA-3).
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior Reviewer: Senior reviewer should have sufficient review experience in the concerned field. Review group normally includes designers with sufficient experience in mechanical design, thermal hydraulic design and process system design. Other than these experts in the field of metallurgy, non-destructive testing, chemistry and Instrumentation are also incorporated. Minimum educational requirement for senior reviewer is Engineering graduate from reputed university and trained in nuclear technology in either Bhabha Atomic Research Centre (BARC) or Nuclear Power Corporation of India Ltd. (NPCIL) training school. Sometimes additional qualification in advance mechanical design, non-destructive testing, quality assurance, etc. are required; – Junior Reviewers: Minimum educational requirement for junior reviewer is Engineering graduate from reputed university and trained in nuclear technology in either BARC or NPCIL training school; – TSO: Specialised knowledge in specific field of Metallurgy, Irradiation effect, simulation, Fuel chemistry etc. are required.
Specialised training, experience and/or education needed for the review of this topic	Junior reviewers are send to different reputed education institutions for specialised training on different aspects of design like Pressure vessel design, piping design, process system design, stress analysis, accident analysis etc. Reviewers are also trained in regulatory requirements, inspection process, non-destructive testing, application of design codes etc.
Level of effort in each review area	There is no limit on effort put for review. However new design is reviewed in very detail and repeat design review calls for review of design in light of design change and changing regulatory/ industry requirement. However, first level review takes around one year and subsequent reviews require time commensurate with replies submitted by applicant. Detailed review is conducted in three tiers. Each tier contains experts of different fields and experience.

Reactor coolant pressure boundary	**Japan NRA**
Design information provided by applicant	– The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors, Article3(1)(ⅱ)(e) ”Structure and equipment of the reactor cooling system equipment”; – The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors, Annex2 “Reactor cooling system facilities”.
Analysis, reviews and/or research performed by the reviewer and scope of review	These activities are to conform to the requirements, standards, criteria, and the like described below.
What type of confirmatory analysis (if any) is performed?	In the establishment permit application stage, adequacy of an applicant's analytic method and the analysis results are verified. Independent evaluation is also performed to comprehend the uncertainties of the analytic method, if needed.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	The following regulatory requirements and guides are applicable to this technical area: – The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors (S53 #77); – The NRA Ordinance on Standards for the Location, Structure and Equipment of Commercial Power Reactors (H25 #5); – The Regulatory Guide of the NRA Ordinance on Standards for the Location, Structure and Equipment of Commercial Power Reactors (#1306193); – The NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities (H25 #6); – The Regulatory Guide of the NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities (#1306194); – The NRA Ordinance on Technical Standards for Nuclear Fuel Material Being Used as a Fuel in Commercial Power Reactors (H25 #7); – Guide for Evaluation of Effectiveness of Preventive Measures Against Core Damage and Containment Vessel Failure of Commercial Power Reactors (#13061915); – Guide for Establish Permit Application of Commercial Power Reactors (#13061919); – The Standard Review Plan on Technical Capability of Severe Accident Management of Commercial NPPs (#1306197); – Guide for Procedure of Construction Work Approval (#13061920).
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Manager and engineer; – Junior: Engineer; – TSO: Researcher. Generally staff who has more than 10-year experience is taken on the task.

Reactor coolant pressure boundary	**Japan** (Cont.) **NRA**
Specialised training, experience and/or education needed for the review of this topic	– Basic training for the examiner for nuclear safety; – Practical application training for the examiner for nuclear safety.
Level of effort in each review area	Resources (hours) are not set up for the individual review area. Regarding the standard processing duration, 2 years are set up for establishment permit of an entire plant, and 3 months per one application are set up for construction work approval. Divided application is granted for construction work approval.

Reactor coolant pressure boundary	**Korea KINS**
Design information provided by applicant	As part of the SAR, the applicant should describe or provide the following related to the reactor coolant pressure boundary (RCPB): – Reactor Coolant Pressure Boundary Materials: - Materials specifications; - Compatibility with Reactor Coolant; - Fabrication and Processing of Ferritic Materials; - Fabrication and Processing of Austenitic Stainless Steel. – In-service Inspection and Testing of Reactor Coolant Pressure Boundary: - System Boundary Subject to Inspection; - Arrangement of Systems and Components to Provide Accessibility; - Examination Categories and Methods; - Inspection Intervals; - Evaluation of Examination Results. – Reactor Coolant Pressure Boundary Leakage Detection Systems: - Leakage Detection Methods; - Leakage Instrumentation in the Main Control Room; - Maximum Allowable Total Leakage; - Intersystem Leakage; - Sensitivity and Response Time; - Operability Testing and Calibration.
Analysis, reviews and/or research performed by the reviewer and scope of review	The Korea Institute of Nuclear Safety (KINS) staff reviews the information provided in the SAR and RAI responses for compliance with the regulations. The scope and level of detail of the staff's safety review is based on the KINS Safety Review Guidelines (SRG) for Light-Water Reactors. The sections of the KINS SRG that are applicable to this area are as follows: – SRG 5.2.3, "Reactor Coolant Pressure Boundary Materials"; – SRG 5.2.4, "Reactor Coolant Pressure Boundary In-service Inspection and Testing"; – SRG 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection".
What type of confirmatory analysis (if any) is performed?	None
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	Nuclear Safety Laws of the Republic of Korea: Regulations on Technical Standards for Nuclear Reactor facilities, Etc. Article 21 (Reactor Coolant Pressure Boundary): 1. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so that the probability of abnormal leakage, rapidly propagating failure, or gross rupture is extremely low; 2. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage; 3. Regarding a reactor pressure vessel, a material surveillance programme shall be established to evaluate periodically the effects of changes in material properties due to irradiation on its structural integrity. And surveillance test specimens shall be installed in it; 4. Requirements for material surveillance test and specimens as provided in the foregoing Paragraph (3) are determined and publicly notified by the NSSC.

	Article 22 (Reactor Coolant System, etc.): 1. The RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operation conditions including anticipated operational occurrences; 2. Reactor coolant system shall be able to maintain constantly the quantity or pressure of the coolant to ensure that the specified design limits are not exceeded during normal operation conditions and anticipated operational occurrences, taking into account its volumetric changes and leakages; 3. Reactor coolant system shall be designed to ensure that concentration of radioactive materials in and water quality of reactor coolants are maintained within the limiting conditions for operation; 4. Reactor coolant system shall be designed to prevent any reverse flow of the coolant to connected systems, and to be isolated from the connected systems. Article 41 (Testability, Monitor ability, Inspectability, and Maintainability) 1. The structures, systems, and components important to safety shall be designed to be tested, monitored, inspected, and maintained in accordance with the importance of safety functions to be performed to ensure that their structural integrity, leak tightness, functional capability, and operability are maintained during the lifetime of the nuclear power plant; 2. For cases where periodic testing, monitoring, inspection and maintenance are limited or not possible to detect the possible faults of components, safety measures shall be made in the design to cope with expected failures; 3. Pressure vessels (excluding auxiliary boilers), piping, major pumps and major valves shall meet the acceptance criteria of pressure retaining test determined and publicly notified by the NSSC. The applicable codes and Standards related to this area are: 1. KEPIC MD (Materials); 2. KEPIC MN (Nuclear- Mechanical); 3. KEPIC ME (Non-destructive Examinations); 4. KEPIC MQ (Welding); 5. KEPIC MI (In-service Inspection).
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Mechanical Engineer; – Material Engineer; – Nuclear Engineer; – Reactor Systems Engineer.
Specialised training, experience and/or education needed for the review of this topic	– Knowledge of reactor system design; – Knowledge of material for primary system; – Knowledge of Code and Standard for safety class 1 components; – Knowledge of and/or experience with welding and non-destructive examination; – Knowledge of Metallography(Metallurgy, Phase transformation, Corrosion);

	– Knowledge of water chemistry; – Knowledge of fracture mechanics; – Knowledge of and/or experience with material degradation mechanism.
Level of effort in each review area	Total: 680 hours. – Material review : 400 hours; – In-service Inspection/In-service Testing review : 80 hours; – Leakage Detection review: 200 hours.

Reactor coolant pressure boundary	Slovak Republic UJD
Design information provided by applicant	– Material specifications, – Strength analysis; – In-service inspection programmes and methods; – System for leakage detection during operations (min. 3 systems); – System qualification and classification; – Requirements for verification and validation; – Documentation of the suitability of metallurgical semi-finished products and welding filler material; – Requirements for management of ageing; – Requirements for processes of procurement, design, manufacture, storage transport, installation, commissioning and operation; – Requirements for technical operating and maintenance procedures, including requirements for the manner and scope of pre-operational and operational checks.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Evaluate that the applicant meets all requirements of the Authority, generally applicable legislation, special regulations and Slovak technical standards; – Review the results of testing, inspection and surveillance.
What type of confirmatory analysis (if any) is performed?	
Technical basis: • **Standards** • **Codes** • **Acceptance criteria (e.g. can come from accident analysis, regulatory guidance)**	
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	Senior: Mechanical Engineer; Junior: Mechanical Engineer; TSO: Mechanical Engineer.
Specialised training, experience and/or education needed for the review of this topic	Experience in Materials Science. Experience with: – In-service inspection; – Methods for testing.
Level of effort in each review area	Review of the submitted design information is a part of approval process which is performed as an administrative procedure based on administrative proceeding code. Based on this act we have 60 days for approval of the submitted documentation. In case that we need more time (for example if we need review from TSO or the other support organisation) we can ask our

Reactor coolant pressure boundary	Slovak Republic UJD
	chairperson about extending the period for approval. In some cases, which are strictly defined in the atomic act the time period for reviewing is longer. These cases are as follows: – Four months if siting of nuclear installation, except repository is concerned; – Six months if nuclear installation commissioning or decommissioning stage is concerned; – One year if building authorisation, siting and closure of repository or repeated authorisation for operation of a nuclear installation are concerned.

Reactor coolant pressure boundary	Slovenia SNSA
Design information provided by applicant	– A list of all components, together with the corresponding applicable codes; – Material specifications; – Fabrication and processing of ferritic materials and austenitic stainless steels; – A description and justification of the results of the detailed analytical and numerical stress evaluations and studies of engineering mechanics and fracture mechanics of all components comprising the reactor coolant pressure boundary subjected to normal conditions, including shutdown conditions, and postulated accident loads; – Leakage detection systems (monitoring, collection, and identification of the leakage); – In-service inspections of the integrity of the primary coolant systems, owing to their importance to safety and the severity of the possible consequences of failure.
Analysis, reviews and/or research performed by the reviewer and scope of review	Evaluate that the applicant has provided complete information to demonstrate that the materials, fabrication methods, inspection techniques and load combinations used conform to all applicable regulations, industrial codes and standards.
What type of confirmatory analysis (if any) is performed?	
Technical basis: • **Standards** • **Codes** • **Acceptance criteria (e.g. can come from accident analysis, regulatory guidance)**	– JV 5, Rules on Radiation and Nuclear Safety Factors: Technical Acceptance; – IAEA Safety Standards.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Mechanical Engineer; – Junior: Mechanical Engineer; – TSO: Mechanical Engineer, Nuclear Engineer.
Specialised training, experience and/or education needed for the review of this topic	Experience in Materials Science. Experience with: – In-service inspection; – Methods for testing.
Level of effort in each review area	Regulator review: 200 hrs. TSO' review time: 400 hrs.

Reactor coolant pressure boundary	United Kingdom ONR
Design information provided by applicant	Pre-Construction Safety Report describing all aspect of the integrity claims and supporting arguments and evidence. In particular this covers: – Integrity Claims, including identification of the highest reliability components where a demonstration over and above nuclear pressure vessel code compliance will be required; – Transient definitions and loading envelopes; – Material selection and forging processes; – Design code assessment; – Beyond design code avoidance of fracture demonstrations for the highest reliability components including fracture assessment, manufacturing inspection qualification and confirmatory fracture toughness testing proposals; – Access requirements for in-service inspection.
Analysis, reviews and/or research performed by the reviewer and scope of review	SI Specific Review: – Integrity claims including identification of the highest reliability components; – Material selection, compositional specification and forging processes; – Transient definition; – Design code principles; – Design code assessment; – Environment effects on fatigue design curves. Beyond design code avoidance of fracture demonstration covering: – Fracture mechanics assessments; – Manufacturing inspection capability and qualification proposals; – Confirmatory fracture toughness testing proposals. In-service inspection access requirements. See SI Step 4 Report Sections 4.2, 4.3, 4.6, 4.7, 4.8, 4.10.
What type of confirmatory analysis (if any) is performed?	Limited confirmatory analyses of: – Beyond design code fracture mechanics assessments; – Design code assessment.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria (e.g. can come from accident analysis, regulatory guidance)**	ONR Safety Assessment Principles: – Integrity of Metal Components and Structures – EMC.1 to EMC.34, with EMC.1 to EMC.3 specifically applicable to the highest reliability components; – Ageing and degradation – EAD.1 to EAD.3. RCC-M Nuclear Pressure Vessel Code.

Reactor coolant pressure boundary	United Kingdom ONR
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	Chartered Engineer Status required for the Regulator in a discipline related to the topic under consideration, with no differentiation in requirement for the Senior or Junior regulator. TSO expertise required in relation to the topic under consideration, but no specific level required.
Specialised training, experience and/or education needed for the review of this topic	Understanding the structural integrity safety principles, and in particular the beyond design code demonstration required for the highest reliability components including detailed consideration of material properties, fracture assessment, and manufacturing inspection.
Level of effort in each review area	TSO support on: material selection and forging processes; confirmatory fracture mechanics assessment; confirmatory design code assessment; review of inspection capability and qualification; design code principles.

Reactor coolant pressure boundary	**United States USNRC**

Design information provided by applicant	As part of the SAR, the applicant should describe or provide the following related to the reactor coolant pressure boundary (RCPB):
	– Materials Specifications;
	– Compatibility of materials with the reactor coolant;
	– Fabrication and processing of ferritic materials and austenitic stainless steel;
	– Integrity of bolting and threaded fasteners;
	– Description of the pre-service and in-service inspection and testing programme;
	– RCPB Leakage detection, including:
	  - Leakage detection instruments and methods;
	  - Leakage detection capability including response time, sensitivity, diversity, redundancy, and seismic qualification;
	  - Determination of leak before break applicability;
	  - Technical specifications, operability, and availability;
	  - Prolonged low level leakage detection programme (monitoring, trending, identification of leakage, responding and leakage management).

Analysis, reviews and/or research performed by the reviewer and scope of review	The NRC staff (1) reviews the information provided in the SAR for compliance with the regulations, (2) issues RAIs as necessary, (3) reviews RAI responses, (4) resolves technical issues with applicants or licensees, and (5) produces a SER documenting its findings. The scope and level of detail of the staff's safety review is based on the guidance of NUREG-0800, SRP. The sections of the SRP that are applicable to this area are as follows: – SRP 5.2.3, "Reactor Coolant Pressure Boundary Materials"; – SRP 5.2.4, "Reactor Coolant Pressure Boundary In-service Inspection and Testing"; – SRP 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection". The above SRP sections also reference additional SRP sections that interface with and supplement this review area. The staff also considers emerging technical and construction issues, operating experience, and lessons learnt related to this category.
What type of confirmatory analysis (if any) is performed?	None
Technical basis: • **Standards** • **Codes** • **Acceptance criteria (e.g. can come from accident analysis, regulatory guidance)**	The applicable NRC Regulatory Requirements include the following: 1. 10 CFR Part 50, Appendix A, Generic Design Criteria (GDC) 1, "Quality Standards and Records"; 2. 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena"; 3. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases"; 4. 10 CFR Part 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary";

Reactor coolant pressure boundary	United States USNRC
	5. 10 CFR Part 50, Appendix A, GDC 30, “Quality of the RCPB”; 6. 10 CFR Part 50, Appendix A, GDC 31, “Fracture Prevention of the RCPB”; 7. 10 CFR Part 50, Appendix A, GDC 32, “Inspection of the RCPB”; 8. 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements”; 9. 10 CFR 50.55a, “Codes and Standards”; 10. 10 CFR 52.47 (b)(1), “Requirement for DC application to contain the proposed inspection, tests, analyses, and acceptance criteria (ITAAC); 11. 10 CFR 52.80(a), “Requirement for COL application to contain the proposed inspection, tests, analyses, and acceptance criteria (ITAAC). The NRC guidance documents that provide an acceptable approach for satisfying the applicable regulatory requirements include the following: 1. Regulatory Guide (RG) 1.29, “Seismic Design Qualification”; 2. RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal”; 3. RG 1.34, “Control of Electroslag Weld Properties”; 4. RG 1.36, “Non-metallic Thermal Insulation for Austenitic Stainless Steel”; 5. RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants”; 6. RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components”; 7. RG 1.44, “Control of the Use of Sensitised Stainless Steel”; 8. RG 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems”; 9. RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel”; 10. RG 1.71, “Welder Qualification for Areas of Limited Accessibility”; 11. RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III”; 12. SECY 05-0197, “Review of Operational Programmes in a Combined Licence Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria”. The applicable codes and Standards related to this area are: 1. ASME B&PV Code Section II; 2. ASME B&PV Code Section III; 3. ASME B&PV Code Section V; 4. ASME B&PV Code Section IX; 5. ASME B&PV Code Section XI.
Skill sets required by (education):	– Materials Engineer;

Reactor coolant pressure boundary	United States USNRC
• **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Chemical Engineer; – Mechanical Engineer; – Reactor Systems Engineer.
Specialised training, experience and/or education needed for the review of this topic	All technical reviewers are required to complete a formal training and qualification programme prior to performing safety reviews independently. Other specialised training, experience, and education that is needed to successfully perform reviews in this technical area include: – Experience in Metallurgy; – Knowledge of boiling water reactor and pressurised water reactor designs; – Knowledge of and/or experience with welding and other special fabrication processes; – Knowledge of and/or experience with non-destructive examination methods (surface and volumetric); – Knowledge of and/or experience with ASME Code requirements for the design, fabrication, inspection, and testing of Class 1 components for nuclear power plants; – Knowledge of operational programme requirements; – Knowledge of reactor coolant pressure boundary leakage detection; – Experience with RG 1.45, Revision 1; – Knowledge of Technical specifications for RCPB leakage detection; – Experience with Prolonged low level leakage detection programme.
Level of effort in each review area	640 hours. RCPB Materials Review: 320 hours; Pre-service/In-service Inspection review: 160 hours; Leakage Detection Review (NRC Staff): 160 hours.

APPENDIX C: REACTOR VESSEL

Summary Table

Country	Is this area reviewed?	Are confirmatory analyses performed?	Expertise of reviewers	Level of effort
Finland	Yes	Yes	Materials scientist/engineer.	130 working days (1040 hours).
France	Yes	Yes	Mechanical engineer, material engineer, risk assessment engineer.	–[1]
India	Yes	Yes	Reviewer should have sufficient review experience in the concerned field. Education requirements provided in Appendix.	–[1]
Japan	Yes	Yes	Civil, Structural and Mechanical engineers. Generally staff who has more than 10-year experience is taken on the task.	–[1]
Korea	Yes	Yes	Mechanical & Material Engineer.	480 hour
Slovak Republic	Yes	No	Technical engineer.	–[2]
Slovenia	Yes	No	Materials engineer, mechanical engineer, nuclear engineer.	600 hours.[3]
United Kingdom	Yes	Yes	Chartered engineer.	-
United States	Yes	Yes	Materials engineer, mechanical engineer.	800 hours.

Notes:

1. In France, India, and Japan, resources (hours) are not set up for each individual review area.
2. In the Slovak Republic, the standard level of effort for the review of submitted documentation is defined by regulation and dependent upon the activity to be approved.
3. In Slovenia, the level of effort was estimated from the analysis, which was prepared in order to assess the resources needed in case of construction of new nuclear power plants.

Reactor Vessel	**Finland STUK**

Design information provided by applicant	– PSAR and TR; – The specification for design; – Preliminary loading specification (pressure and temperature); – Dimensioning; – Preliminary stress analyses reports; – Preliminary brittle fracture analysis; – Material Data File; – Description of Manufacturing; – Preliminary report for radiation embrittlement; – Report for Safe End welds.
Analysis, reviews and/or research performed by the reviewer and scope of review	– STUK's inspectors inspect all documents by themselves. The inspection is performed by specialist from different branch of technology (process, component, strength, manufacturing, quality, non-destructive testing, quality assurance); – Simplified analysis by STUK if needed; – More detailed analysis by TSO if needed.
What type of confirmatory analysis (if any) is performed?	– Material Data File; – Loading and stress analyses.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	– RCCM; – KTA.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	No formal requirements. – Senior: M.Sc./engineer; working experience of sector. – Junior: M.Sc./engineer. – TSO: Specialist of sector; competence of research institute shall be evaluation by audit.
Specialised training, experience and/or education needed for the review of this topic	– Introduction course; – YK Basic professional training course on nuclear safety Finland; – Training for standard; – YTD/SAHA archives tools: Diary tools.
Level of effort in each review area	130 working days

Reactor Vessel	**France** **ASN**
Design information provided by applicant	The information to demonstrate the design of the vessel a N1 nuclear pressure equipment (NPE) is provided by the manufacturer. This information includes embrittlement and ageing management programmes.
Analysis, reviews and/or research performed by the reviewer and scope of review	For pressure vessel N1 NPE, the review is performed by ASN with the help of a notified agreed body (considered as a TSO).
What type of confirmatory analysis (if any) is performed?	This analysis is a part of the conformity assessment performed for each N1 NPE.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	The technical basis of such reviews are regulatory requirements, codes and general standards.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Mechanical, Material, risk assessment; – Junior: Mechanical engineer; – TSO: Mechanical, Material, risk assessment.
Specialised training, experience and/or education needed for the review of this topic	Knowledge of nuclear power plant design and operation, metallurgy, manufacturing process, safety risk analysis, non-destructive tests.
Level of effort in each review area	During this review, ASN is supported by a TSO (Notified agreed body).

Reactor Vessel	**India AERB**
Design information provided by applicant	Design information with respect to Reactor coolant Pressure Boundary should be submitted by applicant in Safety Analysis Report (Section 3.5 of Standard Format And Contents of Safety Analysis Report For Nuclear Power Plants (AERB/NPP/SG/G-9 (draft)). Information should contain relevant data in sufficient detail to provide assurance of the reactor vessel integrity under all plant states. The details should include: – Material of reactor vessel and properties; – Method used for fabrication and manufacturing; – Special controls during manufacturing; – Material acceptance criterion; – Material surveillance programme and its adequacy; – Expected effect of radiation on reactor vessel; – Materials and design of fasteners for the reactor vessel closure; – Non-destructive evaluation procedures; – Lubricants or surface treatments; – Protection provisions for meeting regulatory requirements; – Results of mechanical property and toughness tests; – Pressure – temperature limit and bases of setting operational limit on pressure – temperature for normal, off-normal & test conditions; – Limit curve under different operating conditions; – Detailed fracture toughness requirements for protection against pressurised thermal shock events, (pressurised water reactors only) throughout the life of the plant. Information detailing limits on pressure and temperature for the following conditions should be provided: – Pre-service system hydrostatic tests; – In-service leak and hydrostatic tests; – Normal operation, including heat-up and cool-down; – Reactor core operation.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Reviewers of AERB review the documents about their completeness for containing required information; – Detailed review is carried out by experts of different fields like: Materials, Design adequacy, Layout, provision of inspection, manufacturing requirement, QA aspects, etc.; – Independent stress analysis carried out if required; – Materials properties meeting requirement; – Vessel and equipment support; – Assessment of life based on thermal cycles, irradiation effects; – Provision of in-service inspection including material property assessment

Reactor Vessel	India AERB
	on regular interval as decided during design;
What type of confirmatory analysis (if any) is performed?	– Independent stress analysis carried out if required; – Inspection of vessel and Quality assurance during manufacturing is carried out.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	– Section 6.1, 6.6.9, 6.7.2, 6.27 of Design of Light-Water Reactor Based Nuclear Power Plants (AERB/NPP-LWR/SC/D 2014).
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	The identified reviewer should have a graduate/ master's degree in Mechanical Engineering or Metallurgical Engineering with sufficient review experience for leading the team. The identified junior reviewer should have a graduate/ master's degree in Mechanical Engineering/ Metallurgical Engineering with knowledge of design and Thermal Hydraulics.
Specialised training, experience and/or education needed for the review of this topic	Junior reviewers are send to different reputed education institutions for specialised training on different aspects of design like Pressure vessel design, piping design, process system design, stress analysis, accident analysis, etc. Reviewers are also trained in regulatory requirements, inspection process, non-destructive testing, application of design codes, etc.
Level of effort in each review area	No particular limitation.

Reactor Vessel	Japan NRA
Design information provided by applicant	– The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors, Article3(1)(ⅱ)(c) 4) “Reactor vessel”; – The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors, Annex2 ”Reactor body”.
Analysis, reviews and/or research performed by the reviewer and scope of review	These activities are to conform to the requirements, standards, criteria, and the like described below.
What type of confirmatory analysis (if any) is performed?	In the establishment permit application stage, adequacy of an applicant's analytic method and the analysis results are verified. Independent evaluation is also performed to comprehend the uncertainties of the analytic method, if needed.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria (e.g. can come from accident analysis, regulatory guidance)**	The following regulatory requirements and guides are applicable to this technical area: – The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors (S53 #77); – The NRA Ordinance on Standards for the Location, Structure and Equipment of Commercial Power Reactors (H25 #5); – The Regulatory Guide of the NRA Ordinance on Standards for the Location, Structure and Equipment of Commercial Power Reactors (#1306193); – The NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities (H25 #6); – The Regulatory Guide of the NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities (#1306194); – The NRA Ordinance on Technical Standards for Nuclear Fuel Material Being Used as a Fuel in Commercial Power Reactors (H25 #7); – Guide for Evaluation of Effectiveness of Preventive Measures Against Core Damage and Containment Vessel Failure of Commercial Power Reactors (#13061915); – Guide for Establish Permit Application of Commercial Power Reactors (#13061919); – The Standard Review Plan on Technical Capability of Severe Accident Management of Commercial NPPs (#1306197); – Guide for Procedure of Construction Work Approval (#13061920).
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Manager and engineer; – Junior: Engineer; – TSO: Researcher. Generally staff who has more than 10-year experience is taken on the task.

Reactor Vessel	**Japan** **NRA**
Specialised training, experience and/or education needed for the review of this topic	– Basic training for the examiner for nuclear safety; – Practical application training for the examiner for nuclear safety.
Level of effort in each review area	Resources (hours) are not set up for the individual review area. Regarding the standard processing duration, 2 years are set up for establishment permit of an entire plant, and 3 months per one application are set up for construction work approval. Divided application is granted for construction work approval.

Reactor Vessel	Korea KINS
Design information provided by applicant	– Design of the reactor vessel; – Materials of construction; – Fabrication methods; – Fracture toughness; – Inspection and testing; – Shipping and Installation; – Material Surveillance; – Reactor Vessel Integrity Evaluations; – Pressure and temperature limits; – Pressurised thermal shock (PWRs only); – Charpy upper-shelf energy; – Integrity of bolting and threaded fasteners; – Special fabrication processes; – Special Non-destructive examination (NDE) methods; – Special controls and special processes used for ferritic and austenitic stainless steels.
Analysis, reviews and/or research performed by the reviewer and scope of review	The Korea Institute of Nuclear Safety (KINS) staff reviews the design information provided in the SAR and RAI for compliance with the all applicable regulations and codes and standards. The scope, level, acceptance criteria and review procedures of the staff's safety review are based on the KINS Safety Review Guidelines (SRG) for Light-Water Reactors. The section of the SRG is as follows: – SRG 5.3.1, "Reactor Vessel Material"; – SRG 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurised Thermal Shock"; – SRG 5.3.3, "Reactor Vessel Integrity".
What type of confirmatory analysis (if any) is performed?	– Pressure-Temperature (P-T) limits and associated methodologies and calculations; – Pressurised thermal shock evaluation.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	– Nuclear Safety Laws of the Republic of Korea: Regulations on Technical Standards for Nuclear Reactor facilities, Etc.: - Article 12 (Safety Classes and Standards); - Article 15 (Environmental Effects Design Bases, etc.); - Article 21 (Reactor Coolant Pressure Boundary); - Section 4 (Quality Assurance Regarding Construction and Operation of Reactor Facilities); – Nuclear Safety and Security Commission(NSSC) Notices: - No. 2014-14, "Material Surveillance Criteria for Reactor Pressure Vessel"; - No. 2014-19, "Guidelines for Application of Korea Electric Power Industry Code (KEPIC) as Technical Standards of Nuclear Reactor

Reactor Vessel	Korea KINS
	Facilities”; - No. 2014-23, “Detailed Requirements for Quality Assurance of Nuclear Reactor Facilities”. – Korea Institute of Nuclear Safety(KINS) Regulatory Guides: - KINS/RG-N03.01, “Codes and Standards”; - KINS/RG-N03.13, “Control of Preheat Temperature for Welding of Low-Alloy Steel”; - KINS/RG-N03.14, “Control of Ferrite Content in Stainless Steel Weld Metal”; - KINS/RG-N03.15, “Control of the Use of Sensitised Stainless Steel”; - KINS/RG-N03.16, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components”; - KINS/RG-N03.17, “Non-metallic Thermal Insulation for Austenitic Stainless Steel”; - KINS/RG-N03.18, “Welder Qualification for Areas of Limited Accessibility”. – Applicable codes and Standards: - Korean Electric Power Industry Codes (KEPIC) MD (Materials); - KEPIC MN (Nuclear- Mechanical); - KEPIC ME (Non-destructive Examinations); - KEPIC MQ (Welding) ; - KEPIC MI (In-service Inspection); - ASTM E 185.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Mechanical Engineer; – Material Engineer.
Specialised training, experience and/or education needed for the review of this topic	– Knowledge of reactor design; – Knowledge of material for reactor vessel; – Knowledge of metallography, water chemistry and fracture mechanics; – Knowledge of code and standard for safety class 1 components; – Knowledge of and/or experience with welding and NDEs; – Knowledge of material degradation mechanism including radiation embrittlement; – Experience in reactor vessel integrity evaluations; – Knowledge of operational programme requirements.
Level of effort in each review area	Total: 480 hours.

Reactor Vessel	Slovak Republic UJD
Design information provided by applicant	– Design basis of the reactor vessel; – Reactor vessel materials, material specifications; – Qualification and classification; – The pressure-temperature limits; – The integrity of the reactor vessel, including embrittlement considerations; – Requirements for management of ageing; – Requirements for processes of procurement, design, manufacture, storage transport, installation, commissioning and operation; – Requirements for technical operating and maintenance procedures, including requirements for the manner and scope of pre-operational and operational checks; – Documentation of the suitability of metallurgical semi-finished products and welding filler material.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Evaluate that the applicant meets all requirements of the Authority, generally applicable legislation, special regulations and Slovak technical standards; – Review the results of testing, inspection and surveillance.
What type of confirmatory analysis (if any) is performed?	
Technical basis: • **Standards** • **Codes** • **Acceptance criteria (e.g. can come from accident analysis, regulatory guidance)**	
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Technical Engineer; – Junior: Technical Engineer; – TSO: Technical Engineer.
Specialised training, experience and/or education needed for the review of this topic	Experience with: – Embrittlement process and surveillance of reactor vessel; – Operation procedures, limitations (p-T curves); – Welding techniques; – Non-destructive examination method.

Reactor Vessel	Slovak Republic UJD
Level of effort in each review area	Review of the submitted design information is a part of approval process which is performed as an administrative procedure based on administrative proceeding code. Based on this act we have 60 days for approval of the submitted documentation. In case that we need more time (for example if we need review from TSO or the other support organisation) we can ask our chairperson about extending the period for approval. In some cases, which are strictly defined in the atomic act the time period for reviewing is longer. These cases are as follows: – Four months if siting of nuclear installation, except repository is concerned; – Six months if nuclear installation commissioning or decommissioning stage is concerned; – One year if building authorisation, siting and closure of repository or repeated authorisation for operation of a nuclear installation are concerned.

Reactor Vessel	Slovenia SNSA
Design information provided by applicant	– Design basis of the reactor vessel; – Reactor vessel materials, material specifications; – The pressure – temperature limits; – The integrity of the reactor vessel, including embrittlement considerations; – Special processes used for manufacture and fabrication of components; – Special methods for NDE; – Special controls and special processes used for ferrite steels and austenitic stainless steels; – Fracture toughness; – Reactor vessel fasteners.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Evaluate that the applicant has provided complete information to demonstrate that the materials, fabrication methods, inspection techniques and load combinations used conform to all applicable regulations, industrial codes and standards; – Review the results of testing, inspection and surveillance; – Ensure of adequate safety margins of structural integrity for the components of the reactor vessel.
What type of confirmatory analysis (if any) is performed?	
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	– JV 5, Rules on Radiation and Nuclear Safety Factors: Technical Acceptance; – IAEA Safety Standards.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Mechanical Engineer, Material Engineer, Nuclear Engineer; – Junior: Mechanical Engineer, Nuclear Engineer; – TSO: Mechanical Engineer, Nuclear Engineer, Material Engineer.
Specialised training, experience and/or education needed for the review of this topic	Experience with: – Embrittlement process and surveillance of reactor vessel; – Operation procedures, limitations (p-T curves); – Welding techniques, – NDE method.
Level of effort in each review area	Regulator review: 200 hrs. TSO’ review time: 400 hrs.

Reactor Vessel	**United Kingdom** **ONR**
Design information provided by applicant	Pre-Construction Safety Report describing all aspect of the integrity claims and supporting arguments and evidence. In particular this covers: – Transient definitions and loading envelopes; – Material selection and forging processes; – Design code assessment; – Beyond design code avoidance of fracture demonstrations as this is a highest reliability component including fracture assessment, manufacturing inspection qualification and confirmatory fracture toughness testing proposals; – Access requirements for in-service inspection; – Irradiation damage and through life monitoring.
Analysis, reviews and/or research performed by the reviewer and scope of review	SI Specific Review: – Material selection, compositional specification and forging processes; – Inclusion of a belt line weld in the design; – Transient definition; – Design code principles; – Design code assessment. Beyond design code avoidance of fracture demonstration covering: – Fracture mechanics assessments; – Manufacturing inspection capability and qualification proposals; – Confirmatory fracture toughness testing proposals. In-service inspection access requirements. Irradiation damage and through life monitoring proposals taking account of a reactor specific design feature. See SI Step 4 Report Sections 4.2, 4.3, 4.4, 4.5, 4.8, 4.10.
What type of confirmatory analysis (if any) is performed?	Limited confirmatory analyses of: – Beyond design code fracture mechanics assessments; – Design code assessment**;**
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	ONR Safety Assessment Principles: – Integrity of Metal Components and Structures – EMC.1 to EMC.34, with EMC.1 to EMC.3 specifically applicable to the highest reliability components; – Ageing and degradation – EAD.1 to EAD.3. RCC-M Nuclear Pressure Vessel Code.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	Chartered Engineer Status required for the Regulator in a discipline related to the topic under consideration, with no differentiation in requirement for the Senior or Junior regulator. TSO expertise required in relation to the topic under consideration, but no specific level required.

Specialised training, experience and/or education needed for the review of this topic	Understanding the structural integrity safety principles, and in particular the beyond design code demonstration required for a highest reliability components including detailed consideration of material properties, fracture assessment, and manufacturing inspection. In addition knowledge of irradiation damage mechanisms and surveillance schemes.
Level of effort in each review area	TSO support on: material selection and forging processes; confirmatory fracture mechanics assessment; confirmatory design code assessment; review of inspection capability and qualification; design code principles; irradiation surveillance.

Reactor Vessel	United States NRC
Design information provided by applicant	As part of the Safety Analysis Report, the applicant should describe the following related to the reactor vessel: – Design of the reactor vessel; – Materials of construction; – Fabrication methods; – Fracture toughness; – Inspection and testing; – Shipping and Installation; – Material Surveillance; – Reactor Vessel Integrity Evaluations; – Pressure and temperature limits; – Pressurised thermal shock (PWRs only); – Charpy upper-shelf energy; – Integrity of bolting and threaded fasteners; – Special fabrication processes; – Special NDE methods; – Special controls and special processes used for ferritic and austenitic stainless steels.
Analysis, reviews and/or research performed by the reviewer and scope of review	The NRC staff (1) reviews the information provided in the SAR for compliance with the regulations, (2) issues RAIs as necessary, (3) reviews RAI responses, (4) resolves technical issues with applicants or licensees, and (5) produces a SER documenting its findings. The scope and level of detail of the staff's safety review is based on the guidance of NUREG-0800, SRP. The sections of the SRP that are applicable to this area are as follows: – SRP 5.3.1, "Reactor Vessel Materials"; – SRP 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurised Thermal Shock"; – SRP 5.3.3, "Reactor Vessel Integrity". The above SRP sections also reference additional SRP sections that interface with and supplement this review area. The staff also considers emerging technical and construction issues, operating experience, and lessons learnt related to this category.
What type of confirmatory analysis (if any) is performed?	The staff typically performs an independent analysis to confirm the adequacy of the following submittals: – Pressure-Temperature (P-T) limits and associated methodologies and calculations; – Pressurised thermal shock evaluation.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	The applicable NRC Regulatory Requirements are listed below: 1. 10 CFR Part 50, Appendix A, Generic Design Criteria (GDC) 1, "Quality Standards and Records"; 2. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases"; 3. 10 CFR Part 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary"; 4. 10 CFR Part 50, Appendix A, GDC 31, "Fracture Prevention of the

Reactor Vessel	United States NRC
	Reactor Coolant Pressure Boundary"; 5. 10 CFR Part 50, Appendix A, GDC 32, "Inspection of the Reactor Coolant Pressure Boundary"; 6. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"; 7. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"; 8. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Programme Requirements"; 9. 10 CFR 50.55a, "Codes and Standards"; 10. 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light-water Nuclear Power Reactors for Normal Operation"; 11. 10 CFR 50.61, "Fracture Toughness Requirements for Protection against Pressurised Thermal Shock Events"; 12. 10 CFR 52.47 (b)(1), "Requirement for DC application to contain the proposed inspection, tests, analyses, and acceptance criteria (ITAAC); 13. 10 CFR 52.80(a), "Requirement for COL application to contain the proposed inspection, tests, analyses, and acceptance criteria (ITAAC). The NRC guidance documents that provide an acceptable approach for satisfying the applicable regulatory requirements are listed as follows: 1. RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal"; 2. RG 1.34, "Control of Electroslag Weld Properties"; 3. RG 1.36, "Non-metallic Thermal Insulation for Austenitic Stainless Steel"; 4. RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"; 5. RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components"; 6. RG 1.44, "Control of the Use of Sensitised Stainless Steel"; 7. RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems"; 8. RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel"; 9. RG 1.71, "Welder Qualification for Areas of Limited Accessibility"; 10. RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III"; 11. RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials"; 12. SECY 05-0197, "Review of Operational Programmes in a Combined Licence Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria". The applicable codes and Standards related to this area are: 1. ASME B&PV Code Section II; 2. ASME B&PV Code Section III; 3. ASME B&PV Code Section V; 4. ASME B&PV Code Section IX;

Reactor Vessel	United States NRC
	5. ASME B&PV Code Section XI; 6. ASTM E185.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Materials engineer; – Mechanical Engineer.
Specialised training, experience and/or education needed for the review of this topic	All technical reviewers are required to complete a formal training and qualification programme prior to performing safety reviews independently. Other specialised training, experience, and education that is needed to successfully perform reviews in this technical area include: – Experience in Metallurgy; – Knowledge of boiling water reactor and pressurised water reactor designs; – Understanding of the effects of neutron fluence and radiation embrittlement on steels; – Experience in reactor vessel integrity evaluations(pressurised thermal shock, pressure-temperature limits, charpy upper-shelf energy); – Knowledge of operational programme requirements; – Knowledge of and/or experience with welding and other special fabrication processes; – Knowledge of and/or experience with NDE methods (surface and volumetric); – Knowledge of and/or experience with ASME Code requirements for the design, fabrication, inspection, and testing of nuclear reactor vessels.
Level of effort in each review area	NRC Staff and Contractors: 800 hours.

APPENDIX D:
DESIGN OF THE REACTOR COOLANT SYSTEM

Summary Table

Country	Is this area reviewed?	Are confirmatory analyses performed?	Expertise of reviewers	Level of effort
Finland	Yes	Yes	Materials scientist/engineer.	250 working days (2000 hours).
France	Yes	Yes	Mechanical engineer, Material engineer, Risk assessment engineer.	–[1]
India	Yes	Yes	Reviewer should have sufficient review experience in the concerned field. Education requirements provided in Appendix.	–[1]
Japan	Yes	Yes	Civil, Structural and Mechanical engineers. Generally staff who has more than 10-year experience is taken on the task.	–[1]
Korea	Yes	Yes	Mechanical & Material Engineer, Nuclear Engineer, Reactor Systems Engineer.	2520 hours
Slovak Republic	Yes	No	Technical engineer.	–[2]
Slovenia	Yes	No	Materials engineer, mechanical engineer, nuclear engineer.	920 hours.[4]
United Kingdom	Yes	No		–[3]
United States	Yes	Yes	Materials engineer, mechanical engineer, reactor systems engineer.	2000 hours.

Notes:

1. In France, India, and Japan, resources (hours) are not set up for each individual review area.
2. In the Slovak Republic, the standard level of effort for the review of submitted documentation is defined by regulation and dependent upon the activity to be approved.
3. The UK response to this section is identical to response for 3Reactor Coolant Pressure Boundary.
4. In Slovenia, the level of effort was estimated from the analysis, which was prepared in order to assess the resources needed in case of construction of new nuclear power plants.

Design of the Reactor Coolant System	Finland STUK

Design information provided by applicant	- Reactor Pressure Vessel, Pressuriser, MCP, MCL, Steam Generator;
	– PSAR and TR;
	– Design specification for each component;
	– Preliminary loading specification (pressure and temperature);
	– Dimensioning;
	– Preliminary stress analyses reports;
	– Preliminary brittle fracture analysis;
	– Leak-before-break analysis;
	– Material Data File;
	– Description of Manufacturing;
	– Defence in depth:
	  - BP Concept;
	  - Restrains and anti-whip device;
	  - Unlimited 2A-LOCA = Analysis for RPV Internals without MCL restrains.
	– Material for SG tubes and integrity of SG tubes;
	– Main Steam and Feed water lines inside reactor building:
	  - Material Data File;
	  - Report for Erosion corrosion.

Analysis, reviews and/or research performed by the reviewer and scope of review	– STUK's inspectors inspect all documents by themselves. The inspection is performed by specialist from different branch of technology (process, component, strength, manufacturing, quality, non-destructive testing, quality assurance); – Simplified analysis by STUK if needed; – More detailed analysis by TSO if needed.
What type of confirmatory analysis (if any) is performed?	– Assessment of the BP Concept; – Loading and stress analyses; – Material Data File.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	– RCCM; – KTA.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	No formal requirements. – Senior: M.Sc./engineer, working experience of sector; – Junior: M.Sc./engineer; – TSO: Specialist of sector, Competence of research institute shall be evaluation by audit.

Design of the Reactor Coolant System	**Finland** **STUK**
Specialised training, experience and/or education needed for the review of this topic	– Introduction course; – YK Basic professional training course on nuclear safety Finland; – Training for standard; – YTD/SAHA archives tools: Diary tools.
Level of effort in each review area	250 working days.

Design of the Reactor Coolant System	France ASN
Design information provided by applicant	Design information provided by the manufacturer to demonstrate the conformity of a nuclear pressure equipment (primary and second circuits and other safety systems. *"Equipment shall be designed in such a way as to minimise the risk of loss of integrity, taking account of foreseeable alterations in the materials. The design shall take account of ageing due to irradiation."* Extract of nuclear pressure equipment (NPE) order.
Analysis, reviews and/or research performed by the reviewer and scope of review	According to NPE regulation, for N1 NPE (primary and second circuits) ASN performs an examination of the design, and determinates their conformity with essential safety requirements. For N2 and N3 NPE and for non-nuclear pressure equipment this examination is performed by an independent notified agreed body.
What type of confirmatory analysis (if any) is performed?	A conformity assessment that leads to a certification.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	The technical basis of such assessment are regulatory requirements (essential safety requirements), standards harmonised, codes and general standards.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Mechanical, Material, risk assessment; – Junior: Mechanical engineer; – TSO: Mechanical, Material, risk assessment.
Specialised training, experience and/or education needed for the review of this topic	Knowledge of nuclear power plant design and operation, metallurgy, manufacturing process, safety risk analysis, non-destructive tests.
Level of effort in each review area	During this review, ASN is supported by a TSO (Notified agreed body).

Design of the Reactor Coolant System	India AERB
Design information provided by applicant	Applicant has to submit following details of the Reactor Coolant System as required in section 3.5 of AERB safety guide on “Standard Format And Contents of Safety Analysis Report For Nuclear Power Plants” (AERB/NPP/SG/G-9 (draft)). A description and justification should be provided of the performance and design features those have been implemented to ensure that the various components of the RCS and the subsystems interfacing with the RCS meet the safety requirements for design. This should include, description of the followings (where applicable): – Reactor coolant pumps (RCPs); – Steam generators or boilers; – Reactor coolant piping or ducting; – Main steam line isolation system; – Isolation cooling system of the reactor core; – Main steam line and feed water piping; – Pressuriser; – Pressuriser relief discharge system; – Provisions for main and emergency cooling, – Residual heat removal system, including all components such as pumps, valves and supports etc. Similar regulatory code is available for Pressurised Heavy-water Reactor (PHWR) based Nuclear Power Plants (AERB/NPP-PHWR/SC/D) and requirements are almost similar.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Analysis: Utility carries out loss of coolant accident (LOCA) analysis, Stress analysis etc. and submits as part of review document; – Reviews: Staff members of AERB and experts review the documents submitted by applicant and check compliance with requirements stated in regulatory codes and guides and design codes; – Research: If required, AERB asks for additional analysis reports on areas of regulatory interest. If new concepts are used in design then utility is required to demonstrate the efficacy of the design in simulated condition or in mock up facilities. Reviewers observe the experiment/mock up and review the experimental data after completion of the experiment to corroborate the claim of the applicant about design.
What type of confirmatory analysis (if any) is performed?	AERB has independent capability to carry out Thermal Hydraulic Analysis, Structural Analysis, Stress Analysis, Severe Accident Analysis, PSA etc. If required, AERB carries out confirmatory analysis with validated code.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from**	– Technical Basis: Technical basis of acceptance of design evolves from Accident Analysis. Postulated initiating events are identified and spectrum of events is required to be analysed. Based on the most severe design basis events, the design parameters are selected and used as design parameters after applying design margins. Designers check the selection of design parameters, conservative design approach, use of design codes etc. – Design codes: AERB prefers use of ASME code for design of

<table>
<tr><th>Design of the Reactor Coolant System</th><th>India
AERB</th></tr>
<tr><td>accident analysis, regulatory guidance)</td><td>components. However applicant can use any standard internationally accepted design code for designing components.
– Acceptance Criteria: Reviewers check whether in worst anticipated condition, the design fulfils the intended function with reliability and levels of defence in depth are not challenged. Reactor Coolant pressure boundary integrity is maintained in the extreme possible condition.
– Applicable codes and Guides:
1. Design of Light-Water Reactor Based Nuclear Power Plants (AERB/NPP-LWR/SC/D 2014) as per the following section's requirement:
- 5.12: Materials and Water Chemistry
- 5.15.3: Applicability of LBB
- 6.6 Design of Reactor Coolant System
- 6.7 In-service Inspection of the Reactor Coolant Pressure Boundary
- 6.8 Overpressure Protection of the Reactor Coolant Pressure Boundary
- 6.9 Inventory of Reactor Coolant
- 6.10 Clean-up of Reactor Coolant
- 6.11 Removal of Residual Heat from the Reactor Core
- 6.12 Emergency Cooling of the Reactor Core
- 6.13 Heat Transfer to Ultimate Heat Sink
2. Design of Pressurised Heavy-Water Reactor Based Nuclear Power Plants (AERB/NPP-PHWR/ SC/D (Rev.1))
3. Safety Classification and Seismic Categorisation for Structures, Systems and Components of Pressurised Heavy-Water Reactors (AERB/NPP-PHWR/ SG/D-1)
4. Design Basis Events for Pressurised Heavy-Water Reactors (AERB/SG/D-5).
5. Primary Heat Transport System for Pressurised Heavy-Water Reactors (AERB/NPP-PHWR/SG/D-8)
6. Seismic Qualification of Structures, Systems and Components of Pressurised Heavy-Water Reactors (AERB/NPP-PHWR/ SG/D-23)
7. Quality Assurance in the Design of Nuclear Power Plants (AERB/SG/QA-1)
8. Quality Assurance in the Manufacture of Items for Nuclear Power Plants (AERB/SG/QA-3)</td></tr>
<tr><td>Skill sets required by (education):
• Senior (regulator)
• Junior (regulator)
• TSO</td><td>– Senior Reviewer: Senior reviewer should have sufficient review experience in the concerned field. Review group normally includes designers with sufficient experience in mechanical design, thermal hydraulic design and process system design. Other than these experts in the field of metallurgy, non-destructive testing, chemistry and Instrumentation are also incorporated. Minimum educational requirement</td></tr>
</table>

Design of the Reactor Coolant System	India AERB
	for senior reviewer is Engineering graduate from reputed university and trained in nuclear technology in either Bhabha Atomic Research Centre (BARC) or Nuclear Power Corporation of India Ltd. (NPCIL) training school. Sometimes additional qualification in advance mechanical design, NDT, QA etc. are required; – Junior Reviewers: Minimum educational requirement for junior reviewer is Engineering graduate from reputed university and trained in nuclear technology in either BARC or NPCIL training school; – TSO: Specialised knowledge in specific field of Metallurgy, Irradiation effect, simulation, Fuel chemistry etc. is required.
Specialised training, experience and/or education needed for the review of this topic	Junior reviewers are send to different reputed education institutions for specialised training on different aspects of design like Pressure vessel design, piping design, process system design, stress analysis, accident analysis etc.
Level of effort in each review area	There is no limit on effort put for review. However new design is reviewed in very detail and repeat design review calls for review of design in light of design change and changing regulatory/ industry requirement.

Design of the Reactor Coolant System	**Japan** **NRA**
Design information provided by applicant	– The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors, Article3(1)(ii)(e) ”Structure and equipment of the reactor cooling system equipment”; – The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors, Annex2 “Reactor cooling system facilities”.
Analysis, reviews and/or research performed by the reviewer and scope of review	These activities are to conform to the requirements, standards, criteria, and the like described below.
What type of confirmatory analysis (if any) is performed?	In the establishment permit application stage, adequacy of an applicant's analytic method and the analysis results are verified. Independent evaluation is also performed to comprehend the uncertainties of the analytic method, if needed.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	The following regulatory requirements and guides are applicable to this technical area: – The NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors (S53 #77); – The NRA Ordinance on Standards for the Location, Structure and Equipment of Commercial Power Reactors (H25 #5); – The Regulatory Guide of the NRA Ordinance on Standards for the Location, Structure and Equipment of Commercial Power Reactors (#1306193); – The NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities (H25 #6); – The Regulatory Guide of the NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities (#1306194); – The NRA Ordinance on Technical Standards for Nuclear Fuel Material Being Used as a Fuel in Commercial Power Reactors (H25 #7); – Guide for Evaluation of Effectiveness of Preventive Measures Against Core Damage and Containment Vessel Failure of Commercial Power Reactors (#13061915); – Guide for Establish Permit Application of Commercial Power Reactors (#13061919); – The Standard Review Plan on Technical Capability of Severe Accident Management of Commercial NPPs (#1306197); – Guide for Procedure of Construction Work Approval (#13061920).
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Manager and engineer; – Junior: Engineer; – TSO: Researcher. Generally staff who has more than 10-year experience is taken on the task.

Design of the Reactor Coolant System	**Japan NRA**
Specialised training, experience and/or education needed for the review of this topic	– Basic training for the examiner for nuclear safety; – Practical application training for the examiner for nuclear safety.
Level of effort in each review area	Resources (hours) are not set up for the individual review area. Regarding the standard processing duration, 2 years are set up for establishment permit of an entire plant, and 3 months per one application are set up for construction work approval. Divided application is granted for construction work approval.

Design of the Reactor Coolant System	**Korea** **KINS**
Design information provided by applicant	– Reactor Coolant Pumps (RCPs); – Steam Generators; – Reactor Coolant System Piping; – Main Steam Line Flow Restrictor; – Main Steam Line Isolation System; – Shutdown Cooling System (SCS); – Main Steam Line and Feedwater Piping; – Pressuriser; – Pressuriser Relief Tank; – Valves; – Safety and Relief Valves; – RCS component supports; – Natural circulation cooling (NCC) Analysis, etc.
Analysis, reviews and/or research performed by the reviewer and scope of review	The KINS staff reviews the design information provided in the SAR for compliance with the all applicable regulations and codes and standards. The scope, level, acceptance criteria and review procedures of the staff's safety review are based on the KINS Safety Review Guidelines (SRG) for Light-Water Reactors. The section of the SRG is as follows: – SRG 5.4.1.1, "Reactor Coolant Pump Flywheel Integrity"; – SRG 5.4.2.1, "Steam Generator Materials" and Appendix SRG 5.4.2.1-1 "Monitoring of Secondary Side Water Chemistry in Steam Generators "; – SRG 5.4.2.2 "Steam Generator Tube Programme"; – SRG 5.4.7, "Residual Heat Removal System" and Appendix SRG 5.4.7-1 "Design Requirements of the RHR System"; – SRG 5.4.11, "Pressuriser Relief Tank".
What type of confirmatory analysis (if any) is performed?	KINS staff performs the confirmative analysis by using regulatory safety analysis computer code to verify the results of NCC analysis submitted by the CP/OL applicant.
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	– Nuclear Safety Laws of the Republic of Korea: Regulations on Technical Standards for Nuclear Reactor facilities, etc.: - Article 8, "Handling, Storage and Shipping"; - Article 12, "Safety Classes and Standard"; - Article 15, "Environmental Effects Design Bases, etc."; - Article 16 "Sharing of Structures, Systems, and Components"; - Article 21, "Reactor Coolant Pressure Boundary"; - Article 22, "Reactor Coolant System, etc."; - Article 25, "Control Room, etc."; - Article 29, "Residual Heat Removal System"; - Article 37, "Overpressure Protection"; - Article 41, "Testability, Monitor ability, Inspectability, and Maintainability";

Design of the Reactor Coolant System	Korea KINS
	- Article 43 “Protection during Startup, Shutdown, and Low Power Operations”; - Article 63 “Testing, Monitoring, Inspection and Maintenance”; - Section 4, “Quality Assurance regarding Construction and Operation of Reactor Facilities”. – Nuclear Safety and Security Commission(NSSC) Notices: - 2014-15, “Regulations on Safety Class and Standards by Class of Nuclear Reactor Facilities” of the Nuclear Safety and Security Commission”; - 2014-16, “Regulations on Pre-Operation Inspection of Nuclear Reactor Facility” of the Nuclear Safety and Security Commission”; - 2014-19, “Guidelines on Technical Standards of Nuclear Reactor Facilities of Korean Electric Power Industry Code” of the Nuclear Safety and Security Commission. – NSSC and Korea Institute of Nuclear Safety(KINS) Regulatory Guides: - KINS/RS-N06.05, “ Subsystem and Component Design”; - KINS/RG-N06.01, “Reactor Coolant Pressure Boundary Leakage Detection System”; - KINS/RG-N06.04, “Reactor Coolant Pump Flywheel Integrity”; - KINS/RG-N06.05, “In-Service Inspection of Steam Generator Tubes”; - KINS/RG-N06.07, “Design Requirements of the Residual Heat Removal System”. – The applicable codes and Standards related to this area are: - Korean Electric Power Industry Codes (KEPIC) MD (Materials); - KEPIC MN (Nuclear- Mechanical); - KEPIC ME (Non-destructive examinations); - KEPIC MQ (Welding); - KEPIC MI (In-service Inspection).
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Mechanical Engineer; – Materials Engineer; – Nuclear Engineer; – Reactor Systems Engineer.
Specialised training, experience and/or education needed for the review of this topic	– Detail knowledge of pressurised water reactor designs, systems and operation; – Experience with performing seismic, dynamic, and vibratory analyses of piping, mechanical equipment and supporting system design; – Understanding of codes and Standards (KEPIC, ASME, etc.); – Understanding of the structural integrity safety principles; – Understanding of plant system behaviour during shutdown operation.

Design of the Reactor Coolant System	**Korea** **KINS**
Level of effort in each review area	Total : 2520 hours. – Reactor coolant subsystems review: 800 hours; – Review of the analysis results of NCC and mid-loop operation etc.: 1200 hours.

Design of the Reactor Coolant System	**Slovak Republic UJD**
Design information provided by applicant	– Design basis; – Detailed description all components, which are part of the reactor coolant system; – Resistant to single failure and common cause failure; – Compliance with project requirements arising from codes, standards and regulations; – Reliability analysis; – Interdependencies with other operating systems and structures; – Requirements for testing and maintenance.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Evaluate that the applicant meets all requirements of the Authority, generally applicable legislation, special regulations and Slovak technical standards; – Review the results of testing, inspection and surveillance.
What type of confirmatory analysis (if any) is performed?	
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Technical Engineer; – Junior: Technical Engineer; – TSO: Technical Engineer.
Specialised training, experience and/or education needed for the review of this topic	Experience in evaluation of design; Knowledge about nuclear facilities.
Level of effort in each review area	Review of the submitted design information is a part of approval process which is performed as an administrative procedure based on administrative proceeding code. Based on this act we have 60 days for approval of the submitted documentation. In case that we need more time (for example if we need review from TSO or the other support organisation) we can ask our chairperson about extending the period for approval. In some cases, which are strictly defined in the atomic act the time period for reviewing is longer. These cases are as follows: – Four months if siting of nuclear installation, except repository is

Design of the Reactor Coolant System	Slovak Republic UJD
	concerned; – Six months if nuclear installation commissioning or decommissioning stage is concerned; – One year if building authorisation, siting and closure of repository or repeated authorisation for operation of a nuclear installation are concerned.

Design of the Reactor Coolant System	**Slovenia SNSA**
Design information provided by applicant	A description and justification of the performance and design features that have been implemented to ensure that the various components of the reactor coolant system and the subsystems interfacing with the reactor coolant system meet the safety requirements for design. The design information of reactor coolant pumps, the gas circulators, the steam generators or boilers, the reactor coolant piping or ducting, the main steam line isolation system, the isolation cooling system of the reactor core, the main steam line and feed water piping, the pressuriser, the pressuriser relief discharge system. The provisions for main and emergency cooling and the residual heat removal system, including all components such as pumps, valves and supports.
Analysis, reviews and/or research performed by the reviewer and scope of review	– Evaluate that the applicant has provided complete information to demonstrate that the materials, fabrication methods, inspection techniques and load combinations used conform to all applicable regulations, industrial codes and standards, – Ensure of adequate safety margins of structural integrity for the components.
What type of confirmatory analysis (if any) is performed?	
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	– JV 5, Rules on Radiation and Nuclear Safety Factors: Safety Function, Technical Acceptance, Residual Heat Removal; – IAEA Safety Standards.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Senior: Mechanical Engineer, Nuclear Engineer; – Junior: Mechanical Engineer, Nuclear Engineer; – TSO: Mechanical Engineer, Nuclear Engineer, Material Engineer.
Specialised training, experience and/or education needed for the review of this topic	Knowledge and experience in material science. Experience with performing dynamic analyses of systems and components. Experience with codes and Standards.
Level of effort in each review area	Regulator review: 320 hrs. TSO' review time: 600 hrs.

Design of the Reactor Coolant System	**United Kingdom ONR**
Design information provided by applicant	Repeat of 4.2 ‘Reactor Coolant Pressure Boundary’ from a structural integrity perspective**.**
Analysis, reviews and/or research performed by the reviewer and scope of review	
What type of confirmatory analysis (if any) is performed?	
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	
Specialised training, experience and/or education needed for the review of this topic	
Level of effort in each review area	

Design of the Reactor Coolant System	**United States** **USNRC**
Design information provided by applicant	As part of the SAR, the applicant should provide information regarding performance requirements and design features to ensure the overall safety of the various components and subsystems within or allied with the RCS. This includes a description of the design bases, as well as the necessary evaluations, tests and inspections for each component or subsystem of the RCS. Components and subsystems described by the applicant may include, but are not limited to, the following: – Reactor Coolant Pumps (RCPs); – Steam Generators (PWRs only); – Reactor Coolant Piping; – Main steam line flow restriction; – Pressuriser; – Reactor core isolation cooling (RCIC) system (Boiling Water Reactors only) / isolation condenser system; – Residual heat removal (RHR) system / Passive residual heat removal system / Shutdown cooling mode of the reactor water clean-up system; – Reactor water clean-up system (RWCS); – RCS pressure relief devices / reactor coolant depressurisation systems; – RCS component supports; – Pressuriser relief discharge system; – RCS high-point vents; – Main steam line, feedwater, and auxiliary feedwater piping; The complete listing of components and subsystems to be described is dependent upon the plant design.
Analysis, reviews and/or research performed by the reviewer and scope of review	The NRC staff (1) reviews the information provided in the SAR for compliance with the regulations, (2) issues RAIs as necessary, (3) reviews RAI responses, (4) resolves technical issues with applicants or licensees, and (5) produces a SER documenting its findings. The scope and level of detail of the staff's safety review is based on the guidance of NUREG-0800, SRP. The section of the SRP that is applicable to this area is as follows: – SRP 5.4, "Reactor Coolant System Component and Subsystem Design". The above SRP section also references additional SRP sections that interface with and supplement this review area. The staff also considers emerging technical and construction issues, operating experience, and lessons learnt related to this category.
What type of confirmatory analysis (if any) is performed?	– Design information related to RCS systems are used as inputs to the confirmatory analyses performed under SRP 6.3, "Emergency Core Cooling" and 15, "Transient and Accident Analyses"; – Confirmatory tests and analyses may be performed in pre-operational test programmes of certain RCS components and RCS piping and associated supports; – When necessary, the staff may perform independent calculations to verify that the applicant's design and analyses of piping, components, and supports are valid.

Design of the Reactor Coolant System	**United States USNRC**
Technical basis: • **Standards** • **Codes** • **Acceptance criteria** **(e.g. can come from accident analysis, regulatory guidance)**	The applicable NRC Regulatory Requirements include the following: 1. 10 CFR 50.34, “Content of Applications; Technical Information”; 2. 10 CFR 50.36, “Technical Specifications”; 3. 10 CFR 50.46a, “Acceptance Criteria for Emergency Core Cooling Systems for Light-water Nuclear Power Reactors”; 4. 10 CFR 50.49, “Environmental Qualification for Electric Equipment Important to Safety for Nuclear Power Plants”; 5. 10 CFR 50.55a, “codes and Standards”; 6. 10 CFR 50.63, “Loss of All Alternating Current Power”; 7. 10 CFR Part 50, Appendix A, Generic Design Criteria (GDC) 1, “Quality Standards and Records”; 8. 10 CFR Part 50, Appendix A, GDC 2, “Design Bases for Protection Against Natural Phenomena”; 9. 10 CFR Part 50, Appendix A, GDC 4, “Environmental and Dynamic Effects Design Bases”; 10. 10 CFR Part 50, Appendix A, GDC 5, “Sharing of Structures, Systems, and Components”; 11. 10 CFR Part 50, Appendix A, GDC 14, “Reactor Coolant Pressure Boundary”; 12. 10 CFR Part 50, Appendix A, GDC 15, “Reactor Coolant System Design”; 13. 10 CFR Part 50, Appendix A, GDC 17, “Electric Power Systems”; 14. 10 CFR Part 50, Appendix A, GDC 19, “Control Room”; 15. 10CFR Part 50, Appendix A, GDC-29, “Protection against AOOs”; 16. 10 CFR Part 50, Appendix A, GDC 30, “Quality of RCPB”; 17. 10 CFR Part 50, Appendix A, GDC 31, “Fracture Prevention of the RCPB”; 18. 10 CFR Part 50, Appendix A, GDC 32, “Inspection of RCPB”; 19. 10CFR Part 50, Appendix A, GDC 33, “Reactor Coolant Make Up”; 20. 10 CFR Part 50, Appendix A, GDC 34, “Residual Heat Removal”; 21. 10 CFR Part 50, Appendix A, GDC 36, “Inspection of Emergency Core Cooling System”; 22. 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants”; 23. 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements”. The NRC guidance documents that provide an acceptable approach for satisfying the applicable regulatory requirements include the following: 1. NUREG-1449, “Shutdown and Low Power Operation at Nuclear Power Plants in the United States”; 2. Regulatory Guide (RG) 1.14, “Reactor Coolant Pump Flywheel Integrity”; 3. RG 1.20, “Comprehensive Vibration Assessment Programme for Reactor Internals During Pre-operational and Initial Startup Testing”; 4. RG 1.29, “Seismic Design Classification”; 5. RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal”;

Design of the Reactor Coolant System	**United States USNRC**
	6. RG 1.34, “Control of Electroslag Weld Properties”; 7. RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants”; 8. RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components”; 9. RG 1.44, “Control of the Processing and Use of Stainless Steel”; 10. RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel”; 11. RG 1.65, “Materials and Inspections for Reactor Vessel Closure Studs”; 12. RG 1.71, “Welder Qualification for Areas of Limited Accessibility”; 13. RG 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes”. The applicable codes and Standards related to this area are: 1. ASME Boiler & Pressure Vessel (B&PV) Code Section II; 2. ASME B&PV Code Section III; 3. ASTM A-708.
Skill sets required by (education): • **Senior (regulator)** • **Junior (regulator)** • **TSO**	– Mechanical Engineer; – Materials Engineer; – Reactor Systems Engineer.
Specialised training, experience and/or education needed for the review of this topic	All technical reviewers are required to complete a formal training and qualification programme prior to performing safety reviews independently. Other specialised training, experience, and education that is needed to successfully perform reviews in this technical area include: – Knowledge of boiling water reactor and pressurised water reactor designs; – Experience with performing seismic, dynamic, and vibratory analyses of piping, mechanical equipment; – Experience with codes and Standards (ASME, ASTM, etc.).
Level of effort in each review area	2000 hours.