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

Volume 5: Classification of Structures, Systems and Components

Working Group on the
Regulation of New Reactors

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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, Korea, Latvia, Luxembourg, Mexico, Netherlands, New Zealand, Norway, Poland, Portugal, 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.

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NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 33 countries: Argentina, Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, Romania, Russia, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes 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) is responsible for NEA programmes and activities concerning the regulation, licensing and inspection of nuclear installations with regard to both technical and human aspects of nuclear safety. The Committee constitutes a forum for the effective exchange of safety-relevant information and experience among regulatory organisations. To the extent appropriate, the Committee reviews developments which 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 reviews regulatory aspects of current safety management strategies and safety management practices and operating experiences at nuclear facilities including, as appropriate, consideration of the interface between safety and security with a view to disseminating lessons learnt. In accordance with the *NEA Strategic Plan for 2017-2022*, the Committee promotes 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 promotes transparency of nuclear safety work and open public communication. In accordance with the NEA Strategic Plan, the Committee oversees work to promote the development of effective and efficient regulation.

The Committee focuses on safety issues and corresponding regulatory aspects for existing and new power reactors and other nuclear installations, and the regulatory implications of new designs and new technologies of power reactors and other types of nuclear installations consistent with the interests of the members. Furthermore it examines any other matters referred to it by the Steering Committee for Nuclear Energy. The work of the Committee is collaborative with and supportive of, as appropriate, that of other international organisations for co-operation among regulators and consider, upon request, issues raised by these organisations. The Committee organises its own activities. It may sponsor specialist meetings, senior-level task groups and working groups to further its objectives.

In implementing its programme, the Committee establishes co-operative mechanisms with the Committee on the Safety of Nuclear Installations in order to work with that Committee on matters of common interest, avoiding unnecessary duplications. The Committee also co-operates with the Committee on Radiological Protection and Public Health, the Radioactive Waste Management Committee, and other NEA committees and activities on matters of common interest.

FOREWORD

The Committee on Nuclear Regulatory Activities (CNRA) of the OECD 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 CNRA created the Working Group on the Regulation of New Reactors (WGRNR) at the Bureau meeting of December 2007. Its Mandate was 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. Additionally, the Working Group agreed on the development of a comparison report on the licensing processes for each member state. 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 their input to 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 states 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[1] 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 11 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[2], 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[3], November 2015
- "Report on the Survey of the Design Review of New Reactor Applications: Volume 3: Reactor", NEA/CNRA/R(2016)1[4], March 2016
- "Report on the Survey of the Design Review of New Reactor Applications, Volume 4: Reactor Coolant and Associated Systems", NEA/CNRA/R(2016)3[5], July 2017

1. Follow this link to download the report: www.oecd-nea.org/nsd/docs/2011/cnra-r2011-13.pdf
2. Follow this link to download the report: www.oecd-nea.org/nsd/docs/2014/cnra-r2014-7.pdf
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4. Follow this link to download the report: www.oecd-nea.org/nsd/docs/2016/cnra-r2016-1.pdf
5. Follow this link to download the report: www.oecd-nea.org/nsd/docs/2016/cnra-r2016-3.pdf

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, prepared by Dr Steven Downey (NRC, United States), is based on discussions and input provided by members of the CNRA's Working Group on the Regulations of New Reactors or staff of the regulatory bodies listed below. Mr Janne Nevalainen (STUK, Finland), with the support of Mr Young-Joon Choi (NEA Secretariat), chaired the meetings and supervised the work carried out by the group.

- Christian Carrier, Canadian Nuclear Safety Commission (CNSC), Canada
- Janne Nevalainen, Säteilyturvakeskus (STUK), Finland
- Anne-Cécile Rigail, Autorité de Sûreté Nucléaire (ASN), France
- Yeon-Ki Chung, Korean 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 of Nuclear Regulation (ONR), United Kingdom
- John Monninger, Nuclear Regulatory Commission (NRC), United States
- Steven Downey, NRC, United States

In addition, the Nuclear Regulation Authority (NRA) of Japan responded to the survey in co-operation with the Working Group.

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LIST OF ABBREVIATIONS AND ACRONYMS

ANSI	American National Standard Institute
AOO	Anticipated operational occurrences
ASME	American Society of Mechanical Engineers
ASN	Autorité de Sûreté Nucléaire (France)
CNRA	Committee on Nuclear Regulatory Activities (NEA)
CNSC	Canadian Nuclear Safety Commission
COL	Combined licence
CSA	Canadian Standards Association
DBA	Design-basis accidents
DBE	Design-basis event
DEC	Design-extension conditions
EPRI	Electric Power Research Institute
FTE	Full-time equivalent
GDA	Generic-design assessment
GDC	Generic-design criteria
HF	Human factors
HMI	Human machine interface
HRA	Human reliability analysis
HSE	Health and Safety Executive
IAEA	International Atomic Energy Agency
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
IRSN	Institut de radioprotection et de sûreté nucléaire (France)
ISO	International Organization for Standardization
KEPIC	Korea Electric Power Industry Code
KINS	Korean Institute of Nuclear Safety
LAG	Licence application guide
MCR	Main control room
NPP	Nuclear power plant
NRA	Nuclear Regulation Authority (Japan)
NRC	Nuclear Regulatory Commission (United States)
NSSC	Nuclear Safety and Security Commission (Korea)
ONR	Office of Nuclear Regulation (United Kingdom)
PCSR	Pre-construction safety report

PRA	Probabilistic risk assessment
PSA	Probabilistic safety analysis
PWR	Pressurised water reactor
RAI	Request for additional information
RCC	Règles de conception et de construction
SAPs	Safety assessment principles
SAR	Safety analysis report
SER	Safety evaluation report
SI	Structural integrity
SNSA	Slovenian Nuclear Safety Administration
SRG	Safety review guidelines
SRP	Standard review plan
SSCs	Structures, systems and components
SSE	Safe shutdown earthquake
STUK	Säteilyturvakeskus (Radiation and Nuclear Safety Authority of Finland)
TAG	Technical assessment guide
TSC	Technical support contractor
TSO	Technical Support Organisation
UJD	Úrad Jadrového Dozoru (The Nuclear Regulatory Authority of the Slovak Republic)
WENRA	Western European Nuclear Regulators Association
WGRNR	Working Group on the Regulation of New Reactors
YVL	Ydinturvallisuusohjeet (Regulatory guides on nuclear safety and security, Finland)

EXECUTIVE SUMMARY

At the tenth meeting of the 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 11 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 classification of structures, systems and components category.

The classification of structures, systems and components category includes the following technical topics; classification of systems, structures, and components; plant design for protection against postulated piping failure; seismic and dynamic qualification of safety related mechanical and electrical equipment; and environmental qualification of mechanical and electrical equipment. 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.
- In addition to the regulations, it is a common practice for countries to make use of guidance documents and both domestic and international standards to provide the technical basis for acceptability. Commonly identified standards include IAEA, American Society of Mechanical Engineers (ASME), and International Electrotechnical Commission (IEC).
- The most commonly identified technical expertise needed to perform design reviews related to this category are mechanical engineering. However, a range of other technical disciplines are employed to perform reviews related to this technical category.

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 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 classification of structures, systems and components category.

SURVEY

The Second Phase, or Design Phase, of the licensing process survey conducted by the CNRA Working Group on the Regulation of New Reactors (WGRNR) covers 11 general technical categories that are based on IAEA Safety Guide GS-G-4.1. Under these 11 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 11 general technical categories covered in the survey. This report will present the results of the survey related to the classification of structures, systems, and components 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

Canada

Preamble

The CNSC's licensing process for Class I nuclear facilities is described in REGDOC-3.5.1 Licensing Process for Class I Nuclear Facilities and Uranium Mines and Mills [1]. To assist potential licensees with the process, application guides have been developed for the various licensing stages. The design of the nuclear power plant (NPP) is extensively reviewed during the "application to construct" stage and guidance for this stage is provided in RD/GD-369 Licence to Construct a Nuclear Power Plant [2] (RD/GD-369 was published in August 2011 and is in the process of being updated to reflect the latest versions of more recent published regulatory documents).

The construction licence application guide (LAG) [2] identifies the information that should be submitted in support of an application for a licence to construct a nuclear power plant. The information must be in sufficient detail to allow CNSC staff to make a determination regarding the acceptability of the safety case. The review of the application focuses on determining whether the proposed design, the safety analysis and other required information meet regulatory requirements. The evaluation involves engineering and scientific analysis, taking into consideration national and international standards and best practices in nuclear facility design.

Sections 5 and 6 of the LAG [2] outline the required information for the general design and detailed design of plant structures, systems and components (SSCs) that should be provided in the application documentation. The expectations are based on the CNSC's design requirements as detailed in regulatory document REGDOC-2.5.2 "Design of Reactor Facilities: Nuclear Power Plants" [3] (Note: RD/GD-369 refers to RD-337, however, this regulatory document has now been superseded by REGDOC-2.5.2).

The scope and level of detail of the staff's review of the licence application is based on the guidance provided in the applicable sections of the regulatory documents and on an internal CNSC work instruction for technical assessment. As part of the review, the staff also considers operating experience, accident reports and lessons learnt from the nuclear industry.

Once an application has been formally accepted by the CNSC Commission tribunal and a licence to construct has been issued, CNSC staff reviews the information provided for compliance with the regulatory requirements included in licence and performs confirmatory inspections and analyses, as necessary, to verify compliance.

Survey Response

The WGRNR (NEA) activity for the design aspects of licensing survey aims to gather data from member states and report on the level of technical detail needed for regulatory authorisation. This survey response covers the technical category of "Classification of Structures – Volume 5" Report. This category has four technical topics for which the CNSC has responded. These are:

1. Classification of systems, structures and components (e.g. Functions, includes supports, piping systems);
2. Plant design for protection against postulated piping rupture;
3. Seismic and dynamic qualification of safety related mechanical and electrical equipment;
4. Environmental qualification of mechanical and electrical equipment (e.g. temperature, humidity, radiation, pressure).

The CNSC's response to the survey questions for the four technical topics of Volume 5 is provided in the appendices. A high level summary to provide some additional insights or clarification is provided at the beginning of each table.

Finland

Finland's response to this survey is based on Finnish regulatory review and assessment of the European Pressurised Reactor (EPR) construction licence application (2004). It should be noted that STUK has recently revised its regulatory guides (YVL-Guide) and STUK regulations (2013).

The latest STUK regulations state that the safety functions of a nuclear power plant shall be defined and that the related systems, structures, and components, shall be classified on the basis of their safety significance. Also the requirements set for, and the actions taken to ascertain the compliance with the requirements of, the systems, structures and components implementing safety functions shall be commensurate with the safety class of the item in question. This also applies to connecting systems, structures, and components.

In addition, systems, structures and components that implement, or are related with, safety functions shall be designed, manufactured, installed and used so that their quality level is sufficient considering the safety significance of the item in question. The assessments, inspections, and tests (including environmental qualification), required to verify their quality level shall also consider the safety significance of the item in question.

The latest valid regulatory guide for SCC's is Guide YVL B.2 (2013). In this revision, Safety Class 4 was removed. Also, new system classification category EYT/STUK was introduced for systems belonging to Class EYT (non-nuclear safety). Systems shall be allocated to Class EYT/STUK if the following criteria are met:

1. The system has facility-specific risk importance in consequence of the initiating events caused by its failure.
2. The system protects safety functions, such as fire protection systems, against internal or external threats.
3. The system monitors the radiation, surface contamination or radioactivity of the plant, instruments, workers or the environment (e.g. the environmental radiation monitoring network) but is not assigned to Safety Class 3.
4. The system is necessary for bringing the facility to a controlled state in case of an event involving a design basis category design-extension category (DEC) combination of failures (DEC B) or a rare external event (DEC C).

The quality management requirements applied to the systems, structures and components of different safety classes are given in Guides YVL A.3, *Management system for a nuclear facility*, YVL B.1, *Safety design of a nuclear power plant*, and with regard to components and structures in various fields of technology, in the E Series YVL guides.

However, the changes made in updated regulations and guides do not have significant influence on the radiation or nuclear safety or on the level of detail in which STUK shall review the safety classification documentation.

The survey responses also provide an outline of the regulatory skill sets needed and resources used in each area of the construction application review and assessment.

France

The defence-in-depth principle has to be used to demonstrate that the three basic safety functions – reactivity control, cooling the fuel and confining radioactive substances – are correctly ensured.

The implementation of the defence-in-depth principle can be supported by the introduction of a classification for the safety functions and systems. The aim of this classification is to define general requirements applicable to safety functions and systems with a prioritisation of the requirements depending on the safety importance of the functions and systems.

The applicant should demonstrate that the structures, systems and components can withstand the design basis and beyond design basis conditions, normal and accidental conditions, severe and extreme environmental loads and severe accident.

The applicant should describe the classification approach (methodology, classes of SSCs and exigencies affected to each class) and provide the classification list of the plant equipment.

Equipment needed to achieve the safety demonstration has to be seismic and environmental qualified in which it is required, normal and accidental conditions, including severe accident.

The ASN and its TSO reviews the information provided by the applicant and could perform inspections to verify that the safety requirements described in the safety analysis report are correctly taken into account during design, construction, procurement and assembly specifications of the equipment.

The ASN and TSO staff performing the technical review are engineers.

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 TEPCO’s Fukushima Daiichi NPS accident, all nuclear power plants were stopped. Only the nuclear power plants that conform to the new regulatory requirements could restart.

The new regulatory requirements significantly enhance design basis and strengthen protective measures against natural phenomena which may lead to common cause failure; for example, the 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, strict and thorough measures for fire protection and, countermeasures against internal flooding.

The new regulatory requirements now require preventing core damage under postulated severe accident conditions, such as establishing SSCs, procedures, etc., which make a reactor subcritical 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 loss of large area of NPP due to extreme natural hazards or terrorisms. Applicants should provide information including PRA report and safety analysis reports.

The NRA has issued lots of requirements, standards, and guidelines on the above since its establishment. The NRA staff reviews the design of commercial nuclear power plants in terms of classification of SSCs based on functions and safety significance, taking into account design-basis events and severe accident conditions.

Korea

Information provided in this report is based on the application review of the APR1400 Nuclear Power Plant. Safety reviews of the documents submitted by the applicant are performed by 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 relevant laws and regulations. The review plan is made to perform an in-depth review to be conducted on the important items relating 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 confirmatory audit analysis that presented in the response.

The principal criteria for regulatory review related to classification of structures, systems, and components (SSCs) including other topical areas (plant design for protection against postulated piping rupture, seismic and dynamic qualification of safety related mechanical and electrical equipment, environmental qualification of mechanical and electrical equipment) are provided in the “Regulation on Technical Standards for Nuclear Reactor Facilities, etc.”. This Regulation prescribes the specific requirements for acceptance criteria for the classification of SSCs including above topical areas. In addition, the relevant NSSC Notices prescribe the specific requirements. Korea Electric Power Industry Codes (KEPIC) and Standards endorsed through NSSC Notices can be used as applicable codes and standards for the detailed guidelines for the classification of SSCs including above topical areas.

The KINS also developed safety review guidelines (SRGs) and regulatory guides that prescribe acceptance criteria, and review procedures, regulatory positions applies during a design review for the for the classification of SSCs including above topical areas. Some topical areas such as environmental qualification of mechanical and electrical equipment is verified and confirmed by the KINS on-site audit or vendor inspections process. The staffs performing the design review required to have a review skill in the areas of mechanical, electrical engineering, etc. and should have specialised experiences with the applicable codes and standards related to the classification of SSCs including above topical areas.

Slovak Republic

The information provided is based on Slovak legal framework which accommodates 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 all the structures, systems and components are designed in compliance with the technical codes and standards, and legislative requirements. The applicant also has to demonstrate that the structures, systems, and components, can withstand the design basis conditions and design-extension conditions. SSCs have to be accordingly qualified. It has to be demonstrated that all quality requirements are fulfilled. The applicant has to submit the list of the classified equipment. The main goal of all submitted documentation is to ensure that all legislative requirements are fulfil and that a nuclear facility will be operated safely and the public will be protected against undesirable effects of nuclear facility.

Review of applicants’ submitted documentation is usually performed by regulatory body employees and also with TSO. In case of using support services from TSO there is a condition of TSO independence. This condition resulting from fact, that the Slovak Republic is small and there is no a lot of organisation with relevant skills in this field. So we have to prevent of possibility, that the same organisation will support services for nuclear facility and also for regulatory body.

Slovenia

The information provided is based on the review of a licensing process for approval of classification of structures, system and components (SSCs). The fundamental purpose is for the applicant to demonstrate that SSCs, 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 SSCs changes for example, the licensing system is carried out in the same way, only less intensive.

The basic design bases and the requirements for SSCs safety classification and categorisation are set in Rules on Radiation and Nuclear Safety Factors. They based on WENRA reference levels and IAEA safety standards. The main goal of licensing documentation is to ensure that all legislative requirements are fulfil. This means that the applicant should demonstrate that the SSCs can withstand the design bases and design-extension conditions, operational states and accident conditions including severe accident.

Design information provided by the applicant in this technical category should show that each SSCs shall be classified into a safety class according to its importance to safety. SSCs shall be designed, manufactured and maintained so as to ensure reliability and quality adequate for the importance of the SSC for safety. Detail information of seismic and dynamic qualification of all equipment important to safety and information of a qualification programme for safety-related mechanical and electrical SSCs are requested to confirm the capability of SSCs to achieve their design functions over the entire design service life.

The Slovenian Nuclear Safety Administration evaluates during the licensing process that the applicant has provided complete information to demonstrate that the design, materials, fabrication methods, inspection techniques used conform to all applicable regulations, industrial codes and standards. The TSO`s independent evaluation report is obligatory for approval. Mechanical engineering and electrical engineering are the primary expertise needed to successfully perform SSCs review and assessment.

United Kingdom

The information provided here is relevant to the technical review and assessment of the submissions, pre-construction safety report (PCSR) and its supporting documentation made to the Office for Nuclear Regulation (ONR) for Generic Design Assessment (GDA) applicable to a reactor design(s) intended for construction and subsequent operation in the United Kingdom (UK). The submissions and PCSR are expected to address the categorisation of safety functions and the classification of structures, systems and components (SSCs) that are within the scope of those aspects of reactor design considered in GDA.

The categorisation of safety functions and the classification of SSCs forms an important part of the design information provided by the reactor vendor, as the GDA requesting party, and, where applicable, new licensees to establish the capability of SSCs and the safety functions that they are required to perform under all normal operational modes and, as necessary, anticipated accident conditions within the design basis. It is also expected that SSCs may need to be considered as mitigation against potential accidents beyond the design basis and severe accidents.

Since publication of the original NEA CNRA report on this topic, the ONR has introduced a technical assessment guide (TAG), namely, T-TAST-GD-0094 “Categorisation of Safety Functions and Classification of Structures, Systems and Components”[6]. The purpose of this TAG is to provide advice to

6. T-TAST-GD-0094 is publicly available from ONR’s website at

ONR inspectors on regulatory expectations of the licensee's arrangements for identifying and categorising safety functions and classifying the SSCs that deliver them in accordance with relevant ONR Safety Assessment Principles for nuclear facilities. Guidance is provided on the factors which ought to be considered in each stage of this process and relevant good practice (RGP), including the IAEA's SSG-30 and TECDOC-1787, for the categorisation and classification methodology used. ONR inspectors also use this TAG to assess new licensee's arrangements during GDA or permissioning process for new build or plant modification projects.

The approach outlined in this TAG has been applied in the ongoing GDA assessment of Hitachi-GE's UKABWR and new build construction of the UKEPR at EDF's Hinkley Point C licensed site. This covers the classification of SSCs in mechanical engineering, electrical engineering, instrumentation and control, radiological protection, human factors, and both fault and probabilistic safety analysis that are subject to assessment by ONR as outlined in Appendix A of the original NEA CNRA report on this topic. These assessments are carried out by ONR inspectors according to their technical expertise, seeking specialist advice from external technical support contractors when required.

The ONR approach to assessment of classification of SSCs in new reactor applications is to review the scheme proposed for use in categorisation and classification studies by performing a sample inspection of SSCs identified from GDA submissions and/or the PCSR. This may involve confirmatory analysis on any issues identified where the SSC classification scheme or results of classification studies vary markedly from regulatory expectations.

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 (COLs) issued under Title 10 of the Code of Federal Regulations (10 CFR) Part 52, *Licenses, Certifications, and Approvals for Nuclear Power Plants*. Typically, the most extensive reviews related to this technical category are performed at the design certification stage. New reactor COL applicants generally incorporate most, if not all, of the information related to systems, structures, and components that is included in a certified standard plant design. COL applicants also conduct site-specific analyses associated with certain design parameters, such as environmental and seismic evaluations, to confirm that the standard plant design is suitable for the proposed plant site. If the COL applicant identifies parameters that are not bounded by the standard design, additional analyses are performed to demonstrate that the systems, structures, and components are able to perform their intended functions. Otherwise, the COL applicant may propose a departure from the standard design in order to provide an alternative. In addition to departures, a COL application may also include site-specific systems, structures, and components that are not part of the standard design. As such, the staff's review of this technical category at the COL application stage would focus on site-specific information and departures from the standard plant 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 show that systems, structures, and components are designed to withstand the design basis and specified beyond design basis conditions, normal and abnormal operating conditions, severe and extreme environmental loads, and severe accidents. To accomplish this, the applicant should provide a complete description of the design and analysis procedures

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www.onr.org.uk/operational/tech_asst_guides.htm.

followed to demonstrate that systems, structures, and components can withstand appropriate loads with sufficient margin of safety. The applicant should also provide provisions for design, manufacture, testing, installation, surveillance and operational maintenance to provide assurance of the seismic, environmental, and functional capability of the systems, structures, and components to perform their intended functions.

The regulations related to this technical category require that systems, structures, and components be designed to their respective codes and standards based on the classification of the design. The regulations also require that systems, structures, and components be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. Several regulatory guides have been developed to provide guidance to applicants and licensees on acceptable approaches to meet the regulatory requirements. The NRC staff has endorsed codes and standards related to piping and component designs, as well as procedures for testing, surveillance and inspection.

Once an application has been formally accepted, the 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 mechanical engineering related to systems, structures, and components 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.

Mechanical engineering is the primary expertise needed to successfully perform reviews in the areas of classification, protection against postulated piping rupture, and seismic, dynamic and environmental qualification of mechanical and electrical equipment. In some areas, expertise in electrical engineering, structural engineering, and health physics is also needed to complete a thorough review.

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DISCUSSION

Under the category of classification of structures, systems, and components, 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. For the topic of classification of systems, structures and components (e.g. Functions, includes supports, piping systems), most countries responded that the applicant provides a description of the approach used to classify the various structures, systems, and components (SSCs) of the nuclear plant. When describing the classification approach, it is common for the applicant to describe the design requirements and/or safety requirements for each class, the safety functions to be performed for each class, and the applicability of codes and standards. It is also common for the applicant to provide a list of the SSCs and their classification. The survey responses indicate that an applicant would typically provide the safety classification, seismic classification, and quality group classification for the applicable SSCs.

In the area of plant design for protection against postulated piping ruptures, most countries responded that the applicant a description of the measures used to prevent or mitigate the effects of postulated piping ruptures. Particularly, those ruptures related to moderate and high energy piping systems. Many of the responses also indicated the importance of an applicant describing the structural integrity of plant SSCs, such as the containment, main coolant piping, etc., and how they are designed to protect against the effects of piping breaks. In addition, some countries commonly responded that the applicant also provides an analysis of the effects of various pipe breaks. Commonly identified aspects of the analysis include the location or potential pipe breaks, the environmental effects, and dynamic effects, such as pipe whip and fluid jet impingement.

For the seismic and dynamic qualification of safety related mechanical and electrical equipment, most responses indicate that an applicant identifies all equipment that should be designed to withstand the effects of earthquakes and the full range of normal and accident loadings. For the identified equipment, most responses indicate the applicant is expected to describe the equipment design as well as the seismic and dynamic qualification approach. It is also common for the applicant to describe the tests and analyses used to ensure the integrity, functionality and/or reliability of the applicable equipment. The most commonly identified analysis was a seismic or seismic hazard analysis.

For the environmental qualification of mechanical and electrical equipment (e.g. Temperature, Humidity, Radiation, Pressure), several countries responded that an applicant identifies all equipment important to safety that is to be environmentally qualified. It is common for the applicant to describe the design, location, and applicable environmental conditions of the equipment of interest. It is also common for the applicant to describe the environmental qualification approach, including the use of analyses, calculations, testing, etc., as applicable to ensure that the equipment is capable of performing its safety function. Some countries noted that they require the submittal of a formal environmental qualification programme.

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. All countries review the information provided by the applicant for compliance with the applicable regulatory requirements, guidelines, or codes and standards. Confirmatory analyses are commonly mentioned as part of the design reviews related to this category. However, the types of confirmatory analyses tend to differ among the countries that responded to the survey. Confirmatory analysis methods identified in the survey responses include the use of computer programs, inspections, technical assessments, the verification of test results, and probabilistic analysis.

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 both country-specific and internationally recognised standards related to the technical category. For example, IAEA standards were identified as part of the technical basis for granting regulatory authorisation in Canada, Finland, the Slovak Republic and Slovenia. All four countries commonly identify the use of IAEA standards in relation to the seismic and dynamic qualification of safety related mechanical and electrical equipment. Also, Canada and Slovenia commonly identified the use of IAEA standards in relation to the classification of systems, structures, and components; and the environmental qualification of mechanical and electrical equipment.

Canada, Finland, and the United States all identified the use of ASME codes as part of the technical basis for granting regulatory authorisation. Canada identified the ASME code related to every technical topic. Finland identified the ASME Section III, Division 2, requirements for containment design in relation to the plant design for protection against postulated piping ruptures, while the United Stated identified the ASME Section III in relation to classification of SSCs and ASME AG-1 in relation to seismic and dynamic qualification. Another commonly used consensus standard is the International Electrotechnical Commission (IEC) Standards, which were identified by Canada, Finland, and the United Kingdom as part of the technical basis for granting regulatory authorisation.

Skill Sets Required to Perform Review

Mechanical engineering was the most commonly identified technical skill needed to perform the reviews related to this technical category. Other technical disciplines that were identified on a less consistent basis include plant systems engineers, structural engineers, electrical engineers, materials engineers, and nuclear engineers, as well as human factors, radiation protection, and risk assessment experts.

Specialised Training

Although the specific training requirements may vary, all countries have 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 Classification of SSCs category is provided in the table below. It is noted that in France 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 Classification of SSCs [1]	Basis for Estimate
Canada	517.5 working days	CNSC regulatory framework and licensing experience
Finland	320 working days	European Pressurised Reactor (EPR) construction licence application review
France	-	Resources (hours) are not set up for each individual review area. The effort needed to review a new plant design strongly depends on the degree of novelty of this design
Japan	-	Resources (hours) are not set up for each individual review area
Korea	596.25 working days	APR1400 Nuclear Power Plant application review
Slovak Republic	-	Level of effort defined by regulation and dependent upon the activity to be approved
Slovenia	312.5 working days	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	780 working days	Technical review of a pre-construction safety report and associated documents
United States	537.5 working days	Standard design certification review

Note:

Level of effort provided in this table is the sum of the level of effort provided in the appendices.
All values have been converted to working days at the rate of 8 hours/working day or 225 working days per FTE.

CONCLUSION

This report focused on the results of the design survey related to the “classification of structures, systems, and components. 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 in some manner by all of the regulatory authorities that provided responses.
- In addition to the regulations, it is a common practice for countries to make use of guidance documents and both domestic and international standards to provide the technical basis for acceptability. Commonly identified standards include IAEA, American Society of Mechanical Engineers (ASME) and International Electrotechnical Commission (IEC).
- The most commonly identified technical expertise needed to perform design reviews related to this category are mechanical engineering. However, a range of other technical disciplines are employed to perform reviews related to this technical category.

Additional reports will be issued by the Working Group on the Regulation of New Reactors (WGRNR) in order to discuss the results of the design-phase survey in other technical areas.

APPENDIX A: CLASSIFICATION OF SYSTEMS, STRUCTURES AND COMPONENTS (E.G. FUNCTIONS, INCLUDES SUPPORTS, PIPING SYSTEMS)

Summary Table:

Country	Is this area reviewed?	Are Confirmatory Analyses Performed?	Expertise of Reviewers	Level of Effort
Canada	Yes	Yes	Engineering or scientific degree and work experience in related area (mechanical, nuclear, electrical, structural and etc.). Advanced understanding of nuclear power plant design and safety analysis	0.6 FTE (135 working days)
Finland	Yes	No	Experience with classification principles and knowledge of plant and system design	40 working days
France	Yes	Yes	Engineering	See Note 1
Japan	Yes	Yes	Civil, structural, and mechanical engineers.	See Note 4
Korea	Yes	No	Mechanical engineering, materials engineering, nuclear engineer, plant systems engineer	300 hours
Slovak Republic	Yes	Yes	Technical Engineer	See Note 2
Slovenia	Yes	No	Mechanical engineer	800 hours (Note 5)
United Kingdom	Yes	Yes	Mechanical engineering, electrical engineering, radiological protection, human factors, probabilistic assessment,	~ 34 person-months (5 440 hours) (Note 3)
United States	Yes	Yes	Mechanical engineering, structural engineering	1 400 hours

Notes:

1. In France, the effort needed to review a new plant design strongly depends on the degree of novelty of this design. For Flamanville 3, the review has lasted one year.
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 estimate of hours for the UK includes 100 person weeks for work related to human factors. This 100 week estimate covers all individual topics in the survey.
4. In Japan, resources are not set up for the individual review area.
5. 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.

Canadian Nuclear Safety Commission
High Level Summary for Classification of Structure, Systems and Components (e.g. Functions, includes supports, piping systems)

The CNSC requires that structures, systems and components be classified using a consistent and clearly defined classification method. Its position is that a well-defined, controlled and documented safety classification process is a keystone in establishing and maintaining the overall safety case for a nuclear facility. The purpose is to ensure that the SSCs are then designed, manufactured, constructed, installed, commissioned, operated, tested, inspected and maintained such that their quality and reliability is commensurate with their importance to safety. This approach helps to ensure that engineering design rules are applied in a manner that recognises and supports the need for safety important SSCs to function as designed throughout the lifetime of the plant.

The applicant's safety classification philosophy, principles and associated design requirements are examined and CNSC staff looks for early examples of applied safety classification to confirm the application of the safety classification principles. The method for classifying the safety significance of SSCs important to safety is expected to be based primarily on deterministic methodologies, complemented (where appropriate) by probabilistic methods and engineering judgement. The CNSC's expectations with respect to safety classification have evolved and the organisation has moved to adopt a more systematic method as described in IAEA specific safety guide SSG-30 [4] and accompanying TECDOC-1787 [5].

The regulatory framework related to the technical category of safety classification is not intended to be overly prescriptive and provides for flexibility in the safety classification scheme. The safety class of an SSC should be linked to the selection of such things as specific pressure boundary classification, electrical and I&C equipment class, structure, environmental and seismic classification levels. Other regulatory documents may require that SSCs be designed to their respective codes and standards based on the assigned safety class.

The classification method should also address special cases, such as where there is sharing of the structures/components between two or more systems, or a system performs multiple functions. This is done since some SSCs could potentially be vulnerable to fault propagation due to cross-links, or common cause events there is a potential for physical interaction (e.g. pipe whip, jet impingement) or functional interaction between the SSCs (e.g. depressurisation of heat transport system for the emergency core cooling injection, initiation of the emergency water supply, or shutdown cooling). Also, the boundary of some important systems can be a function of the operating configuration of the plant.

Classification of systems, structures and components	**Canada** **CNSC**
Design Information Provided by Applicant	As part of an application for a licence to construct [2], the Licence Application Guide (LAG) section 5.3 states that an applicant is to describe the approach adopted in the design for the classification of the structures, systems and components important for the safety of the plant. It elaborates to say that the approach taken should be consistent with the expectations of section 7.1 of REGDOC-2.5.2 (RD-337) and with the identified codes and standards to be used. The application should include the proposed criteria for deciding on the appropriate design requirements for each class such as: 1. appropriate codes and standards to be used in the design, manufacturing, construction, testing and inspection of individual SSC; 2. system-related characteristics such as the degree of redundancy, the diversity, the separation, the reliability expectations, the environmental qualification expectations (and seismic qualification expectations); 3. availability requirements for particular SSC for on-demand duty, as well as for reliability for the prescribed mission time; 4. quality assurance requirements. The application's description of the classification design requirements should also address special cases, such as where: 1. There is sharing of the structures/components between two or more systems, or a system performs multiple functions. 2. Some SSC could potentially be vulnerable to fault propagation due to cross-links, or common cause events. 3. There is a potential for physical interaction (e.g. pipe whip, jet impingement) or functional interaction between the SSC (e.g. depressurisation of heat transport system for the emergency core cooling injection, initiation of the emergency water supply, or shutdown cooling). 4. The boundary of some important systems can be a function of the operating configuration of the plant. The classification of systems, structures and components should provide the criteria for the level of design detail included in the application as related to the SSC. The description provided should be sufficient to permit an independent assessment of the adequacy of the classification approach.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	The Canadian Nuclear Safety Commission (CNSC) staff conducts the following activities: • reviews the information provided by the applicant for compliance with the CNSC regulations, • asks for clarifications for issues uncovered during staff review and makes requests for additional information as necessary, • reviews the additional responses and resolves technical issues with applicants or licensees. The scope of the review consists of the following: The scope and level of detail for the staff's review is based on the type of application received (e.g. licence to construct, licence to operate (initial or renewal). In addition,

	CNSC staff can either perform a detailed review or adapt a graded approach based on risk and complexity. The scope of the review is aligned with CNSC REGDOC 2.5.2, section 7.1 *Classification of structures, systems and component* [3]. The CNSC has developed specific Work Instructions (WIs) that are used during the technical assessment stage of the Environmental Assessment and Licensing process. They are internal documents that provide instructions to CNSC staff on the conduct of an assessment. They also inform potential applicants, and the public, about the criteria used to assess licence applications for new nuclear power plants.
What type of confirmatory analysis (if any) is performed?	The CNSC performs a technical assessment of the design and safety analysis to verify that the safety classification methodology is correctly incorporated into the design.
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	The applicable CNSC documents are listed below: • Physical Design, Design of Reactor Facilities: Nuclear Power Plants REGDOC-2.5.2. • Design of Small Reactor Facilities RD-367. • Licence Application Guide Licence to Construct a Nuclear Power Plant RD/GD-369. • Reliability Programs for Nuclear Power Plants (RD/GD-98). The applicable Codes and Standards related to this area are: • IAEA SSG-30 Safety Classification of Structures Systems and Components. • IAEA TECDOC-1787 Application of Safety Classification of Structures Systems and Components. • ASME Boiler and Pressure Vessel Code, Section III. • N285.0/N285.6 series, general requirements for pressure-retaining systems and components in CANDU nuclear power plants/Material standards for reactor components for CANDU nuclear power plants. • B51-14 – Boiler, pressure vessel, and pressure piping code. • N287.1, General requirements for concrete containment structures for nuclear power plans. • N291-15, Requirements for safety-related structures for nuclear power plants N289.1, General requirements for seismic design and qualification of CANDU nuclear power plants. • N290.8, Technical specification requirements for nuclear power plant components. • N290.13, Environmental qualification of equipment for CANDU nuclear power plants. • N290.14-15, Qualification of digital hardware and software for use in instrumentation and control applications for nuclear power plants. • N293-12, Fire protection for nuclear power plants. • Joint Canada-United States Guide for Approval of Type B(U) and Fissile Material Transportation Packages (RD-364). • National Building Code of Canada 2015 (NBC). • National Fire Code of Canada 2015 (NFC). CNSC expect that classification of SSCs and events will meet the criteria regulatory documents listed documents above. In view of the multitude of classification, the highest class from the various classification schemes applies to the structure, system or component.

	The identified events shall be classified, based on the results of deterministic safety analysis, probabilistic safety analysis and engineering judgement, into the following three classes of events: anticipated operational occurrences (AOO); design-basis accidents (DBA) and design-extension conditions (DEC). The criterion for determining safety importance of SSC is based on: 1. safety function(s) to be performed; 2. consequence(s) of failure; 3. probability that the SSC will be called upon to perform the safety function; 4. the time following a PIE at which the SSC will be called upon to operate, and the expected duration of that operation.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	Engineering or scientific degree and work experience in related area (mechanical, nuclear, electrical, structural and etc.). Advanced understanding of nuclear power plant design and safety analysis.
Specialised Training, Experience and/or Education Needed for the Review of this topic	Basic understanding of the IAEA safety classification methodology and knowledge of regulatory documents as listed above. Understanding of plant system interactions and dependencies.
Level of Effort in Each Review Area	0.6 FTE (225 days per FTE)

Classification of systems, structures and components	**Finland** **STUK**
Design Information Provided by Applicant	• Classification principles used (safety classification, quality classification, seismic classification against safe shutdown earthquakes [SSE]). • Classification of functions. • Classification lists of systems, structures and components. • Probabilistic risk analysis (PRA) of classification.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	Confirm that classification is in line with principles presented in YVL Guides. Safety functions shall be defined, and systems, structures and components (SSC) shall be classified on the basis of their safety significance (safety classes 1, 2, 3 and 4, and Class EYT). SSCs are classified between S1 against SSE and S2, which are not required for SSE. Further on such SSCs in seismic class S2 are identified so that their collapse of other loss of functionality will be rejected in order not to endanger S1 classified functions. When a structure or a component contributes to the accomplishment of a safety function on which the system's classification is based, they shall be assigned to the same safety class as the system. Classification has to fulfil defence-in-depth principle, i.e. the safety shall be ensured by means of successive levels of protection independent of each other, and this principle shall extend to the operational and structural safety of the plant (operational and structural classification).
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 2.1, Nuclear power plant systems, structures and components and their safety classification. • YVL Guide 2.6, Seismic events and nuclear power plants. • YVL Guide 2.8, Probabilistic safety analysis in safety management of nuclear power plants. • YVL Guide 3.3, Nuclear facility piping. • YVL Guide 4.1, Concrete structures for nuclear facilities. • YVL Guide 4.2, Steel structures for nuclear facilities. • YVL Guide 5.6, Air-conditioning and ventilation systems and components of nuclear facilities.
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	Experience with classification principles and knowledge of plant and system design.
Level of Effort in Each Review Area	Regulator’s review: 40 working days.

Classification of systems, structures and components	**France ASN**
Design Information Provided by Applicant	The applicant provides the following information in the safety analysis report (SAR): • Description of the classification approach (methodology, classes of functions and SSCs and requirements for each class) • SSCs Safety classification • Quality group classification of SSCs regarding RCC code (Design and construction rules for French NPP), sections M (mechanical), E (electric) • Seismic classification • Classification of mechanical SSC regarding to pressure and radiation of the fluid inside (cf. Ministerial Order of 12th December 2005 related to Nuclear pressure equipment – *ESPN Order)* • Safety requirements concerning each type of classification
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	A comprehensive review of the safety file (SAR + supporting documents) provided by the applicant is performed by the TSO and ASN. Compliance with regulations and codes is reviewed in detail. Considerations can also be given to experience feedback. For example, for Flamanville 3 NPP, the safety classification approach has been assessed and submitted to ASN advisory committee.
What type of confirmatory analysis (if any) is performed?	The ASN staff could perform inspections on design specifications to verify that the safety requirements related to classification described in the SAR are correctly take into account.
Technical basis • Standards • Codes • Acceptance criteria (e.g, can come from Accident analysis, regulatory guidance)	The applicable requirements on this topic is: • Section B.2.1 “classification of the safety functions, barriers, structures and systems” of the technical guidelines for design and construction of the next generation of NPP. • Ministerial Order of 12th December 2005 dedicated on Nuclear pressure equipment. • DRAFT regulatory guide “Reactor (PWR) design”. The applicable codes related to this topic is: • RCC code, sections M (Design and construction rules for mechanical components and E (Design and construction rules electric components).
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	Engineering

Specialised Training, Experience and/or Education Needed for the Review of this topic	Specialised training and experience to understand the plant systems interactions and dependences. The staff performing the technical review at IRSN (TSO) has a long experience (more than 10 years) on this topic.
Level of Effort in Each Review Area	The effort needed to review a new plant design strongly depends on the degree of novelty of this design. For Flamanville 3, this review of the classification of SSC has lasted a full year.

Classification of systems, structures and components	**Japan** **NRA**
Design Information Provided by Applicant	As part of the safety analysis report the applicant should describe the following: • Seismic classification of structures, systems, and components (SSCs) • Safety (including reliability) classification of SSCs • Materials and structures classifications of SSCs
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	These activities are to conform to the standards, criteria, and the like described below.
What type of confirmatory analysis (if any) is performed?	In the establishment permit application stage, acceptability of applicant's analytic method and the analysis results are confirmed. Independent evaluation for demonstration of the analysis results is performed, as needed, (cross check analysis).
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	Technical bases established by legislation and regulation : The regulatory requirement guides employed by the NRA Nuclear Safety Commission are listed below: • The NRA Ordinance on Standards for the Location, Structure and equipment of Commercial Power Reactors Examination Guide for Nuclear Reactor Siting and Criteria and its Application. • The NRA Ordinance on Technical Standards for Commercial Power Reactors Facilities. The regulatory guides that provide an acceptable approach to meeting the applicable regulatory requirements ports compiled by “the Special Committee on Examination of Reactor Safety” of the Nuclear Safety Commission are listed below: • The Regulatory Guide of the NRA Ordinance on Standards for the Location, Structure and Equipment of Commercial Power Reactors. • The Regulatory Guide of NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities. • Examination Guide for Classification of Importance of Safety Functions of Light Water Power Reactor Facilities.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Senior: Director for Nuclear Safety Examination. • Junior: Nuclear Safety examiner. • TSO: None since Japan Nuclear Energy Safety Organisation (JNES) was integrated into NRA as of April, 2014. Generally the staff who have more than 10 years’ experience are taken on the task, although no specific skill set is required.

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	As part of the safety analysis report the applicant should describe the following: • Seismic classification of structures, systems and components (SSCs). • Safety (including reliability) classification of SSCs. • Materials and structures classifications of SSCs.

Classification of systems, structures and components	**Korea** **NSSC and KINS**
Design Information Provided by Applicant	As part of the SAR, the applicant should describe or provide the following related to the classification of structures, systems and components: • Safety related systems such as pressure boundary vessels, heat exchangers, tanks, pumps, pipes, etc. • Related quality group classifications. • KEPIC/ASME codes and standards. • Date of flow systems like quality assurance requirement.
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 (request for additional information) 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 3.2.1, "Seismic Classification". • SRG 3.2.2, "Quality Group Classification".
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 12 (Safety Classes and Standards). • Article 13 (External Events Design Bases). • Article 14 (Protection against Fire, etc.). • Article 33 (Fuel Handling and Storage Facilities). **<Applicable Codes and Standards>** The applicable Codes and Standards related to this area are: • KEPIC MN (Nuclear- Mechanical). • KEPIC MD (Materials).
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Mechanical Engineer. • Materials Engineer. • Nuclear Engineer. • Plant System Engineer.
Specialised Training, Experience and/or Education Needed for the Review of this topic	• Experience in plant system engineering. • Knowledge of architectural engineering. • Knowledge of materials for reactor vessel. • Understanding of Codes and Standards (KEPIC, ASME, etc.).
Level of Effort in Each Review Area	Total: 300 hours

Classification of systems, structures and components	**Slovak Republic UJD**
Design Information Provided by Applicant	• Principles of categorisation of selected facilities (qualified facilities) into safety classes and requirements for the creation of a list of selected facilities categorised into safety classes. • Categorisation of selected facilities into safety classes i to iv based on deterministic methods. And if unavoidable, probabilistic methods and engineering assessments may also be used, taking into account: a. Safety functions performed, b. Consequences of their failure, c. The likelihood of their activity being required during their failure, d. The duration of the expected trigger event during which their activity may be required. • Calculations and calculation results to prove the resistance of selected facilities to seismic. • Activity and environmental influences during all test, operation and emergency conditions. • Considered in their design. • Design test results – they prove that facility is able to fulfil its role. • Accompanying technical documentation.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	Review of the submitted documentation, if it conforms to atomic act and regulations. Evaluate if the systems, structures and components are in compliance with all requirements arising from applicable regulations, codes and standards. Confirm that the systems, structures and components: • are able to manage their roles under normal, transient and accident conditions; • that the facilities have been properly classified to identify their importance to safety.
What type of confirmatory analysis (if any) is performed?	
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	Regulatory body regulations, Slovak Technical Standards, 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 classification 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 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.

Classification of systems, structures and components	**Slovenia** **SNSA**
Design Information Provided by Applicant	• Information of SSCs safety class according to its importance to safety. SSCs shall be designed, manufactured and maintained so as to ensure reliability and quality adequate for the importance of the SSC for safety. • Information of the SSCs classification into safety classes according to their importance for safety based on nuclear safety analyses carried out employing deterministic methods and supplemented with probabilistic methods and engineering judgement as appropriate. • Information of : – regulations and standards to be applied in design, manufacture, installation and inspection; – requirements for emergency power supply and SSC compatibility with anticipated ambient conditions; – availability/unavailability of systems necessary to achieve a safety function upon initiating events postulated in safety analyses employing deterministic methods; – Quality assurance requirements
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	• Reviews: Review of application material + TSO' independent evaluation report. • Scope of review: Determine adequacy of SSCs classification.
What type of confirmatory analysis (if any) is performed?	
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	• Rules on Radiation and Nuclear Safety Factors • SNSA Practical Guidelines • IAEA Safety Standards
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	Categorisation shall be undertaken by a group of experts familiar with the plant and proficient, as a minimum, in PSA, other types of safety analyses, plant operation, specification of design bases and system design. • Senior: mechanical engineer, • Junior: mechanical engineer • TSO: mechanical engineer

Specialised Training, Experience and/or Education Needed for the Review of this topic	• Understanding of plant system interactions and dependencies • Basic knowledge of codes and standards
Level of Effort in Each Review Area	Regulator: 300 hrs TSO' review time: 500 hrs

<table>
<tr><th>Classification of systems, structures and components</th><th>United Kingdom
ONR</th></tr>
<tr><td>Design Information Provided by Applicant</td><td>
<u>Electrical Engineering</u>
Submission on classification of systems structures and components
<u>Radiological Protection</u>
EDF and AREVA provided the following information.

Sub-chapters 12.2 and 12.3 of the Pre-construction safety report (PCSR) included shielding provisions which took account of the following.
“Realistic source term” – based on feedback on mean corrosion product and fission product activity levels from French NPPs, and used to estimate occupational doses. “Biological protection / shielding design source term” - based on maximum corrosion product and fission product activity levels from French NPPs, and used as a design parameter for buildings, systems and shielding provisions in the UK EPR.
An overview document that summarised radiological zoning and bulk shielding across the nuclear island was submitted in response to a Generic Design Assessment (GDA) Issue on radiological zoning and bulk shielding.

Responses to technical queries from ONR on shielding and the overview document (that summarised radiological zoning and bulk shielding) were submitted in the form of additional information and documentation.

<u>Human Factors (HF)</u>

Allocation of function criteria
System specifications of HMIs to support operator actions
Classification of C&I platforms that drive HMIs (PICS, SICS etc.)
System requirements for control rooms (MCR, RSS) and scenarios for use
Building, system and equipment specifications
Identification of safety significant operator actions – via HRA in PSA
HRA
Detailed substantiation of key human based safety claims (pre and post-fault)

<u>Mechanical Engineering (Mech. Eng.)</u>
Design information supplied by the requesting party including:

Safety case
Responses to Technical Queries
Responses to Cross cutting Regulatory Observation
Classification of Structures, Systems and Components document.

Specifically our assessment covered:

Check valves
Isolation valves that provide a confinement safety function

<u>Probabilistic Safety Analysis (PSA)</u>
A Probabilistic Safety Analysis (PSA) for the UK EPR was provided by the applicant as part of the safety submission for ONR’s Generic Design Assessment (GDA). However, no information was provided regarding the input of the PSA into the classification of Structures, Systems and Components. A GDA issue was raised regarding the methodology developed and applied for categorising safety function and classifying Structures, Systems and Components (see GDA Issue GI-UKEPR-CC-01 in http://www.hse.gov.uk/newreactors/2011-gda-issues-epr.htm).
</td></tr>
</table>

	As part of the resolution of this GDA issue, the applicant is expected to submit a methodology for classifying the safety significance of a structure, system or components and relevant examples of its applications. <u>SI</u> Pre-Construction Safety Report describing all aspect of the methodology adopted for the Classification and Categorisation of systems, structures and components and examples of how it is applied to certain systems
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	<u>Electrical Engineering</u> • Assessment in conjunction with other disciplines of classification requirements. • Assessment of defined requirements for equipment of each safety classification <u>Radiological Protection</u> *ONR and ONR's Technical Support Contractor (TSC) reviewed the following topic areas.* • Biological shielding design source term used as a design parameter for buildings, systems and shielding provisions in the UK EPR. • Shielding source terms - full power operation; during shutdown; other sources. • Shielding materials – concrete bulk shielding; neutron shielding provision; gamma shielding; liquid shielding. • Radiological classification of areas (zoning) – including bulk shielding. • Calculation methods – computational codes; application of codes in shielding assessments. • Shielding provisions for protection of the public from direct radiation, • Shielding provisions for protection of workers from direct radiation – bulk shielding provisions; local shielding and penetration assessments; temporary shielding provisions. <u>HF</u> • HF assessment of plant wide ergonomics and main HMIs • Assessment of HRA operator actions • Research into HRA data for digital interfaces • Assessment of the overall HF Integration programme as part of ALARP evaluation relating to human error <u>Mechanical Engineering</u> • Review of requesting party material • Determined adequacy of categorisation and classification methodology • Technical meetings with requesting party <u>PSA</u> No specific PSA assessment has been carried out in this area within GDA up to date. However, assessment of the methodology developed and applied for categorising safety function and classifying structures, systems and components is expected in the future. As identified in ONR's standards and criteria in the Safety Assessment Principles (SAPs) ECS.2, probabilistic methods should be used where appropriate to complement the deterministic methods for classifying the safety significance of a structure, system or components. <u>SI</u> • SI Specific Review: • Identification of the highest reliability components, and incorporation in the safety case (components where the reliability needs to be shown to be sufficiently high that gross failure can be discounted in the safety case).

<table>
<tr><td></td><td>

Review of criteria used to allocate nuclear pressure vessel class.
Equivalence of industrial pressure vessel standards with additional controls to nuclear pressure vessel standards.
See SI Step 4 Report Sections 4.1, 4.9.

</td></tr>
<tr><td>What type of confirmatory analysis (if any) is performed?</td><td>
<u>Electrical Engineering</u>
None
<u>Radiological Protection</u>
ONR's TSC performed independent confirmatory analysis on a few examples of shielding calculation submissions from EDF and AREVA which confirmed that EDF and AREVA's calculations were reproducible.

<u>HF</u>
None

<u>Mechanical Engineering</u>
Not required

<u>SI</u>
None
</td></tr>
<tr><td>Technical basis
Standards
Codes
Acceptance criteria
(e.g. can come from Accident analysis, regulatory guidance)</td><td>
<u>Electrical Engineering</u>
IEC standards.
ONR Safety Assessment Principles.
<u>Radiological Protection</u>
Full references for references referred to below are given in the "References" section of the radiological protection assessment report entitled " Generic Design Assessment – New Civil Reactor Build: Step 4 Radiological Protection Assessment of the EDF and AREVA UK EPR™ Reactor", available on www.hse.gov.uk/newreactors/reports/step-four/technical-assessment/ukepr-rp-onr-gda-ar-11-025-r-rev-0.pdf

The key pieces of legislation on the protection of workers and members of the public are IRR99 (Ref. 17), REPPIR (Ref. 18) and EPR10 (Ref. 19), and the key pieces of guidance are in the ACOP and guidance to IRR99 (Ref. 21) and in guidance to REPPIR (Ref. 22).
Guidance on radiation shielding is available in RP.6 and paras 493 to 495 of the SAPs (Ref. 4), and in TAG T/AST/002 on radiation shielding (Ref. 29).
Guidance on radiological zoning is in RP.3 and para. 485 of the SAPs (Ref. 4) and in paras 4.6 and 4.7 of the TAG on Radiological Protection (Ref. 33).

<u>Human Factors</u>

International Organization for Standardization (ISO) standards:

ISO 11064: Ergonomic design of control centres (2008).
ISO 9241: Ergonomics of human-system interaction (2002).
ISO 80416: Basic principles for graphical symbols for use on equipment (2005).
ISO 7000: Graphical for use on equipment (2004).
ISO 14617: Graphical symbols for diagrams (2004).
ISO 13406: Ergonomic requirements for work on flat panel display screens.
ISO 15534: Ergonomic design for the safety of machinery (2000).
ISO 14738: Safety of machinery – anthropometric requirements for the design of workstations at machinery (2002).
ISO 6385: Ergonomics principles in the design of work systems (1990).

</td></tr>
</table>

	Nuclear Regulatory Commission NUREGS: • NUREG-0711: Human Factors Engineering Programme Review Model (2004). • NUREG-0700: Human-System Interface Design Review Guidelines (2002). International Electro-technical Commission (IEC) standards • IEC 80416: Basic principles for graphical symbols for use on equipment (2002). • IEC 60073: Basic and safety principles for man-machine interfaces, marking and identification (2002). • IEC 60447: Man-machine interface – actuating principles; (2004). • IEC 60960: Functional design criteria for SPDS; (1988). • IEC 60964: Design for control rooms of nuclear power plants • IEC 61227: NPPs – Control rooms – Operator controls. • IEC 61771: NPPs – MCR – Verification and validation. • IEC 61772: NPPs – MCR – Application of visual display units (1995). • IEC 62241: NPPs – MCR – Alarm functions and presentation. Institute of Electrical and Electronics Engineers (IEEE) standards • IEEE 1023: guide of application of human factors engineering to systems, equipment and facilities of nuclear power generating systems (1988). Electric Power Research Institute (EPRI) Guide: • EPRI: Human Factors Guidance for Control Room and Digital Human-System Interface. Design and Modification (2005). Relevant HF SAPs and TAGs – see Table 1 below. Mechanical Engineering • Comparison to UK primary legislation and associated regulations. • Comparison to HSE Safety Assessment Principles and Technical Assessment Guides. • Comparison to UK expectations for relevant good practice. Probabilistic Safety Analysis • *[SAP] Safety Assessment Principles for Nuclear Facilities*. 2006 Edition Revision 1. HSE. January 2008. www.onr.org.uk/saps/saps2006.pdf. *[TAST/30] Nuclear Safety Technical Assessment Guide.* NS-TAST-GD-030 Revision 5, June 2016. www.onr.org.uk/operational/tech_asst_guides/ns-tast-gd-030.pdf Structural Integrity) • ONR Safety Assessment Principles. • Safety Classification and Standards – ECS.1 to ECS.3. • Integrity of Metal Components and Structures – EMC.1 to EMC.34, with EMC.1 to EMC.3 specifically applicable to the highest reliability components.

<table>
<tr>
<td>Skill Sets Required by (Education)
Senior (regulator)
Junior (regulator)
TSO</td>
<td><u>Electrical Engineering</u>
Regulator:
University degree in electrical engineering and proven knowledge of nuclear safety.
Knowledge of electrical safety.
<u>Radiological Protection</u>
ONR training requirements for ONR Principal Inspector / Inspector of Nuclear Safety.
Technical Support Contractor – Experienced senior consultant.
<u>Human Factors</u>
HF & HRA expertise.
Safety case & ALARP knowledge.
<u>Mechanical Engineering</u>
Senior: Mechanical Engineer.
Junior: Mechanical Engineer.
<u>Probabilistic Safety Assessment</u>
ONR training requirements for ONR Principal Inspector/Inspector of Nuclear Safety.
Technical Support Organisation – Experienced senior consultant.
<u>SI</u>
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.</td>
</tr>
<tr>
<td>Specialised Training, Experience and/or Education Needed for the Review of this topic</td>
<td><u>Electrical Engineering</u>
Knowledge of nuclear plants and safety systems.
Knowledge of electrical system analysis.
Knowledge of equipment standards for electrical equipment.
<u>Radiological Protection</u>
Experience needed:
Radiological protection.
Shielding.
<u>Mechanical Engineering</u>
Knowledge of UK regulatory regime and processes.
Understanding of safety requirements for nuclear equipment and facilities.

<u>Probabilistic Risk Assessment</u>
Adequate knowledge of nuclear engineering and experience in PRA analyses.

<u>SI</u>
Understanding of nuclear classification and categorisation principles.
Understanding of the structural integrity safety principles related to the highest reliability components.
Understanding of nuclear pressure vessel classification principles.</td>
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</table>

Level of Effort in Each Review Area	<u>Electrical Engineering</u> • Included in overall electrical assessment. <u>Radiological Protection</u> • ONR – Approx. 2 person-months. • Technical Support Contractor – Approx. 7 person-months. <u>HF</u> Extensive TSC support – total effort for all HF assessments for UK EPR at Step 4 around 100 person weeks. This covers all the individual topics listed in this survey. <u>Mechanical Engineering</u> ONR- 200 hrs.

ONR Table 1: Safety Assessment Principles and Technical Assessment Guides used as an Assessment Basis for GDA Step 4 HF Assessments

Work Stream	Relevant HF SAP applied	Relevant non-HF SAP applied	Relevant TAG applied
Work Stream 1 – Substantiation of human based safety actions	EHF.2 EHF.3 EHF.4 EHF.5 EHF.6 EHF.10	SC.4 SC.6 EKP.1 EKP.2 EKP.3 EKP.4 EKP.5 ESS.9 FA.7 NT.2	T/AST/005 – ND Guidance on the demonstration of ALARP (Ref. 7). T/AST/051 – Guidance on the purpose, scope and content of Nuclear Safety Cases (Ref. 7). T/AST/063 – Human Reliability Analysis (Ref. 7).
Work Stream 2 – Generic Human Reliability Assessment	EHF.5 EHF.7 EHF.10	SC.5 ERL.1 FA.13	T/AST/063 – Human Reliability Analysis (Ref. 7).
Work Stream 3 – Engineering systems	EHF.1 EHF.2 EHF.3 EHF.6 EHF.7 EHF.10	ECS.3 ECS.5 ERL.2 EMT.1 EMT.4 EMT.6 ELO.1 EMC.8 ESS.15 ESS.26	T/AST/009 – Maintenance, inspection and testing of safety systems, safety-related structures and components (Ref. 7). T/AST/058 – Human Factors Integration (Ref. 7). T/AST/059 – Human Machine Interface (Ref. 7).
Work Stream 4 – Human Factors Integration	EHF.1 EHF.2 EHF.3 EHF.4 EHF.5 EHF.6 EHF.7 EHF.8 EHF.9 EHF.10	MS.4 SC.4 SC.7	T/AST/005 – ND Guidance on the demonstration of ALARP (Ref. 7). T/AST/058 – Human Factors Integration (Ref. 7).

ONR Table 1: Safety Assessment Principles and Technical Assessment Guides used as an Assessment Basis for GDA Step 4 HF Assessments

Work Stream	Relevant HF SAP applied	Relevant non-HF SAP applied	Relevant TAG applied
Work Stream 5 – Plant-wide generic Human Factors assessment	EHF.1 EHF.2 EHF.3 EHF.4 EHF.5 EHF.6 EHF.7 EHF.8 EHF.9 EHF.10	SC.4 EKP.1 EKP.4 ELO.1 ESS.3 ESS.13 ESS.14 ESS.15 ESR.1	T/AST/059 – Human Machine Interface (Ref. 7).

Classification of systems, structures and components	**United States NRC**
Design Information Provided by Applicant	As part of the safety analysis report the applicant should describe the following: • Seismic classification of structures, systems, and components (SSCs). • Quality group classifications of SSCs and relation to ASME Code Section III. • Safety classification of SSCs. • Regulatory Treatment of Non-Safety Systems (RTNSS). • Non-safety related SCCs.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	The Nuclear Regulatory Commission (NRC) staff (1) reviews the information provided in the SAR for compliance with the regulations, (2) issues requests for additional information (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, Standard Review Plan (SRP). The sections of the SRP that are applicable to this area are as follows: • SRP 3.2.1, "Seismic Classification". • SRP 3.2.2, "System Quality Group Classification". • SRP 19.3, "Regulatory Treatment of Non-Safety Systems for Passive Advanced Light Water Reactors". 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 audits of design specifications to verify that information related to classification is correctly translated from the SAR into the design specifications.
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 50.55a, as it relates to compliance with published Codes and Standards. 2. 10 CFR Part 50, Appendix A, Generic Design Criteria (GDC) 1, "Quality Standards and Records". 3. 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena". 4. 10 CFR Part 50, Appendix A, GDC 61, "Fuel Storage and Handing and Radioactivity Control". 5. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants". 6. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants". 7. 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants". The NRC guidance documents that provide an acceptable approach to meeting the applicable regulatory requirements are listed as follows: • Regulatory Guide (RG) 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants". • RG 1.29, "Seismic Design Classification".

	• RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants. • RG 1.151, "Instrument Sensing Lines". • RG 1.189, "Fire Protection for Nuclear Power Plants". • RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)". 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: • ASME Boiler and Pressure Vessel Code, Section III.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Mechanical Engineering. • Structural Engineering (for structural portions of seismic classification).
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: • Understanding of plant system interactions and dependencies. • Knowledge of ASME Code.
Level of Effort in Each Review Area	1 400 hours

APPENDIX B: PLANT DESIGN FOR PROTECTION AGAINST POSTULATED PIPING RUPTURE

Summary Table:

Country	Is this area reviewed?	Are Confirmatory Analyses Performed?	Expertise of Reviewers	Level of Effort
Canada	Yes	Yes	Engineering or scientific degree and work experience in, Mechanical Engineering or Plant Systems Engineering, and knowledge of the aircraft impact assessment responsibilities and processes	0.5FTE (112.5 working days)
Finland	Yes	Yes	Experience with classification principles and knowledge on containment design	10 working days
France	Yes	Yes	Mechanical engineering, Plant systems engineering	See Note 1
Japan	Yes	Yes	Civil, structural and mechanical engineers.	See Note 3
Korea	Yes	Yes	Mechanical engineering, materials engineering, nuclear engineer, plant systems engineer	2 400 hours
Slovak Republic	Yes	Yes	Technical engineer	See Note 2
Slovenia	Yes	No	Mechanical engineer	300 hours (Note 4)
United Kingdom	Yes	No		~100 person days (800 hours)
United States	Yes	Yes	Mechanical engineering, plant systems engineering	1 200 hours

Notes:

1. In France, The effort needed to review a new plant design strongly depends on the degree of novelty of this design. It can take one to several years.
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 Japan, resources are not set up for the individual review area.
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.

Canadian Nuclear Safety Commission
High Level Summary for Plant Design against Postulated Piping Rupture

The applicant must implement a pressure boundary programme and hold itself responsible for all aspects of pressure boundary design, registration and inspections. A pressure boundary programme is understood to be comprised of the many programmes, processes and procedures and associated controls that are required to ensure compliance with CSA standard N285.0 (for CANDU). This standard defines the technical requirements for the design, procurement, fabrication, installation, modification, repair, replacement, testing, examination and inspection of pressure-retaining and containment systems, including their components and supports.

The CNSC expects that all pressure-retaining SSCs will be protected against overpressure conditions, and shall be classified, designed, fabricated, erected, inspected, and tested in accordance with established standards. All pressure-retaining SSCs of the reactor coolant system and auxiliaries shall be designed with an appropriate safety margin to ensure that the pressure boundary will not be breached, and that fuel design limits will not be exceeded in operational states, or DBA conditions. For Design Extension Conditions (DECs), relief capacity must be sufficient to provide reasonable confidence that pressure boundaries credited in severe accident management will not fail.

The pressure boundary design needs to minimise the likelihood of flaws in pressure boundaries and include timely detection of flaws in pressure boundaries important to safety. All pressure boundary SSCs shall be designed to withstand static and dynamic loads anticipated in operational states, and DBAs. Where two fluid systems operating at different pressures are interconnected, failure of the interconnection must be considered. Both systems must be either designed to withstand the higher pressure, or provision made so that the design pressure of the system operating at the lower pressure will not be exceeded.

Plant design for protection against postulated piping rupture	**Canada** **CNSA**
Design Information Provided by Applicant	As part of an application for a licence to construct [2], the LAG section 5.4 states that an applicant is to describe the basis for the design of the pressure-retaining systems, components and their supports. The information provided in this subsection should meet the expectations of section 7.7 of REGDOC-2.5.2 (RD-337)). The information provided should include general design considerations and an explanation of the assessment methodology used, including the codes and standards employed. The pressure boundary code classification and design of pressure-retaining SSCs should be aligned with safety classification, nationally recognised codes and standards, or with codes and standards accepted by national or international institutions. The application should include a high level description of the pressure boundary design registration process, including proposed Authorised Inspection Agencies, pressure boundary quality assurance processes, identification of major process steps and interfaces with external authorities. The description should include the basis for pressure boundary code classification of such components. It should also include, directly or by reference, other support processes that are an integral part of the design such as: 1. specification and traceability of the materials of construction; 2. requirements for quality assurance; 3. qualifications and certifications of designers; fabricators; authorised inspectors and examination personnel;

	4. the codes and standards to be used for examination and pressure testing; 5. documentation and records; 6. in-service inspection; 7. maintenance and testing of pressure-retaining SSC. The application should include information concerning general design considerations, such as the methodology used to address protection against postulated piping failures for medium- and high-energy systems.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	The Canadian Nuclear Safety Commission (CNSC) staff conducts the following activities: • Reviews the information provided by the applicant for compliance with the CNSC regulations; • Asks for clarifications for issues uncovered during staff review and makes requests for additional information as necessary; • Reviews the additional responses and resolves technical issues with applicants or licensees. The scope of the review includes the following: • Verification that the design layout of all SSCs required to shut down, cool and contain the NPP are separated and as far as practicable, do not lie in the close proximity of high energy systems within or outside containment. • Verification that pressure boundary components are designed with sufficient margins, have proper barriers or pipe whip restraints or else have LBB qualification. • Verification that phenomena such as water hammer, creep damage, flow accelerated corrosion and fatigue are ruled out as possible degradation mechanisms. • Verification of a capable leak detection system as part of LBB qualification. In addition, as part of the application for a licence to construct, the applicant is expected to provide detailed design documentation regarding design of reactor shut down/control system, nuclear heat removal system and the containment system and how they are protected from high energy line break effects such as pipe whip, fluid jet impingement and environmental effects such flooding and spraying both inside or outside the containment. The submission should provide the following information: • SSCs design documentation to include protection against postulated pipe ruptures, unless otherwise justified by barriers or pipe whip restraints. • All pressure-retaining SSCs to be protected against overpressure conditions. • All pressure-retaining SSCs of the reactor coolant system and auxiliaries to be designed with an appropriate safety margin to ensure that the pressure boundary will not be breached in operational states, or DBA conditions. Must have significant margins against ASME code service limits. • The criteria for determining the location and configuration of postulated breaks and cracks in high- and moderate-energy piping inside and outside of containment. • The methods used to define the jet thrust reaction at the break or crack location and the jet impingement loading on adjacent safety-related SSCs. • LBB analyses in lieu off pipe whip restraints, jet impingement barriers and shields. • Must demonstrate that pressure boundary systems have sufficient protection against fatigue damage/failures. • The design criteria for protection against environmental effects (EQ for elements such as heat, humidity and radiation) of postulated piping failures. • All pressure boundary piping and vessels to be separated from electrical and

	control systems to the greatest extent practicable. A description of how pressure-retaining components whose failure will affect nuclear safety are designed to permit inspection of their pressure boundaries throughout the design life.
What type of confirmatory analysis (if any) is performed?	A technical assessment is completed to verify that the applicant's design activities meet the prescribed guidance and acceptance criteria as stated in CNSC regulatory documents.
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	The applicable CNSC Regulatory Requirements are listed below: • Design of Reactor Facilities: Nuclear Power Plants Regulatory Document REGDOC-2.5.2. The applicable Codes and Standards related to this area are: • American Society of Mechanical Engineers (ASME), ASME Boiler and Pressure Vessel Code, New York, 2010. • CSA Group, N285.0/N285.6 Series, General requirements for pressure-retaining systems and components in CANDU nuclear power plants/Material Standards for reactor components for CANDU nuclear power plants, Toronto, Canada. • CSA Group, N287.3 CANDU concrete containment design rules, Toronto, Canada. The protection of safety important SSCs from rupture of pressure boundary components are governed by ASME code section III design rules, along with Leak-Before-Break analyses and the guidelines for such analyses are specified in US NRC standard review plan NUREG-0800, chapter 3, section 3.6.3.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	Engineering or scientific degree and work experience in: • Mechanical Engineering. • Plant System Engineering (particularly for environmental effects). • Knowledge of the AIA responsibilities and processes as applied to a specific jurisdiction.
Specialised Training, Experience and/or Education Needed for the Review of this topic	All technical reviewers are required to have formal training and knowledge of pressure boundary design principles based on ASME code, section III and rules of fracture mechanics prior to performing safety reviews independently. Other specialised training, experience, and education that is needed to successfully perform reviews in this technical area include: • Background in pipe fracture analysis and understanding of associated dynamic effects. • ASME Code Section III knowledge.
Level of Effort in Each Review Area	0.5 FTE (225 days per FTE)

Plant design for protection against postulated piping rupture	**Finland STUK**
Design Information Provided by Applicant	Requirements for structural integrity and ultimate capacity of containment have been approved in preliminary safety assessment report.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	Classification of containment has been reviewed in accordance with corresponding requirements for structural integrity and ultimate capacity requirements.
What type of confirmatory analysis (if any) is performed?	Best practice of structural design of containment.
Technical basis • Standards • Codes • Acceptance criteria (e.g., can come from Accident analysis, regulatory guidance)	Containment: ASME Code, Section III. Div. 2, subsection CC (ACI 359), concrete containments.
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	Experience with classification principles and knowledge on containment design.
Level of Effort in Each Review Area	10 working days.

<table>
<tr><th>Plant design for protection against postulated piping rupture</th><th>France
ASN</th></tr>
<tr><td>Design Information Provided by Applicant</td><td>Safety objectives:
The applicant have to show that a piping ruptures do not cause damage to :
the integrity of the main primary system,
the reactor shutdown and the evacuation of the residual power,
the prevention and mitigation of radioactive releases to an acceptable level.
Method:
For all the facilities containing high and moderate energy fluid system, the applicant postulates breaks.
The applicant justifies :
the type of breaks (guillotine, longitudinal breaks or craking),
the localisation of the breaks,
the method for determining the effect of blowdown jets and reactive forces and pipe whip effects.
For all the postulated breaks, the applicant studies the gravity of the accident :
internal consequences, i.e. the partial or total loss of flow or pressure of the fluid in the pipes or reservoirs,
external consequences, i.e.
the thermohydraulic consequences of the fluid (liquid or gaseous) to the materials (contaminated fluid, increase of the temperature, the pressure, the humidity, etc.),
the mechanical consequences induced by blowdown jets and reactive forces and pipe whip effects,
the consequences of the induced internal flooding (sprinkling or immersion of materials).
The objective in to demonstrate the absence of common mode failure. Where it is impossible, the applicant proposes :
organisational measures to ensure the safety (delete the common mode) using penalising hypothesis,
material measures to ensure the safety (delete the common mode): qualification of material to environmental conditions (pressure, temperature, humidity, etc.), geographical localisation of material important for the safety (separation by distance or orientation), physical separation (by concrete wall) or adding of protection materials like anti-whipping support frames, shock absorber, etc.</td></tr>
<tr><td>Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review</td><td>IRSN (TSO) and ASN review all the step of the studies provided by the applicant: type and localisation of breaks, internal and external consequences. At the end, ASN has to give its position on the pertinence and the sufficiency of the organisational and material measures proposed by the applicant.</td></tr>
<tr><td>What type of confirmatory analysis (if any) is performed?</td><td>Inspections can be performed to ensure :
that structures, systems and components important to safety be designed to accommodate the environmental and dynamic effects of postulating piping failure,</td></tr>
</table>

	• the relevance of organisational measures (field tests in the control room and in the facilities).
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	• Technical guidelines for design and construction of the next generation of NPP; • 10 CFR 50 – Appendice A – Critère général de conception : Critère n° 4 : Bases de conception relatives à l'environnement et aux projectiles; • Regulatory Guide: RG 1.46 : Protection contre les fouettements de tuyauteries; • Standard Review Plan N° 3.6.1 et 3.6.2. DRAFT regulatory guide "Reactor (PWR) design"
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Mechanical engineering, • Plant system engineering.
Specialised Training, Experience and/or Education Needed for the Review of this topic	Specialised training and experience to understand the plant systems interactions and dependences. The staff performing the technical review at IRSN (TSO) has a long experience (more than 10 years) on this topic.
Level of Effort in Each Review Area	The effort needed to review a new plant design strongly depends on the degree of novelty of this design. It can take one to several years.

Plant design for protection against postulated piping rupture	**Japan** **NRA**
Design Information Provided by Applicant	In the establishment permit application stage, the following information is provided in the description regarding the safety design of nuclear reactor facility: • Importance classification of safety functions, • Seismic design, • Design policy.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	Identical with scope provided in Subsection of "Classification of SSCs".
What type of confirmatory analysis (if any) is performed?	In the establishment permit stage, adequacy of an applicant's analytic method and the analysis results are verified. Independent evaluation is also performed to demonstrate the analysis results, if needed, (cross check analysis).
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	• The NRA Ordinance on Standards for the Location, Structure and equipment of Commercial Power Reactors. • The NRA Ordinance on Technical Standards for Commercial Power Reactors Facilities. • The Regulatory Guide of the NRA Ordinance on Standards for the Location, Structure, and Equipment of Commercial Power Reactors. • The Regulatory Guide of NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Senior: Director for Nuclear Safety Examination. • Junior: Nuclear Safety examiner. • TSO: None since Japan Nuclear Energy Safety Organisation (JNES) was integrated into NRA as of April, 2014.
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) is not set up for the individual review area. Regarding the standard processing duration, 2 years is set up for the basic design of an entire plant, and 3 months per one application is set up for detailed design. Divided application is granted for the detailed design.

Plant design for protection against postulated piping rupture	**Korea** **NSSC and KINS**
Design Information Provided by Applicant	As part of the SAR, the applicant should describe or provide the following related to the Plant Design for Protection against Postulated Piping: • High energy piping system. • Moderate energy piping system. • Essential Equipment and Component Description. • Protection methodology for postulated piping rupture. • Classification of postulated piping rupture. • Determination of Rupture Locations associated with postulated rupture of piping. • Analytical methods for calculating of jet thrust reaction at the postulated pipe break. • Dynamic analysis methodology for piping systems. • Leak before break (LBB) analysis results.
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(request for additional information) 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 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment". • SRG 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping". • SRG 3.6.3, "Leak-Before-Break Evaluation Procedures". • Appendix SRG 3.6.1-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment". • Appendix SRG 3.6.2-1, "Determination of Locations Associated with the Postulated Piping Rupture in Fluid Systems Inside and Outside Containment". • Appendix SRG 3.6.3-1, "Procedure of Dynamic Fracture Test".
What type of confirmatory analysis (if any) is performed?	KINS staff performs the confirmative analysis by using the regulatory computer programme for safety analysis to verify the results of LBB 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 13 (External Events Design Bases). • Article 15 (Environmental Effects Design bases, etc.). **<Korea Institute of Nuclear Safety(KINS) Regulatory Guides>** The applicable Regulatory Guides: • KINS/RS-N04.06, "Protection of Postulated piping ruptures". • KINS/RG-N04.14, "Plant Design for Protection Against Postulated Piping Failures • in Fluid Systems Outside Containment". • KINS/RG-N04.15, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping". • KINS/RG-N04.02, "Leak-Before-Break Evaluation Procedures".

	The applicable regulatory technical reports related to this area are: • NUREG-1061, Vol.3, “Evaluation of Potential for pipe break”. • NUREG/CR-4575, “Predictions of J-R curves with large crack growth from small specimen data”. **<Applicable Codes and Standards>** The applicable Codes and Standards related to this area are: • KEPIC MN (Nuclear- Mechanical). • KEPIC MI (Inservice Inspection).
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Mechanical Engineer. • Materials Engineer. • Nuclear Engineer. • Plant System 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. • Knowledge of reactor coolant system design. • Knowledge of material for reactor coolants system. • Knowledge of LBB analysis and fracture mechanics. • Knowledge of pressurised water reactor designs, systems and operation. • Understanding Codes and Standards (KEPIC, ASME, etc.).
Level of Effort in Each Review Area	Total : 2 400 hours • Review of Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment: 800hours. • Review of Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping: 800hours. • Review of adequacy on LBB analysis results: 800 hours.

Plant design for protection against postulated piping rupture	**Slovak Republic** **UJD**
Design Information Provided by Applicant	• Analyses of the responses of the nuclear facility at small, medium and large leaks of primary circuit coolant due to a burst in the main circulation piping, a burst in the main steam piping and feed water piping. • Deterministic or probabilistic safety analyses, which prove, that the sensitivity of the project's design to a postulated trigger event is minimised. • Demonstrate, that the suitable preventive and alleviative measures for potential flooding, pipe swing, influence of media flow or leakage of liquids from damaged systems, assemblies, and components or other facilities in a nuclear facility are exist. • Identification of the safety systems, components and structures needed for fulfilment of the based safety functions. • Results of performed strength calculation.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	Review of the submitted documentation, if it conforms to atomic act and regulations. Evaluate if the systems, structures and components are in compliance with all requirements arising from applicable regulations, codes and standards. Confirm that the systems, structures and components: • are able to manage their roles under normal, transient and accident conditions; • that the facilities have been properly classified to identify their importance to safety.
What type of confirmatory analysis (if any) is performed?	
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	Regulatory body regulations, Slovak Technical Standards, 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 nuclear technology and analysis. 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 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.

Plant design for protection against postulated piping rupture	**Slovenia** **SNSA**
Design Information Provided by Applicant	• Determination of piping rupture locations. • Pipe break analyses and analyses of the dynamic effects associated with the postulated rupture of piping. • Plant procedures and preventive measures in case of postulated piping rupture.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	• Review of application material + TSO' independent evaluation report. • Determine adequacy of rupture locations. • Confirm that there is appropriate protection of SSCs components for safe reactor. • Shutdown or mitigation of the consequences of a postulated pipe rupture.
What type of confirmatory analysis (if any) is performed?	
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	• Rules on Radiation and Nuclear Safety Factors. • SNSA Practical Guidelines. • IAEA Safety Standards.
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	Background in pipe break analysis and understanding of dynamic effects associated.
Level of Effort in Each Review Area	Regulator: 100 hrs TSO' review time: 200 hrs

<table>
<tr><th>Plant design for protection against postulated piping rupture</th><th>United Kingdom
ONR</th></tr>
<tr><td>Design Information Provided by Applicant</td><td><u>Human Factors (HF)</u>
Identification of operator actions required to respond to pipe rupture (e.g. LOCA) – via PSA.
Human reliability analysis (HRA) for safety significant operator actions.
Detailed qualitative substantiation (by task analysis) of operator actions.
<u>SI</u>
Pre-Construction Safety Report describing the claims placed on pipework integrity and the direct and indirect consequences of pipework failure.</td></tr>
<tr><td>Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review</td><td>Assessment of HRA.
Assessment of qualitative substantiations of operator actions.
<u>SI Specific Review</u>:
Identification of pipework where a highest reliability claim needs to be established as the consequences of gross failure has been discounted.
Provision of challenge to arguments that moderate energy pipework will only leak and not rupture.
See SI Step 4 Report Sections 4.1, 4.9, 4.12.</td></tr>
<tr><td>What type of confirmatory analysis (if any) is performed?</td><td>None</td></tr>
<tr><td>Technical basis
Standards
Codes
Acceptance criteria
(e.g. can come from Accident analysis, regulatory guidance)</td><td>ONR Safety Assessment Principles:
Relevant Safety Assessment Principles & Technical Assessment Guides (see Table 1) for Human Factors.
Integrity of Metal Components and Structures – EMC.1 to EMC.34, with EMC.1 to EMC.3 specifically applicable to the highest reliability components.</td></tr>
<tr><td>Skill Sets Required by (Education)
Senior (regulator)
Junior (regulator)
TSO</td><td>Senior regulator & Junior Regulator – 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.
Technical Supp – Expertise required in relation to the topic under consideration, but no specific level required.</td></tr>
</table>

Specialised Training, Experience and/or Education Needed for the Review of this topic	• HF & HRA expertise. • Safety case & ALARP knowledge. • Understanding of the structural integrity safety principles and in particular the need to ensure that the consequences of gross piping failure are not discounted unless the pipework is shown to be in the highest reliability category.
Level of Effort in Each Review Area	Very considerable – around 100 person days. No TSO support required.

Plant design for protection against postulated piping rupture	**United States** **NRC**
Design Information Provided by Applicant	As part of the safety analysis report (SAR) the applicant should describe the design bases and design measures used to ensure that the containment vessel and all essential equipment inside or outside the containment, including components of the reactor coolant pressure boundary, have been adequately protected against environmental effects (flooding, spray wetting, and other adverse environmental effects) and dynamic effects (e.g. blowdown jet and reactive forces and pipe whip effects) resulting from postulated rupture of piping located either inside or outside of containment. This description should include the following: • The plant design for protection against high- and moderate-energy fluid system piping failures outside containment. • The criteria for determining the location and configuration of postulated breaks and cracks in high- and moderate-energy piping inside and outside of containment. • The methods used to define the jet thrust reaction at the break or crack location and the jet impingement loading on adjacent safety-related structures, systems, and components. • The design criteria for pipe whip restraints, jet impingement barriers and shields, and guard pipes. • The design criteria for protection against environmental effects of postulated piping failures. • The pipe break hazards analysis report, which describes and summarises the results of the plant design for protection against postulated piping failures. • The analyses used to eliminate from the design basis the dynamic effects of certain pipe ruptures and demonstrate that the probability of pipe rupture is extremely low under conditions consistent with the design basis for the piping.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	The Nuclear Regulatory Commission (NRC) staff (1) reviews the information provided in the SAR for compliance with the regulations, (2) issues requests for additional information (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, Standard Review Plan (SRP). The sections of the SRP that are applicable to this area are as follows: • SRP 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment". • BTP 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment". • SRP 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping". • BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment". • SRP 3.6.3, "Leak-Before-Break Evaluation Procedures". 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 a review or inspection (depending on whether "design acceptance criteria" are used in the initial design certification) of the pipe break hazards analysis report to ensure that structures, systems and components important to safety be designed

	to accommodate the environmental and dynamic effects of postulated piping failures.
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) 2, "Design Bases for Protection Against Natural Phenomena" 2. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases"
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Mechanical Engineering. • Plant System Engineering (particularly for environmental effects).
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, or education that is needed to successfully perform reviews in this technical area includes a background in pipe break analysis and understanding of dynamic effects associated.
Level of Effort in Each Review Area	1 200 hours

APPENDIX C: SEISMIC AND DYNAMIC QUALIFICATION OF SAFETY RELATED MECHANICAL AND ELECTRICAL EQUIPMENT

Summary Table:

Country	Is this Area Reviewed?	Are Confirmatory Analyses Performed?	Expertise of Reviewers	Level of Effort
Canada	Yes	Yes	Mechanical Engineering, Structural Engineering	0.7 FTE (157.5 working days)
Finland	Yes	Yes	Experience with classification principles and knowledge on seismic qualification and safety design	70 working days
France	Yes	Yes	Engineering	See Note 1
Japan	Yes	Yes	Civil, structural and mechanical engineers	See Note 4
Korea	Yes	No	Mechanical engineer, seismic engineer	650 hours
Slovak Republic	Yes	No	Technical Engineer	See Note 2
Slovenia	Yes	No	Mechanical engineering, electrical engineering	600 hours (Note 5)
United Kingdom	Yes	No	Knowledge of equipment and systems	
United States	Yes	Yes	Mechanical engineering, structural engineering	1 200 hours

Notes:

1. In France, the effort needed to review a new plant design strongly depends on the degree of novelty of this design. It can take from one to two years.
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. For the UK, the level of effort for this technical topic is included in the hours provided for the overall electrical assessment.
4. In Japan, resources are not set up for the individual review area.
5. 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.

Canadian Nuclear Safety Commission
High Level Summary for Seismic and Dynamic Qualification of Safety Related Mechanical and Electrical Equipment

Seismic and dynamic qualifications are considered to be part of equipment qualification. Seismic qualification (SQ) ensures that all seismically credited SSCs important to safety in a Nuclear Power Plant are designed, installed and maintained to perform their safety function during and/or after (as needed and pre-defined) a design basis earthquake or site design earthquake and also ensures an adequate margin against review level earthquakes. These credited SSCs must be available during and after an earthquake to ensure the reactor can be safely shutdown indefinitely, decay heat can be removed and containment remains functional.

The Canadian approach requires that SSCs important to safety meet more restrictive design requirements than those imposed by the National Building Code of Canada (NBCC). The CSA standard imposes the design-basis earthquake (DBE) as the level of earthquake safety to be considered. All other non-nuclear SSCs of an NPP must be designed to the NBCC.

Seismic and dynamic Qualification of Safety Related Mechanical and Electrical Equipment	**Canada CNSC**
Design Information Provided by Applicant	As part of an application for a licence to construct [2], the LAG section 5.6.3 states that an applicant is to describe how the plant design protects SSC (including building structures) from earthquake damage. It should also demonstrate how the approach followed meets the expectations of section 7.13 of REGDOC-2.5.2 (RD-337). The description should explain the seismic design and qualification of SSCs and the seismic qualification of equipment, and refer to the applicable national (such as CSA) and international (such as IAEA) codes and standards that have been used. The seismic qualification programme should take into account considerations such as: 1. seismic input, which includes the design response spectra, design time history, selection and determination of design basis ground motion and critical damping values; 2. for seismic qualification by testing, the test equipment requirements, test input response spectra and acceptance criteria should be included; 3. seismic analysis for building structures, taking into account the seismic analysis method, procedure used for modelling, soil-structure interaction, development of floor response spectra, and combination of modal responses; 4. seismic analysis methodology for sub-systems, including structures and components that do not have an interface with the soil structure interaction analyses; 5. seismic qualification of equipment in order to demonstrate its capability to perform designated safety functions during a design basis seismic event. The applicant should also describe seismic instrumentation systems necessary to determine and record site-specific seismic responses. Certain SSCs and equipment may be credited to mitigate the consequences or to monitor the conditions following a beyond design-basis earthquake (BDBE). The ability of the credited equipment to operate in the BDBE environment should be assessed to a reasonable degree of confidence The equipment credited for mitigation of the consequences of BDBEs and for accident management is not required to be qualified to the same level of confidence as that for DBEs.

	As part of the nuclear plant in Canada, the applicant should describe the following: • Identification of all safety related mechanical and electrical components along with their seismic classification/categorisation. • All instrumentation, electrical equipment, and mechanical components (other than pipes), including their supports, that should be designed to withstand the effects of earthquakes and the full range of normal and accident loadings. • The acceptance criteria used for seismic analysis and testing. • The methods and procedures used to ensure the structural integrity and functionality of mechanical and electrical equipment for operation in the event of a design-basis earthquake. • The results of tests and analyses that demonstrate adequate seismic qualification.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	The Canadian Nuclear Safety Commission (CNSC) staff conducts the following activities: • reviews the information provided by the applicant for compliance with the CNSC regulations; • asks for clarifications for issues uncovered during staff review and makes requests for additional information as necessary; • reviews the additional responses and resolves technical issues with applicants or licensees. The scope of the review consists of the following: • Verification that all SSCs required to shut down, cool and contain the NPP are seismically qualified. • Verification that non- seismically qualified SSCs do not cause structural/functional failure of seismically qualified SSCs under DBE conditions. • Verification that seismic fragility levels are evaluated for SSCs important to safety by analysis or, where possible, by testing. • Verification that a beyond design-basis earthquake (BDBE) is identified that meets the requirements for identification of Design Extension Condition (DEC) as described in section 7.3.4 of REGDOC-2.5.2. Verify SSCs credited to function during and after a BDBE shall be demonstrated to be capable of performing their intended function under the expected conditions by calculating a high confidence of low probability of failure (HCLPF) under BDBE conditions for these SSCs. The acceptance criterion for BDBE should demonstrate that the plant HCLPF is at least 1.67 times the DBE.
What type of confirmatory analysis (if any) is performed?	The staff verifies that the applicant's design activities (dynamic analyses and equipment seismic qualification test results) meet the prescribed acceptance criteria as stated in CNSC regulatory documents.
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	The applicable CNSC Regulatory Requirements are listed below: 1. Design of Reactor Facilities: Nuclear Power Plants Regulatory Document REGDOC-2.5.2, 2014, clause 7.13 The applicable Codes and Standards related to this area are: • CSA Group, N289 series on seismic design and qualification of nuclear power plants. • CSA Group, N291, Requirements for Safety-Related Structures for CANDU Nuclear Power Plants, Toronto, Canada. • IEEE, 344, IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, Piscataway, New Jersey, 2004.

	• The acceptance criteria for seismic design by analysis shall meet ASME service level "C" stress limits for DBE loading. For electrical components, for functionality testing during the DBE, the test response spectrum (TRS) shall envelope the owner specified required response spectrum (RRS).
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	Engineering or scientific degree and work experience in the seismic field. • Mechanical Engineering. • Structural Engineering.
Specialised Training, Experience and/or Education Needed for the Review of this topic	All technical reviewers are required to have formal training and knowledge of seismic design prior to performing safety reviews independently. Other specialised training, experience, and education that is needed to successfully perform reviews in this technical area include: • Background in the dynamic qualification of equipment. Background in seismic analysis of components
Level of Effort in Each Review Area	0.7 FTE (225 days per FTE)

Seismic and dynamic Qualification of Safety Related Mechanical and Electrical Equipment	**Finland STUK**
Design Information Provided by Applicant	• Safety classification for SSE is based on safe shut down requirements of the plant. Seismic hazard analysis is the basis in supporting PSA for fine tuning the safety classification. • Reported qualification procedure combined from dynamic analyses and testing against external vibrations. • Structural design principles.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	• Common mode failure studies of SSCs for the assessment of safety classification SSE. • Inspection of qualification procedure and review of corresponding standards. • Inspections at testing facilities. • Specialist statements from a Technical Support Organisation.
What type of confirmatory analysis (if any) is performed?	Seismic PSA and fragility studies
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	• IAEA, • KTA • YVL 2.6 • YVL 2.8 • IEC/IEEE
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	No official requirements.

Specialised Training, Experience and/or Education Needed for the Review of this topic	Experience with classification principles and knowledge on seismic qualification and safety design.
Level of Effort in Each Review Area	Regulator's review: 50 working days. TSO's statements: 20 working days.

0Seismic and dynamic Qualification of Safety Related Mechanical and Electrical Equipment	**France** **ASN**
Design Information Provided by Applicant	The applicant provides the following information in the safety analysis report: • Seismic qualification approach, including the full range of normal and accidents loadings. Seismic qualification includes both function and reliability; • List of structures and components (mechanical, electrical and instrumentation control) that should be designed to withstand the effect of earthquakes. Information required as support of the SAR: • Methods used to ensure de structural integrity and functionality of mechanical and electrical equipment in case of earthquake; • The results of test and analysis; • The criteria used for seismic analysis.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	A comprehensive review of the safety file (SAR and support documents) provided by the applicant is performed by the TSO. The following aspects are regarded in detail: • Adequacy of the safety criteria established. • Results of the test and analyses and compliance with the safety criteria. • Considerations can also be given to experience feedback.
What type of confirmatory analysis (if any) is performed?	The ASN staff could perform inspections to verify that the safety requirements related to seismic qualification described in the SAR are correctly take into account during in design, construction, procurement and assembly specifications. ASN and TSO staff could participate in seismic qualification tests of SCCs.
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	The applicable requirements on this topic is: • Section B.2.2.1 “qualification of equipment” of the technical guidelines for design and construction of the next generation of NPP. • ASN guide 2/ 01 concerning the construction of earthquake-resistant civil works. • ASN guide concerning PWR design- DRAFT. The applicable code related to this topic is: • Basic safety rule 2001-01 of 31st may 2001 – determination of the seismic risk of the surface basic nuclear installations. Acceptance criteria are defined by the applicant and reviewed by TSO.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator)	Engineering

• TSO	
Specialised Training, Experience and/or Education Needed for the Review of this topic	Specialised training and experience to understand the plant systems interactions and dependences. The staff performing the technical review at IRSN (TSO) has a long experience (more than 10 years) on this topic.
Level of Effort in Each Review Area	The effort needed to review a new plant design strongly depends on the degree of novelty of this design. It can take from one to two years.

Seismic and dynamic Qualification of Safety Related Mechanical and Electrical Equipment	**Japan** **NRA**
Design Information Provided by Applicant	In the establishment permit application stage, the following information is provided in the description regarding the safety design of nuclear reactor facility: • Importance classification of safety functions, • Seismic design, and • Design policy.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	Identical with scope provided in Subsection of “Classification of SSCs”.
What type of confirmatory analysis (if any) is performed?	In the establishment permit stage, adequacy of an applicant's analytic method and the analysis results are verified. Independent evaluation is also performed to demonstrate the analysis results, if needed, (cross check analysis).
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	• The NRA Ordinance on Standards for the Location, Structure and equipment of Commercial Power Reactors. • The NRA Ordinance on Technical Standards for Commercial Power Reactors Facilities. • The Regulatory Guide of the NRA Ordinance on Standards for the Location, Structure, and Equipment of Commercial Power Reactors. • The Regulatory Guide of NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Senior: Director for Nuclear Safety Examination. • Junior: Nuclear Safety examiner. • TSO: None since Japan Nuclear Energy Safety Organisation (JNES) was integrated into NRA as of April, 2014.
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) is not set up for the individual review area. Regarding the standard processing duration, 2 years is set up for the basic design of an entire plant, and 3 months per one application is set up for detailed design. Divided application is granted for the detailed design.

Seismic and dynamic Qualification of Safety Related Mechanical and Electrical Equipment	**Korea** **NSSC and KINS**
Design Information Provided by Applicant	As part of the SAR, the applicant should describe or provide the following related to the seismic and dynamic qualification of safety related mechanical and electrical equipment: • Seismic Qualification Criteria. • Seismic and Dynamic Qualification for Electrical Equipment and Instrumentation: - Methods and Procedures for Qualifying Seismic Category I and II: Electrical Equipment and Instrumentation. - Methods and Procedures of Analysis or Testing of Supports of Electrical: Equipment and Instrumentation. • Seismic and Dynamic Qualification for Mechanical Equipment Including Motors: - Methods and Procedures for Qualifying Seismic Category I Mechanical: Equipment Including Motors. - Design Adequacy for Supports. - Qualification of Seismic Category II Mechanical Equipment. • Seismic Qualification Records of Mechanical and Electrical Equipment.
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 (request for additional information) 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 3.10, "Seismic and Dynamic Qualification for Mechanical and Electrical Equipment." • Appendix SRG 3.10-1, "Review Guide of Seismic Qualification Report."
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 12 (Safety Classes and Standards); - Article 13 (External Event Design Bases); - Article 15 (Environmental Effects Design Bases, etc.); - Article 21 (Reactor Coolant Pressure Boundary); - Article 70 (Design Control). **<Korea Institute of Nuclear Safety (KINS) Regulatory Guides>** KINS Regulatory Standard and Regulatory Guide • KINS/RS-N03.00 3.5 Equipment Qualification. • KINS/RG-N03.02 Seismic Qualification of Mechanical and Electric Equipment. **<Applicable Codes and Standards>** The applicable Codes and Standards related to this area are: • KEPIC END 1100 (Qualifying Class 1E Equipment for Nuclear Power Generating Stations).

	• KEPIC END 2000 (Seismic Qualification of Class 1E Equipment for Nuclear Power Generation Stations). • KEPIC MF (Qualification of Active Mechanical Equipment Used in Nuclear Facilities).
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Mechanical Engineer. • Seismic Engineer.
Specialised Training, Experience and/or Education Needed for the Review of this topic	• Experience with performing seismic, dynamic, and vibratory analyses of mechanical equipment and supporting system design. • Understanding Codes and Standards (KEPIC, ASME, etc.). • Knowledge of mechanical integrity evaluations.
Level of Effort in Each Review Area	Total : 650 hours • Seismic Qualification Criteria review: 50 hours • Seismic Qualification method review: 400 hours • Seismic Qualification Records review: 200 hours

Seismic and dynamic Qualification of Safety Related Mechanical and Electrical Equipment	**Slovak Republic** **UJD**
Design Information Provided by Applicant	• Requirements for facilities qualification. • Calculations and calculation results to prove the resistance of selected facilities to seismic activity. • Analysis which demonstrate that selected facilities are designed so, that during earthquakes, it is possible to: a) Safely shut down the nuclear facility and maintain it in a subcritical state, b) Remove residual heat from spent nuclear fuel or radioactive waste, c) Maintain leaks of radioactive substances below specified levels. • The design must also take into account: a) A)maximum expected acceleration given for the site's location, based on an assessment of the location's seismic loading performed during the siting of the nuclear facility, specified as seismic level 1 and seismic level 2; b) Requirements for earthquake-resistant nuclear facility systems, components and structures or parts thereof that must correspond to their safety function and presumed effects of an earthquake according to specified seismic level 1 and seismic level 2. • Seismic adequate evaluation of mechanical and electrical equipment. • Seismic categorisation of mechanical and electrical equipment into two seismic class.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	Review of the submitted documentation, if it conforms to atomic act and regulations. Evaluate if the systems, structures and components are in compliance with all requirements arising from applicable regulations, codes and standards. Confirm that the systems, structures and components: • Are able to manage their roles in case of seismic event; • That the facilities have been properly classified to identify their importance to safety.
What type of confirmatory analysis (if any) is performed?	
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	National regulations, IAEA safety guidelines, Slovak Technical standards

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 equipment evaluation 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 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.

Seismic and Dynamic Qualification of Safety Related Mechanical and Electrical Equipment	**Slovenia** **SNSA**
Design Information Provided by Applicant	• Detail information of seismic and dynamic qualification of all equipment important to safety. • Information of a qualification programme for safety-related SSCs to confirm the capability of SSCs to achieve their design functions over the entire design service life. The SSC-qualification programme shall include collection, documentation and maintenance of information to confirm the capability of SSCs to achieve their design functions over the entire design service life. The qualification programme shall consider operating conditions such as vibration, temperature, pressure, water-jet impacts, electromagnetic disturbances, irradiation, moisture, earthquakes and combinations thereof. Operating conditions shall cover normal operating conditions over the entire design service life and conditions of anticipated operating occurrences or accidents. • Information concerning the methods of test and analysis employed to ensure the functionality of mechanical and electrical equipment under the full range of normal and accident loadings (seismic).
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	• Determine the adequacy of the information provided by the applicant and TSO' independent evaluation report. • The review of the analyses and technical report that have been adequately demonstrated that the mechanical and electrical equipment are capable of performing their safety function under significant stresses.
What type of confirmatory analysis (if any) is performed?	
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	• Rules on Radiation and Nuclear Safety Factors. • SNSA Practical Guidelines. • IAEA Safety Standards.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	Senior: mechanical engineer, electrical engineer. Junior: mechanical engineer, electrical engineer. TSO: mechanical engineer, electrical engineer.
Specialised Training, Experience and/or Education Needed for the Review of this topic	Background in the dynamic qualification of equipment.

Level of Effort in Each Review Area	Regulator: 200 hrs, TSO' review time: 400 hrs.

Seismic and Dynamic Qualification of Safety Related Mechanical and Electrical Equipment	**United Kingdom ONR**
Design Information Provided by Applicant	<u>Electrical Engineering</u> • Design and Construction rules for nuclear islands. • Pre-construction safety report. <u>Mechanical Engineering</u> Design information supplied by the requesting party including: • Safety case. • Responses to Technical Queries. • Selected System Design Manuals.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	<u>Electrical Engineering</u> • Review of submitted documentation. <u>Mechanical Engineering</u> • Review of requesting party material. • Determined adequacy of seismic classification for sampled equipment important to safety. • Technical meetings with requesting party.
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)	<u>Electrical Engineering</u> • IEC standards. • ONR Safety Assessment Principles. <u>Mechanical Engineering</u> • Comparison to UK primary legislation and associated regulations. • Comparison to HSE Safety Assessment Principles and Technical Assessment Guides. • Comparison to UK expectations for relevant good practice.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	Regulator: • University degree in electrical engineering and proven knowledge of nuclear safety. • Knowledge of electrical safety.
Specialised Training, Experience and/or Education Needed for the Review of this topic.	• Knowledge of nuclear plants and safety systems. • Knowledge of electrical system analysis. • Knowledge of equipment standards for electrical equipment.

Level of Effort in Each Review Area	Included in overall electrical assessment

<table>
<tr><th>Seismic and Dynamic Qualification of Safety Related Mechanical and Electrical Equipment</th><th>United States
NRC</th></tr>
<tr><td>Design Information Provided by Applicant</td><td>As part of the safety analysis report, the applicant should describe the following:
• All instrumentation, electrical equipment, and mechanical components (other than pipes), including their supports, that should be designed to withstand the effects of earthquakes and the full range of normal and accident loadings.
• The criteria used for seismic analysis and testing.
• The methods and procedures used to ensure the structural integrity and functionality of mechanical and electrical equipment for operation in the event of a Safe Shutdown Earthquake.
• The results of tests and analyses that demonstrate adequate seismic qualification.
• An implementation program, if seismic and qualification testing is incomplete at the time of the combined licence application.</td></tr>
<tr><td>Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review</td><td>The Nuclear Regulatory Commission (NRC) staff (1) reviews the information provided in the SAR for compliance with the regulations, (2) issues requests for additional information (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, Standard Review Plan (SRP). The sections of the SRP that are applicable to this area are as follows:
• SRP 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components".
• SRP 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment".

The staff also considers emerging technical and construction issues, operating experience, and lessons learnt related to this category.</td></tr>
<tr><td>What type of confirmatory analysis (if any) is performed?</td><td>The staff verifies that the applicant's design, procurement, construction, and preoperational activities, meet the prescribed acceptance criteria thru the verification of Inspection, Testing, Analysis, and Acceptance Criteria (ITAAC) (10CFR 52.99).</td></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 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 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".
5. 10 CFR Part 50, Appendix A, GDC 15, "Reactor Coolant System Design".
6. 10 CFR Part 50, Appendix A, GDC 30, "Quality of Reactor Coolant Pressure Boundary".
7. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants".
8. 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

The NRC guidance documents that provide an acceptable approach for satisfying the applicable regulatory requirements are listed as follows:
1. Regulatory Guide (RG) 1.61, "Damping Values for Seismic Design of Nuclear</td></tr>
</table>

	Power Plants”. 2. RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to safety for Nuclear Power Plants”. 3. RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis”. 4. RG 1.100, Revision 2, “Seismic Qualification of Electrical and Mechanical Equipment for Nuclear Power Plants”. 5. RG 1.148, “Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants”. 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: • ANSI/IEEE Std 344-1987. • ASME AG-1, “Code on Nuclear Air and Gas Treatment”.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Mechanical Engineering. • Structural Engineering.
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: • Background in the dynamic qualification of equipment. Background in seismic analysis of components.
Level of Effort in Each Review Area	1 200 hours

APPENDIX D: ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT (E.G. TEMPERATURE, HUMIDITY, RADIATION, PRESSURE)

Summary Table:

Country	Is This Area Reviewed?	Are Confirmatory Analyses Performed?	Expertise of Reviewers	Level of Effort
Canada	Yes	Yes	Engineering or scientific degree and work experience in related area (mechanical, nuclear, electrical, structural and etc.)	0.5FTE (112.5 working days)
Finland	Yes	No	University degree in process and nuclear engineering, adequate working experience in design / research	200 working days
France	Yes	Yes	Engineering	See Note 1
Japan	Yes	Yes	Civil, structural and mechanical engineers	See Note 4
Korea	Yes	No	Mechanical engineer, electric engineer, I&C engineer	1 420 hours
Slovak Republic	Yes	No	Technical Engineer	See Note 2
Slovenia	Yes	No	Mechanical engineering, electrical engineering	800 hours (Note 5)
United Kingdom	Yes	No	Knowledge of the equipment and systems	See Note 3
United States	Yes	Yes	Mechanical engineering, electrical engineering, health physics	500 hours

Notes:

1. In France, the effort needed to review a new plant design strongly depends on the degree of novelty of this design.
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. For the UK, the level of effort for this technical topic is included in the hours provided for the overall electrical assessment.
4. In Japan, resources are not set up for the individual review area.
5. 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.

Canadian Nuclear Safety Commission
High Level Summary for Environmental Qualification of Mechanical and Electrical Equipment

Environmental qualification (EQ) is a process followed by the nuclear industry which will generate and maintain evidence to demonstrate capability of SSCs important to safety to perform designated safety functions on demand under postulated service conditions, when exposed to harsh environment resulting from a design-basis accident (DBA). EQ Programme comprises a set of planned and co-ordinated activities that establishes auditable assurance that the SSCs required to perform safety functions during the plant life will meet or exceed the functional and performance requirements under accident conditions, taking into consideration plant-specific normal and accident environmental and operating conditions and their ageing effects.

The CNSC requires that NPP designers provide assurance that SSCs important to safety are qualified using the following defined methods; qualification by testing, by analysis, by operating experience or with a combined qualification.

Provision for condition monitoring to assess variables that indicate the physical state of the equipment, and its ability to perform its intended function must be included in the design. Environmental monitoring measures environmental stressors, such as temperature, radiation and operational cycling during normal operating conditions.

Environmental Qualification of Mechanical and Electrical Equipment	**Canada CNSC**
Design Information Provided by Applicant	As part of an application for a licence to construct [2], the LAG section 5.6 states that an applicant is to describe the procedure adopted for equipment qualification, and should confirm that the items important for plant safety will meet the design requirements, and will remain fit for purpose when subjected to the range of individual or combined environmental challenges identified throughout the lifetime of the plant. **5.6.1 Environmental qualification** This subsection requires the applicant to describe the environmental qualification programme. It should comprise a set of planned and co-ordinated activities that will ensure that equipment can perform its intended safety functions under the environmental conditions defined for all plant states in which it is credited. Refer to section 5.2.8, Identification of plant states and operational configurations, for identification of plant states. The programme should be verifiable. The information presented here should include a complete list of the equipment (mechanical, electrical, instrumentation and control and post-accident monitoring) required to be environmentally. It should also include the designated functional requirements, the definition of the applicable environmental parameters, and the documentation of the qualification process used to demonstrate that the required equipment is capable of meeting the expectations of appropriate sections REGDOC-2.5.2 (RD-337). A sample of the equipment qualification documentation should be submitted. Certain SSCs and equipment may be credited to mitigate the consequences or to monitor the conditions following a BDBA or a severe accident (DEC). The ability of the credited equipment to operate in the BDBA (DEC) environment should be assessed to a

	reasonable degree of confidence. As per CNSC Regulatory Document REGDOC-2.5.2 [3] the design shall include an equipment environmental qualification (EQ) program. Development and implementation of this programme shall ensure that the safety functions can be carried out. The environmental conditions to be accounted for shall include those expected during normal operation, and those arising from AOOs and DBAs. Operational data and applicable design assist analysis tools, such as the deterministic safety analysis (complemented by probabilistic safety analysis where appropriate), shall be used to determine the envelope of environmental conditions. The equipment qualification programme for SSCs important to safety shall include the consideration of ageing effects due to service life. Equipment qualification shall also include consideration of any unusual environmental conditions that can reasonably be anticipated, and that could arise during normal operation or AOOs (such as periodic testing of the containment leak rate). Equipment and instrumentation credited to operate during Design Extension Conditions (DECs) shall be demonstrated, with reasonable confidence, to be capable of performing their intended safety function(s) under the expected environmental conditions. The designer should identify the EQ-related standards and codes (e.g. CSA, IEEE and ASME). The latest editions of the applicable standards for use in the equipment qualification are preferred; any deviations should be justified. The design should address protective barriers, if applicable. When protective barriers are designed to isolate equipment from possible harsh environmental conditions, the barriers themselves should be addressed in a qualification programme.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	The Canadian Nuclear Safety Commission (CNSC) staff: • review the information provided for compliance with regulations, • issue requests for additional information as necessary, • review applicant's responses, and • resolve technical issues with applicant. The staff also considers emerging technical issues, operating experience, and lessons learnt related to this category.
What type of confirmatory analysis (if any) is performed?	CNSC staff conduct a technical assessment to verify that the applicant's design and design activities meet the requirements as stated in CNSC regulatory documents.

Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	The applicable CNSC Regulatory Requirements related to this area are listed below: • CNSC Regulatory Document REGDOC-2.5.2, Physical Design, Design of Reactor Facilities: Nuclear Power Plants, August 2013, Draft. The applicable Codes and Standards related to this area are listed below: American Society of Mechanical Engineers (ASME): • QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants, New York, 2002. CSA (Canadian Standards Association) • N290.13, Environmental qualification of equipment for CANDU nuclear power plants, February 2005, Reaffirmed 2015. EPRI (Electric Power Research Institute) • Technical Report rev. 1, Nuclear Power Plant Equipment Qualification Reference Manual, Palo Alto, California, 2010. IAEA (International Atomic Energy Agency) • Safety Reports Series No. 3, Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving and Reviewing, Vienna, 1998. IEC (International Electrotechnical Commission) • 60 780 ed 2.0, Nuclear Power Plants — Electrical Equipment of the Safety System — Qualification, Geneva, 1998. IEEE (Institute of Electrical and Electronics Engineers, Inc.) • Standard 323, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations, Piscataway, New Jersey, 2003. • Standard 627, Qualification of Equipment Used in Nuclear Facilities, Piscataway, New Jersey, 2010.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	Engineering or scientific degree and work experience in related area (mechanical, nuclear, electrical, structural and etc.)
Specialised Training, Experience and/or Education Needed for the Review of this topic	Specialised training, experience, and education that is needed to successfully perform reviews in this technical area include: • Knowledge of requirements for environmental equipment qualification, • Background in Digital I&C, and. Knowledge of radiation dose and dose rates to determine the radiation environment.
Level of Effort in Each Review Area	0.5 FTE (225 days per FTE)

Environmental Qualification of Mechanical and Electrical Equipment	**Finland STUK**
Design Information Provided by Applicant	Preliminary safety analysis report description chapter V.V and topical reports TR66.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	Regulatory review of above mentioned documentation.
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)	Government degree 733. Regulatory guides: YVL 1.0, YVL 2.0, YVL 5.X and YVL 7.X.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	No official requirements, but in practice. Senior Inspectors: university degree in process and nuclear engineering, adequate working experience in design/research.
Specialised Training, Experience and/or Education Needed for the Review of this topic	Altogether 5 week training course on nuclear safety.
Level of Effort in Each Review Area	Regulator review 200 working days.

Environmental Qualification of Mechanical and Electrical Equipment	**France** **ASN**
Design Information Provided by Applicant	The applicant provides the following information in the safety analysis report: • Methodology used to define Temperature, Humidity, radiation and Pressure conditions during accidental transients in the different part of the NPP and results of the application of this methodology; • Environmental qualification approach, considering environmental conditions which materials and equipment would be exposed to in the plant, including severe accident conditions. Environmental qualification includes both function and reliability. The debris generation during accident conditions has to be taken into account in the qualification approach; • Methods of qualification and the standards covering ambient conditions for reference as well as for severe accident situations have to be defined and their representativeness has to be justified; • List of structures and components (mechanical, electrical and instrumentation control) needed to achieve the safety demonstration and the conditions (normal and accident) in which this equipment is required. The location of each piece of equipment. Information required as support of the SAR: • The criteria used for the analyses. • The results of test and analysis. • For each SCCs, a synthesis note of its qualification (NSQ), which includes the specified qualification programme for this SCC, the criteria established to pronounce the qualification, the results of test/ calculi and the analysis.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	A comprehensive review of the safety file (SAR and support documents) provided by the applicant is performed by the TSO. The following aspects are assessed in detail: • Adequacy of the safety criteria established. • Results of the test and analyses and compliance with the safety criteria. • Completeness of the NSQs. Considerations can also be given to experience feedback.
What type of confirmatory analysis (if any) is performed?	The ASN staff could perform inspections to verify that the safety requirements related to environmental qualification described in the SAR are correctly take into account in design, construction, procurement and assembly specifications. ASN and TSO staff could participate on environmental qualification tests of SCCs.

Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	The applicable requirements on this topic is: • Section B.2.2.1 “qualification of equipment” of the technical guidelines for design and construction of the next generation of NPP. • DRAFT regulatory guide “Reactor (PWR) design”. There is no standard or code in France on this topic. Acceptance criteria are defined by the applicant and reviewed by TSO.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	Engineering
Specialised Training, Experience and/or Education Needed for the Review of this topic	Specialised training and experience to understand the plant systems interactions and dependences. The staff performing the technical review at IRSN (TSO) has a long experience (more than 10 years) on this topic.
Level of Effort in Each Review Area	The effort needed to review a new plant design strongly depends on the degree of novelty of this design.

Environmental Qualification of Mechanical and Electrical Equipment	**Japan NRA**
Design Information Provided by Applicant	In the establishment permit application stage, the following information is provided in the description regarding the safety design of nuclear reactor facility: • Importance classification of safety functions, and • Design policy.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	Identical with scope provided in Subsection of "Classification of SSCs".
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 demonstrate the analysis results, if needed, (cross check analysis).
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	• The NRA Ordinance on Standards for the Location, Structure and equipment of Commercial Power Reactors. • The NRA Ordinance on Technical Standards for Commercial Power Reactors Facilities. • The Regulatory Guide of the NRA Ordinance on Standards for the Location, Structure, and Equipment of Commercial Power Reactors. • The Regulatory Guide of NRA Ordinance on Technical Standards for Commercial Power Reactor Facilities.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Senior: Director for Nuclear Safety Examination. • Junior: Nuclear Safety examiner. • TSO: None since Japan Nuclear Energy Safety Organisation (JNES) was integrated into NRA as of April, 2014.
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) is not set up for the individual review area. Regarding the standard processing duration, 2 years is set up for the basic design of an entire plant, and 3 months per one application is set up for detailed design. Divided application is granted for the detailed design.

Environmental Qualification of Mechanical and Electrical Equipment	**Korea** **NSSC and KINS**
Design Information Provided by Applicant	As part of the SAR, the applicant should describe or provide the following related to the environmental qualification of mechanical and electrical equipment: **<Mechanical>** • Equipment Identification and Environmental Conditions. • Qualification Test and Analysis. • Mechanical and Electrical Equipment Environmental Design and Qualification for Normal Operation, During and After a Design-Basis Accident. • Qualification Test Results: - Instrumentation and Electrical Equipment; - Mechanical Equipment; - Class 1E Equipment Loss of Ventilation Effects. • Chemical Spray, Humidity, Submergence, and Power Supply Voltage and Frequency Variation. • Radiation Environmental Qualification. • Environmental Qualification Records. **<Electrical>** • Design Bases. • Equipment Description. • Applicable Codes and Classification. • Equipment Identification and Location. • Environmental Conditions (Normal, Accident and Post-accident Environments). • Required Operational Time during Design Basis Accidents. • Environmental Qualification Program. • Environmental Qualification Tests and Analysis Results.
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 (request for additional information) 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 and Regulatory Guide that are applicable to this area are as follows: • KINS SRG 3.11, "Environmental Qualification for Mechanical and Electrical Equipment." • KINS Reg. Guide 3.3, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants".
What type of confirmatory analysis (if any) is performed?	None. (KINS staff performs the review or inspection of environmental qualification results submitted by the 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 12 (Safety Classes and Standards). - Article 13 (External Event Design Bases). - Article 15 (Environmental Effects Design Bases, etc.). - Article 26 (Protection System). - Article 40 (Use of Qualified Equipment). - Article 70 (Design Control). - Article 78 (Test Control). - Article 84 (Quality Assurance Records). - Section 4 (Quality Assurance Regarding Construction and Operation of Reactor Facilities). **<Nuclear Safety and Security Commission(NSSC) Notices>** • No. 2016-10, "Regulation on Safety Classification and Applicable Codes and Standards for Nuclear Reactor Facilities". • No. 2016-13, "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-N03.03,"Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants". • KINS/RG-N03.04, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment". • KINS/RG-N03.05, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants". • KINS/RG-N03.06, "Qualification Test of Electric Cables, Field Splices, and Connections for Nuclear Power Plants". • KINS/RG-N03.07, Environmental Qualification of Connection Assemblies for Nuclear Power Plants". • KINS/RG-N03.08, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants". • KINS/RG-N03.09, "Qualification for EMI in Safety Related I&C and Electrical Systems". • KINS/RG-N03.10, "Environmental Qualification of Safety Related Computer based I&C systems in Nuclear Power Plants". • KINS/RG-N03.11, "Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants". • KINS/RG-N03.12, "Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants". **<Applicable Codes and Standards>** • KEPIC EN (Electric and I&C). • KEPIC END 1100(Qualification of Class 1E Equipment for Nuclear Power Generating Stations). • KEPIC MFA Appendix B (Guide for Qualification of Nonmetallic Parts).
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Mechanical Engineer. • Electric Engineer. • I&C Engineer.

Specialised Training, Experience and/or Education Needed for the Review of this topic.	**<Mechanical>** • Experience with Ageing analyses of mechanical and Electrical equipment. • Understanding Codes and Standards (KEPIC, ASME, etc.). • Knowledge of radiation degradation mechanism. • Knowledge of water chemistry. • Knowledge of metallography. • Knowledge of and/or experience with material degradation mechanism. **<Electrical>** • Knowledge of electrical, I&C engineering. • Knowledge of material for electrical component. • Knowledge of applicable code and standards. • Related terminology (for example, ageing, qualified life, design-basis event [DBE], common-mode failure, common-cause failure, dedication, harsh environment, mild environment). • Knowledge of ageing principles and methods. Knowledge of Commercial Grade Item Dedication
Level of Effort in Each Review Area	**<Mechanical>** Total: 650 hours • Environmental Qualification Criteria review: 50 hours. • Environmental Qualification method review: 400 hours. • Environmental Qualification Records review: 200 hours. **<Electrical>** Total: 770 hours • Environmental Qualification Programme (EQP) review: 90 hours. • Environmental Qualification Reports (EQR) review: 680 hours.

Environmental Qualification of Mechanical and Electrical Equipment	**Slovak Republic** **UJD**
Design Information Provided by Applicant	• Calculations and calculation results to prove the resistance of selected facilities to environmental influences during all test, operation and emergency conditions considered in their design. • Requirements for quality of qualified equipment. • Qualification for their required functionality and presumed effects of their environmental for conditions considered in their design, during their commissioning, operation, decommissioning and during accidents. The qualification method shall correspond to the safety class of the selected Facilities.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	Review of the submitted documentation, if it conforms to atomic act and regulations. Evaluate if the systems, structures and components are in compliance with all requirements arising from applicable regulations, codes and standards. Confirm that the systems, structures and components: • Are able to manage their roles in the condition of working environment. • That the facilities have been properly classified to identify their importance to safety.
What type of confirmatory analysis (if any) is performed?	
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	National regulations, IAEA safety guidelines, Slovak Technical standards.
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 equipment evaluation. 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 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.

Environmental Qualification of Mechanical and Electrical Equipment	**Slovenia** **SNSA**
Design Information Provided by Applicant	• Detail information of environmental qualification of mechanical and electrical SSCs important to safety. • Information of a qualification programme for safety-related mechanical and electrical SSCs to confirm the capability of SSCs to achieve their design functions over the entire design service life. The SSC-qualification programme shall include collection, documentation and maintenance of information to confirm the capability of SSCs to achieve their design functions over the entire design service life. The qualification programme shall consider operating conditions such as vibration, temperature, pressure, water-jet impacts, electromagnetic disturbances, irradiation, moisture, earthquakes and combinations thereof. Operating conditions shall cover normal operating conditions over the entire design service life and conditions of anticipated operating occurrences or accidents. • Information concerning the methods of test and analysis employed to ensure the functionality of mechanical and electrical equipment under the full range of normal and accident loadings.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	• Determine the adequacy of the information provided by the applicant. • The review of the Technical Support Organisation's independent evaluation report. • The reviews of the analyses that have been adequately demonstrated that the electrical, mechanical and Instrumentation and Control (I&C) equipment are capable of performing their safety function under significant environmental stresses.
What type of confirmatory analysis (if any) is performed?	
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	• Rules on Radiation and Nuclear Safety Factors. • SNSA Practical Guidelines. • IAEA Safety Standards.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	Senior: mechanical engineer, electrical engineer. Junior: mechanical engineer, electrical engineer. TSO: mechanical engineer, electrical engineer.
Specialised Training, Experience and/or	Background in Environmental Engineering. Background in Digital I&C.

Education Needed for the Review of this topic	Background in the Environmental Qualification.
Level of Effort in Each Review Area	Regulator: 300 hrs. TSO' review time: 500 hrs.

Environmental Qualification of Mechanical and Electrical Equipment	**United Kingdom ONR**
Design Information Provided by Applicant	• Design and Construction rules for nuclear islands. • Pre-construction safety report.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	Review of submitted documentation.
What type of confirmatory analysis (if any) is performed?	None carried out.
Technical basis • Standards • Codes • Acceptance criteria (e.g. can come from Accident analysis, regulatory guidance)	• IEC standards. • ONR Safety Assessment Principles.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	Regulator: • University degree in electrical engineering and proven knowledge of nuclear safety. • Knowledge of electrical safety.
Specialised Training, Experience and/or Education Needed for the Review of this topic	• Knowledge of nuclear plants and safety systems. • Knowledge of electrical system analysis. Knowledge of equipment standards for electrical equipment.

Level of Effort in Each Review Area	Included in overall electrical assessment

Environmental Qualification of Mechanical and Electrical Equipment	**United States NRC**
Design Information Provided by Applicant	As part of the safety analysis report, the applicant should describe the following: • The electrical equipment that is important to safety. • Mechanical and electrical equipment associated with systems essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. • Equipment for which postulated failure might affect the safety functions of safety-related equipment or mislead an operator. • Equipment that is essential to prevent significant releases of radioactive material to the environment. • The location of each piece of equipment, inside and outside containment. • The normal and accident environmental conditions for each piece of equipment. • The process for determining the suitability of environmentally sensitive mechanical equipment. • The Environmental Qualification programme and its implementation.
Analysis, Reviews and/or Research Performed by the Reviewer and Scope of Review	The Nuclear Regulatory Commission (NRC) staff (1) reviews the information provided in the SAR for compliance with the regulations, (2) issues requests for additional information (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, Standard Review Plan (SRP). The sections of the SRP that are applicable to this area are as follows: • SRP 3.11, "Environmental Qualification of Mechanical and Electrical Equipment". 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 audits of procurement specifications to verify that the correct information is translated from the SAR into the procurement specifications.
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 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants". 2. 10 CFR 50.67, "Accident Source Term". 3. 10 CFR Part 50, Appendix A, Generic Design Criteria (GDC) 1, "Quality Standards and Records". 4. 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena". 5. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases". 6. 10 CFR Part 50, Appendix A, GDC 23, "Protection System Failure Modes". 7. 10 CFR Part 50, Appendix B, Section III, "Design Control". 8. 10 CFR Part 50, Appendix B, Section XI, "Test Control". 9. 10 CFR Part 50, Appendix B, Section XVII, "Quality Assurance Records". 10. 10 CRF Part 52.47, "Content of Applications, Technical Information". The NRC guidance documents that provide an acceptable approach for satisfying the

	applicable regulatory requirements are listed as follows: 1. Regulatory Guide (RG) 1.40, “Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants”. 2. RG 1.63, “Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants”. 3. RG 1.73, “Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants”. 4. RG 1.89, “Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants”. 5. RG 1.97, “Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants”. 6. RG 1.131, “Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants”. 7. RG 1.156, “Environmental Qualification of Connection Assemblies for Nuclear Power Plants”. 8. RG 1.158, “Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants”. 9. RG 1.180, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems”. 10. RG 1.183, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Plants”. 11. SECY 05-0197, “Review of Operational Programmes in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria”. 12. RG 1.1000, Rev. 3, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants”. Note: Guidance documents are not a substitute for regulations, and compliance with guidance documents is not required.
Skill Sets Required by (Education) • Senior (regulator) • Junior (regulator) • TSO	• Mechanical Engineering. • Electrical Engineering. • Health Physics.
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 requirements for equipment qualification. • Background in Digital Instrumentation &Control. • Knowledge of radiation dose and dose rates to determine the radiation environment.
Level of Effort in Each Review Area	500 hours.