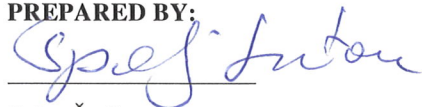




The Third NEK Periodic Safety Review Program

Report Number:
NEK ESD-TR-03/20
Rev. 1

PREPARED BY:



Srđan Špalj
ESD SA Engineer

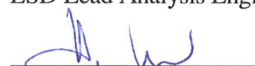
Date: 9.12.2020

REVIEWED BY:



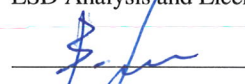
Rudolf Prosen
ESD Lead Analysis Engineer

Date: 9.12.2020



Aleksandra Antolović
ESD Analysis and Licensing Superintendent

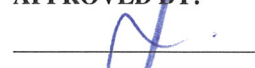
Date: 9.12.2020



Božidar Krajnc
ESD Director

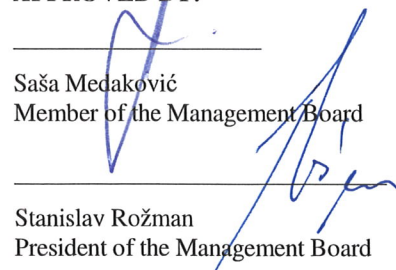
Date: 10/12/2020

APPROVED BY:



Saša Medaković
Member of the Management Board

Date: 11/12/2020


Stanislav Rožman
President of the Management Board


Date: 11.12.2020



Document No.: **NEK ESD-TR-03/20, Rev. 1**


TITLE: The Third NEK Periodic Safety Review Program

Rev.	Summary of Change	Approval	Date
1	<p>Changes were based on the URSJV comments provided in SNSA letters 3570-7/2020/12 [16] and 3570-7/2020/13 [17]:</p> <ul style="list-style-type: none">• Scope and methodology for review of each safety factor shall follow the guidelines from IAEA Safety Standards, 'Periodic Safety Review for Nuclear Power Plants', Specific Safety Guide No. SSG-25 [20] and draft of URSJV "Praktične smernice, Vsebina in obseg občasnega varnostnega pregleda sevalnega ali jedrskega objekta", PS 1.01, Izdaja 2, 2020 [14], [15].		

		Document Number	NEK ESD-TR-03/20, Rev.1
Title	The Third NEK Periodic Safety Review Program		
Assigned Reviewer(s)		Assigned by: (LAD Superintendent Name and Signature)	A. Antolovič 
Comment No. / Page	Comment	Response	
	Other comments were resolved during the preparation of revision 1.		


Reviewed by:

A. Antolovič
(Name)


(Signature)


Date: 9/12/2020

R. Prosen
(Name)


(Signature)

Date: 9/12/2020

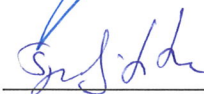
B. Krajnc
(Name)


(Signature)

Date: 10/12/2020

Response Prepared by:

S. Špalj
(Name)


(Signature)

Date: 9/12/2020

ABSTRACT

The purpose of a Periodic Safety Review (PSR) is to verify, by means of a comprehensive evaluation against modern standards, that the plant is as safe as originally intended and that no structures, systems, components, human activities or administrative processes can be found that could limit the life of the plant in the foreseeable future. Although the routine safety reviews and assessments ensure safety within the design basis, there is a need to consider the cumulative effects of aging, modifications, operating experience, changes in the standards and technical developments. Such tasks can be achieved by a dedicated systemic safety review against the current standards that takes all applicable factors into account at defined intervals. The PSR also identifies whether phenomena such as aging will have an effect on the facility that may compromise safety within the period before the next PSR.

In practice, in most countries operating nuclear plants PSR is required by law (either top level act or license condition) and, typically, a PSR is conducted every ten years.

The PSR project for Krško NPP (NEK) is required per national Ionising Radiation Protection and Nuclear Safety Act, Ref [3], considering the requirements from the Regulation JV9, Ref. [4].

The purpose of the present report is to provide a program for the 3rd NPP Krško Periodic Safety Review. The report considers inputs which represent a referential ground for the development of the program for the 3rd NEK PSR. They are:

1. The program, experience and results from the 1st and 2nd NPP NEK PSR,
2. The development of national legislation and requirements relevant for the PSR, which took place since the end of 2nd PSR,
3. Changes in the international perception of the PSR resulting in changes in the relevant international guiding documents reflecting on PSR.
4. Various plant specific and international inputs related to plant itself, industry, vendor country regulatory requirements (NRC) and different international research and development programs as described in Section below.

This program defines all relevant elements of the 3rd Periodic Safety Review that need to be agreed and confirmed by the Slovenian Nuclear Safety Administration (SNSA) before commencing the review. It includes the definition of objectives and scope of the review, selection of methods and assessment criteria as well as managerial and procedural / quality assurance arrangements related to PSR. The required resources and time schedule are also defined in the document.

This document provides the basis for the organization and control of the 3rd PSR process. It also provides a basic guidance to the PSR team regarding methodological approaches to be used in the assessment.

TABLE OF CONTENTS

TABLE OF CONTENTS	5
LIST OF ACRONYMS	7
LIST OF TABLES	9
LIST OF FIGURES	10
1. INTRODUCTION	11
2. BACKGROUND INFORMATION	12
2.1. Objectives of PSR3	12
2.2. Basis for the 3 rd PSR	16
2.2.1. National Requirements / Standards	16
2.2.2. International Requirements / Standards	17
3. DEFINITION OF THE 3RD PSR SCOPE AND METHODOLOGY	18
3.1. Available Inputs to the 3 rd PSR	18
3.2. Scope of the 3 rd PSR	20
3.2.1. Special Considerations	22
3.2.2. Review of the Safety Factors	22
3.2.3. Plant	24
3.2.4. Safety Analyses	31
3.2.5. Performance and Feedback of Experience	37
3.2.6. Review of the Management	40
3.2.7. Environment	47
3.2.8. Security	49
3.2.9. Radiation protection	50
3.3. Methodological Approach	52
3.3.1. Basic Methodology for Review of Plant Safety Factors	52
3.3.2. Ranking of Safety Issues	57
3.3.3. Identification and Prioritization of Corrective Measures	81
3.3.4. Implementation Action Plan	87
3.3.5. Global Assessment of Plant Status	88
3.4. PSR3 Documentation	89
3.4.1. Topical Reports	89
3.4.2. Final Broad and Detailed Ranking of Safety Issues with Identification and Prioritization of Corrective Measures	90
3.4.3. Implementation Action Plan	90
3.4.4. Summary Report and Global Assessment of Plant Status	91

3.4.5.	Status Reports	91
3.5.	PSR Project Organization	91
3.5.1.	Management	91
3.5.2.	External Members of the Project	92
3.5.3.	Responsibilities	92
3.5.4.	Project Interfaces	92
3.6.	PSR Program and Schedule	96
3.6.1.	Availability of Resources and Needed Expertise	96
3.6.2.	Milestones and Time Scales	96
4.	CONCLUSIONS	97
5.	REFERENCES	98

LIST OF ACRONYMS

AE	Severe Accident Management Equipment
AI	Authorized Institutions
AMES	Aging Materials Evaluation and Studies
AMSAC	ATWS Mitigating System Actuation Circuitry
ANS	American Nuclear Society
AOP	Abnormal Operating Procedures
ASCE	American Society of Civil Engineers
ASSET	Assessment of Safety Significant Events Team
ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design Basis Accident
CAP	Corrective Actions Program
CCF	Common Cause Failures
CDF	Core Damage Frequency
CSNI	Committee on Safety of Nuclear Installations
DBA	Design Basis Accident
DEC	Design Extension Conditions
EDMG	Extensive Damage Mitigation Guidelines
EPRI	Electric Power Research Institute
EOP	Emergency Operating Procedures
FHA	Fire Hazard Analysis
FPAP	Fire Protection Action Plan
FSAR	Final Safety Analysis Report
GALL	Generic Aging Lessons Learned
GSI	Generic Safety Issue
HRA	Human Reliability Analysis
IAEA	International Atomic Energy Agency
INES	International Nuclear Events Scale
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination for External Events
IPSART	International Probabilistic Safety Assessment Review Team
ISA	Integrated Safety Assessment
ISI	Inservice Inspection
LOCA	Loss of Coolant Accidents
LWR	Light Water Reactor
MCR	Main Control Room
NEA	Nuclear Energy Agency
NEK	Nuklearna elektrarna Krsko (Krsko Nuclear Power Plant)
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
ORAM	Outage Risk Assessment and Management
OSART	Operational Safety Review Team
OSF	Operational Safety Feature
PIE	Postulated Initiating Event
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSHA	Probabilistic Seismic Hazard Analysis
PSR	Periodic Safety Review

PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Westinghouse Owners Group (former WOG)
QA	Quality Assurance
RCP	Regulatory Conformance Program Compliance Review
RCS	Reactor Coolant System
RETS	Radiological Effluent Technical Specifications
RPV	Reactor Pressure Vessel
SAMG	Severe Accidents Management Guidelines
SAR	Safety Analysis Report
SBO	Station Blackout
SNSA	Slovenian Nuclear Safety Administration
SGTR	Steam Generator Tube Rupture
TS	Technical Specifications
URSJV	Uprava Republike Slovenije za jedrsko varnost (SNSA)
USAR	Updated Safety Analysis Report
WANO	World Association of Nuclear Operators
WOG	Westinghouse Owners Group (renamed to PWROG)

LIST OF TABLES

Table 3-1: Safety Factors Interface Matrix	56
Table 3-2: Assessment Criteria for Broad Ranking of Safety Issues	64
Table 3-3: Primary Safety Function Capability	65
Table 3-4: Degradation of a Level of Defense	66
Table 3-5: Risk Significance Attribute.....	69
Table 3-6: Depth of Defense Significance Attribute.....	70
Table 3-7: Functionality Impact Attribute	71
Table 3-8: Ranking of Issues with a Direct Link to Plant Safety with Respect to Primary Attributes	71
Table 3-9: Ranking of Issues Representing Re-evaluation of the Nuclear Safety Basis with Respect to Primary Attributes	73
Table 3-10: Primary Attribute Operational Safety Feature Impact (Ranking of Non Nuclear Safety Issues and Issues Related to “Soft Factors”).....	77
Table 3-11: Severity Scale for Prior Risk Information	78
Table 3-12: Severity Scale for Qualitative Cost Evaluation	78
Table 3-13: Summary of Issue Categories and Expected Responses.....	79
Table 3-14: Internal NEK PSR Organizational Interfaces	93
Table 3-15: Draft NEK 3 rd PSR Schedule.....	96

LIST OF FIGURES

Figure 3-1: Safety Factor Assessment Process 54

Figure 3-2: Overall Approach to PSR Issue Ranking 61

Figure 3-3: Detailed Ranking Process..... 62

Figure 3-4: Detailed Ranking Process (continued) 63

Figure 3-5: PSR Project Management Organization..... 95

1. Introduction

The routine reviews of nuclear power plant operation (including reviews of modifications to hardware and procedures, significant events, operating experience, plant management and personnel competence) and special reviews following major events of safety significance are the primary means of ensuring the plant safety. While the routine safety reviews and assessments ensure safety within the design basis, there is a need to consider the cumulative effects of aging, modifications, operating experience, changes in the standards and technical developments. Such tasks can be achieved by a dedicated systemic safety review against the current standards that encompass all applicable factors into account at defined intervals. The PSR also identifies whether phenomena such as aging will have an effect on the facility that may compromise safety within the period before the next PSR. In the current, internationally accepted, safety philosophy periodic safety reviews (PSRs) are comprehensive reviews aimed at the verification that an operating NPP remains safe when judged against current safety objectives and practices and that adequate arrangements are in place to maintain an acceptable level of safety.

In practice, in most countries operating nuclear plants PSR is required by law (either top level act or license condition). Its aim is to provide additional safety related insights, typically, after 10 years of operation and/or to support plant modifications.

The 3rd PSR project (PSR3) for NEK has been initiated per the requirements of national Ionising Radiation Protection and Nuclear Safety Act, Ref [3], considering also the requirements from the Regulation JV9, Ref. [4], as well as practical directives provided in Ref. [14]. PSR3 project for NEK will cover all changes either from current national and/or international safety standards/practices or plant design and operational arrangements and history from the 2nd PSR to the freeze date December 31, 2020. PSR2 review resulted in the action plan [31] with 225 issues included in the Corrective Action Program (CAP) with the deadline for implementation 30. 5. 2019 [38]. Only 5 issues were not resolved up to this date and the extension was approved by SNSA [39].

The purpose of the present report is to provide a program for the 3rd NPP Krško Periodic Safety Review (PSR3). The report considers inputs which represent a referential ground for the development of the program for the 3rd Krško PSR. They are:

1. The program, experience and results from the 1st and 2nd NPP Krško PSR,
2. The development of national legislation and requirements relevant for the PSR, which took place since the end of 2nd PSR,
3. Changes in the international perception of the PSR resulting in changes in the relevant international guiding documents reflecting on PSR.
4. Various plant specific and international inputs related to plant itself, industry, vendor country regulatory requirements (NRC) and different international research and development programs as described in Section 3.1, below.

This program defines all relevant elements of the review that need to be agreed upon with the SNSA before commencing the review. It includes the definition of objectives and scope of the review, selection of methods and assessment criteria as well as managerial and procedural arrangements related to PSR. The required resources and time schedule are also determined.

2. Background Information

2.1. Objectives of PSR3

The PSR3 project for NEK has been initiated per requirements of national Ionising Radiation Protection and Nuclear Safety Act, Ref [3], considering also the requirements from the Rules on the safety assurance of radiation and nuclear facilities (JV9) Ref. [4], as well as practical directives provided in Ref. [14]. The PSR3 project will cover all changes either from current national and/or international safety standards/practices or plant design and operational arrangements and history from the 2nd PSR (PSR2) to the so called freeze date: December 31st, 2020. This date meets the requirements for 10 years period between periodic safety reviews, since the freeze date for 2nd PSR was December 31st, 2010.

Krško NPP undertook the 1st PSR following, approximately, twenty years of plant operation and subsequent PSRs would be performed every ten years until the end of the plant operation. Ten years is considered to be an appropriate interval for such reviews in view of the likelihood, within this period, of the following:

- Changes in national and international safety standards, operating practices, technology, underlying scientific knowledge or analytical techniques;
- The potential for the cumulative effects of plant modifications to adversely affect safety or the accessibility and usability of the safety documentation;
- Identification of significant aging effects or trends;
- Accumulation of relevant operating experience;
- Changes in how the plant is, or will be, operated;
- Changes in the natural, industrial or demographic environment in the vicinity of the plant;
- Changes in staffing levels or in the experience of staff;
- Changes in the management structures and procedures of the plant's operating organization.

Implementation of PSR3 reflects the current, internationally accepted, safety policy with regard to PSR and fulfils obligations that follow from the IAEA nuclear safety convention. The overall goals of the PSR3 project are defined in compliance with the basic role of PSR as described in the related guide issued by IAEA [20].

The objectives of the PSR3 are formulated as follows:

- (1) Demonstrate that the licensing basis remains valid.
- (2) Demonstrate that plant conforms to current national and / or international safety standards and practices.
- (3) Demonstrate the adequacy of the arrangements that are in place to maintain plant safety until the next PSR.
- (4) Compare current level of safety in the light of modern standards and knowledge, and identify where improvements would be beneficial for minimizing deviations at justifiable costs.

Krško NPP has implemented many modifications related to plant hardware and procedures or safety related documentation. Many changes were intended to improve reliability, operability

and maintainability of the plant, some of them have been directly related to safety. These changes resulted from the advancements in technology, lessons learned from operational experience, and changes in safety standards. All changes in the plant are subject to strict control (Safety Screening and Evaluation) in accordance with Slovenian legislative (ZVISJV-1 [3] and JV9 [4]) and USA regulations 10 CFR 50.59 to ensure plant operation within licensing basis and that safety of the plant is not compromised. This gives a high confidence that the level of safety intended in the original design is not likely to have been reduced due to these changes.

Since the 1st PSR NEK has initiated comprehensive aging management and equipment qualification programs, among many other activities related to maintaining of the safe operation of the plant. Therefore, within the scope of the 2nd PSR, the special attention was made to safety factors Aging Management and Equipment Qualification. Comprehensive assessment and review of seismic hazard study was performed based on results from state of the art geophysical and geological research.

After the Fukushima accident NEK immediately initiated relevant activities at the plant level and issued the addendum to the original scope of the 2nd PSR program, requiring review and evaluation of the NEK response to Fukushima accident, taking into consideration special NEK reports issued following Fukushima accident ([22], [23]) and, in particular, the decrees that were issued by SNSA as a response to Fukushima accident ([24] and [25]). The post-Fukushima activities performed in NEK until June 30th, 2012, were reviewed within the PSR2, as well as the documents produced by NEK, in response to abovementioned decisions and SNSA letter 3570-11/2011/9 [26].

NEK systematically addresses the changes in the safety standards and safety related knowledge. One of the relevant safety programs, which provide a logical framework for this process, is the Regulatory Conformance Program (RCP). The RCP has been initiated in 1995, and is regularly updated. This program identifies and addresses the relevant changes in the U.S. regulations. In general, these are considered to reflect the actual internationally accepted safety philosophy. The activities conducted within the RCP program are broadly equivalent to those normally covered by PSRs.

In addition, the implementation of the PSR2 action plan [31] resulted in numerous improvements important for the plant safety:

- Containment buckling capacity analysis
- Seismic qualification program (ED-18)
- Mechanical equipment qualification program (ED-17)
- Electro-magnetic compatibility qualification program (ED-19)
- Equipment Survivability (ES) for severe accidents
- Test for the MCR integrity per GL 2003-01 Control Room Habitability
- New procedure ESP-2.405 (Guidance for Hazards Screening) and systematic review of internal and external hazards documented in technical reports ([113] and [114])
- Extensive Damage Mitigation Guidelines (EDMG)
- Beginning of constructing the Technical Support Center (TSC) and Operational Support Centre (OSC) in the frame of the safety upgrade project (SUP)
- Revision and improvements of probabilistic safety analyses (PSA)
 - Combinations of Fire and other events
 - Containment Event Tree Upgrade
 - Lightning probability and intensity
 - Recalculation of uncertainty of initial events
 - Introduction of PSA models for all modes of operation
 - High Energy Line Breaks - PSA upgrade

-
- Heavy Load Drop analysis
 - Revision of MAAP model for NEK and upgrade to version MAAP 5.03
 - Update of Severe Accident Management Guidelines (SAMG) to include Shutdown states
 - Numerous revisions of the USAR from the various aspects: meteorological data, severe winds, atmospheric discharges, gas pipelines, fire hazard analyses, plant on-site power, legislation, regulatory guides
 - Improvements in the field of emergency preparedness
 - Revision of NEK Code of safety and business ethics
 - NEK leadership self-assessments
 - Improvements of the plant internal communication

The PSR3 shall take into account also the major plant improvements contained in the Safety Upgrade Project (SUP) [40] and Severe Accident Management Equipment (AE) installation, which is a plant response to the Fukushima accident. To a certain extent it was already reviewed within PSR2 scope, as mentioned above. The SUP contains the upgrade of safety measures to address Design Extension Conditions (DEC) and partially Beyond Design Bases Accident (BDBA).

Safety Upgrade Program (SUP) [40] is approved by SNSA [27] and shall be completed by December 31. 2021. SUP consists of the following main projects and modifications, which are grouped into 3 phases:

1. Phase 1 (2012-2014) – This phase was completed and consisted of modifications related to Containment integrity – Pressure & Hydrogen control:
 - a. 1008-VA-L Passive Containment Filtered Venting System – PCFVS
 - b. 1002-GH-L Passive Autocatalytic Recombiners – PAR

Both modifications were implemented during the 2013 Outage.
2. Phase 2 (2014-2021) – includes the following modifications:
 - a. 1025-RC-L Pressurizer PORV Bypass Valves
 - b. 1028-SF-L Spent Fuel Pool Alternative Cooling
 - c. 1029-RH-L Reactor Cooling System and Containment Alternative Cooling
 - d. BB1 Project with the following subset of modifications:
 - i. 1007-XI-L Emergency Control Room (ECR)
 - ii. 1027-NA-L ECR/TSC Support Systems
 - iii. 1137-EE-L Upgrade of Bunkered building 1 Electrical Power Supply
 - iv. 1140-RC-L Replacement of the ICCMS control cabinets
 - v. 1053-PC-L Upgrade of NEK Communications Systems
 - vi. 1058-VA-L ECR/TSC HVAC Habitability systems
 - vii. NEK KFSS Upgrade and Construction of the Simulator Emergency Control Room
 - viii. 1069-TZ-L Technical security for the ECR and TSC
 - e. 1026-NA-L Flooding Protection of NSSS complex
 - f. 1056-NA-L Reconstruction of the Operating Support Center (OSC)
3. Phase 3 is planned to be completed till the end of the 2021 except for the Spent Fuel Dry Storage facility (SFDS) due to the NEK's request for rescheduling. Modifications shall ensure the rest of Safety Upgrade Program functionalities (alternate sources of water for heat sink and water injection pumps):
 - a. 1024-BS-L Bunkered Building 2 with Auxiliary Systems

-
- b. 1005-SI-L Alternative Safety Injection system
 - c. 1010-AF-L Alternative Auxiliary Feedwater system
 - d. 1030-EE-L BB2 Emergency Electrical Power Supply

Part of Phase 3 safety upgrade project is also a construction and use of Spent Fuel Dry Storage facility (SFDS) with design lifetime of minimum 100 years with DEC seismic, flooding and other design and construction requirements. In addition, high temperature resistant Reactor Coolant Pump seals will be installed (modification 1239-RC-L).

One of the important benefits expected from the NEK PSR, is that it is capable to provide a global check and transparent demonstration of plant safety. This enables NEK to increase public confidence, particularly at international level. As noted above, in many areas relevant to safety, which are covered by the actual safety oversight activities, the review is not expected to identify significant safety concerns. However, in order to add credibility to the results, the review will address all relevant safety aspects, as recommended in internationally accepted PSR guides Ref [7] and good practice documents.

From organizational point of view, the PSR can be divided into following stages:

- Preparation of the review,
- Review,
- Regulatory assessment,
- And preparation of Implementation Action Plan.

A brief overview of these stages is provided below.

The preparation stage is intended to set up the required framework for carrying out the review. Relevant key elements of the PSR are defined and agreed with the SNSA.

The review stage includes gathering of plant specific data and the assessment of current safety of the plant. The main outcome of this stage is a list of plant specific safety concerns and safety merits identified for each relevant safety area. The corrective actions are also proposed for the identified shortcomings. For the PSR3, the review stage is proposed to be divided into two basic phases:

- Safety factors review phase
- Global assessment and prioritization phase

The first phase includes a systematic evaluation of all safety areas (factors) using current methods and actual plant data. Comparison with current safety standards and practices is the basic element of this assessment. This step focuses on the identification of safety concerns (issues) but it also includes an immediate evaluation of issues with regard to their safety significance and indicates potential corrective actions. In the case of the PSR3, implementation of this phase will rely on significant input provided from the current safety oversight activities that have regularly been performed at NEK.

The second phase includes global assessment of identified strengths and shortcomings using the methodology described further in the document. Corrective measures are evaluated and implementation action plan is proposed. The decision making process involves all relevant aspects i.e. risk impact, costs involved, implementation time and trouble, etc., and is supported

by cost-benefit analysis. The review process shall be documented and topical reports and summary report will be submitted to the SNSA for assessment.

The same methodological approach adopted for the 2nd NEK PSR [1], will be applied for PSR3. This approach is intended to be systematic, rigorous, transparent and auditable. These attributes are explained below.

Systematic. An overall framework is presented in this report to be agreed at the start of the review. The work required will then be detailed, apportioned, and progressed against this framework but with an overview being retained throughout.

Rigorous. As a general rule, checks will be made by the reviewer to confirm the correctness of the material received. This refers, for example, to the validity of assumptions, the integrity of calculations, and the accuracy of descriptive material.

Transparent. To the extent possible, the information considered and the reasons for the conclusions reached will be given to a level of detail necessary for these to be checked by an independent reviewer.

Auditable. Proper controls will be applied to the review process and associated documentation. It should be noted that the necessary procedures are in place at NEK to ensure that the operational information required for the PSR is being recorded and hence will be available to the review team, including SNSA assessment of topical reports as described above.

2.2. Basis for the 3rd PSR

2.2.1. National Requirements / Standards

Regarding the national requirements, the primary bases, considered by this program, for the 3rd PSR are:

- Zakon o varstvu pred ionizirajočimi sevanji in jedrski varnosti (Uradni list RS, št. 76/17 in 26/19) (ZVISJV-1), (Ionising Radiation Protection and Nuclear Safety Act, Official Gazette of the Republic of Slovenia, No. 76/17 and 26/19), [3]
- Pravilnik o zagotavljanju varnosti po začetku obratovanja sevalnih ali jedrskih objektov (Uradni list RS, št. 81/16 in 76/17) (JV9), (Rules on the safety assurance of radiation and nuclear facilities (JV9), Official Gazette of the Republic of Slovenia, No. 81/16 and 76/17), [4]
- Pravilnik o dejavnikih sevalne in jedrske varnosti (JV5), Uradni list RS, št. 74/16 in 76/17 (Rules on radiation and nuclear safety factors (JV5), Official Gazette of the Republic of Slovenia, No. 74/16 and 76/17), [5]
- Osnutek URSJV praktične smernice, Vsebina in obseg občasnega varnostnega pregleda sevalnega ali jedrskega objekta, PS 1.01, Izdaja 2, 2020 (Content and scope of Periodical Safety Review of a radiation or nuclear facility, draft, 2020), [14]

2.2.2. International Requirements / Standards

Considerable experience has been accumulated in conducting PSRs worldwide. International organizations such as IAEA summarized the most recent PSR practices and provide guidance regarding organization, conduct, and methodology of the review. The basic findings and recommendations are taken into account in setting up the scope and methodology for the NEK PSR. The approach for the 3rd NEK PSR follows, as close as possible, the recommendations given in:

- International Atomic Energy Agency, IAEA Safety Standards, ‘Periodic Safety Review for Nuclear Power Plants’, Specific Safety Guide No. SSG-25, IAEA, Vienna, 2013 [20]

This document is considered to reflect the current international experience in the assessment of overall safety of NPPs. In this approach, the process is divided into a number of elements, each devoted to a different safety area / factor and progresses in steps, as is described in the sections to follow.

3. Definition of the 3rd PSR Scope and Methodology

3.1. Available Inputs to the 3rd PSR

In general, the inputs to the PSR3 consist of, but not limited to, plant documents, national/international standards and requirements, operational experience and results of research and development programs. The following lists are not exhaustive:

- **Plant Specific Documents**

- Updated Safety Analysis Report (USAR) [43]
- Operating License [42]
- Technical Specifications (TS) [44]
- Design Extension Conditions Technical Specifications (DEC TS) [45]
- Radiological Effluent Technical Specifications (RETS) [46]
- MD-1 Notranje usmeritve in cilji - petletni razvojni načrt [47]
- MD-2 Sistem vodenja - procesna organizacija [67]
- MD-4 Program človeškega ravnanja [48]
- MD-5 NEK Aging Management Program [96]
- MD-7 Management of Dose Reduction Program [49]
- MD-21 Plant Performance Monitoring Program [126]
- ED-1 Design Modification Control Program [50]
- ED-12 Environmental Qualification Program [91]
- ED-17 Mechanical Equipment Environmental Qualification Program (MEQ) [92]
- ED-18 Seismic Qualification Program [93]
- ED-19 Electromagnetic Compatibility Program [94]
- Emergency Preparedness Plan [143]
- Severe Accident Management Guidelines (SAMG) [51]
- Extensive Damage Mitigation Guidelines (EDMG), [52], [53]
- QA Program [57]
- Reload License Submittal
- Fire Protection Manual/Analysis/Plan
- Process Control Program
- Station Blackout (SBO) Analysis
- NEK PRS2 implementation Action Plan [31]
- NEK PSA Studies
- Records from Routine Safety Oversight Process and Special Reviews
- Reports to SNSA (daily, weekly, monthly, quarterly, yearly, licensee reportable events)

-
- Inspection Reports from SNSA
 - Outage Assessment Reports from Authorized Institutions
 - Event Reports
 - Surveillance Testing and ISI Program
 - NEK Technical Reports
 - System Health Reports
 - International missions reports (IAEA OSART, IAEA EPREV, EU WENRA)
 - Regulatory Conformance Program (RCP)
 - Documentation from Safety Upgrade Program (SUP) projects and modifications,
- **National/International Standards and Requirements**
 - National – ZVISJV-1, JV9, JV5, JV7, JV10, JV4
 - IAEA
 - EU
 - WENRA
 - OECD-NEA
 - U.S. NRC

- **Operational Experience Feedback**

Experience feedback (plant specific and international) is covered in NEK by application of Operating Experience Assessment Program (OEAP) defined by plant procedure ADP-1.1.200 [63]. The primary objective of OEAP is to ensure that lessons learned from operating experience are used to prevent occurrences of such events and to improve plant/personnel safety and plant reliability. OEAP covers two areas of activity:

- Events occurring on-site
 - Events occurring off-site
- **Results of Research and Development Programs**
 - PWROG (Pressurized Water Reactor Owners Group)
 - EPRI (Electric Power Research Institute)
 - CAMP (Code Applications and Maintenance Program for codes RELAP5, TRACE, SNAP)
 - MUG (MAAP User Group)
 - CSARP (Cooperative Severe Accident Research Program – MELCOR code)
 - Other R&D activities such as seismic, flooding, Sava river temperature changes and climatological changes

3.2. Scope of the 3rd PSR

The scope of 3rd NEK PSR is defined in terms of safety factors and shall follow the national requirements of JV9 [4] and IAEA recommendation of SSG-25 [20]. According to SSG-25 [20] the 14 PSR safety factors have been selected. These 14 safety factors are very similar to the requirements of SNSA Regulation JV9 [4] except for the following differences:

- Safety culture (*varnostna kultura*) is a separate safety factor (no. 11).
- Safety factor 16 *Radiativni odpadki* (radioactive waste and spent fuel) is a separate factor which is considered in the safety factors 1, 2 and 8 of SSG-25 [20].
- Safety factor 17 *fizična zaščita* (physical security) is additional factor which is not included in SSG-25 [20]. A review of the physical security of nuclear power plants is generally not included in the PSR because of the sensitivity of the subject and the need to ensure confidentiality. The effectiveness of security arrangements to prevent unauthorized actions that could jeopardize nuclear safety should be reviewed periodically by the appropriate national authorities.
- Radiation protection is a separate safety factor 18 (*varstvo pred sevanji*). Radiation protection is not regarded as a separate safety factor in SSG-25 [20] since it is related to most of the other safety factors. The arrangements for radiation protection and their effectiveness should generally be reviewed as specific aspects of the safety factors relating to: plant design; actual condition of SSCs important to safety; safety performance; and procedures.

The 14 safety factors from SSG-25 [20] are divided into five subject areas (the order and numbering of the listed safety factors does not indicate an order of importance) to facilitate the review:

Plant

- (1) Plant design,
- (2) Actual condition of structures, systems and components (SSCs) important to safety,
- (3) Equipment qualification,
- (4) Aging.

Safety analysis

- (5) Deterministic safety analysis,
- (6) Probabilistic safety analysis,
- (7) Hazard analysis.

Performance and feedback of experience

- (8) Safety performance,
- (9) Use of experience from other plants and research findings.

Management

- (10) Organization, the management system and safety culture,
- (11) Procedures,
- (12) Human factors,
- (13) Emergency planning.

Environment

(14) Radiological impact on the environment

The 18 safety factors from JV9 [4] (Priloga 9) are divided into seven subject areas and are listed below in Slovenian language and translated to English. The correspondence to IAEA SSG-25 is given in parentheses:

Objekt (Plant)

- (1) Projekt objekta - Plant Design (SSG-25 SF 1)
- (2) Dejansko stanje SSK - Actual Condition of SSCs (SSG-25 SF 2)
- (3) Kvalifikacija opreme – Equipment Qualification (SSG-25 SF 3)
- (4) Staranje objekta – Aging (SSG-25 SF-4)

Varnostne analize (Safety Analysis)

- (5) Deterministične varnostne analize objekta – Deterministic Safety Analysis (SSG-25 SF-5)
- (6) Verjetnostne varnostne analize objekta – Probabilistic Safety analysis (SSG-25 SF-6)
- (7) Analize ogroženosti in možnih nevarnosti glede na jedrsko in sevalno varnost - Hazard Analyses (SSG-25 SF-7)

Obratovanje in uporaba obratovalnih izkušenj (Performance and Feedback of Experience)

- (8) Obratovalne izkušnje in obratovalni kazalniki lastnega objekta - Safety Performance (SSG-25 SF-8)
- (9) Obratovalne izkušnje drugih objektov ter ugotovitve znanosti in tehnologije za obdobje pregleda - Use of Experience from Other Plants and Research Findings (SSG-25 SF-9)

Vodenje (Management)

- (10) Sistemi vodenja in organiziranost upravljavca – Organization and the Management System (SSG-25 SF-10)
- (11) Varnostna kultura - Safety Culture (SSG-25 within SF-10)
- (12) Pisni postopki upravljavca - Procedures (SSG-25 SF-11)
- (13) Vpliv dejavnosti osebja – človeški dejavnik - Human Factors (SSG-25 SF-12)
- (14) Načrt zaščite in reševanja - Emergency planning (SSG-25 SF-13)

Okolje(Environment)

- (15) Radiološki vplivi na okolje - Radiological Impact on the Environment (SSG-25 SF-14)
- (16) Radioaktivni odpadki in izrabljeno jedrsko gorivo (SSG-25 within SF-1, SF-2 and SF-8)

Fizična zaščita (Security)

- (17) Fizična zaščita - Security (SSG-25 N/A)

Varstvo pred sevanji (Radiation Protection)

- (18) Varstvo pred sevanji- Radiation Protection (SSG-25 within SF-1, SF-2, SF-8 and SF-11)

3.2.1. Special Considerations

The scope of the PSR3 review shall follow the guidance of JV9 [4] with the following exceptions:

- The results of IAEA Pre-SALTO mission, as agreed between NEK and SNSA [98], shall be considered as an input to the review of the Safety Factor 4 (Aging Management) if Pre-SALTO final report is issued by December 31, 2021. The issues found as a result of IAEA Pre-SALTO mission shall be considered as PSR3 issues and will go directly into the ranking process.
- Fire PSA (a part of Safety Factor 6 - Probabilistic safety analysis) is covered by a separate evaluation as a response to findings of IAEA expert mission [109] This was agreed between NEK and SNSA [30]. The issues found as a result of this mission shall be considered as PSR3 issues and will go directly into the ranking process.
- Safety Factor 17 (Security) - The threat assessment is prepared by Ministry of internal affairs every year. Physical Security Plan is then reviewed accordingly, agreed by SNSA and approved by Ministry of internal affairs. This review is conducted every year and this aspect will not be included in the PSR3 review.
- The review of NEK Decommissioning Plan shall be performed at least during the PSR interval per requirement of [5] (JV5). The decommissioning plan was revised and the third revision was issued in 2019 [28] and accepted by SNSA [29]. NEK decommissioning plan will not be reviewed again during the PSR3 as agreed between NEK and SNSA [30].
- Compliance with licensing and regulatory requirements shall be evaluated as a separate subject during PSR3, even though it is not a required safety factor. This subject consists of two parts: 1. domestic legislation and licensing requirements, and 2. NEK Regulatory Conformance Program Compliance Review (RCP) which demonstrate the continued compliance with U.S. regulatory requirements. RCP is included as an input to the Safety Factor 1 (Plant Design).

3.2.2. Review of the Safety Factors

The review of safety factors should determine the status of each safety factor at the time of the PSR and should assess future safety at the nuclear power plant at least until the next PSR and, particularly for Safety Factors 1 (Plant Design), 2 (Actual Conditions of SSCs) and 4 (Aging), should make the effort to assess long-term operation safety, i.e., until the end of extended lifetime. The review should consider whether there are any foreseeable circumstances that could threaten safe operation of the nuclear power plant. If such circumstances are identified, NPP Krško should take appropriate action to ensure that the licensing basis remains valid.

The review of safety factors should identify findings of the following types:

- Positive findings (that is, strengths): Where current practice is equivalent to good practices as established in current codes and standards, etc.
- Negative findings (that is, deviations): Where current practices are not of a standard equivalent to current codes and standards or industry practices, or do not meet the current licensing basis, or are inconsistent with operational documentation for the plant or operating procedures.

Deviations should be further divided into:

- Deviations for which no reasonable and practicable improvements can be identified;
- Deviations for which identified improvements are not considered necessary;
- Deviations for which safety improvements are considered necessary.

In the case of deviations for which no reasonable and practicable improvements can be identified, the reasons should be documented and the issue revisited after an appropriate period of time to determine whether a practicable solution is available.

For deviations for which identified improvements are not considered necessary, the reasons should be documented and the action considered completed.

Negative findings for which safety improvements are necessary, including updating/or extending of plant documentation or operating procedures, should be categorized and prioritized according to their safety significance. The categorization and prioritization of safety improvements may be performed on the basis of deterministic analyses, probabilistic safety assessment, engineering judgement, etc. Safety improvements from the safety factor reviews, together with safety improvements resulting from the global assessment, should be included in the implementation action plan.

If a review identifies a finding that poses an immediate and significant risk to the health and/or safety of workers or the public or to the environment, corrective action should not await the completion of the PSR. NEK shall take urgent steps to reduce the immediate and significant risk and, where relevant, should submit details of these steps to the SNSA.

The level of detail of the review could vary from safety factor to safety factor. In the NEK PSR3 the review of each safety factor shall be done in accordance to SSG-25 [20] Chapter 5 (SAFETY FACTORS IN A PERIODIC SAFETY REVIEW) and URSJV practical guidelines (praktične smernice, Vsebinska in obsega občasnega varnostnega pregleda sevalnega ali jedrskega objekta, PS 1.01, Izdaja 2, 2020) [14] where the suggestion for objective, scope and methodology is given. Moreover, the relevant national and international regulation, and other applicable documents for the review of safety factors shall be taken from the Annex of SSG-25 [20] and/or URSJV practical guidelines PS 1.05, [18], [19].

In order to integrate the results of the reviews of individual safety factors, a global assessment of safety at the plant shall be performed. The global assessment shall consider all findings and proposed improvements from the safety factor reviews and possible interfaces between different safety factors in accordance with SSG-25, Appendix I [20].

The results of the review shall be documented and the documentation shall be submitted to the SNSA during the PSR as required. The documentation shall include:

- Reports on the review of each safety factor;
- A report documenting the results of the global assessment;
- The final PSR report, including information on the proposed safety improvements and integrated implementation plan and
- A summary of the reports on safety factors and the global assessment.

In the following paragraphs the general objective and the tasks for the review of each safety factor are given

3.2.3. Plant

3.2.3.1. Safety Factor 1 - Plant Design

The objective of the review of the design of the NPP Krško is to determine the adequacy of the design and its documentation in an assessment against current national, international standards, requirements and practices.

Plant SSCs important to safety should have appropriate characteristics and should be configured in such a way as to meet the requirements for plant safety and performance, including the prevention and mitigation of events that could jeopardize safety. Adequate design information, including the design basis, should be available to provide for the safe operation and maintenance of the plant and to facilitate plant modifications.

A PSR should ensure that all significant documentation relating to the original or reconstituted design basis has been obtained, securely stored and updated to reflect all the modifications made to the plant and procedures since its commissioning. The review should also assess whether the project is being altered in a way to practically eliminate the core melt accidents and for accidents that cannot be practically eliminated, solutions shall be in place to assure that only limited measures are needed for the public protection, what is the requirements of the basic Slovenian national legislation JV5 [5]. The NEK's Regulatory Conformance Program (RCP), which evaluates the compliance with US NRC regulations, is also one of the inputs to this Safety Factor.

The review of the nuclear power plant design (including site characteristics) will include the following tasks:

- Review of the list of SSCs important to safety for completeness and adequacy including their safety classification.
- Review to verify that design and other characteristics are appropriate to meet the requirements for plant safety and performance for all plant conditions and the applicable period of operation, including:
 - the prevention and mitigation of events (faults and hazards) that could jeopardize safety,
 - the application of defense in depth and engineered barriers for preventing the dispersion of radioactive material (integrity of fuel, cooling circuit and containment building),
 - safety requirements (for example, on the dependability, robustness and capability of SSCs important to safety),
 - design codes and standards.
- Identification of differences between standards met by the nuclear power plant's design (for example, the standards and criteria in force when it was built) and modern nuclear safety and design standards
- Review of adequacy of the design basis documentation.
- Review for compliance with plant design specifications.
- Review of the safety analysis report or licensing basis documents following plant modifications and in light of their cumulative effects and updates to the site characterization.

-
- Review of plant SSCs important to safety to ensure that they have appropriate characteristics and are combined and segregated in such a way as to meet the requirements for plant safety and performance, including the prevention and mitigation of events (PIE's) that could jeopardize safety.
 - Verify current status of the human-machine interface to underline upgrade of the modern standards requirements between two PSR:
 - design of the control room and other workstations;
 - analysis of human information requirements and task workload.

The review of this safety factor shall also take into account also the major plant improvements contained in the Safety Upgrade Project (SUP) [40] and Severe Accident Management Equipment (AE) installation, which is a plant response to the Fukushima accident. To a certain extent it was already reviewed within PSR2 scope, as mentioned above. The SUP contains the upgrade of safety measures to address Design Extension Conditions (DEC) and partially Beyond Design Bases Accident (BDBA).

Methodology

The review should be performed systematically by means of a clause-by-clause review of national and international requirements and standards listed in the PSR basis document and other requirements and standards identified as relevant during the course of the review. Where this would assist the review, the evolution of these requirements and standards from the versions used for the original design and the second NEK PSR should be evaluated to assess the impact of changes on the plant design.

In the review, consideration should be given to subdivision into topics according to plant systems, such as reactor core, reactor coolant system, containment system, instrumentation and control systems, electrical power systems and auxiliary systems.

In some cases, comparison with requirements and standards may be best carried out by means of a high level or programmatic review. If this approach is to be adopted, the PSR basis document should clearly indicate this intention and, where appropriate, this should be agreed with the regulatory body.

The review of this safety factor should be carried out for all SSCs important to safety. The review should seek to identify deviations between the plant design and current safety requirements and standards (including relevant design codes) and to determine their safety significance. If a suitable list of SSCs is not available, one should be developed by the operating organization as part of the PSR.

The review should consider the adequacy of defense in depth in the plant design. This should include an examination of:

- The degree of independence of the levels of defense in depth;
- The adequacy of delivery of preventive and mitigatory safety functions;
- Redundancy, separation and diversity of SSCs important to safety;
- Defense in depth in the design of structures (for example, review of the integrity of fuel, cooling circuit and containment building).

Where the plant has undergone a significant number of modifications over its lifetime or in the period since the last PSR, the cumulative effects of all modifications on the design should be

examined (for example, review of the loading on electrical supplies or post-trip cooling demands on water supplies).

The PSR should verify that significant documentation relating to the original and/or reconstituted design basis has been obtained, securely stored and updated to reflect all the modifications made to the plant since its commissioning. Recommendations on meeting the requirements of Ref [85] for document control are provided in Ref [86].

A design re-evaluation should be undertaken if the design information is inadequate or there is significant uncertainty over the adequacy of an SSC important to safety to fulfil its safety function (for example, in view of its actual condition (see safety factor 2)).

The review of the human-machine interface should examine the actual condition of the plant using, for example, plant walkdowns by specialists.

If deficiencies in the procedures and processes or in the design of the human-machine interface represent a potentially significant adverse contribution to risk, the PSR should make proposals for corrective actions to be considered in the global assessment. These may include improvements in procedures, enhanced training or redesign of human-machine interfaces.

3.2.3.2. Safety Factor 2 - Actual Condition of Systems, Structures and Components

The actual condition of SSCs important to safety within the nuclear power plant is an important factor in any review of the safety of the plant design. Hence, it is important to document thoroughly the condition of each SSC important to safety. Additionally, knowledge of any existing or anticipated obsolescence of plant systems and equipment should be considered part of this safety factor

The objective of the review of this safety factor is to determine the actual condition of SSCs important to safety and so to consider whether they are capable and adequate to meet design requirements, at least until the next PSR. In addition, the review should verify that the condition of SSCs important to safety is properly documented, as well as reviewing the ongoing maintenance, surveillance and in-service inspection programs, as applicable

The review of the condition of the SSCs of the nuclear power plant should consider the following aspects for each SSC:

- Existing or anticipated ageing processes;
- Operational limits and conditions;
- Current state of the SSC with regard to its obsolescence;
- Implications of changes to design requirements and standards on the actual condition of the SSC since the plant was designed or since the last PSR (for example, changes to standards on material properties);
- Plant programmes that support ongoing confidence in the condition of the SSC;
- Significant findings from tests of the functional capability of the SSC;
- Results of inspections and/or walkdowns of the SSC;
- Maintenance and validity of records;
- Evaluation of the operating history of the SSC;

-
- Dependence on obsolescent equipment for which no direct substitute is available;
 - Dependence on essential services and/or supplies external to the plant;
 - Verification of the actual state of the SSC against the design basis.

Methodology

The actual condition of the SSCs important to the safety of the nuclear power plant should be reviewed using knowledge of any existing or anticipated ageing processes or of obsolescence of plant systems and equipment, modification history and operating history. The implications of changes to design standards since the plant was designed or since the last PSR should be examined during the review of plant condition.

Inputs to the review of this safety factor should be made available from the ageing management program. However, if this program does not provide adequate information, the necessary inputs should be derived at an early stage of the PSR.

Where data are lacking, they should be generated or derived by performing special tests, plant walkdowns and inspections as necessary. The validity of existing records should be checked to ensure that they accurately represent the actual condition of the SSCs important to safety, including any significant findings from ongoing maintenance, tests and inspections.

It may not always be possible to determine the actual condition of SSCs important to safety in some areas of the plant owing to, for example, plant layout or operating conditions that may preclude inspection. Such instances should be highlighted and the safety significance of the resultant uncertainty in the true condition of the SSCs should be determined. These uncertainties may be reduced by considering evidence from similar components from other plants or facilities that are subject to similar conditions and/or knowledge of the relevant ageing processes and operating conditions.

For practical purposes, the review may group SSCs important to safety according to functional systems or type.

After determining the actual condition of the SSCs important to safety, each SSC should be assessed against the current design basis (or updated design basis: see safety factor 1) to confirm that design basis assumptions have not been significantly challenged and will remain so until the next PSR.

Where consistency with the design basis has been significantly challenged, the PSR should make proposals for corrective action (for example, additional inspections or tests, further safety analysis or the replacement of components). These proposals should then be considered further in the global assessment.

Taking into account previous NEK PSR and that the plant design at that time has been considered and justified against safety and design standards applicable at that time, it will only be necessary to consider changes in standards since the previous PSR.

3.2.3.3. Safety Factor 3 - Equipment Qualification

Plant equipment important to safety (that is, SSCs) should be properly qualified to ensure its capability to perform its safety functions under all relevant operational states and accident conditions, including those arising from internal and external events and accidents (such as loss of coolant accidents, high energy line breaks and seismic events or other vibration conditions).

The qualification should adopt a graded approach consistent with the safety classification of the SSC and should be an ongoing activity.

The objective of the review is to determine whether equipment important to safety is qualified to perform its designated safety function taking into account all environmental conditions (e.g. seismic, vibration, temperature, pressure, jet impingement, electromagnetic interference, irradiation, humidity, and combinations thereof) over the lifetime of the plant and when required in normal operating conditions, design basis accidents and beyond design basis accidents. This safety factor was reviewed in detail for the first time within the 2nd NEK PSR. The review shall confirm that all qualification programs (ED-12 Environmental Qualification Program [91], ED-17 Mechanical Equipment Environmental Qualification Program (MEQ) [92], ED-18 Seismic Qualification Program [93] and ED-19 Electromagnetic Compatibility Program [94]) are properly managed and applied within NEK processes.

The review of equipment qualification should include an assessment of the effectiveness of the plant's equipment qualification programs. These programs should ensure that plant equipment (including cables) is capable of fulfilling its safety functions for the period until at least the next PSR. The review should also consider the effects of ageing degradation of equipment during service and of possible changes in environmental conditions during normal operation and predicted accident conditions since the programs was devised.

Qualification of plant equipment important to safety should be formalized using a process that includes generating, documenting and retaining evidence that equipment can perform its safety functions during its installed service life. This should be an ongoing process, from its design through to the end of its service life. The process should take into account plant and equipment ageing and modifications, equipment repairs and refurbishment, equipment failures and replacements, any abnormal operating conditions and changes to the safety analysis. Although many parties (such as designers, equipment manufacturers and consultants) will be involved in the equipment qualification process, the operating organization has the ultimate responsibility for the development and implementation of an adequate plant specific equipment qualification program.

As a part of this safety factor the following shall also be reviewed:

- Whether installed equipment meets the qualification requirements.
- Adequacy of the equipment qualification record.
- Procedures to maintain qualification throughout the installed service life of the equipment.
- Procedures which ensure SSC modifications and additions do not compromise equipment qualification.
- A surveillance program and a feedback procedure to ensure that aging degradation of qualified equipment remains insignificant.
- Monitoring of actual environmental conditions and identification of 'hot spots' of high activity or temperature.
- Protection of qualified equipment from adverse environmental conditions.

Methodology

Plant equipment should be classified, designed, manufactured and qualified according to its importance to safety on the basis of relevant safety requirements and standards. At a minimum, the PSR should verify that the standards and requirements in use for equipment qualification at the plant remain valid. The review should also include assessment of the following:

- Changes in the equipment classification resulting from design modifications;
- Qualification for all designed environmental conditions;
- The availability of equipment that is required to fulfil safety functions;
- Quality management provisions that ensure that an effective qualification program is in place.
- How is equipment qualification master list maintained and its revisions controlled.
- How are qualification requirements maintained and its revisions controlled.

The review of equipment qualification shall determine:

- Whether adequate assurance of the required equipment performance was initially provided;
- Whether current equipment qualification specifications and procedures are still valid (for example, initial assumptions regarding the service life of equipment and the environmental conditions);
- Whether equipment performance has been preserved by ongoing application of measures such as scheduled maintenance, condition monitoring, testing and calibration and whether such programs have been properly documented.

The review should evaluate the results of plant tests, inspections and walkdowns and other investigations carried out to assess the current condition of installed qualified equipment (see Safety Factor 2). This part of the review should seek to identify any differences from the qualified configuration (for example, abnormal conditions such as missing or loose bolts and covers, exposed wiring or damaged flexible conduits). The walkdowns and inspections should be carried out to verify that the installed equipment matches the required qualification described in the safety documentation and should provide an input to the review of the adequacy of the plant's procedures for maintaining equipment qualification.

3.2.3.4. Safety Factor 4 - Aging

The objective of the review of aging is to determine whether aging aspects affecting SSCs important to safety are being effectively managed and whether an effective aging management program is in place so that all required safety functions will be delivered for the design lifetime of the plant and, if it is proposed, for long term operation.

The review of aging should include the review of the NEK Aging Management Program (MD-5) [96] and National Action Plan on the NEK Ageing Management Program [97].

The review shall evaluate both programmatic and technical aspects. The following programmatic aspects of the aging management program shall be evaluated:

- The timely detection and mitigation of aging mechanisms and/or aging effects;

-
- The comprehensiveness of the program, i.e. does it address all SSCs important to safety;
 - The effectiveness of operating and maintenance policies and/or procedures for managing the aging;
 - Evaluation and documentation of potential aging degradation that may affect the safety functions of SSCs important to safety;
 - Management of the effects of aging on those parts of plant that will be required for safety when the nuclear reactor has ceased operation, for example the spent fuel storage facilities;
 - Performance indicators;
 - Record keeping.

The review shall evaluate the following technical aspects:

- Aging management methodology;
- The operating organization's understanding of dominant aging mechanisms and phenomena, including knowledge of actual safety margins;
- Availability of data for assessing aging degradation, including baseline data and operating and maintenance histories;
- Acceptance criteria and required safety margins for SSCs important to safety;
- Operating guidelines aimed at controlling and/or moderating the rate of aging degradation;
- Methods for monitoring aging and for mitigation of aging effects;
- Awareness of the physical condition of SSCs important to safety and any features that could limit service life;
- Understanding and control of aging of all materials (including consumables, such as lubricants) and SSCs that could impair their safety functions;

Methodology

The aging management program should be reviewed to confirm that it provides for the timely detection and prediction of ageing degradation that might affect the safety functions and service lives of SSCs important to safety, and that it identifies appropriate measures for the maintenance of these functions. Program descriptions, evaluation of programs and technical bases for programs, plans for the reliability and availability of SSCs important to safety, the detection and mitigation of ageing effects, and the actual physical condition of structures and components should be examined. The review should focus on the integrated performance of the systems important to safety and on the results of periodic inspection and testing programs and trends in important safety parameters.

The review should examine whether effective control of ageing degradation is achieved by means of a systematic ageing management process in accordance with the requirements established in Refs [37], [106] and the recommendations provided in Ref [103]. Such a process consists of the following ageing management tasks, which should be carried out on the basis of a proper understanding of the ageing of the SSCs important to safety:

-
- Operation within operating guidelines with the aim of minimizing the rate of ageing degradation;
 - Inspection and monitoring consistent with the applicable requirements with the aim of timely detection and characterization of any ageing degradation;
 - Assessment of observed ageing degradation in accordance with appropriate guidelines in order to assess the integrity and functional capability of the structure or component;
 - Maintenance (that is, repair or replacement of parts) to prevent or remedy unacceptable ageing degradation.

The review should assess whether:

- A systematic, effective and comprehensive ageing management program is in place;
- Any non-safety-classified SSCs whose failure might inhibit or adversely affect a safety function are addressed to an adequate extent;
- All relevant aging degradation mechanisms are identified, and the models used to predict the evolution and advancement of ageing degradation are properly supported in accordance with current accepted practices pertaining to ageing degradation;
- Adequate measures are taken to monitor and control ageing processes;
- The aging management program will ensure continued safe operation for at least the period until the next PSR.

In addition, the issues found as a result of IAEA Pre-SALTO mission shall be considered as PSR3 issues and will go directly into the ranking process. For the purpose of tracking, these issues shall have distinctive identification added to the regular PSR3 identifications. The Pre-SALTO mission was agreed between NEK and SNSA [98]. In this context, the review shall take into consideration the differences between US NRC ([99] and [100]) and IAEA ([101], [102], [103]) practices regarding aging management. The condition for taking the Pre-SALTO results as an input to PSR3 review is the finalization of Pre-SALTO final report by December 31, 2021.

3.2.4. Safety Analyses

3.2.4.1. Safety Factor 5 - Deterministic Safety Analysis

Deterministic safety analysis should be conducted for each nuclear power plant, in order to confirm the design basis for SSCs important to safety and to evaluate the plant behavior for postulated initiating events.

The objective of the review of this safety factor is to determine to what extent the existing deterministic safety analysis is complete and remains valid when the following aspects have been taken into account:

- The actual plant design, including all modifications of SSCs since the last update of the safety analysis report or the last PSR;
- Current operating modes and fuel management;
- The actual condition of SSCs important to safety and their predicted state at the end of the period covered by the PSR;
- The use of modern, validated computer codes;

-
- Current deterministic methods;
 - Current safety standards and knowledge (including research and development outcomes), insofar as these standards and knowledge are reasonably applicable to NEK as Gen II PWR type NPP;
 - The existence and adequacy of safety margins.

The review of the deterministic safety analyses should include the following tasks:

- Review of the application of analytical methods, guidelines and computer codes used in the existing deterministic safety analysis and comparison with current standards and requirements, insofar as such standards and requirements are reasonably applicable to NEK as Gen II PWR type of NPP and taking into account the backfitting requirements (mandatory vs. voluntary) laid out in the standards;
- Review of the current state of the deterministic safety analysis (original analysis and updated analysis) for the completeness of the set of postulated initiating events forming the design basis, with consideration given to feedback of operating experience from plants of a similar design, insofar as such operating experience feedback is reasonably applicable to NEK as Gen II PWR type of NPP;
- Evaluation of whether the assumptions made in performing the deterministic safety analysis remain valid given the actual condition of the plant;
- Evaluation of whether the actual operational conditions of the plant meet the acceptance criteria for the design basis;
- Evaluation of whether the assumptions used in the deterministic safety analysis are in accordance with current regulations and standards, insofar as such regulations and standards are reasonably applicable to NEK as Gen II PWR type of NPP;
- Review of the application of the concept of defense in depth;
- Evaluation of whether appropriate deterministic methods have been used for development and validation of emergency operating procedures and the accident management program at the plant;
- Evaluation of whether calculated radiation doses and releases of radioactive material in normal and accident conditions meet regulatory requirements and expectations;
- Analysis of the functional adequacy and reliability of systems and components, the impact on safety of internal and external events, equipment failures and human errors, the adequacy and effectiveness of engineering and administrative measures to prevent and mitigate accidents.

Safety requirements relevant for the review of deterministic safety analysis are established in Refs [108], [106] and recommendations are provided in Ref [107].

Methodology

The review of deterministic safety analysis should provide a systematic re-examination of how operating experience feedback, new knowledge (for example, of physical phenomena) and changes in analysis and modelling techniques affect safety at the nuclear power plant.

The existing deterministic safety analysis should be reviewed against the current national and international requirements, standards and good practices to verify that the design basis for SSCs

important to safety is correct and that plant behavior for postulated initiating events is properly addressed to a current standard. However, it shall be taken into account if such standards and requirements are reasonably applicable to NEK as Gen II PWR type of NPP and taking into account the backfitting requirements laid out in the standards.

The review should seek to identify (or confirm) any major weaknesses as well as strengths of the plant design in relation to the application of defense in depth, and should evaluate the importance of systems and measures for preventing or controlling accidents.

The capabilities of the plant in its current state, and where relevant with account taken of planned safety improvements, should be demonstrated to be within regulatory requirements and expectations for both normal operation and accident conditions.

If it is necessary to repeat the analysis, consideration should be given to using current analytical methods, particularly with regard to computer codes for transient analyses. If the earlier approach is still used, its continuing validity should be verified explicitly in the review, including the assumptions used, the degree of conservatism applied and inherent uncertainties in the analysis.

The review should include an evaluation of the supporting analyses for design extension conditions. This should determine whether the arrangements aimed at preventing or mitigating severe core damage continue to be sufficient and whether any improvements are reasonable and practicable.

3.2.4.2. Safety Factor 6 - Probabilistic Safety Analysis

A review of the probabilistic safety assessment (PSA) should be conducted to identify weaknesses in the design and operation of the plant and, as part of the global assessment, to evaluate and compare proposed safety improvements.

The objectives of the review of the PSA are to determine:

- The extent to which the existing PSA study remains valid as a representative model of the nuclear power plant;
- Whether the results of the PSA show that the risks are sufficiently low and well balanced for all postulated initiating events and operational states;
- Whether the scope (which should include all operational states and identified internal and external hazards), methodologies and extent (i.e. Level 1, 2 or 3) of the PSA are in accordance with current national and international standards and good practices;
- Whether the existing scope and application of PSA are sufficient.
- Whether the USAR Chapter 19 adequately reflects the actual status of PSA analyses

The review of the PSA should include the following aspects:

- The existing PSA, including the assumptions used, the fault schedule, the representations of operator actions and common cause events, the modelled plant configuration and consistency with other aspects of the safety case;
- Whether accident management program for accident conditions (design basis accident conditions and design extension conditions) are consistent with PSA models and results;
- Whether the scope and applications of the PSA are sufficient;

-
- The status and validation of analytical methods and computer codes used in the PSA;
 - Whether the results of PSA show that risks are sufficiently low and well balanced for all postulated initiating events and operational states, and meet relevant probabilistic safety criteria;
 - Whether the existing scope and application of the PSA are sufficient for its use to assist the PSR global assessment, for example, to compare proposed improvement options.

Safety requirements relevant for the review of PSA are established in Refs [108], [106] and recommendations are provided in Refs [110] and [111].

Methodology

The PSA should be reviewed to confirm that the modelling reflects the current design and operating features, takes account of all relevant operating experience, includes all modes of operation and, where relevant, has a scope agreed with the regulatory body.

The PSA should be reviewed for completeness against an appropriate set of postulated initiating events and hazards.

The extent to which hazards are represented in the PSA should be reviewed to verify that omissions are based on site specific justifications and that these omissions do not weaken the overall risk assessment for the plant.

The analytical methods and computer codes used in the PSA should be reviewed to verify that the methods used and validation standards adopted continue to be appropriate.

If it is necessary to repeat parts of the PSA, consideration should be given to using current PSA methodology (analytical methods and computer codes). If the earlier approach is still used, its continuing validity should be verified explicitly in the review, including the assumptions used, the degree of conservatism applied and inherent uncertainties in the analysis.

The extent to which the potential for unidentified cross-links and the effects of common cause events are taken into account in the model should be reviewed, as these are often not adequately considered in plants of earlier design.

The human reliability analysis carried out in the PSA should be reviewed to ensure that the actions are modelled on a plant specific and scenario dependent basis, and that current methods are applied.

The results of the PSA should be compared with relevant probabilistic safety criteria (for example, for system reliability, core damage and releases of radioactive material) defined for the plant or set by the regulatory body.

The history of updates to the PSA to reflect changes in plant status should be reviewed. Ideally a living PSA should be maintained; however, where this is not practicable, the PSA should be kept sufficiently up to date throughout the lifetime of the plant to make it useful for safety decision making.

In addition, the fire PSA (a part of Safety Factor 6 - Probabilistic safety analysis) is covered by a separate evaluation as a response to findings of IAEA expert mission [109]. The issues found as a result of this mission shall be considered as PSR3 issues and will go directly into the ranking process. For the purpose of tracking these issues shall have distinctive identification added to the regular PSR3 identifications.

3.2.4.3. Safety Factor 7 - Hazard Safety Analysis

To ensure the delivery of required safety functions and operator actions, SSCs important to safety, including the control room and the emergency control center, should be adequately protected against relevant internal and external hazards.

The objective of the review of hazard safety analysis (HSA) is to determine the adequacy of protection of the nuclear power plant Krško against internal and external hazards with account taken of the actual plant design, actual site characteristics, the actual condition of SSCs and their predicted state at the end of the period covered by the PSR, and current analytical methods, safety standards and knowledge.

For each internal or external hazard identified, the review should evaluate the adequacy of the protection, with account taken of the following:

- The credible magnitude and associated frequency of occurrence of the hazard;
- Current safety standards;
- Current understanding of environmental effects;
- The capability of the plant to withstand the hazard as claimed in the safety case, based on its current condition and with allowance given to predicted ageing degradation;
- The appropriateness of procedures to cover operator actions claimed to prevent or mitigate the hazard.

A list of relevant internal and external hazards that may affect plant safety is already established for NPP Krško. The screening of internal and external hazards is presented in the reports NEK ESD-TR-18/16 [113] and NEK ESD-TR-07/17 [114]. These reports should be reviewed for completeness.

The following representative internal hazards that may affect plant safety should be reviewed (additional site specific internal hazards should be included under this safety factor if appropriate):

- Fire (including measures for prevention, detection and suppression of fire);
- Flooding;
- Pipe whip;
- Missiles and drops of heavy loads;
- Steam release;
- Hot gas release;
- Cold gas release;
- Deluge and spray;
- Explosion;
- Electromagnetic or radio frequency interference;
- Toxic and/or corrosive liquids and gases;
- Vibration;
- Subsidence;

-
- High humidity;
 - Structural collapse;
 - Loss of internal and external services (cooling water, electricity, etc.);
 - High voltage transients;
 - Loss or low capacity of air conditioning (which may lead to high temperatures).

The following representative external hazards that may affect plant safety should be reviewed (additional site specific internal hazards should be included under this safety factor if appropriate):

- Floods, including tsunamis;
- High winds, including tornadoes;
- Fire;
- Meteorological hazards (extreme temperatures, extreme weather conditions, high humidity, drought, snow, buildup of ice);
- Sun storm;
- Toxic and/or corrosive liquids and gases, other contamination in the air intake (for example, industrial contaminants, volcanic ash);
- Hydrogeological and hydrological hazards (extreme groundwater levels, seiches);
- Seismic hazards;
- Volcano hazards;
- Aircraft crashes, external missiles;
- Explosion;
- Biological fouling;
- Lightning strike;
- Electromagnetic or radio frequency interference;
- Vibration;
- Traffic;
- Loss of internal and external services (cooling water, electricity, etc.).

Methodology

For each relevant hazard, the review should verify, by means of current analytical techniques and data, that the frequency of occurrence and/or the consequences of the hazard are sufficiently low so that either no specific protective measures are necessary, or the preventive and mitigatory measures in place are adequate.

The analytical methods, safety standards and information used for the hazard analysis should be up to date and valid. If this is not the case, the analysis should be repeated or revised as necessary. The analysis and/or methods should take account of the plant design, site characteristics, the condition of SSCs important to safety (both at present and predicted for the

end of the period covered by the PSR) and relevant international practice [107]. Amongst other things, changes in plant design, the prevailing climate, the potential for floods and earthquakes, and transport and/or industrial activities near the site should be considered.

In considering the risk of a particular hazard, consideration should be given to experience of hazards and operating practices at nuclear power plants and at other facilities, both in the State and in other States.

Knowledge gained from actual events, in particular those that have occurred at nuclear power plants, should be identified. Any experience from managing such events (for example, external floods, seismic events and tornadoes) should be used to improve existing procedures at the plant.

The adequacy of the procedures used to prevent a hazard or to mitigate its consequences should be reviewed, including the extent to which these are tested and rehearsed (Refs [115] to [124]). The adequacy of the preventive and mitigatory measures can be evaluated by deterministic safety analysis (safety factor 5) or probabilistic safety analysis (safety factor 6).

3.2.5. Performance and Feedback of Experience

3.2.5.1. Safety Factor 8 - Safety Performance

Safety performance is determined from assessment of operating experience, including safety related events, and records of the unavailability of safety systems, radiation doses and the generation of radioactive waste and discharges of radioactive effluents. NPP Krško has in place top-level program MD-21 Plant Performance Monitoring Program [126] and the review shall assess the implementation of the program and introduction of aggregate performance indicators.

The objective of the review of safety performance is to determine whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including:

- Safety related incidents, low level events and near misses;
- Safety related operational data;
- Maintenance, inspection and testing;
- Replacements of SSCs important to safety owing to failure or obsolescence;
- Modifications, either temporary or permanent, to SSCs important to safety;
- Unavailability of safety systems;
- Radiation doses (to workers, including contractors);
- Off-site contamination and radiation levels;
- Discharges of radioactive effluents;
- Generation of radioactive waste;
- Compliance with regulatory requirements.

The review of safety performance is closely linked to the review of the use of experience from other plants and research findings (safety factor 9), but the review of safety performance should be restricted to operating experience at the plant under review.

Where safety performance indicators are used, the review should consider their adequacy and effectiveness, applying trend analysis and comparing performance levels with those for other plants in the State or in other States.

The review should consider the effectiveness of the processes and methodology used to evaluate and assess operating experience and trends. The findings of the reviews of other safety factors should be taken into account when undertaking this task.

Records of radiation doses and radioactive effluents should be reviewed to determine whether these are within prescribed limits, as low as reasonably achievable and adequately managed. Although radiation risks will need to be considered in all safety factors, the review of this safety factor should examine specifically data on radiation doses and radioactive effluents and the effectiveness of the radiation protection measures in place. Here the review should take into account the types of activity being undertaken at the plant, which may not be directly comparable with those at other nuclear power plants in the State or in other States.

Methodology

Where available, the review should utilize a set of safety performance indicators, which should cover in a systematic manner all aspects of operation important to safety. These indicators should provide information on both positive and negative aspects of safety performance. The sets of safety performance indicators developed by the IAEA, by certain Member States and by WANO could be used for this purpose. References [127] and [128] provide recommendations and guidance on the use of safety indicators for verifying compliance with the requirements for safe plant operation established in Ref [37]. Reference [37] requires that the operating experience at the plant be evaluated in a systematic way and that operating experience be used as an input to the PSR.

The review should also examine any other records of operating experience from the review period that are relevant to safety but have not been considered on the basis of the plant's safety performance indicators.

The review of safety performance should evaluate the adequacy of the plant's safety performance methodologies and processes with regard to:

- The identification and classification of safety related events;
- Root cause analysis of incidents and feedback of results;
- Methods for the selection and recording of safety related operational data, including data on maintenance, testing and inspection;
- Trend analyses of safety related operational data;
- Trend analyses regarding component replacements owing to failures or obsolescence;
- Feedback of safety related operational data to the operating regime (for example, for training purposes);
- The qualification of workers;
- The quality of procedures and results;

-
- Records of radiation doses and radioactive effluents;
 - Off-site and on-site contamination and radiation levels;
 - Compliance with regulatory requirements;
 - Implementation of corrective actions following events.

The analysis of trends over the operating lifetime of the plant or since the last PSR should be reviewed to identify potential future safety concerns (for example, precursors to accidents) or deteriorating safety performance. Where relevant, the results of the previous PSR should be examined to detect any long term trends in deteriorating safety performance.

Consideration should be given to the effects of any changes in operation at the plant (for example, the use of a new design of fuel) on safety performance. In particular, the review should evaluate the continuing relevance of the current indicators and other safety performance methods in the context of current and future operations, and ensure that only relevant data and records are used.

The PSR should include a review of the effectiveness of the operating organization's process for the routine evaluation of operating experience. However, where a common process is applied by the operating organization at several plants, and this process has been reviewed by a recent PSR at another plant, this element of the review could be confined to reviewing how the process is applied at the plant under review. Reference [129] provides detailed recommendations on reviewing the effectiveness of the process for the feedback of operating experience.

The use of performance indicators also enables comparisons to be made with other nuclear power plants and provides an opportunity for operating organizations to benefit from each other's experience. The extent to which this is being undertaken should be examined.

In cases where there are significant findings relating to the effectiveness of the feedback process, the PSR should carry out a full review of relevant operating experience at the plant over the review period.

Where the review indicates a weak performance or trend, possible root causes (for example, deficiencies in procedures, training or safety culture) should be identified.

For the purpose of providing data for other safety factors and for consideration in the global assessment, the results of the routine evaluations should be summarized (using, for example, indicators or trends) to provide an overall assessment of the safety performance for each year of the plant's operation over the review period. Trends should be reported and, where necessary, further analysis should be undertaken to highlight any potential safety problems.

3.2.5.2. Safety Factor 9 - Use of Experience from Other Plants and Research Findings

Experience from other nuclear power plants, and sometimes from non-nuclear facilities, together with research findings, can reveal previously unknown safety weaknesses or can help in solving existing problems. Reference [37] requires the operating organization to obtain and evaluate information on operating experience at other plants and to derive lessons for its own operations. This should include information from other plants for which the operating organization is responsible and wider experience, including relevant information from non-nuclear facilities.

The objective of the review of this safety factor is to determine whether there is adequate feedback of relevant experience from other nuclear power plants and from the findings of research and whether this is used to introduce reasonable and practicable safety improvements at the plant or in the operating organization [129], [130].

The review should identify operating experience reports and other information that may be important to nuclear safety at other plants owned by the operating organization, together with relevant experience and national and international research findings from nuclear and non-nuclear facilities. The review should pay the attention on the experiences from the plants with approved long-term operation and should consider their good practices and should take measures to avoid their bad experiences. With that respect the review should focus on the existing plant data base regarding long-term operating experience. It should be verified that this information has been properly considered within the plant's routine evaluation processes and that appropriate action has been taken.

The review of this safety factor is closely related to the review of safety performance (safety factor 8). However, unlike the review of safety performance, the review of the use of experience from other plants and research findings should seek to identify good practices and lessons learned elsewhere and take advantage of improved knowledge derived from research.

Methodology

The review of the use of experience from other plants and research findings should:

- Verify that arrangements are in place for the feedback of experience relevant to safety from other nuclear power plants and from relevant non-nuclear facilities;
- Review the effectiveness of such programs for the timely feedback of operating experience and for their output;
- Review the processes for assessing and, if necessary, implementing research findings and findings from operating experience relevant to safety.

Arrangements have been established for the dissemination of operating experience at nuclear power plants by the IAEA, the Nuclear Energy Agency of the OECD and various plant owners' groups. The operating organization should have in place a process for receiving, analyzing and acting upon such operating experience. The PSR should provide a summary of the findings from this process and should evaluate the effectiveness of the process. Where the review of effectiveness indicates significant shortcomings in the process, appropriate measures should be taken, including a repeat review of relevant events and information.

Arrangements for the dissemination of research findings may not be as well established as those from operating experience. The PSR should therefore pay particular attention to the adequacy of these arrangements and the timely implementation of research findings.

3.2.6. Review of the Management

According to the JV9 [4] Management is divided into 5 Safety Factors while SSG-25 [20] divides Management into 4 Safety Factors, what was already mentioned in the Chapter 3.2. Safety Factor 10 (Safety Management Systems and Organization) and Safety Factor 11 (Safety Culture) are treated as a single safety factor in SSG-25 [20], therefore, the attention shall be paid to the interface assessment of these two safety factors.

The operating organization is required to have in place a management system that ensures that policies and objectives are implemented in a safe, efficient and effective manner. Similarly, the

organization should have a strong safety culture so that all individuals carry out duties important to safety correctly, with alertness, due thought, full knowledge, sound judgement and a proper sense of accountability.

3.2.6.1. Safety Factor 10 - Safety Management Systems and Organization

The objective of the review of management systems for safety is to determine whether the management systems are adequate for the safe operating of the nuclear power plant [67]. Additionally, it should be assessed whether the management systems are adequate to achieve and enhance nuclear safety by ensuring that other demands on the licensee are not considered separately from nuclear safety requirements, to help preclude their possible negative impact on nuclear safety. The management system is a set of interrelated or interacting elements that establishes policies and objectives [47] and which enables those objectives to be achieved in a safe, efficient and effective manner. It integrates the principles of quality management, quality assurance and quality control and ensures that safety is not compromised by considering the implications of all actions.

The review of the organization and management system should include a review of the following elements or program against national and international standards:

- Policy statements of the operating organization;
- The documentation of the management system;
- The adequacy of arrangements for managing and retaining responsibility for activities or processes important to safety that have been outsourced (for example, maintenance and engineering services and safety analysis);
- The roles and responsibilities of individuals managing, performing and assessing work;
- The processes and supporting information that explain how work is to be specified, prepared, reviewed, performed, recorded, assessed and improved.

In addition, the review of the organization and management system should verify the following:

- There are adequate processes in place for managing organizational change.
- There is a human resource management process in place that ensures the availability of adequate, qualified human resources, including succession planning.
- There is adequate control of documents, products and records and this information is readily retrievable.
- There is adequate control of purchasing of equipment and services where this affects plant safety:
 - There are adequate processes in place to check the quality of suppliers' management systems that are intended to ensure that equipment and services supplied to the nuclear power plant are fit for purpose and provided in an effective and efficient manner.
 - There are established procedures for identifying inadequate, counterfeit, fraudulent or suspicious items and there are activities in the supply chain to prevent the entry of such items
- There are adequate communication policies in place.
- There are adequate facilities for training and training program are well structured.

-
- There are formal arrangements in place for employing suitably qualified internal and external technical, maintenance or other specialized staff.
 - There are adequate processes in place for feedback of operating experience to the staff, including experience relating to organizational and management failures.
 - There are suitable arrangements in place for maintaining the configuration of the nuclear power plant and operations are carried out in accordance with the safety analysis of the plant.
 - There are programs in place for ensuring continuous improvement, including self-assessment and independent assessment.

Methodology

Regular and systematic reviews of the management system are necessary to ensure that the safety policies, goals and objectives of the organization are being met as required [47]. These reviews should include evaluation of how the tasks are being undertaken and completed. This can be achieved by the review of independent audits carried out on behalf of senior management, task observations, self-assessments and supporting corrective action plans.

The review should examine whether regular management system reviews have been conducted at sufficient intervals and whether the following have been covered:

- Outputs from all forms of assessment (audits, self-assessments and task observations);
- Results delivered and objectives achieved by the operating organization and its processes;
- Non-conformances and corrective and preventive actions;
- Lessons learned from other organizations;
- Opportunities for improvement.

The review should also examine whether weaknesses and obstacles have been identified, evaluated and remedied in a timely manner. It should also examine whether the need to make changes to, or improvements in, policies, goals, strategies, plans, objectives and processes has been properly identified in the management system reviews.

Where the scope of the regular management system reviews has not addressed any of the aspects mentioned above, the PSR should undertake a detailed review of the omitted tasks.

3.2.6.2. Safety Factor 11 - Safety Culture

The review of Safety Culture is an assessment of commitment to safety and strengthening of safety culture and should include the following tasks:

- A review of the safety policy to verify that it states that safety takes precedence over production and to confirm that this policy is effectively implemented;
- A review of procedures to ensure that nuclear and radiation safety are properly controlled and that appropriate measures are applied consistently and conscientiously by all staff;

-
- An assessment of the extent to which a questioning attitude exists and conservative decision making is undertaken in the organization;
 - Verification that there is a strong drive to ensure that all events that may be instructive are reported and investigated to discover root causes and that timely feedback is provided to appropriate staff on findings and remedial actions;
 - Verification that unsafe acts and conditions are identified and challenged in a constructive manner wherever and whenever they are encountered by plant employees and external staff (contractors);
 - Verification that the organization has a learning culture and that it strives continuously for improvements and new ideas, and benchmarks against and searches out best practices and new technologies;
 - Verification that there is an established and effective process for communication of safety issues;
 - Verification that there is a process in place for prioritization of safety issues, with realistic objectives and timescales, that ensures that these issues receive proper resources;
 - Verification that there is a method in place for achieving and maintaining clarity of the organizational structure and managing changes in accountability for matters affecting safety;
 - Verification that there is adequate training in safety culture, particularly for managers.

In the scope of the regular and systematic reviews of the management system, the evaluation of the above mentioned tasks shall be performed with regards to the safety culture. This can be achieved by the review of independent audits carried out on behalf of senior management, task observations, self-assessments and supporting corrective action plans.

A safety culture assessment can also be performed by interviewing all levels of personnel at NPP and organizations supporting an NPP and should address all the tasks identified for this safety factor. The objective of the review of safety culture is to verify the development and the current status of the safety culture at the NEK and determine if it is still adequate for its safe operation till next PSR.

3.2.6.3. Safety Factor 12 - Procedures

Procedures important to the safety of the nuclear power plant should be comprehensive, validated, formally approved, appropriately distributed and subject to rigorous management control. In addition, the procedures should be unambiguous and relevant to the actual plant (with modifications taken into account); they should reflect current operating practices and due consideration should be given to human factor aspects (for example, whether they are user friendly).

The objective of the review of procedures is to determine whether the operator's processes for managing and following procedures and for maintaining compliance with operational limits and conditions are adequate, effective and maintain plant safety.

The review should examine the following types of procedures:

- Operating procedures for normal and abnormal conditions (including anticipated operational occurrences, design basis accident conditions and post-accident conditions);

-
- Procedures for the management of design extension conditions, including accidents with significant core degradation (for example, EEOP, SAMG, EDMG, AESP);
 - Maintenance, testing and inspection procedures;
 - Procedures for issuing work permits;
 - Procedures for controlling modifications to the plant design, procedures and hardware, including the updating of documentation;
 - Procedures for controlling the operating configuration;

Methodology

The review of procedures should:

- Verify that there is an effective process in place for formal approval and documentation of all safety related procedures.
- Verify that there is a formal system in place for development and modification of any procedure governing activities affecting safety, including adequate arrangements for tracking changes.
- Evaluate audits, self-assessments, safety performance and events to determine whether there is adequate understanding and acceptance of these procedures by managers and staff.
- Determine whether procedures are followed.
- Evaluate the adequacy of these procedures in comparison with good practices.
- Determine whether arrangements for regular review and maintenance of these procedures are in place and are adequate.
- Verify that procedures are structured and written with consideration given to human factors. For example, it should be checked whether the procedures are user friendly and can be readily understood and implemented by all staff who need to use them.
- Evaluate processes to update procedures to allow for changes in the assumptions made and/or the limits and conditions arising from the safety analysis, plant design and operating experience.
- Verify that the analysis and justification of the accident management procedures are documented.
- Verify that an appropriate process is in place for the categorization of procedures in accordance with their significance to safety.
- Examine whether there is adequate involvement in the development of procedures by the staff who will use them.
- Evaluate the distribution process for the control, copying and removal of obsolete versions of procedures, so that only the last approved edition is used.

The review of this safety factor should focus on those procedures that have the highest safety significance and need not necessarily include a full review of every procedure. The safety significance of procedures can be determined from deterministic safety analysis and/or PSA. For procedures assigned lower safety significance, a sampling approach could be followed to

review the overall adequacy of procedures (and the management processes used to develop and control them).

3.2.6.4. Safety Factor 13 - Human Factors

Human factors influence all aspects of the safety of a nuclear power plant. The review should examine the human factors at the plant and within the operating organization to determine whether these correspond to accepted good practices and to verify that they do not present an unacceptable contribution to risk. In particular, the review should determine whether operator actions claimed to be in support of safety are feasible and properly supported.

The objective of the review of this safety factor is to evaluate the various human factors that may affect the safe operation of the nuclear power plant and to seek to identify improvements that are reasonable and practicable.

The review of human factors should consider the procedures and processes in place at the nuclear power plant to ensure the following:

- Adequate staffing levels exist for operating the plant, with due recognition given to absences, shift working and restrictions on overtime;
- The sufficient number of the personnel and their competencies for work related to radiation or nuclear safety is regularly verified and documented;
- Qualified staff are available on duty at all times;
- Adequate programs are in place for initial training, refresher training and upgrading training, including the use of simulators;
- Operator actions needed for safe operation have been assessed to confirm that assumptions and claims made in safety analyses (for example, PSA, deterministic safety analysis and hazard analysis) are valid;
- Human factors in maintenance are assessed to promote error-free execution of work;
- Adequate competence requirements exist for operating, maintenance, technical and managerial staff;
- Staff selection methods (for example, testing for aptitudes, knowledge and skills) are systematic and validated;
- Appropriate fitness for duty guidelines exist relating to hours, types and patterns of work, good health and substance abuse;
- Policies exist for maintaining the know-how of staff and for ensuring adequate succession management in accordance with good practices;
- Adequate facilities and programs are available for staff training.

It shall be noted that the elements of the review related to the human-machine interface, which is in the original scope of the review of this safety factor in the IAEA SSG-25 [20], is moved to the scope of the review of Safety Factor 1 (Plant Design).

Methodology

The review of human factors should include the above tasks and should take account of recognized national and international good practices.

The review should be carried out with the assistance of properly qualified specialists. Because of the difficulties associated with carrying out an objective review of what is essentially the performance of its own staff, the operating organization may decide that specific elements of the review should be carried out by external consultants.

3.2.6.5. Safety Factor 14 - Emergency Planning

A nuclear power plant is required to be designed and operated to prevent or otherwise minimize releases of radioactive substances that could give rise to risks to workers or the public or to the environment. Emergency planning for the possibility of such releases is a prudent and necessary action, not only for the plant but also for local and national authorities

The objective of the review of emergency planning is to verify (a) whether the NEK still has adequate plans, staff, facilities and equipment for dealing with emergencies [143] and (b) whether the NEK organization's arrangements have been adequately coordinated with local and national systems and are regularly exercised [143].

The 3rd PSR will include an overall review to check that the emergency planning at the plant continues to be satisfactory and to check that emergency plans are maintained in accordance with current safety analyses, accident mitigation studies and good practices. The compliance with the applicable international (IAEA) standards needs to be verified (see references [146] and [147]). Additionally, PSR3 should verify that the operating organization has considered significant changes at the nuclear power plant site and in its use, organizational changes at the plant and changes in the maintenance and storage of emergency equipment, and developments around the site that could influence emergency planning.

To adequately verify the current status of NEK Emergency Planning the following tasks will be performed:

- Review the adequacy of on-site equipment and facilities for emergencies.
- Review the adequacy of on-site emergency support centers.
- Review of the efficiency of communications in emergency cases, in particular the interaction with organizations outside the plant.
- Review the current status of the content and efficiency of emergency training, performed exercises and check records of experience from these exercises.
- Review the arrangements for regular reviews of emergency plans and procedures and their regularly update.
- Review the changes in the maintenance and storage of emergency equipment.
- Review the effects of any recent residential and industrial developments around the site (3 and 10 km area).
- Review the Evacuation Time Estimates report [145]

Records of emergency exercises between two PSRs should be reviewed to verify competence of on-site and off-site staff, the required functional capabilities of equipment (including communications equipment) and the adequacy of emergency planning.

NEK interactions with relevant organizations such as SNSA, police, fire departments, hospitals, ambulance services, local authorities, public welfare authorities and information media should be re-evaluated.

The review of the adequacy of on-site equipment and facilities for emergencies and off-site emergency facilities or locations should include walkdowns of relevant areas on and off the site.

The content and effectiveness of emergency training and exercises should be evaluated by reviewing the records of these exercises with respect to, for example, their frequency and results, and the actions taken in case of deficiencies. These can be compared with current national and international guidelines and good practices.

Arrangements for regular reviews of emergency plans and procedures and their periodic updates between PSRs will be evaluated by reviewing the operating organization's management processes.

3.2.7. Environment

3.2.7.1. Safety Factor 15 - Radiological Impact on the Environment

NPP Krško shall have in place an established and effective monitoring program that provides data on the radiological impact of the nuclear power plant on its surroundings. The objective of the review of the radiological impact of the nuclear power plant on the environment is to determine whether the plant has an adequate program for monitoring of the radiological impact of the plant on the environment, which ensures that emissions are properly controlled and are as low as reasonably achievable.

Radiological monitoring data should be compared with the values measured before the NPP plant was put into operation and/or historical values examined in the last PSR. In the event of significant deviations, an explanation should be provided by the operating organization, with account taken of relevant factors external to the nuclear power plant. Where environmental data have not been provided since the start of operation of the plant or since the last PSR, these data should be submitted to the SNSA for information.

The review should establish whether the monitoring program is appropriate and sufficiently comprehensive. In particular, the review should verify that the radiological impact of the plant on the environment is not significant compared with that due to other sources of radiation.

As part of the review it should be verified that:

- Concentrations of radionuclides in air, water (including river water and groundwater), soil, agricultural and marine products and animals are being monitored by NPP Krško or by an independent organization and are trended, and appropriate corrective actions are taken in the event that action levels are exceeded;
- Potential new sources of radiological impact have been recognized;
- Sampling and measurement methods are consistent with current standards;
- Records of discharges of effluents are being monitored and trended and appropriate actions are taken to remain within established limits and to keep such discharges as low as reasonably achievable;
- On-site monitoring is undertaken at locations and using methods that have a high probability of the prompt detection of a release of radioactive material to the environment;

-
- Off-site monitoring for contamination levels and radiation levels is adequate and corrective actions are taken to keep such levels as low as reasonably achievable;
 - Actions have been taken to clean up contamination where reasonable and practicable;
 - Alarm systems to respond to unplanned releases of radioactive material from on-site facilities are suitably designed and available and will remain available in the future;
 - Appropriate data have been published on the environmental impact of the plant;
 - Changes in the use of areas around the site have been taken into account in the development of monitoring programs.
 - The monitoring equipment is adequate and the plans exist for replacing the obsolete or inadequate equipment
 - The analyses of liquid discharges with regards to the recent changes in the environment, particularly to HE Brežice construction, are performed
 - The independent survey, for different monitoring points on the yearly basis, is performed and reviewed

The review should also look for potential new sources of radiological impact by examining relevant plant modifications and the actual conditions of SSCs important to safety.

3.2.7.2. Safety Factor 16 - Radioactive waste and spent fuel

Radioactive waste and spent fuel is a separate Safety Factor 16 in JV9 [4]. This factor is not considered as a separate safety factor in SSG-25 [20] since it is related to most of the other safety factors. The radioactive waste and spent fuel should generally be reviewed as specific aspects of the safety factors relating to: Plant design (SF 1), Actual condition of SSCs important to safety (SF 2) and Safety performance (SF 8).

The objective of the review is to determine whether the plant has in place appropriate processes for radioactive waste and spent fuel management, and also, if the appropriate decommissioning plan is in place. The assessment shall be made if the radioactive waste management is appropriately described in the USAR. The reviewer shall pay the attention if the radioactive waste and spent fuel management process is in line with the requirements of ZVISJV-1, Article 121 [3] and [9], and in line with the national radioactive waste management program.

Data on the generation of radioactive waste should be reviewed to determine whether operation of the plant is being optimized to minimize the quantities of waste being generated and accumulated, taking into account the national policy on radioactive discharges and international treaties, standards and criteria, etc. Therefore, at least the following elements shall be reviewed:

- Radioactive waste and spent fuel management program;
- Criteria for acceptance into the storage facility and possible deviations from these criteria during the storage including the quality control for identification of compliance with these criteria
- Compliance of radioactive waste and spent fuel management with the national radioactive waste and spent fuel management program;
- Generation of radioactive waste;

-
- Accumulation of radioactive waste and spent fuel (inventory and reporting);
 - Procedures for management of radioactive waste and compliance with the Radioactive Waste Management Program and national regulations;
 - Record keeping;
 - Control of the integrity of RAO packaging
 - Compliance with regulatory requirements and national strategy;
 - Cumulative effect of performed modifications, good practices, operation experiences and abnormal events with regards to the radioactive waste and spent fuel management;
 - Progress in development and technology of radioactive waste and spent fuel management and evaluation of applicability to NPP Krško.

Regarding spent fuel, the strategy for the spent fuel storage and conduct of an engineering assessment of the condition of the storage facilities, the records management and the inspection regimes being used shall be reviewed. The following elements shall be taken into consideration

- The condition and operation of spent fuel storage facilities;
- The effect on the spent fuel storage strategy for the nuclear power plant;
- Management of the effects of aging on the spent fuel storage facilities;
- Procedures for spent fuel management and compliance with the Radioactive Waste Management Program national regulations
- The inspection regimes.
- Record keeping
- Performance indicators
- Compliance with regulatory requirements and national strategy;

In addition, the review of NEK Decommissioning Plan shall be performed at least during the PSR interval per requirement of [5] (JV5). NEK decommissioning plan was revised and the third revision was issued in 2019 [15] and accepted by SNSA [16]. NEK decommissioning plan will not be reviewed again during the PSR3 as agreed between NEK and SNSA [17].

3.2.8. Security

3.2.8.1. Safety Factor 17 - Security

This is additional safety factor required by SNSA Regulation JV9 [4] and is not included in requirements of SSG-25 [20]. A review of the physical security of nuclear power plants is generally not included in the PSR because of the sensitivity of the subject and the need to ensure confidentiality.

Safety measures and security measures have in common the aim of protecting human life, health and physical security of the plant. While safety is protection against hazards (accidents that are unintentional), security is a state of feeling protected against threats that are deliberate and intentional.

The effectiveness of security arrangements to prevent unauthorized actions that could jeopardize nuclear safety should be reviewed periodically by the appropriate national authorities.

The threat assessment is prepared by Ministry of internal affairs every year. NEK Physical Security Plan is then reviewed accordingly, agreed by SNSA and approved by Ministry of internal affairs. This review is conducted every year and this aspect will not be included in the PSR3 review.

A review of the physical security of nuclear power plant will be performed separately to ensure confidentiality of this subject.

3.2.9. Radiation protection

Radiation protection is a separate Safety Factor 18 in JV9 [4]. Radiation protection is not regarded as a separate safety factor in SSG-25 [20] since it is related to most of the other safety factors. The arrangements for radiation protection and their effectiveness should generally be reviewed as specific aspects of the safety factors relating to: Plant design (SF1), Actual condition of SSCs important to safety (SF2), Safety performance (SF8) and Procedures (SF12).

3.2.9.1. Safety Factor 18 - Radiation protection

Radiation protection procedures must be established and implemented according to ZVISJV-1 [3], and other corresponding Slovenian legislative acts ([4] to [13]) for the control, guidance and protection of personnel. The radiation protection staff of the plant must carry out routine monitoring of on-site radiological conditions at all accessible locations, and monitor the short term and cumulative exposure of personnel to radiation. The individual risk of fatality to any worker on the NPP attributable to doses of radiation received should be shown to be As Low As Reasonably Achievable (ALARA) and in accordance with national regulations and international guidance documents.

The design of the nuclear power plant should be such as to ensure that authorized dose limits and dose constraints for site personnel and the public will not be exceeded over specified periods (e.g. monthly, quarterly or annually) in operational states (normal operation and anticipated operational occurrences) and decommissioning. For workers who do not enter the designated areas (supervised areas and controlled areas), the authorized dose constraints should be set at the same level as the individual dose limit for members of the public. To ensure that a design both reduces doses to levels that are as low as reasonably achievable and represents best practice, design targets should be set for the individual dose and collective dose to workers and for the individual dose to those members of the public who will receive the greatest doses.

Procedures for normal operation (including preventive and corrective maintenance, inspection, condition monitoring and surveillance testing activities) should be established in a way to avoid or minimize risk from any hazard to plant personnel associated with performed activities. The criteria for the review of the radiation protection procedures are the same as already stated in the section for Safety Factor 12 (Procedures).

References [3] and [37] establishes the requirements for a radiation protection program, including requirements on the assessment of occupational exposure and on the management of radioactive waste and effluents arising from the operation of a nuclear power plant. References [160], [161] and [152] provide relevant recommendations and further guidance.

The objective of the review is to check if the radiological protection measures are incorporated into the design and that radiation protection is carried out at all stages of the lifetime of the plant, from design and construction to operation and decommissioning. It is also required to assess the adequacy and efficiency of radiation protection program, existing procedures and evaluate the actions for minimizing risk from radiation exposure and from radiation doses to the personnel and public.

The review shall include at least the following elements:

- The main source of radiation for all plant conditions (normal, abnormal accident) for which precautionary design measures should be adopted
- The magnitudes, locations, possible transport mechanisms and transport routes of the sources of potential radiation exposure under all plant conditions
- Procedures for radiation protection, including procedures for on-site transport of radioactive material;
- Radiation protection approach including the control of contamination and radiation exposure on the location;
- Evaluation of characterization of sources of exposure and other elements of radiation protection e.g. shielding, hot spots and categorization of areas;
- Level of radiation and contamination in the controlled and monitored areas of the object;
- Radiation doses received by workers, contractors and visitors including the analysis of adequacy dose barriers;
- The assessment of doses received by the population due to plant operation including assessment of models for dispersion of radioactive materials and transport paths;
- The review how the description of strategy for radiation protection is included in USAR, with the description of methods and measures for providing radiation protection of workers and members of public;
- The assessment of the most exposed processes and areas from the view of radiation exposure;
- The assessment of activities for the next 10 year period which can result in larger exposure;
- The assessment of radiation protection measures related to the capacity of radiation waste storage, and measures related to the manipulation and inspection of waste packages;

Methodology

The purpose of the review is to assess the adequacy of radiation barriers, dose and contamination limits with regards to the national and international regulations, standards and good practices. The review of radiation doses records shall verify if these limits are in concurrence with the ALARA approach. Within this safety factor it is essential to comprehensively review the effectiveness of radiation protection program.

The review of the availability of equipment for radiation protection shall be made, including the process radiation monitors, effluent monitors, portable monitors and contamination monitors and, also, their maintenance and calibration.

The radiation protection procedures shall be reviewed and evaluated.

It is necessary to perform the trend analysis for collective and individual radiation exposure, for NEK workers, as well as for contractors, visitors and public.

In addition, in line with the requirement of ZVISJV-1 [3], the review of NEK radiation protection assessment reports (references [162] and [163]) shall be performed within the PSR, according to the article 41 of ZVISJV-1 and as defined in [8]. NEK completed the report on the review of radiation protection assessment [164] in the year 2020 and, also, the independent expert evaluation of this report was made [165]. It is necessary to include already available report on the review of NEK radiation protection assessment in early phase of the PSR3 process i.e. in the review phase. Radiation protection assessment report is considered as mandatory based on ZVISJV's articles 40 – 41 and shall be implemented during the PSR project. However, the time for completion of this action shall not exceed the 5 year period. This action will be reported at the same way as other PSR issues. At this way, the requirement of ZVISJV-1 [3], which states, in the article 112, that the radiation protection assessment report shall be reviewed within the PSR, will be fully respected.

3.3. Methodological Approach

3.3.1. Basic Methodology for Review of Plant Safety Factors

This section describes the basic methodological aspects and the related assessment criteria applicable to the first phase of the review. This phase focuses on the identification of safety status of the plant, considering the specified safety factors.

When the previous PSR (2nd) has been undertaken the safety factors have been considered and justified against safety and design standards applicable at that time, therefore, it will only be necessary to consider changes in standards and plant processes, programs, procedures and records since the 2nd PSR.

The assessment of each of the relevant safety factors (described in section 3.2) will be focused on finding the differences in

- review elements defined by JV9 [4] and SSG-25 [20], in comparison with previous definitions, or
- modern standards / norms / requirements as compared to those evaluated in the 2nd PSR, or
- plant processes / programs / procedures / SSCs as assessed and evaluated in the 2nd PSR.

The process is described in the Figure 3-1.

Preliminary, broad, ranking of the identified safety issues is also conducted in this phase. Deterministic, formal and rule-based assessment (against internationally accepted safety requirements and standards) is the main approach applied at this stage of the review. Engineering rules and deterministic criteria based on good practice are applied. Some safety weaknesses may also be identified using probabilistic safety assessment, however its use at this step is limited.

The deterministic assessment concentrates on establishing if the current design and operation of the plant reflect the good practice, that suitable management policies and procedures are in place, and that the current safety analysis is comprehensive, suitably validated and demonstrates the required level of safety. Engineering rules and deterministic criteria based on good practice applied in this assessment depend on safety factor under consideration.

The general approach shall be in the compliance with the methodology given in the IAEA SSG-25 [20], as already mentioned in the previous section. Other relevant sources can also be used, as needed. Discussion of specific aspects of the methodology that is provided in this report is limited. However, reference to the methodology given in a particular standard, or parts thereof, will in many cases be sufficient to define the methodological approach. The most relevant references related to the subjected review of particular safety factor shall be systematically provided in the NEK PSR reports for subjected safety factor.

It shall be noted that the assessment criteria used at this stage of the review are not very precise. In most cases, the point at which a particular element of the review shows sufficient deviation from the 'ideal' to be classified as a 'safety issue' will be a question of expert judgment.

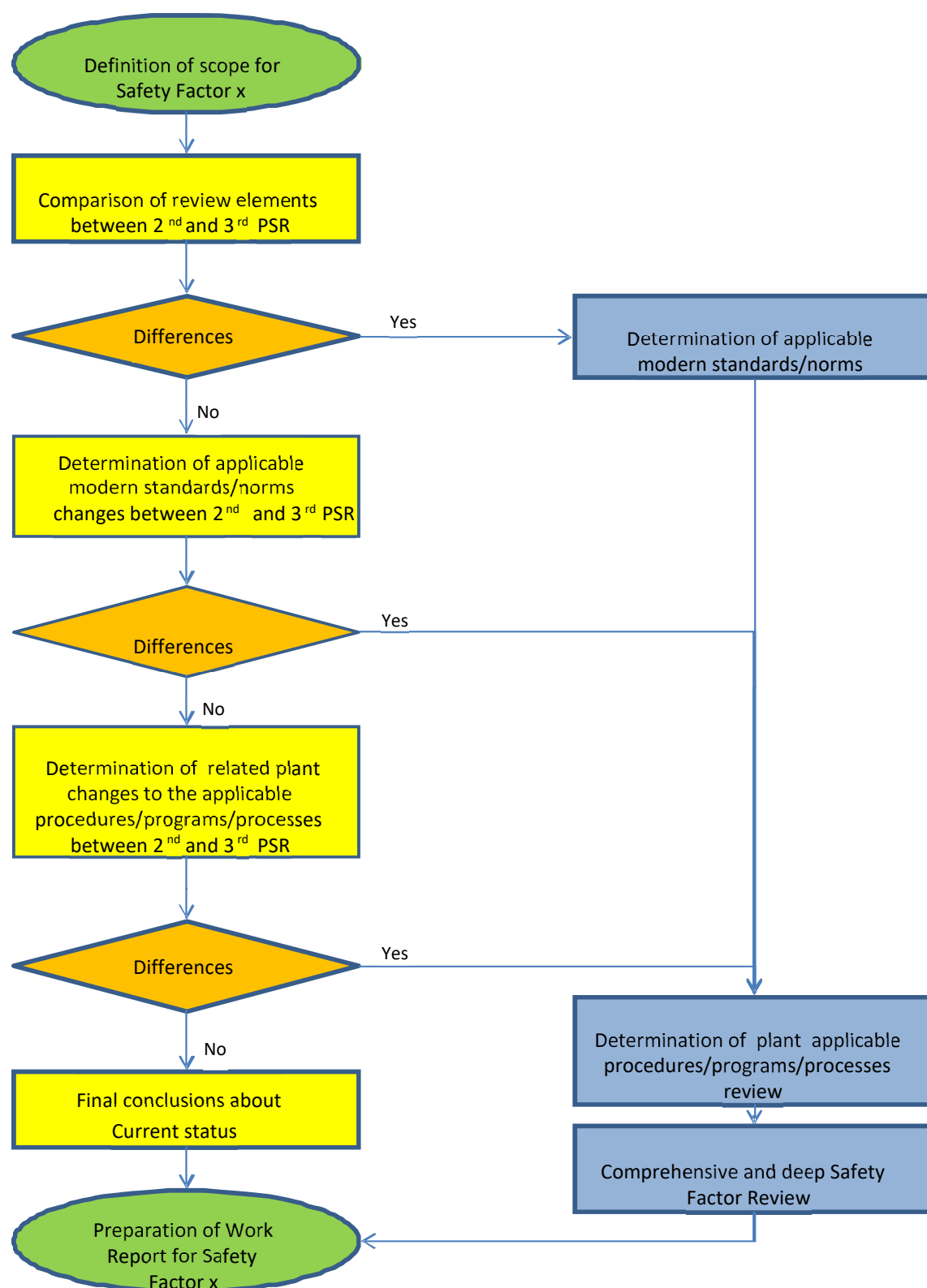


Figure 3-1: Safety Factor Assessment Process

3.3.1.1. Broad Objectives of the Review

In broad terms, the objectives specific to this step of the review can be defined as follows:

1. To achieve a good understanding of the current design and operational status of the plant, and to confirm that this is substantiated by the supporting documentation;
2. To confirm that the current design and operation of the plant reflect good practice;
3. To confirm that suitable management policies, procedures and other measures are in place to support safe operation of the plant;
4. To confirm that the current safety analysis is comprehensive, suitably validated, reflects good practice and demonstrates that the plant is safe;
5. To confirm that the design and operation of the plant are consistent with each other and with the assumptions and requirements of the safety analysis.

In addition to the specific objectives identified for each safety factor, these broad objectives should be born in mind throughout the review process.

3.3.1.2. Review Elements

A list of the elements to be reviewed shall be done in accordance to SSG-25 Chapter 5 and Annex of SSG-25 [20] as already mentioned in the Section 3.2. These elements are generic and include all aspects relevant to safety of NPPs.

It should be noted that many of these elements have an association with more than one Safety Factor. SSG-25 [20] in the Appendix I shows the main interfaces for all Safety Factors and emphasizes the importance of interface consideration. This matrix was expanded for the scope required by JV9 and presented in the Table 3-1.

For each safety factor from the first column, marked are safety factors to which the considered factor provides an input. This interface matrix is not exhaustive. Some other interfaces may also be found. (One may claim that every Safety Factor is, theoretically, interfaced with every other Safety Factor.) However, the Table 3-1 is considered to show the main interfaces for all Safety Factors. Awareness of these interfaces should prevent that common issues and subjects of review are overlooked during review and that interfacing areas between different Safety Factors can overlap.

During the review process additional links could be found between factors or elements or some review elements may be covered from different perspectives (in different safety factors). However, it should be noted that interfaces related to inputs and outputs between safety factors need to be discussed and appropriately covered if PSR will be divided between several PSR teams (contractors) to prevent that some of review elements will be forgotten by mistake in planning and contracting process.

Table 3-1: Safety Factors Interface Matrix

Safety factors receiving input	Safety factors providing input																	
	(1) Plant Design	(2) Actual condition of SSCs	(3) Equipment qualification	(4) Aging	(5) Deterministic safety analysis	(6) Probabilistic safety analysis	(7) Hazard analysis	(8) Safety performance	(9) Use of experience from other plants and research findings	(10) Safety Management Systems and Organization	(11) Safety culture	(12) Procedures	(13) Human factor	(14) Emergency planning	(15) Radiological impact on the environment	(16) Radioactive waste and spent fuel	(17) Security	(18) Radiation protection
(1) Plant Design		X	X	X	X	X	X	X		X	X	X		X	X	X		X
(2) Actual condition of SSCs	X		X	X	X	X	X			X	X	X		X	X	X		X
(3) Equipment qualification	X	X		X	X	X	X							X				
(4) Aging	X	X	X									X		X		X		
(5) Deterministic safety analysis	X	X	X	X												X		X
(6) Probabilistic safety analysis	X	X	X	X	X											X		X
(7) Hazard analysis	X	X	X		X	X										X		X
(8) Safety performance	X	X			X	X	X		X	X	X	X	X		X	X		X
(9) Use of experience from other plants and research findings		X								X	X				X	X		X
(10) Safety Management Systems and Organization		X		X	X			X	X		X	X			X	X		X
(11) Safety culture		X		X	X			X	X			X			X	X		X
(12) Procedures	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X		X
(13) Human factor	X	X	X	X	X	X	X	X	X	X	X	X						
(14) Emergency planning	X				X	X	X	X	X	X	X							
(15) Radiological impact on the environment	X	X				X		X	X			X						X
(16) Radioactive waste and spent fuel	X	X				X		X	X			X		X				X
(17) Security																		
(18) Radiation protection	X			X					X	X	X			X	X	X		

3.3.1.3. Assessment Approach

The assessment concentrates on the broad objectives (as described above). The following aspects are considered:

- understanding of the plant status,
- following good practices in the design and operation,
- adequacy of measures supporting plant operation,
- demonstration of plant safety through safety analysis,
- and consistency of design, operation and safety analysis.

Aspects related to supporting programs, which have been developed and implemented at the plant to ensure safe operation, will also be addressed.

A specific methodology or ‘checklist’ of points can be given, but in other cases only general guidance can be provided. The latter means that reviewers will often have to apply their own judgment of what is important based upon the identified objective, the general guidance and their previous findings.

3.3.1.4. Preliminary Ranking of Safety Issues

Preliminary ranking of safety issues and the identification of potential corrective measures are conducted initially during the assessment of the current safety status of the plant. This process will be repeated and extended during the global assessment. The preliminary ranking of the safety issues and the identification of the corrective measures conducted at this step will be the main responsibility of the PSR analyst / reviewer who identifies the issue. The NPP staff most familiar with the design and operation of the plant will provide input to the identification of the potential corrective measures and to the assessment of their feasibility / impact on normal plant operation.

For each safety factor a list of findings will be established of which a subset will be considered safety issues. Some of the findings and issues may refer to aspects of the design or operation of the plant where the basic requirements have been exceeded.

Preliminary ranking is based on the principles of broad ranking, described in Section 3.3.2.2. It is done by the reviewer assigned to the particular safety factor. Preliminary rank is assigned to each issue in the topical report devoted to the corresponding safety factor and it is an input into the final issue ranking process.

3.3.2. Ranking of Safety Issues

3.3.2.1. Introduction to PSR Issues Ranking

The review with respect to particular safety factors concentrates on identification of safety concerns trying to identify those safety issues that are sufficiently serious that either immediate remedial action or some other interim measure is considered necessary. The methodology adopted for this, first, phase (as described in Section 3.2 above) is tailored to identify any gaps or issues that may exist regarding the specific safety factor when current standards and good practices are considered.

The process of ranking of safety issues identified during the review of particular safety factors, involves certain degree of subjective expert judgment and the subdivision of issues into predefined categories of significance is performed.

The starting point for the ranking process should be a clear written statement on each safety issue. This should as a minimum include the following:

- A title and identification reference,
- The best possible description of the issue,
- The safety factor (or factors) to which the issue relates,
- Other design and operational features that have a bearing on the safety issue,
- Any other information that might aid further consideration of the issue.

In evaluating the safety significance of any PSR issue, it is useful to consider its impact on the defense in depth (DID). This impact can be evaluated qualitatively, to various degrees of details, or quantitatively. The PSA impact can, for example, be considered as a quantitative evaluation of impact on defense in depth. This is because the DID barriers and functions are directly represented as function events in the event tree sequences in the PSA model. Therefore, any degradation / unavailability of a DID element would reflect as an increase of a quantitative risk measure (e.g. CDF) calculated by means of a PSA model. Regardless of particular technique applied, the safety significance of considered issue must be the measure of its impact on the DID.

‘Defense in depth’ is defined as a philosophy which ensures that successive measures are incorporated into the design and operating practices for nuclear plants to compensate for potential failures in protection and safety measures. This idea of multiple levels of protection is the central feature of defense in depth.

The principle of defense in depth is implemented primarily by means of a series of barriers which should in principle never be jeopardized, and which would have to be violated in turn before harm could occur to people or the environment. These barriers are physical, providing for the confinement of radioactive material at successive locations. The barriers may serve operational and safety purposes, or may serve safety purposes only. Power operation is only allowed if this multi-barrier system is not jeopardized and is capable of functioning as designed.

The physical barriers to the release of radioactivity include the fuel matrix and cladding (barrier 1), the primary coolant pressure boundary (barrier 2), and the containment structure (barrier 3). The reliability of the physical barriers is enhanced by applying the concept of defense in depth to them in turn. In this concept each of them is protected by a series of measures.

Each physical barrier is designed conservatively, its quality is checked to ensure that the margins against failure are retained, its status is monitored, and all plant processes capable of affecting it are controlled and monitored in operation. Human aspects of defense in depth are brought into play to protect the integrity of the barriers, such as quality assurance, administrative controls, safety reviews, independent regulation, operating limits, personnel qualification and training, and safety culture. Design provisions including both those for normal plant systems and those for engineered safety systems help to prevent undue challenges to the integrity of the physical barriers, to prevent the failure of a barrier if it is jeopardized, and to prevent consequential damage of multiple barriers in series. Safety system designers ensure to the extent practicable that the different safety systems protecting the physical barriers are functionally independent under accident conditions.

The protection measures are applied at five levels of defense. These are:

- Level 1. Prevention of abnormal operation and failures (*Prevention*)
- Level 2. Control of abnormal operation and detection of failures (*Control*)
- Level 3. Control of accidents within the design basis (*Protection*)
- Level 4. Control of severe conditions including prevention of accident progression and mitigation of the consequences of a severe accident (*Accident management and mitigation*)
- Level 5. Mitigation of the radiological consequences of significant external releases of radioactive materials (*Off-site counter-measures*).

The approach to safety issue ranking in the NEK PSR is developed in such a way as to be able to address and cover the issues coming from the review of all PSR safety factors.

In its essence, the approach is based on assessing the depth of remaining defense or remaining mitigation capability, provided that considered safety issue remains unaddressed. This approach is described in detail in reference [41]. The approach maps the issue of concern into the defense-in-depth structure (e.g. by failing or reducing the affected barrier capability) or into the accident sequences (e.g. by failing or reducing the affected mitigation function capability). Thus, both deterministic and probabilistic methods can be used.

The principles and elements of this approach were, to various extents and levels of detail, used worldwide.

The approaches where remaining defense depth is estimated by counting of levels of defense were used to assess the safety of existing nuclear power plants and their elements are described in the IAEA publications such as [80], [81] and [77].

Similarly, the approaches where remaining mitigation capability is estimated (qualitatively, in terms of orders of magnitude) by mapping of the issue of concern into the affected accident sequences are used in the US NRC Significance Determination Process (SDP), described in references [82], [83] and [84].

Also, it should be pointed that similar approach is used in the industry risk informed applications (recognized also by the regulators) such as risk informed in-service inspection to determine the remaining mitigation capability (or conditional risk) following an assumed failure or degradation of a pipe segment.

The overall process of PSR issues ranking in NEK PSR3 will be divided into two major steps:

- 1. Broad ranking (*preliminary and final*);**
- 2. Detailed ranking**

Broad ranking is done in two phases: preliminary and final. The preliminary rank (importance category) is assigned by a reviewer devoted to the specific safety factor who is the originator of an issue of concern (section 3.3.1.4) The preliminary ranks of all identified issues are provided in the topical report for the corresponding factor. When the topical reports (safety factor reports) are available for all the safety factors, all issues, together with assigned preliminary ranks, are collected and input into the final issue ranking process. This process is

carried out by the group of analysts which is devoted to issue ranking / corrective measures prioritization.

The first step is a final broad ranking, by which all PSR issues will be classified into three general categories of importance: High (H), Medium (M) and Low (L). The results of the broad ranking will be used for the initial pre-screening of the issues which will be input directly to Corrective Action Program (CAP). Pre-screened and directly input to CAP will be issues of the two types:

- Issues including Technical Specification violation, violation of current licensing basis or any found shortfall in plant design (or actual condition of an SSC) which requires a Justification for Continued Operation (JCO). These issues would require immediate attention and would further be processed through the CAP. In case that mandatory action was implemented during the PSR project it will be reported latter in the same way as other PSR issues. These are expected to be the majority of issues with high (H) importance.
- Issues desirable to be resolved, which can be resolved at minimum effort and in a short time frame. Although these can come from any importance category, it is expected that most of them would come from low significance (L) category, relating to matters such as procedure changes, corrections to plant drawings or documents and similar. Requiring the minimum effort, they will be further dealt through the CAP mechanisms and procedures.

Pre-screened issues (expected to be mostly from H category and L category with minimum effort and short implementation time) will be added to the CAP directly. This (as well as other PSR-related) CAP items will be flagged as “PSR Issues” or similar means for tracking of the status of PSR issues implementation will be provided.

All remaining issues (i.e. those surviving pre-screening) will be subject to detailed ranking (expected to be, mostly, M category L category with longer implementation time issues), which is the second step in the final issue ranking process. Based on the results of detailed ranking, particular issues will then be added into the CAP. The approach is illustrated by Figure 3-2.

Note that, issues with minimum effort and short implementation time can come from any importance category and this is indicated in the figure. In reality, however, it is expected that most of those would come from L category, as pointed in the above discussion.

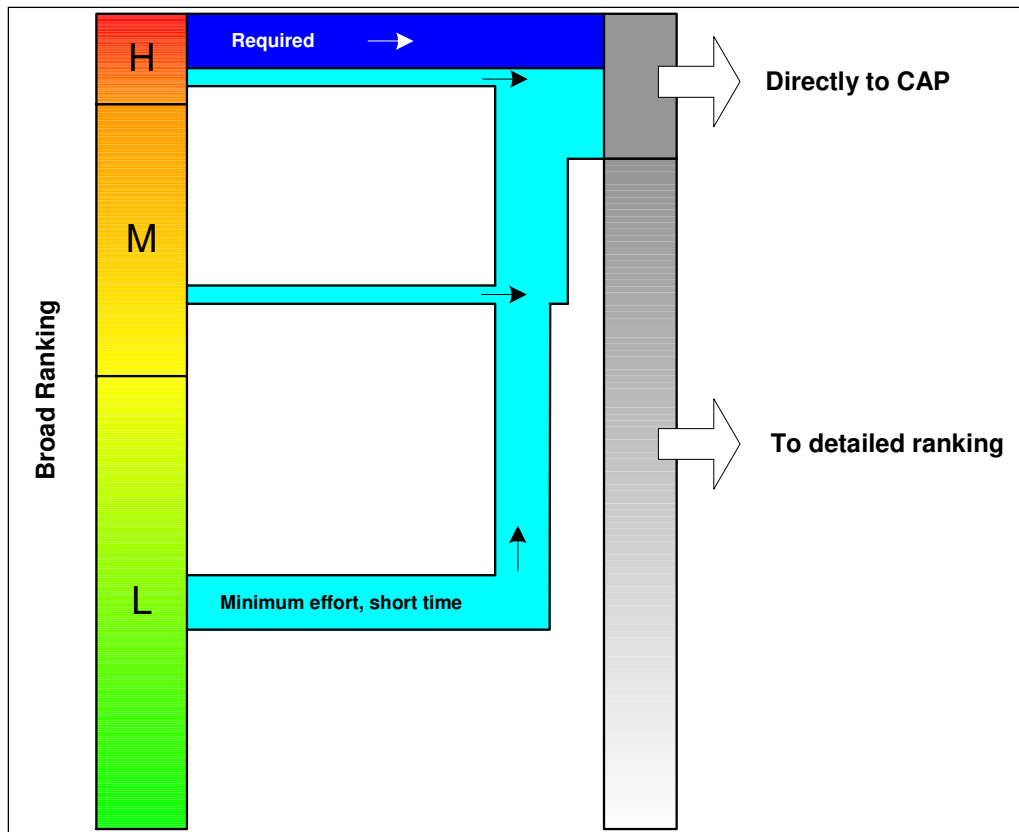


Figure 3-2: Overall Approach to PSR Issue Ranking

For the purpose of detailed ranking, each issue will first be characterized in terms of the three general attributes:

- Direct link to nuclear safety (whether, for considered issue, a direct link to nuclear safety can be established);
- Re-evaluation of nuclear safety basis (whether an issue represents re-evaluation of nuclear safety basis);
- Non nuclear safety issue or issue related to “soft factors”.

Each issue should be related to at least one of these three general attributes. Some issues may relate to more than one.

For each general attribute, a separate ranking path applies. Ranking with respect to certain general attribute results with certain severity number (referred also to as a rank). Severity number is assigned by evaluating the issue of concern through three layers of ranking, with respect to primary attributes (first layer), second layer attribute and third layer attribute. In the case that more than one ranking path applies, the highest severity number is taken as representative.

The detailed ranking process is shown in Figure 3-3 and Figure 3-4.

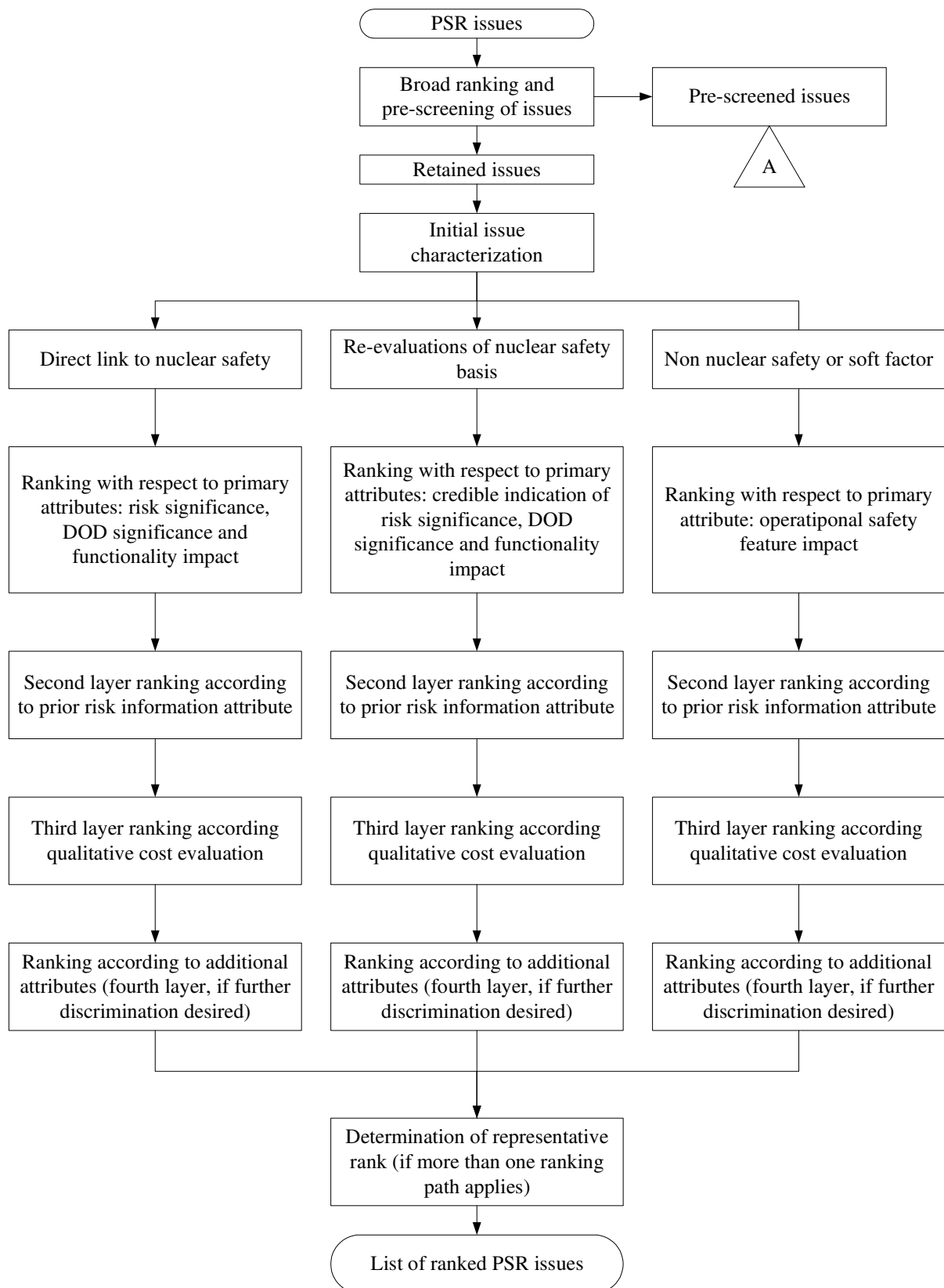


Figure 3-3: Detailed Ranking Process

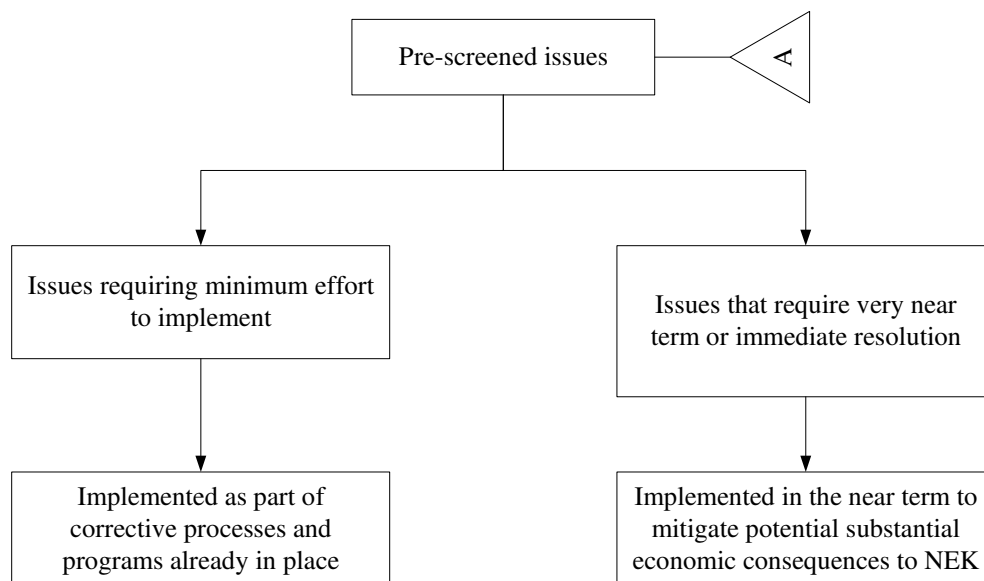


Figure 3-4: Detailed Ranking Process (continued)

3.3.2.2. Broad Ranking

As a result of the review of particular safety factors, certain number of observations are expected to be made regarding specific factors. These observations are broadly ranked by judgment, based on the principles outlined below. The approach is reliant on expert judgment and is considered suitable for a broad ranking. Assessment criteria used are derived from the defense in depth concept.

These principles (i.e. the principles of “broad ranking”) are applied already in the “preliminary ranking” of issues. However, the preliminary ranking is done during the plant status review, by the reviewer devoted to the respective safety factor. The preliminary rank is assigned to each issue in the corresponding topical report. All the issues, together with preliminary ranks, are then collected for the final broad ranking (which is done by the analysts devoted to the ranking and prioritization of issues and corrective measures). It should be noted that, in this stage, the preliminary rank of an issue can be reconsidered / adjusted for the reasons such as:

- Importance of an issue (in terms of its impact on the DID) may differ when seen in the light of some other issues, from other safety factors (the information which may have not been known to the reviewer of the particular safety factor, which assigned the preliminary rank);
- Several issues from different safety factors (topical reports) may, e.g., have the same underlying cause, and may be combined into a single issue for the purpose of final ranking / definition of corrective measures.

Following below is the description of broad ranking principles.

3.3.2.2.1 Safety significance categories

In the broad ranking process, subdivision of significance into three categories is used:

- High importance – issues requiring immediate attention and specification of corrective actions. In most cases, these would be the issues including Technical Specification violation, violation of current licensing basis or any found shortfall in plant design (or

actual condition of an SSC) which requires a Justification for Continued Operation (JCO).

- Medium importance – interim measures acceptable, coupled with other corrective measures in the longer term, considering also practicability arguments.
- Low importance – continued operation permitted. Some form of corrective measures in the longer term recommended, depending on the practicability arguments.

This subdivision is considered realistic and also corresponds to the levels of action possible.

3.3.2.2.2 Assessment Approach

The method and criteria are derived from defense in depth concept. In this approach the safety of a nuclear power plant is considered to depend on:

- (1) The Postulated Initiating Events (PIEs) and hazards being adequately identified and analyzed (and included within the design basis);
- (2) The design being sufficiently robust and including multiple physical barriers to the release of radioactivity;
- (3) The design and operation of the plant including multiple levels of defense against plant upset conditions;
- (4) The staff training and operational performance matching best practice;
- (5) The safety culture, as exemplified by the attitude of the management and staff, promoting safe operation.

Each safety issue is assessed against the above 5 broad criteria, and in the case of criteria (2) and (3) this assessment considers the impact of the issue on each ‘barrier’ and each ‘level of defense’ individually.

Table 3-2, which is based upon these 5 assessment criteria, provides a methodology for ranking the safety issues into ‘low’, ‘medium’, and ‘high’ above mentioned categories of significance to safety. It should be noted that when more than one objective criterion is applicable to the issue, the highest safety significance ranking is assigned to the issue.

Table 3-2: Assessment Criteria for Broad Ranking of Safety Issues

Ref. #	Description of assessment criteria	Safety significance		
		Low	Medium	High
1	Impact on plant risk of a new PIE or of increased frequency of a PIE	Small	Significant	Major
2	Impact of issue on a physical barrier to the release of radioactivity	Affected	Degraded	Seriously degraded
3	Impact of issue on one or more levels of defense	Affected	Significantly affected	Lost
4	Level of staff training and operational performance	Warrants improvement	Inadequate	Unacceptable
5	Level of safety culture	Warrants improvement	Inadequate	Unacceptable

Assignment of categories for the above 5 broad criteria involves an engineering judgment. For criteria (1), (2), (4) and (5) the categorization is relatively straightforward.

However, additional explanations are needed for criterion (3) that addresses the degradation of all levels of defense.

The levels of defense aim firstly to prevent damage to the barriers and the plant, and secondly to mitigate the consequences of any damage. The assessment therefore involves a judgment of the effectiveness of the primary safety functions (controlling the reactivity, cooling the reactor core, confining the radioactive material).

First, the capability of safety function affected by considered issue is categorized as ‘robust’, ‘adequate’, and ‘inadequate’. The specific criteria used in the above-mentioned categorization are defined in Table 3-3.

Table 3-3: Primary Safety Function Capability

<i>‘Robust’</i>	Where the primary safety function can still be achieved with redundancy and diversity comparable to the design basis, and with sufficient margin for the design capacity and protection against common cause failure (CCF).
<i>‘Adequate’</i>	Where the primary safety function can still be achieved with full capacity, but without the complete redundancy, protection against CCF, diversity or design margins associated with current standards and practices.
<i>‘Inadequate’</i>	Where the primary safety function cannot be achieved, or is unlikely to be achieved

Next, considering the capabilities of safety functions, the performance of levels of defense under accident conditions is mapped into one of the categories: ‘affected’, ‘significantly affected’, and ‘lost’. The categories are defined in Table 3-4. Design Basis Accident (DBA) have, statistically, higher level of occurrence than Beyond Design Basis Accidents (BDBA), therefore are ranked with higher category of Level-of-Defense degradation in the mentioned table.

Table 3-4: Degradation of a Level of Defense

Issue	Level of Defense	Impact
Low	Affected	Issue affects function capability in certain <u>design basis (DB)</u> accident sequences. However, capability is still “ <u>robust</u> ”. OR Issue affects function capability in certain <u>beyond design basis (BDB)</u> accident sequences. However, capability is still “ <u>adequate</u> ”.
Medium	Significantly affected	Issue affects function capability in certain <u>DB</u> accident sequences. However, capability is still “ <u>adequate</u> ”. OR Issue affects function capability in certain <u>BDB</u> accident sequences and its capability is considered “ <u>inadequate</u> ”.
High	Lost	Issue affects function capability in certain <u>DB</u> accident sequences and its capability is considered “ <u>inadequate</u> ”.

Each PSR issue will be assessed against the 5 criteria presented in Table 3-2. In the case of item (2) and (3) consideration will be given to the impact of the issue on each of the three barriers and each of the levels of defense individually. The levels of defense (item 3) aim firstly to prevent damage to the barriers and the plant, and secondly to mitigate the consequences of any damage. In general terms, assessment of all 5 criteria can be undertaken using a deterministic approach. Additionally, assessment against criterion (1) can be undertaken with the assistance of probabilistic techniques.

It should be noted that the impact of new PIEs, or of PIEs with an increased frequency of occurrence (criterion 1), may be ‘bounded’ by that of other PIEs, or such PIEs may place additional demands on safety systems and the operations staff. Therefore, they may on the one hand cause minimal impact on the overall plant risk, or on the other hand constitute a significant new contributor to risk.

Based on the broad ranking, pre-screening of issues is performed.

Some identified safety issues may necessitate conditions that require immediate or very near-term resolution. For example, they may require that the plant promptly go into an outage or extend an outage in progress, a power reduction, or equipment damage or degradation which, as required by TS, results in a near term power reduction or outage. All such safety issues do not require further, detailed, ranking as they must be accomplished in the near term to mitigate potential substantial economic consequences to NEK. Most of the issues with significance declared as “high” in the broad ranking are expected to be pre-screened in this manner.

On the other hand, pre-screening also eliminates from consideration in the detailed ranking process all safety issues identified as desirable (note that this covers any kind of issues which are “required” to be implemented) and requiring minimum effort to implement. As an example

of the type of issues considered here, minor corrections to plant procedures that can be implemented as a part of corrective processes and programs that normally take place in the plant are screened from further consideration in the ranking process.

3.3.2.3. Detailed Ranking

All the issues which remained after pre-screening, based on the broad ranking, will be the subject of the detailed ranking.

In detailed ranking, each issue is first characterized in terms of the three general attributes:

- Direct link to nuclear safety (whether, for considered issue, a direct link to nuclear safety can be established);
- Re-evaluation of nuclear safety basis (Whether an issue represents re-evaluation of nuclear safety basis);
- Non nuclear safety issue or issue related to “soft factors”.

Each issue should be related to at least one of these three general attributes. Some issues may relate to more than one. Ranking with respect to each of the three general attributes is done through a separate ranking path (Figure 3-3). Each ranking path ends with certain severity number (referred also to as a rank). The severity (rank) scale is for all three paths “calibrated” in a way that particular severity number (e.g. “4”) in any of the three ranking paths corresponds to the same level of safety significance.

Severity number is, in any of the three paths, assigned by evaluating the issue of concern through three layers of ranking, with respect to primary attributes (first layer), second layer attribute and third layer attribute.

In the case that, for a specific issue, more than one ranking path applies, the highest rank is taken as a representative (i.e. as a final rank for this issue).

The process results with a list of PSR issues, ranked according to their safety significances.

3.3.2.3.1 Ranking of Issues where a Direct Link to Nuclear Safety Is Established

The following generic attribute categories are derived to allow for an efficient safety issue ranking method for NEK PSR issues where a direct link to plant safety can be established:

Primary attributes:

- Risk significance
- Depth of defense (DOD) significance
- Additional attribute Functionality

Second layer attribute:

- Prior risk information (historical information, precursor data)

Third layer attribute:

- Qualitative cost evaluation

Additional attributes can also be defined at fourth layer, should further discrimination be necessary.

The above attributes reflect both the deterministic and probabilistic safety evaluation principles discussed in the previous section.

In the first phase of the ranking process for issues where a direct link to risk or degradation of defense-in-depth is determined, a distinction is made as to the relative significance of the attributes.

A detailed evaluation of each safety issue identified as having a direct link to plant safety is performed to map each issue into one of the most significant attribute categories related to risk, defense-in-depth and system functionality.

Prior risk information and costs of resolution are not considered as significant as the three attributes directly related to plant safety. However, prior risk information is considered more significant than the costs of resolution since this attribute is related to the potential for precursor events which may result in risk or degradation of defense-in-depth.

A three-layer ranking method is utilized for determining the relative significance of the issues where first, risk, depth of defense and system functionality are evaluated; second, prior risk information is determined; and finally, qualitative cost categories are assigned.

The risk, degradation of DOD and system functionality evaluations provide the dominant discriminator between safety issues. In other words, if a safety issue is ranked above another based on the evaluation of risk, defense-in-depth and functionality alone, it retains its dominance independent of prior risk information and qualitative cost evaluation. For example, if a safety issue is identified as resulting in a ΔCDF greater than $1.0E-5/ry$, it is always ranked above another safety issue which results only in a system-level degradation of functionality but does not have a quantifiable impact on CDF.

Impact on system level functionality is questioned only for issues with small risk impact and small DOD impact. The issues with high risk / DOD impact by their nature would involve functionality degradation at one or several systems / functions.

The prior risk attribute is evaluated next and dominates the qualitative cost evaluation attribute. Finally, the qualitative cost evaluation provides the third discriminator for safety issues ranked identically based upon risk, degradation of defense-in-depth, functionality and prior risk information.

Once the attribute category and severity ranking within the attribute category are identified, no further evaluation need to be performed in this phase.

Severity ranking with respect to primary attributes of issues with a direct link to plant safety is defined in the Table 3-5.

3.3.2.3.1.1 Primary Attribute Risk Significance

Risk significance evaluation is performed by means of NEK PSA. It should be clear that risk-based evaluation is not applicable to issues of all types, i.e. some issue types cannot be mapped to PSA terms. Furthermore, it needs to be recognized that risk-based evaluation in PSA terms is a complex task and it cannot, from practical point of view, be performed for every issue. Therefore, a risk based evaluation of a PSR issue is NOT a requirement. However, if it is a choice of ranking analyst to perform a risk based evaluation of a particular issue, then the guidance from Table 3-5 applies.

It is expected that primary attribute normally used for the issue ranking will be the Depth-Of-Defense (DOD). The DOD determination is straightforward and simple. The detailed risk evaluation based on NEK PSA is expected to be done on selected cases in order to determine the risk impact of an issue which scored high severity based on simple DOD principle and thus obtain more realistic measure.

Since the risk based PSA evaluation is expected to be a detailed evaluation, as compared to the simple and basic DOD technique, the risk significance result (where available) would take precedence over DOD significance.

Table 3-5: Risk Significance Attribute

State	Δ CDF and Δ LERF criteria,/yr	Severity Number
Very high risk significance	Δ CDF $> 10^{-04}$ or Δ LERF $> 10^{-05}$	6
High risk significance	$10^{-05} < \Delta$ CDF $< 10^{-04}$ or $10^{-06} < \Delta$ LERF $< 10^{-05}$	5
Moderate risk significance	$10^{-06} < \Delta$ CDF $< 10^{-05}$ or $10^{-07} < \Delta$ LERF $< 10^{-06}$	4
Low risk significance	Δ CDF $< 10^{-06}$ and Δ LERF $< 10^{-07}$	1-3, depending on functionality attribute

The question which PSR issue ranking analyst asks is: *Assuming that an issue remains non-addressed, how much would it change current base case (long term averaged) risk estimate (in terms of Δ CDF and / or Δ LERF, as applicable)?*

3.3.2.3.1.2 Primary Attribute Depth of Defense (DOD) Significance

Depth of Defense (DOD) attribute can acquire one among the four states defined by Table 3-6. The background for these four states is provided in ESD-TR-17/13 [41], which describes the DOD method in detail.

The DOD is simple and straightforward method which relies on criteria for acceptability that are based on the number of lines of defense, their strength, and the consequences of their failure. These criteria are thus equivalent in concept to risk criteria, but are based on deterministic principles

Table 3-6: Depth of Defense Significance Attribute

State	Description	Severity Number
NT	Not Tolerable. Operation without resolving the issue is not justified. Resolution required in the shortest time.	6
TS	Tolerable Short term. Operation is justified for a limited time while the issue is being resolved, possibly with compensating measures.	5
TL	Tolerable Long term. Operation is justified in longer time frame. However, the issue needs to be addressed in longer term in terms of additional arguments, analyses or compensatory measures. Practicability arguments are considered.	4
MS	Modern Standard With respect to DOD, plant is considered being in compliance with modern standards. The issue is limited to particular system or function.	1-3, depending on functionality attribute

In the case of PSR issue significance evaluation, the question which ranking analyst asks is: *Assuming that an issue remains non-addressed, how many lines of defense (LOD) would remain in the critical accident sequence, i.e. which would be the smallest depth of defense for a considered consequence?*

3.3.2.3.1.3 Additional Primary Attribute Functionality Impact for Issues within “MS” Category

In the case that issue maps into “MS” DOD category (or risk significance category “ $\Delta CDF < 10^{-06}$ and $\Delta LERF < 10^{-07}$ ”), it is limited to particular function or system (or it may, even, be multiple systems) but without having relevant impact on nuclear safety (due to adequate number of remaining LODs). Issues in this category are additionally discriminated by functionality attribute. The severity scale is provided by Table 3-7.

Table 3-7: Functionality Impact Attribute

Status	Severity Number
Affects safety system or feature involved in <u>design basis</u> (DB) accident sequence	3
Affects system or feature involved in <u>beyond design basis</u> (BDB) accident sequence	2
Impact limited to non-safety system not involved in BDB accident sequence	1

3.3.2.3.1.4 Ranking of Issues with a Direct Link to Plant Safety with Respect to Primary Attributes

With respect to primary attributes (risk significance, DOD significance and functionality), the issues are ranked into 6 categories according to Table 3-8.

Within each categories, the issues are further discriminated by means of second layer (prior risk information) and third layer (qualitative cost evaluation) attributes.

Table 3-8: Ranking of Issues with a Direct Link to Plant Safety with Respect to Primary Attributes

Risk significance (if available)	DOD Significance	Functionality Impact	Severity Number
$\Delta\text{CDF} > 10^{-04}$ or $\Delta\text{LERF} > 10^{-05}$	NT		6
$10^{-05} < \Delta\text{CDF} < 10^{-04}$ or $10^{-06} < \Delta\text{LERF} < 10^{-05}$	TS		5
$10^{-06} < \Delta\text{CDF} < 10^{-05}$ or $10^{-07} < \Delta\text{LERF} < 10^{-06}$	TL		4
$\Delta\text{CDF} < 10^{-06}$ and $\Delta\text{LERF} < 10^{-07}$	MS	Affects safety system or feature involved in DB accident sequence	3
		Affects system or feature involved in BDB accident sequence	2
		Impact limited to non-safety system not involved in BDB accident sequence	1

3.3.2.3.2 *Ranking of Issues Representing Re-evaluation of Nuclear Safety Basis*

Issues involving re-evaluations of the nuclear safety basis and potential identification of new risk, degradation of defense-in-depth or degradation of system functionality are ranked separately from issues where a direct link can be established to the nuclear safety.

The re-evaluation of safety basis is related to potential modifications to plant safety evaluations, which may result in identification of new risk, identification of weakness in defense-in-depth or identification of issues with system level functionality.

Following are examples of categories of safety issues which can imply the re-evaluation of safety:

- Probabilistic safety assessment
- Hazard analyses
- Deterministic design basis accident analysis
- Re-evaluation of anticipated operational occurrences
- Updating information on local meteorological conditions
- Updating information on off-site population distribution

The following generic attribute categories are derived to allow for an efficient issue ranking method for NEK PSR issues where a re-evaluation of the nuclear safety basis is recommended:

Primary attributes:

- Credible indication of risk significance (potential identification of new risk),
- Credible indication of DOD significance
- Discrepancy regarding international standards for nuclear safety evaluation,
- Credible indication of system functionality issues

Second layer attribute:

- Prior risk information (historical information, precursor data)

Third layer attribute:

- Qualitative cost evaluation

Additional attributes can also be defined at fourth layer, should further discrimination be necessary.

In the first phase of the ranking process for issues involving a re-evaluation of the nuclear safety basis only, a distinction is made as to the relative significance of potential changes to the plant risk profile, DOD and/or system functionality. Table 3-9 provides a severity ranking scale for the re-evaluation of the nuclear safety basis attribute.

Similarly to the issues with direct link to nuclear safety, it is pointed that obtaining a an indication of risk significance of a particular PSR issue is NOT a requirement. However, if it is a choice of ranking analyst to obtain such an indication (which may require to perform a limited risk based evaluation or PSA for a particular issue), then the guidance from Table 3-9 applies.

Also, since the risk based PSA evaluation (for obtaining risk indication) is expected to be a more detailed evaluation, as compared to the simple and basic DOD technique, the risk indication result (where available) would take precedence over DOD significance.

Table 3-9: Ranking of Issues Representing Re-evaluation of the Nuclear Safety Basis with Respect to Primary Attributes

Credible indication of risk significance (if available) <u>Or</u> discrepancy regarding international standards for nuclear safety evaluation	Credible indication of DOD significance <u>Or</u> discrepancy regarding international standards for nuclear safety evaluation	Credible indication of functionality impact	Severity Number
$\Delta\text{CDF} > 10^{-04}$ or $\Delta\text{LERF} > 10^{-05}$	NT		5
$10^{-05} < \Delta\text{CDF} < 10^{-04}$ or $10^{-06} < \Delta\text{LERF} < 10^{-05}$ or major discrepancy regarding international standards for nuclear safety evaluation	TS or major discrepancy regarding international standards for nuclear safety evaluation		4
$10^{-06} < \Delta\text{CDF} < 10^{-05}$ or $10^{-07} < \Delta\text{LERF} < 10^{-06}$ or discrepancy regarding international standards for nuclear safety evaluation	TL or discrepancy regarding international standards for nuclear safety evaluation		3
$\Delta\text{CDF} < 10^{-06}$ and $\Delta\text{LERF} < 10^{-07}$	MS	Affects safety system or feature involved in DB accident sequence	2
		Affects system or feature involved in BDB accident sequence or non-safety system	1

Similarly to the ranking of issues with direct link to nuclear safety, a three-layer ranking method is utilized for determining the relative significance of the attributes where first, a qualitative evaluation of the potential change in the plant risk profile, DOD or system functionality is performed; second, prior risk information is determined; and finally, qualitative cost categories are assigned. The potential change in the plant risk profile, DOD or system functionality provides the dominant discriminator between safety issues which represent nuclear safety basis re-evaluation only. In other words, if a safety issue is ranked above another based on the

evaluation of potential change to the plant risk profile (DOD or system functionality) alone, it retains its dominance independently of prior risk information and qualitative cost evaluation.

For example, if a safety issue is identified as having a potential to result in a significant change in the plant risk profile, it is always ranked above another safety issue which has a potential only to affect particular system functionality without quantifiable impact on risk. The prior risk attribute is evaluated next and dominates the qualitative cost evaluation attribute. Finally, the qualitative cost evaluation provides the third discriminator for safety issues.

In the second and third phases, the evaluation of prior risk information and qualitative cost is performed as described for the issues with direct link to nuclear safety. Also, as was the case for the evaluation of issues identified with a direct link to nuclear safety, additional attributes are evaluated (if desired) only in the event that sufficient discrimination has not resulted from the ranking of issues based on the re-evaluations of the safety basis, prior risk information, and qualitative cost evaluation attributes.

3.3.2.3.3 Ranking of Non Nuclear Safety Issues and Issues Related to “Soft Factors”

Non nuclear safety related issues are those which relate to factors such as occupational hazards (e.g. hazards to plant workers), including aspects of radiation protection and waste management.

Often termed “Soft Factors” are the factors such as:

- Organization, management systems and safety culture
- Human factors, including, to some extent, also plant procedures
- Use of experience from other plants and research findings.

The above types of issues are ranked in terms of impact they have on the features provided to ensure safe plant operation. These features are referred to as “Operational Safety Features” (OSF). For evaluating the significance of this issue type, the following four OSF categories are considered:

- Operating Organization
- Normal Operating and Administrative Procedures
- Safety Management Systems
- Radiological Protection and Other Occupational Hazards

Assessment of significance of considered issue is based on its impact on relevant OSFs. Considered issue can relate to one or to multiple OSFs among the four OSFs specified above. Each OSF is considered to consist of certain elements which must be provided to ensure its functionality.

Following below is description of elements of each of the four OSFs.

- Operating Organization
 - The operating organization should exert full responsibility for the safe operation of the NPP through a strong organizational structure, under the line authority of the Station Manager. The Station Manager ensures that all elements for safe operation are in place, including an appropriate complement of Suitably Qualified and Experienced Persons.

-
- Safety review procedures should be maintained by the operating organization to provide a continuing surveillance and audit of plant operational safety, with feedback and corrective action programs.
 - Operation of the plant must be conducted by authorized personnel, according to strict administrative controls and observing procedural discipline.
 - Programs must be in place for training and retraining for operations, maintenance and specialist technical support staff, to enable them to perform their duties safely and efficiently. Training is particularly intensive for control room staff, and involves the use of plant simulators and some form of ongoing assessment.
 - A change review process should be in place at the NPP to carefully assess any changes to plant design, operation, or procedures prior to implementation, to ensure they are not detrimental to safety.
 - Normal Operating Procedures
 - Normal operation is controlled by detailed, validated and formally approved procedures. These procedures should cover all plant activities including plant configuration management, response to alarms and indications, plant security matters, refueling and fuel route operations, outage activities, as well as the key operational stages of start-up, full and low-power operation, and controlled shutdown.
 - The activities representing regular preventive and corrective maintenance, inspection, condition monitoring and surveillance testing of safety-related SSCs must be carried out in accordance with written procedures that are detailed, validated and formally approved.
 - Safety Management Systems
 - A strong safety culture should be established, implemented and continuously monitored. This should be achieved through policy requirements, visible management commitment and individual responsibility
 - Measures must be in place to ensure that events significant or potentially significant to safety are detected and evaluated in depth, and that any necessary corrective measures are taken promptly and information on them is disseminated. This applies both to events occurring at the NPP in question, and also related experience from other similar NPPs around the world.
 - An operational Quality Assurance program must be established and implemented by the operating organization to assist in ensuring satisfactory performance in all plant activities important to plant safety.
 - A structured Self-Assessment process should be in place for all safety related activities, ensuring that personnel involved in these activities detect problems concerning safety and performance, and have responsibility for correcting those problems in the first instance.
 - Peer review programs should involve periodic review of operational practices and programs by expert personnel independent of the NPP operating organization

-
- Radiological Protection and Other Occupational Hazards
 - Radiation protection procedures must be established and implemented, for the control, guidance and protection of personnel. The radiation protection staff of the plant must carry out routine monitoring of on-site radiological conditions at all accessible locations, and monitor the short term and cumulative exposure of personnel to radiation.
 - The individual risk of fatality to any worker on the NPP attributable to doses of radiation received should be shown to be As Low As Reasonably Achievable (ALARA) and in accordance with national regulations and international guidance documents.
 - Procedures for normal operation (including preventive and corrective maintenance, inspection, condition monitoring and surveillance testing activities) should be established in a way to avoid or minimize risk from any hazard to plant personnel associated with performed activities.

Therefore, the significance of an issue is assessed and ranked on the basis of the impact that issue has on one or more of the elements of relevant OSFs. The evaluator considers the impact is from the two aspects:

- Whether an element of relevant OSF is appropriately provided (e.g. whether an adequate procedure or program or database is provided), and
- Whether provided OSF element is appropriately complied with (e.g. whether a procedure or program is appropriately implemented and a database appropriately administered).

With respect to the first aspect, severity number (rank) is assigned depending on whether an element is missing or there is a gap or there is a minor area which warrants improvement, etc (in decreasing order of severity / rank).

With respect to the second aspect, severity number is assigned depending on whether there is a widespread failure to comply or there are frequent failures to comply or there are occasional failures to comply, etc. (in decreasing order).

The primary attribute for issue ranking is referred to as “Operational Safety Feature Impact”. The ranking with respect to this attribute is performed according to Table 3-10.

Table 3-10: Primary Attribute Operational Safety Feature Impact (Ranking of Non Nuclear Safety Issues and Issues Related to “Soft Factors”)

Operational Safety Feature Impact	Severity Number
There is a significant shortfall in the provision of the Operational Safety Feature identified, a major element is missing or is so poorly implemented that it will not provide the element of Operational Safety that was intended or could even be detrimental to Operational Safety. OR There is widespread failure of compliance with the Operational Safety Features provided.	5
There are gaps in the provision of the Operational Safety Feature identified, or there are topic areas that are not fully addressed, or where procedures are poorly established or confusing. OR There are frequent failures in the level of Operational Safety Feature procedural compliance.	4
There are minor areas in which the provision of Operational Safety Feature could be improved in order to bring the Feature in line with best international practice, such that Operational Safety would be enhanced by the improvement. OR There are small and occasional deficiencies in Operational Safety Feature procedural compliance.	3
Minor issue related to Operational Safety Feature which does not influence safety-related SSC, plant security or workers safety	2
Impact limited to operational excellence, external confidence or public image of the plant	1

Operational excellence is evaluated by identified violations of procedures, deviations from defined programs, failure to meet goals, and other NEK performance indicators. External confidence relates to addressing comments from international missions such as those conducted by the IAEA and the World Association of Nuclear Operators (WANO).

3.3.2.3.4 Second and Third Layer Attributes for Further Discrimination

For additional discrimination within a group of issues with same rank, the following attributes can be used:

- Second Layer: Prior risk information
- Third Layer: Qualitative cost evaluation

Priori risk information takes precedence over the qualitative cost evaluation.

If still additional discrimination is needed, some additional (fourth layer) attributes may be defined and used. Prior risk information and qualitative cost evaluation are presented below.

Prior Risk Information

For all safety issues, the evaluation of the prior risk information attribute considers problems encountered with respect to a particular safety issue identified in NEK experience, similar Westinghouse designed plants (not NEK specific), or generic industry-wide sources exclusively.

Table 3-11 provides the severity scale for the prior risk information attribute.

Table 3-11: Severity Scale for Prior Risk Information

Attribute	Description	Severity Number
Prior Risk Information	Issue identified as plant specific problem	3
	Issue identified as Westinghouse generic problem (no plant specific identification)	2
	Issue identified as generic industry-wide problem exclusively	1

Expert judgment and consultation with plant subject matter experts (SME) are utilized to determine the plant specific, Westinghouse, and generic industry-wide information sources for a given safety issue and whether or not it is directly applicable to NEK.

Qualitative Cost Evaluation

For all safety issues, the qualitative cost evaluation attribute considers preliminary order of magnitude costs associated with issue resolution. By necessity, due to the large number of potential corrective measures to be considered, plant SMEs provide the primary input to these order of magnitude cost evaluations. Table 3-12 provides the severity scale for the qualitative cost evaluation attribute.

Table 3-12: Severity Scale for Qualitative Cost Evaluation

Attribute	Description	Severity Number
Qualitative Cost Evaluation	Less than 10,000 EUR	4
	Between 10,000 EUR and 100,000 EUR	3
	Between 100,000 and 1,000,000 EUR	2
	Greater than 1,000,000 EUR	1

Additional Attributes

Additional attributes, which may be evaluated if desired, can be defined. As discussed previously, additional attributes are only considered in the ranking process if desired when insufficient discrimination between two or more safety issues has occurred when considering the primary attributes, prior risk information, and qualitative cost evaluation attributes.

The definition of the safety issue itself as well as a review of plant documentation determines the applicability of any additional attributes for a given safety issue. Expert judgment and consultation with plant SMEs is utilized. If an additional attribute is found to be applicable for a given safety issue, it will be ranked as one. A zero ranking is given for non-applicability.

3.3.2.3.5 Summary of Ranked Issue Categories and Expected Responses

As described earlier, each issue is ranked through one or more (as applicable) of the three ranking paths (direct link to nuclear safety, nuclear safety basis re-evaluation and non-nuclear safety / soft factors), which is shown in Figure 3-3. The severity (rank) scale is for all three paths “calibrated” in a way that particular severity number in any of the three ranking paths corresponds to the same level of safety significance. In the case that, for a specific issue, more than one ranking path applies, the highest rank is taken as a final rank for the considered issue).

The process results with a list of PSR issues, ranked according to their safety significances. The ranked list contains the issues with 6 major ranks, obtained according to the 6 severity numbers based on the primary attributes from the relevant ranking paths.

Table 3-13 summarizes, for each of the 6 major rank categories, types of issue expected to fall within a specified major rank category, based on its impact on plant risk, defense-in-depth and operational safety. For each category, expected plant response is outlined together with indicative time frame.

Table 3-13: Summary of Issue Categories and Expected Responses

Rank / Severity	Expected Resolution	Issue Types
6	Some kind of remedial action immediately, full resolution within 1 fuel cycle with appropriate corrective action.	Issues with very high risk confirmed by PSA or non-tolerable degradation of defense-in-depth
5	The resolution of open issues is expected within 5 to 8 years (8 year only for complex and expensive actions/ plant upgrades. (JV9, Ref. [4], article 45).	Issues with moderate to high risk confirmed by PSA or with DOD degradation which is tolerable only for a limited time. Issues with credible indication of high risk significance or non-tolerable DOD degradation which can be confirmed by safety re-evaluation. Issues with major element of operational safety feature missing or confirmed widespread failure to comply with operational safety features

4	The resolution of open issues is expected within 5 to 8 years (8 year only for complex and expensive actions/ plant upgrades. (JV9, Ref. [4], article 45).	<p>Issues with low to moderate risk confirmed by PSA or with DOD degradation which is tolerable for a longer time.</p> <p>Issues with credible indication of moderate risk or DOD degradation with limited tolerability, which can be confirmed by safety re-evaluation.</p> <p>Issues with major discrepancy regarding international standards for nuclear safety evaluation.</p> <p>Issues with some gaps in operational safety features missing or frequent failures to comply with operational safety features</p>
3,2,1	The resolution of open issues is expected within 5 to 8 years (8 year only for complex and expensive actions/ plant upgrades. (JV9, Ref. [4], article 45), or re-visited at next PSR	<p>Issues limited to system or function level with acceptable plant level risk and DID status.</p> <p>Issues with credible indication of low risk or low DOD degradation (with longer time tolerability), which can be confirmed by safety re-evaluation.</p> <p>Issues with certain discrepancy regarding international standards for nuclear safety evaluation.</p> <p>Issues with small deficiencies / improvements possibilities in operational safety features or occasional compliance problems.</p>

For issues within rank categories 6 and 5 appropriate corrective actions will be proposed and input into CAP. The implementation is expected in a very short term or within a time frame of five years after the regulatory approval of the PSR report, in accordance with SNSA Regulation JV9 [4], article 45.

For issues within rank category 4, corrective actions will be defined based on practicability arguments and global assessment. (Note: the purpose of global assessment in the PSR is to demonstrate that plant can be safely operated considering the issues which were found and corrective actions which were proposed.) Specified corrective actions are expected to be performed within the time frame of 5 years after the regulatory approval of the PSR report, according to SNSA Regulation JV9 [4], except for expensive and complex actions. For those, the implementation could be extended to maximum of 8 years, based on the regulatory approval, as defined by JV9 regulation [4].

For issues within rank category 3 and lower, corrective actions will be considered only if justified by practicability arguments and global assessment. These issues are limited to a system or a function level, while plant level safety status is compliant to modern standards (i.e. plant sustains the issue with risk being low and defense in depth strong and according to licensing basis)..

3.3.3. Identification and Prioritization of Corrective Measures

3.3.3.1. General Aspects

By definition, any measures that are considered ‘mandatory’ (i.e. those required by the SNSA with the relevant argumentation) have to be implemented and so fall outside the prioritization process. The remaining measures need to be prioritized before the implementation action plan can be established.

As a general principle, the highest priority is given to those measures that provide the greatest benefit in terms of plant risk reduction. Other factors, such as cost and time for implementation, will affect the program but will not influence the priority.

The overall process of identification and prioritization of corrective measures involves the following stages:

- Obtaining a good understanding of the issues and their safety significance (according to the issue ranking results)
- Identifying “mandatory” measures
- Identifying all other potential corrective measures
- Assessing the positive and negative impact of these measures
- Coming to a decision on whether each potential measure is worthwhile
- Prioritizing the worthwhile corrective measures
- Including the prioritized measures in the implementation action plan.

There is no preferred methodology to use for identification of measures. Lateral thinking based upon a detailed knowledge of the plant design, operation and associated safety analysis should give the best results. A strong input from the NPP staff most familiar with the design and operation of the plant is anticipated for this activity. In some cases, individual measures may enable resolution of several separate safety issues; and in other cases, several unrelated measures may collectively be required to resolve a single safety issue.

The types of corrective measures possible will differ depending on the type of deficiency and the stage in the overall review process. To assist in consideration of their treatment, deficiencies can be grouped into those of ‘information’, ‘design’, ‘operation’, and ‘safety culture’.

3.3.3.2. Identification of Corrective Measures

The types of corrective measures possible differ depending on the type of deficiency. Some discussion is provided below for each of the basic types of potential deficiencies indicated in the previous section. Typical corrective actions are indicated.

Deficiencies of information

Lack or poor quality of information may or may not constitute an actual threat to safety. The initial concern is that it introduces uncertainty, and this should be removed at the earliest opportunity. Lack or poor quality of design documentation and / or operational records will be identified during the course of the ‘assessment of the current plant safety’. Where possible, suitable corrective action will be initiated at the time of identification. Lack of analysis will also be identified during the ‘assessment of the current plant safety’. However, care will be required to ensure that any new analysis fully addresses the deficiency (particularly that requiring detailed modeling of the plant). It may be more appropriate, therefore, to delay the start of any such analysis until the ‘assessment of the current plant safety’ has been completed.

Types of potential actions:

- Regenerate existing documentation (documents, drawings, schedules, etc.) where the quality is poor
- Validate existing documentation against plant
- Commission the preparation of new documentation where missing
- Commission additional analysis (transient analysis, PSA, etc.)

Deficiencies of design

Deficiencies of design imply that some form of ‘upgrading’ is required. However, it should be recognized that the requisite design changes may not always be feasible for existing plants due to physical and other constraints.

Types of potential actions:

- Upgrade the capacity, reliability and / or redundancy of equipment,
- Install new equipment (safety systems, support systems, fire barriers, etc.),
- Qualify existing equipment for extended functions, or introduce other design features to reduce the qualification requirement.

Deficiencies of operation

Deficiencies of operation imply that suitable programs, procedures and other measures are not in place, are inadequate, are not supported by suitable facilities, or are not being followed. In most cases, the necessary corrective measures are self-evident, and these can usually be implemented.

Types of potential actions:

- Modify the existing procedures or develop new procedures,
- Modify operator training and maintenance practices,
- Modify test and inspection requirements (coverage and intervals),
- Reduce loads to equipment,
- Introduce new operational restrictions.

Deficiencies of safety culture

Deficiencies of safety culture relate to the organization and administration of the plant, QA and other human factors aspects. Safety culture is more associated with the general attitude displayed by the management and staff rather than to any ‘measurable’ quantity.

Types of potential actions:

- Change management and administrative policies and practices,
- Clarify roles, responsibilities and authority of groups and individuals,
- Implement training of all staff in safety culture matters.

A good understanding of each safety issue is a necessary prerequisite for identification of the appropriate corrective measures. For certain issues, particularly those regarding safety culture, this includes an understanding of the underlying causes.

It should be noted that operational corrective measures should complement engineering measures and not impose an unacceptably long-term burden on the operators.

3.3.3.3. Principles of Assessment of Corrective Measures

In general, measures proposed to overcome deficiencies of information and safety culture can be implemented with confidence that the results will be beneficial to safety. In the case of measures proposed to overcome deficiencies of design and operation, however, detailed consideration is required of the ‘negative’ as well as the ‘positive’ aspects of implementing the proposed measures. A useful philosophy to adopt is “if in doubt, don’t”.

The prioritization of corrective measures according to systematic criteria is an integral part of the assessment process in the PSR. This process provides relevant insights regarding the relative priorities of potential corrective measures identified for each of the specific safety issues. The prioritization can be performed based on deterministic and / or probabilistic criteria. Deterministic approach is qualitative and due to its simplicity applicable for initial prioritization. Probabilistic approach is more sophisticated, and therefore is typically applied for final assessment, if necessary. It should be noted that the latter approach might not be suitable for issues that are difficult to represent in the PSA.

The available prioritization techniques are not always capable to take into consideration all relevant benefits and drawbacks associated with the implementation of the selected measure(s). Therefore, the prioritization will be amended by a qualitative assessment of aspects that are not amenable to quantification. The final decision-making process will take into account all types of aspects.

The final decision on whether a particular potential corrective measure, or which of several potential measures, should be implemented, will be taken on the basis of informed expert and engineering judgment. Both the prioritization results (based on deterministic criteria and / or probabilistic criteria) and qualitative aspects will be taken into account. The two parts together, quantitative and qualitative enable the reviewers to reach a strong final position, which can be defended.

Prioritization of corrective measures

For the initial assessment, the prioritization process can follow an approach based on simple rules, such as identification of “mandatory” measures, identification of measures which are beneficial and can be easily implemented, etc.

For the remaining measures, determination of the priorities will be done using both deterministic and probabilistic techniques, as necessary. In the latter, the respective reduction in risk calculated for each proposed measure can be used to confirm their relative priorities. Following below is a brief description of principles of deterministic and probabilistic approaches to prioritizing corrective measures.

Deterministic approach

Deterministic approach is based on assigning the ‘worth’ to each proposed measure (can be done quantitatively or qualitatively) based in the assessed ability of the corrective measure to restore or improve the ‘integrity of barriers’ and ‘levels of defense’ as well as the ‘effectiveness of the overall implementation action plan’. Considerations of the defense-in-depth and safety barriers principles, should be accorded to relevant national and international requirements and standards, including Ref. [4] (SNSA Regulation JV9), Ref. [14] (Praktične smernice), Ref. [21] (WENRA document), [68], [74], [75], [106], [107], [108] and other applicable documents.

The “worth” assigned to a proposed corrective measure can be used in the prioritization. The proposed measures should be assessed, given a ‘worth’ based on their individual contribution

to risk reduction, and then further weighted via an integrated assessment of all proposed measures. This additional ‘weighting’ of a proposed measure will be assessed on the basis of its contribution to the effectiveness of the overall implementation action plan. For example, some measures can resolve a particular issue but can have other adverse effects on plant safety or can defeat effectiveness of other proposed measures.

The overall ‘worth’ can then be used in comparison with those obtained for other corrective measures to confirm the relative priorities.

Probabilistic approach

General framework for the use of probabilistic safety analysis approach to prioritizing corrective measures and for establishing probabilistic safety criteria, as necessary, should be established in accordance with national and international requirements and standards, as provided in Ref. [4] (SNSA Regulation JV9), Ref. [14] (Praktične smernice), Ref. [21] (WENRA document), Ref.[75], [108], [110], [111] and other relevant documents.

The impact of each potential corrective measure can be assessed using the plant PSA in terms of its risk reduction capability (the greater the risk reduction, the greater the relative worth of the measure). The risk reduction worth is a relevant parameter to assess whether particular corrective measure is worth implementing.

The level of risk is considered in terms of appropriate risk measures such as the frequency of particular off-site doses, large and early off-site releases, and the extent of plant damages (significant core degradations).

Quantitative evaluation of corrective measures

Comparison of alternative measures should be based on a systematic analysis, which considers both the beneficial aspects (values) and the costs (impacts) anticipated from a proposed measure.

This formal decision method can help in addressing all relevant aspects contributing to the decision and in decomposing the problem into manageable portions. The analysis can also provide a record of the decision rationale and a framework for sensitivity testing of data and assumptions. The results of value-impact are not to be considered binding.

The techniques applied in current international practice are described in various documents (e.g. Ref. [76]). A value-impact analysis-type should be employed which considers both the benefits (values) and costs (impacts) of a particular course of action. In general, the benefits (values) of risk reduction by particular corrective measure take into consideration all the societal consequences. The basic attributes taken into consideration include public health effects (accident and routine) and occupational health (accident and routine) as well as impacts on off-site and on-site property.

The costs (impacts) are those associated with implementing a particular potential corrective measure. These constitute the sum of all costs resulting from implementation (including capital expenditure and loss of revenue e.g. due to any extended outage for the implementation), and future operating costs (which may be increased or decreased). Two indices are normally used to evaluate particular option by means of value-impact analysis: the ‘Net Value’ and the ‘Cost-Benefit Ratio’. Both of them are applied in the process of corrective measures prioritization. The ‘Net Value’ is used to summarize the balance between the favorable and unfavorable consequences of the proposed measure. The basic perspective of the net value index is overall economic efficiency.

The 'Cost Benefit Ratio' is used to account for the relation between the costs (impacts) and benefits (values). It is useful in providing guidance on the "practicability" of implementing each potential corrective measure.

It should be noted that values of all the attributes, which enter a value-impact analysis, need to be expressed in monetary terms (in order to be able to sum them up for comparison of measures). For those associated with radiation exposure an appropriate person to dollar conversion factor needs to be defined for this purpose. The remaining attributes can be quantified monetarily in a direct manner.

In most cases the proposed corrective measures affect the release category frequency with dose conversion factor basically unaffected. If the proposed measure/action affects health effects through a mitigation of consequences a new conversion factor reflecting the 'after-backfitting' conditions need to be estimated.

Both the 'net value' and the 'value-impact ratio' are used to support decision. The 'net value' is an absolute measure which combines values and impacts quantified in monetary terms. It indicates the magnitude of the proposed measure's contribution toward the specified goals. When faced with a choice between two mutually exclusive corrective measures, the 'optimal' decision is to select the measure with the largest 'net value'.

Qualitative evaluation of corrective measures

Some examples of aspects that may be difficult to quantify but should be considered in final decision are given below:

Design benefits:

- Improved separation and segregation,
- Reduced dependency on the operators,
- Reduced workload on operator,
- Improved resistance to hazards,
- Improved single failure tolerance.

Completion drawbacks:

- Risk that the bases for the existing design and operation are not fully understood such that safety may be compromised following implementation of the measure;
- Complexity and novelty of the measure and hence any uncertainty associated with its performance or reliability;
- Risk arising from the short term unfamiliarity of the operations and maintenance staff with the measure following its implementation (i.e. giving rise to an increased potential for human error);
- Increase in the burden on the operating, maintenance, testing or inspection staff following implementation of the measure or increase in manpower;
- Loss of good physical access for maintainers / operators;
- Conventional risk to workers.

Implementation drawbacks:

-
- Risk that safety is compromised during implementation of the measure (i.e. due to an increased risk of a PIE occurring or to a reduced capability to take protective actions and mitigate its consequences);
 - Engineering complexity of the implementation process;
 - Engineering novelty of the proposed solution (not been done elsewhere);
 - Time taken to implement;
 - Conventional risk to implementers;
 - Extent use of critical resource.

3.3.3.4. Methodology for identification and prioritization of corrective measures in NEK PSR

Due to the diversity in potential safety factors, a number of possible corrective measures may be identified in NEK PSR. In some cases, multiple corrective measures may be postulated to provide resolution for a given safety issue. Identification of the optimal corrective measure for a particular deficiency will allow for a time optimal reduction of residual plant risk and enhancement of defense-in-depth.

As a general principle, the highest priority will be given to those measures that provide the greatest benefit in terms of plant risk reduction (i.e. those related to the top ranked issues).

Allocation of resources between the issues where a direct link to plant nuclear safety can be established, those which are a re-evaluation of the nuclear safety basis only and those related to non nuclear safety / “soft” factors will be determined based upon severity, costs, and ease of remediation. The corrective measures for the issues with rank / severity 3 or lower are expected to be exempted from further consideration since the costs are not commensurate with the minimal benefit of resolution. They will be considered only if justified by practicability arguments and global assessment.

Corrective measures will first be prioritized based on the ranking that a given safety issue(s) obtained. For example, corrective measures which reduce residual plant risk will receive the highest priority.

If the available corrective measure can yield a reduction in residual plant risk, a quantitative approach to prioritization will be employed. All other corrective measures will be qualitatively evaluated utilizing expert judgment and consultation with plant subject matter experts.

Prioritization of Corrective Measures for Safety Issues with Rank / Severity 6 and 5

Depending on the number of safety issues and proposed measures, a cost benefit approach will be utilized to provide relative rankings of available corrective measures for safety issues with rank 6 and 5. Corrective measures’ rank will not be affected by prioritization. The benefit will have been previously determined through the PSA and / or DOD evaluation. Refer to Section 3.3.2 for more detail. If necessary, the costs associated with a corrective measure will be derived based on the generic cost estimates. Such costing analysis estimates, while not expected to be exactly reflective of actual NEK costs, should still be sufficient to provide for a fine level of discrimination between the various corrective measures. It is anticipated that only a small percentage of safety issues will be found to be mapped into the extreme and moderate events risk attributes. Therefore, it is expected that a quantitative evaluation will not expend significant resources but still allow for the most defensible rationale for prioritization of corrective measures and the reduction of residual plant risk.

The guidelines for quantitative cost analyses, as necessary, can be found in Reference [76]. Cost analyses for the various corrective measures will be performed according to standard

engineering practices, considering also large accumulated experience of NEK staff with plant modifications. This involves an initial design evaluation of the corrective measure, identification of equipment and materials necessary for the modification, and an assessment of the work areas within the plant in which the proposed modification will take place. In addition to the cost of physical modifications, the cost analyses will include costs for engineering and quality assurance, radiation exposure, health physics (HP) support, and radioactive waste disposal. Nuclear power plant costs associated with re-writing operating and test procedures, staff training, and other technical tasks will also be considered.

Prioritization of Other Corrective Measures

First, a more detailed qualitative evaluation will be performed for the costs associated with all identified corrective measures associated with issues of rank 4, which were not previously ranked by the quantitative cost evaluation described above. This more detailed qualitative evaluation will consider the same cost categories as were quantitatively evaluated for rank 6 and 5.

Plant subject matter experts will provide the primary input for these qualitative cost evaluations. In addition to utilizing qualitative cost to prioritize other corrective measures, priority will be given to corrective measures, which provide resolution for multiple safety issues. Also, priority will be given to corrective measures which can be completed more expeditiously in the event that adequate discrimination cannot be obtained by considering cost of implementation and the potential for resolution of multiple safety issues alone.

Cost evaluation and other practicability arguments for measures associated with issues of rank 4 will be input to the final selection of measures (if alternative strategies proposed) and to definition of implementation timing.

Finally, it will be decided whether there are measures related to the issues with rank 3 or lower which are practicably justified or justified by the global assessment of plant status. Such measures, if any, will be added to the implementation action plan.

3.3.4. Implementation Action Plan

The basic objective of the implementation action plan is to ensure that all corrective measures are implemented in the intended manner and on the time-scale necessary to permit safe continued plant operation.

To this effect, the requirements of the implementation action plan are no different from those of any other complex plant modification program. The plan will, therefore, need to include the normal features and controls applicable to such a program. These should include, for example, those that ensure:

- The overall targets and the milestones required to achieve these targets are identified,
- The interactions between activities are established and clearly indicated,
- The impact on plant safety is known and agreed to be acceptable,
- The impact on normal plant operation is known and agreed to be acceptable,
- The resources required to achieve the milestones are identified and put in place,
- The necessary monitoring is in place to ensure that potential problems are identified and resolved early.

The dates planned for the detailed safety submission required to satisfy the SNSA of the acceptability of specific proposals are key items to be identified in the plan.

Measures from the PSR implementation action plan will be entered into the NEK Corrective Action Program (CAP) and their monitoring and implementation will be controlled by CAP mechanisms.

3.3.5. Global Assessment of Plant Status

The objective of the global assessment is to present an overall risk (or safety) judgment of the plant's ability for continued operation that includes a balanced view of the significant PSR results including unresolved shortcomings, corrective actions and/or safety improvements and the plant strengths identified in the review of all PSR safety factors.

The procedure explained in Section 3.3.1 describes the way of identification of differences between the safety status of a nuclear power plant and current safety standards and practices. Some differences may actually be strengths because the safety status of a plant on particular issues may be better than currently required.

By the time of the global assessment is to be performed, it can be assumed that the following will be available:

- A list of safety issues identified during the assessment of the current plant safety through the set of specified safety factors;
- A measure of safety significance assigned to each issue (by issue ranking process);
- Proposed corrective actions for issues.

The risk associated with continued operation in the presence of all unresolved shortcomings for all the safety factors should be assessed in their totality. This is important because it is possible that each shortcoming when considered in isolation may appear acceptable, but when taken into account together with others may prove to be unacceptable. This is particularly relevant when considering human and organizational factors (safety culture is represented by the combination of many individual factors, any one of which in isolation may appear unimportant).

Use of the PSA makes it possible to clearly estimate and compare the maximum achievable safety benefit from any combination of proposed solutions and to compare the total risk with the accepted risk targets. Examination of the PSA results for each option also provides engineering insights and understanding about how that benefit will be achieved. Critical examination of each measure within the integrated PSA framework can identify synergistic effects and unexpected negative impacts, and it can put all of this information in a proper perspective.

It shall be pointed out that PSA information is clearly helpful, but the uncertainties in data and technique do not allow decisions to be made on the basis of PSA results alone. There are also some issues which will be difficult to address in a PSA. Therefore, the deterministic arguments are considered equally important to provide acceptable justification for the safety case.

There is no obvious or verified procedure available at present other than a review and the use of expert judgment to deterministic consideration of the total effect on the safe plant operation of all unresolved shortcomings and all corrective actions and strengths identified. Generally, the justification approach can be based on any combination of rational arguments. However, expert judgment becomes a major input to the process.

The acceptability of the plant status considering all safety issues identified will be evaluated as a part of NEK PSR global assessment, taking into account also possible corrective measures and possible improvements.

3.4. PSR3 Documentation

The following documents shall be produced during the conduct of the PSR to provide the information required by different stages of the process described in this document above:

- Topical reports for each Safety Factor;
- Final Broad and Detailed Ranking of Safety Issues with Identification and Prioritization of Corrective Measures;
- Implementation Action Plan;
- Summary Report and Global Assessment of Plant Status.

3.4.1. Topical Reports

The Topical Reports provide details on the assessment of each Safety Factor. The presentation of the assessment covered by topical reports ensures that all the necessary information to assess the results of the review is available. The topics will correspond to the main safety factors (as listed in Section 3.2). Topical report will be written for a separate Safety Factor. In any case, it shall include the preliminary ranking for each safety issue found during the review process.

A table of contents for Topical Report is presented below:

- Introduction
- Objectives
- Scope and limitations
- Assessment methodology
 - Definition of Review Elements
 - Review of Applicability of Modern Standards
 - Definition of Plant Status between Two PSRs
 - Comprehensive Safety Factor Review
- Assessment results
 - Assessment Findings
 - Comparison of Review Elements
 - Determination of New Standards, Norms, Legislative and Regulatory Acts
 - Conformance to the Requirements
 - Identification of New Issues
 - Preliminary Broad Ranking of Issues
 - Interfaces Impact Assessment
- Conclusions and recommendations
- References
- Appendices.

The content of topical report, as presented above, is in line with the recommendations of IAEA SSG-25 [20], and, also, reflects the positive experience from the 2nd PSR review.

The section on the assessment method, according to IAEA SSG-25 [20], describes how the actual assessment is performed. Typical aspects addressed here include the selection of items to assess (safety issues), judgments regarding safety importance, involvement of internal experts and/or external independent reviewers, use of deterministic or probabilistic methods, etc.

The section related to the results presents safety issues identified in the review, their preliminary ranking, prioritization and discussion of possible corrective measures. The results obtained using different approaches, if applicable, are discussed. Corrective measures finally selected are described.

Details of the presentation need to be sufficient to provide a convincing demonstration that all aspects relevant to safety have been evaluated and no safety significant issues remain unresolved. References to supporting material and a brief concluding summary shall be provided. Supporting references include the existing studies used in the review and/or other detailed documents developed for the review. They are considered as part of the external documentation of the PSR.

3.4.2. Final Broad and Detailed Ranking of Safety Issues with Identification and Prioritization of Corrective Measures

This Report shall provide:

- the final ranking of safety issues collected from topical reports from the review phase of NEK PSR3, using the methodology presented within this report
- identification and prioritization of corrective measures to be considered for the inclusion into the PSR3 Implementation Action Plan.

3.4.3. Implementation Action Plan

The basic objective of the implementation action plan (IAP) is to ensure that corrective measures relevant to safety are implemented in the intended manner and on the timescale necessary to permit safe continued plant operation.

The methodological and technical approach taken in order to meet these objectives follows the requirements given in this document (section 3.3).

The prioritized list of candidate corrective measures from the ranking report (section 3.4.2) formed the basis for the PSR3 Implementation Action Plan (IAP) and input in Corrective Action Plan (CAP), which is described in this technical report. Summarized input shall be given taking into account corrective measures with rank above 3.

Corrective actions for issues which are the parts of the Implementation Action Plan shall be performed within a time frame required in JV9 [4] (5 years except for complex actions for which the implementation may be extended to 8 years). It is expected that the resolution of the corrective actions will be divided into three time implementation categories, the same as for PSR2:

- Time implementation category I: 0-1 year
- Time implementation category II: 1-3 year
- Time implementation category III: 3-5 year

It should be noted that NEK, through the CAP process, will establish a reasonable schedule for implementation considering the available human resources, predetermined budget other technical priorities in NEK.

3.4.4. Summary Report and Global Assessment of Plant Status

The Summary and Global Assessment Report will include conclusions concerning whether the plant fulfils regulations, safety standards and guides. Implications of possible deviations and also future corrective measures to be taken as identified by the review will be discussed. The report will include all relevant information needed to justify continued operation of the plant for the period up to the next review.

The summary report will be written in English and contain sufficient information that the SNSA will be able to evaluate the assessment. The summary report will be the subject of the independent review by the authorized institution (AI) before the SNSA approve it. Independent evaluation must be clear, without conditional statements regarding reviewed subject and must demonstrate that plant is safe as it was designed and is capable for continuous safe operation. In case that conclusions derived in summary report are unclear or need additional information to be independently reviewed, the AI need to verify their bases from topical reports. The level of needed background review is judgment of AI to be able for clear and unconditional evaluation conclusions.

The extraction of summary report, with the basic description of PSR process and conclusion of global assessment, shall be prepared and written in Slovene.

3.4.5. Status Reports

A written status reports for each safety factor shall be prepared on a monthly basis, in order to provide schedule update. These reports will be the bases for NEK progress report which shall be delivered to SNSA on half-year basis, as required by JV9 [4].

3.5. PSR Project Organization

Overall PSR project organization including management of the project, responsibilities and project interfaces shall be defined in the plant Project Management Manual (PMM) document which will be developed at the start of the project. PMM shall provide, in more details, the involved parties within the NPP Krško organizational, technical and administrative requirements to assure a smooth implementation of the project in accordance with NEK QA Program and procedures. In addition, the applicable Codes & Standards and QA Requirements will be summarized.

The project organization, documents and other deliverables must be clear, traceable and in accordance with NEK QA requirements.

3.5.1. Management

Overall PSR Project Management Organization is shown on the Figure 3-5 and it will also be a part of associated PMM document discussed above, which will provide the details of managerial aspects.

3.5.2. External Members of the Project

External members of NEK PSR project will be defined on an on-going basis, related to the task that has to be supported by them. Contributions, deliverables, responsibilities, communication with NEK and schedule commitments will be precisely defined per each separate task according to NEK technical specification which will be issued related to this particular task.

3.5.3. Responsibilities

NEK internal responsibilities for PSR project will be precisely defined in the PMM document. Responsibilities for external PSR Project Team members will be defined precisely for each task that will be contributed outside NEK, according to the technical specification related to this task. For each safety factor the Sponsor from NEK Licensing Department is assigned with the function to coordinate the activities between SF review group/responsible engineer and the PSR3 project responsible engineer. She/he will also be responsible for assigning and tracing the actions within CAP system once the implementation action plan is finished. Responsible engineers from NEK Licensing Department are, at the same time, the Sponsors for their own Safety Factor. The responsibilities will be contributed to the NEK PSR team members as described in Table 3-14 and Figure 3-5.

3.5.4. Project Interfaces

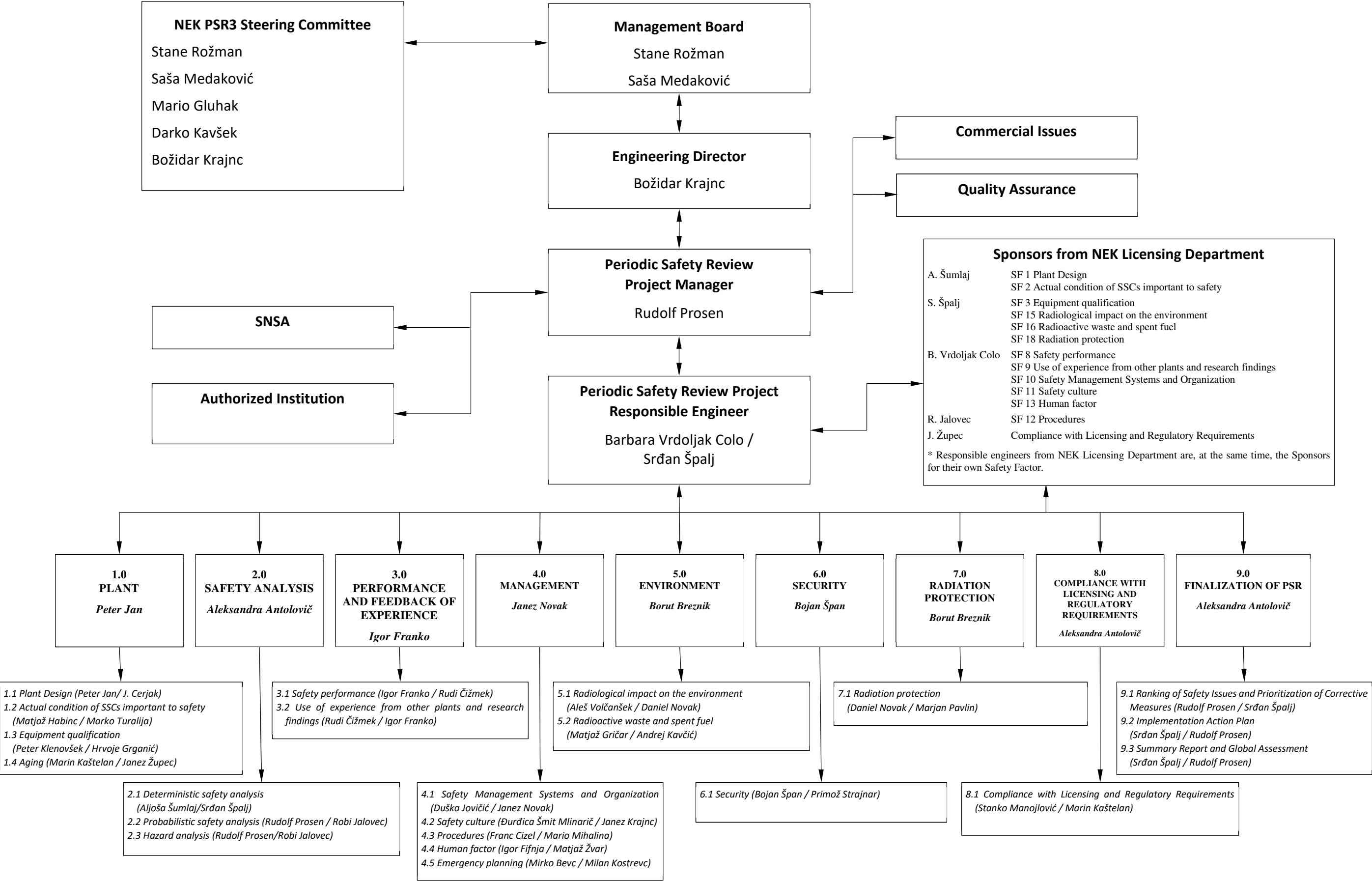
Internal NEK interfaces related to each of Safety Factor are presented in Table 3-14 and will also be provided in PMM document. External interfaces of NEK PSR project will be defined in an on-going basis, related to the task that has to be supported by external members. Contributions, deliverables, responsibilities, communication with NEK and schedule commitments will be precisely defined per each separate task according with NEK technical specification which will be issued related to this particular task.

Table 3-14: Internal NEK PSR Organizational Interfaces

No.	SF No.	Issue Description	Responsible engineer	Functional Support Area							
				Design eng.	System eng.	Operation	Maintenance	Training	Licensing /analysis	QA	ISEG
1.0		Plant	P. Jan								
1.1	1	Plant Design	P. Jan / J. Cerjak	X	X					X	
1.2	2	Actual condition of structures, systems and components (SSCs) important to safety	M. Habinc / M. Turalija		X		X			X	
1.3	3	Equipment qualification	P. Klenovšek / H. Grganič	X			X			X	
1.4	4	Aging	M. Kaštelan / J. Župec	X	X		X		X	X	
2.0		Safety analysis	A. Antolovič								
2.1	5	Deterministic safety analysis	A. Šumlaj/ S. Špalj						X		X
2.2	6	Probabilistic safety analysis	R. Prosen/ R. Jalovec						X		X
2.3	7	Hazard analysis	R. Prosen/ R. Jalovec						X		X
3.0		Performance and feedback of experience	I. Franko								
3.1	8	Safety performance	I. Franko / R. Čižmek			X		X			X
3.2	9	Use of experience from other plants and research findings	R. Čižmek / I. Franko			X		X			X
4.0		Management	J. Novak								
4.1	10	Safety Management Systems and Organization	D. Jovičić / J. Novak							X	
4.2	11	Safety culture	Đ. Šmit Mlinarič/ J. Krajnc			X		X		X	
4.3	12	Procedures	F. Cizel / M. Mihalina	X	X	X	X			X	
4.4	13	Human factor	I. Fifnja / M. Žvar	X		X		X		X	
4.5	14	Emergency planning	M. Bevc / M. Kostrevc			X			X	X	

No.	SF No.	Issue Description	Responsible engineer	Functional Support Area							
				Design eng.	System eng.	Operation	Maintenance	Training	Licensing /analysis	QA	ISEG
5.0		Environment	B. Breznik								
5.1	15	Radiological impact on the environment	A. Volčanšek / D. Novak	X	X	X	X			X	X
5.2.	16	Radioactive waste and spent fuel	M. Gričar / A. Kavčič	X	X	X	X			X	X
6.0		Security	B. Špan								
6.1	17	Security	B. Špan / P. Strajnar								
7.0		Radiation protection	B. Breznik								
7.1	18	Radiation protection	D. Novak / M. Pavlin	X	X	X		X	X	X	
8.0		Compliance with Licensing and Regulatory Requirements	A. Antolovič								
8.0		Compliance with Licensing and Regulatory Requirements	S. Manojlović / M. Kaštelan								
9.0		Finalization of PSR	A. Antolovič								
9.1		Ranking of Safety Issues and Prioritization of Corrective Measures	R. Prosen / S. Špalj								
9.2		Implementation Action Plan	S. Špalj / R. Prosen								
9.3		Summary Report and Global Assessment	S. Špalj / R. Prosen								

Figure 3-5: PSR Project Management Organization



3.6. PSR Program and Schedule

3.6.1. Availability of Resources and Needed Expertise

The detailed plan of needed internal / external resources and expertise will be established by NEK Management when overall 3rd NEK PSR Program will be accepted and approved by SNSA according to ZVISJV requirement [3]. Management of the PSR Project with all tasks in details will be planned / scheduled and described in NEK PSR3 PMM document.

3.6.2. Milestones and Time Scales

Final milestones and project management plan will be agreed between NEK and Slovenian Nuclear Safety Administration. NEK shall submit the PSR3 reports to SNSA by June 30 2023 as required in [38]. Taking into account this date the draft schedule is prepared and provided in Table 3-15.

Table 3-15: Draft NEK 3rd PSR Schedule

1	SNSA Approval of 3 rd NEK PSR Program	December 2020
2	Preparation of Specifications and Required Inputs for 3 rd NEK PSR Program	Dec. 2020 – Jan. 2020
3	Safety Factors Review Phase	Feb. 2021 - Jan. 2022
4	Final Broad and Detailed Ranking of Safety Issues / Identification and Prioritization of Corrective Measures	Feb. 2022 – Apr. 2022
5	Preparation of Implementation Action Plan (IAP)	May 2022 - Jul. 2022
6	Preparation of Summary Report and Global Assessment of Plant Status	Aug. 2022 – Oct. 2022
7	Independent Review of Summary Report and Global Assessment	Nov. 2022 – May 2023
8	Submittal of NEK PSR Report to SNSA	Jun. 30. 2023

4. Conclusions

This report presents the program for the 3rd PSR of Krsko NPP, taking into account the changes in IAEA safety guide for conducting of PSRs made since the 2nd Krsko NPP PSR has been performed, as well as Slovenian regulatory framework (JV9), which was established in the meantime. The report summarizes all PSR project organizational and management issues.

The 3rd PSR project for NEK will cover all changes either from current national and/or international safety standards/practices or plant design and operational arrangements since the 2nd PSR to the, so called, freeze date: December 31st, 2020. This date meets the requirements for 10 years period between periodic safety reviews, since the freeze date for 2nd PSR was December 31st, 2010.

The general scope of safety factors review shall follow the national requirements of JV9 [4] and IAEA recommendation of SSG-25 [20]. According to JV9 [4], 18 PSR safety factors have been selected. The following special considerations shall be taken into account:

- The results of IAEA Pre-SALTO mission, as agreed between NEK and SNSA [54], shall be considered as an input to the review of the Safety Factor 4 (Aging Management) if Pre-SALTO final report is issued by December 31, 2021. The issues found as a result of IAEA Pre-SALTO mission shall be considered as PSR3 issues and will go directly into the ranking process.
- Fire PSA (a part of Safety Factor 6 - Probabilistic safety analysis) is covered by a separate evaluation as a response to findings of IAEA expert mission [109]. This was agreed between NEK and SNSA [30]. The issues found as a result of this mission shall be considered as PSR3 issues and will go directly into the ranking process.
- Safety Factor 17 (Security) - The threat assessment is prepared by Ministry of internal affairs every year. Physical Security Plan is then reviewed accordingly, agreed by SNSA and approved by Ministry of internal affairs. This review is conducted every year and this aspect will not be included in the PSR3 review.
- The review of NEK Decommissioning Plan shall be performed at least during the PSR interval per requirement of [5] (JV5). The decommissioning plan was revised and the third revision was issued in 2019 [28] and accepted by SNSA [29]. NEK decommissioning plan will not be reviewed again during the PSR3 as agreed between NEK and SNSA [30].
- Compliance with licensing and regulatory requirements shall be evaluated as a separate subject during PSR3, even though it is not a required safety factor. This subject consists of two parts: 1. domestic legislation and licensing requirements, and 2. NEK Regulatory Conformance Program Compliance Review (RCP) which demonstrate the continued compliance with U.S. regulatory requirements. RCP is included as an input to the Safety Factor 1 (Plant Design).

5. References

- [1] The Second NEK Periodic Safety Review Program, ESD-TR09/09, Revision 3, NEK, 2010
- [2] Nuclear Power Plant Krško Periodic Safety Review Project Summary Report, Report Number PSR-NEK-8.0, Revision 0, NEK, 2004
- [3] Zakon o varstvu pred ionizirajočimi sevanji in jedrski varnosti (Uradni list RS, št. 76/17 in 26/19) (ZVISJV-1), (Ionising Radiation Protection and Nuclear Safety Act, Official Gazette of the Republic of Slovenia, No. 76/17 and 26/19)
- [4] Pravilnik o zagotavljanju varnosti po začetku obratovanja sevalnih ali jedrskih objektov (Uradni list RS, št. 81/16 in 76/17) (JV9), (Rules on the safety assurance of radiation and nuclear facilities (JV9), Official Gazette of the Republic of Slovenia, No. 81/16 and 76/17)
- [5] Pravilnik o dejavnostih sevalne in jedrske varnosti (JV5), Uradni list RS, št. 74/16 in 76/17 (Rules on radiation and nuclear safety factors (JV5), Official Gazette of the Republic of Slovenia, No. 74/16 and 76/17)
- [6] Uredba o mejnih dozah, referenčnih ravneh in radioaktivni kontaminaciji (UV2) (Uradni list RS, št. 18/18), 2018
- [7] Pravilnik o uporabi virov sevanja in sevalni dejavnosti (Uradni list RS, št. 27/18), 2018
- [8] Pravilnik o posebnih zahtevah varstva pred sevanji in načinu ocene doz (Uradni list RS, št. 47/18), 2018
- [9] Pravilnik o ravnanju z radioaktivnimi odpadki in izrabljenim gorivom, (Uradni list RS, št. 49/06 in 76/17 – ZVISJV-1), 2018
- [10] Pravilnik o zagotavljanju usposobljenosti delavcev v sevalnih in jedrskih objektih, (JV4), Uradni list RS, št. 32/11, 76/17 – ZVISJV-1 in 162/20, 2011
- [11] Pravilnik o pogojih in metodologiji za ocenjevanje doz pri varstvu delavcev in prebivalstva pred ionizirajočimi sevanji (SV5), Uradni list RS, št. 115/03 in 83/16
- [12] Pravilnik o obveznostih izvajalca sevalne dejavnosti in imetnika vira ionizirajočih sevanj (SV8), Uradni list RS, št. 3/17, 8/17, 76/17 – ZVISJV-1, 43/18 in 47/18
- [13] Pravilnik o ukrepih varstva pred sevanji na nadzorovanih in opazovanih območjih (SV8A), Uradni list RS, št. 47/18
- [14] URSJV praktične smernice, Vsebina in obseg občasnega varnostnega pregleda sevalnega ali jedrskega objekta, PS 1.01, Osutek izdaje 2, 2020
- [15] Dopis NEK ING.DOV-336.20, Komentarji na osutek druge izdaje praktične smernice PS 1.01 »Vsebina in obseg občasnega varnostnega pregleda sevalnega ali jedrskega objekta«, 12. 11. 2020
- [16] URSJV dopis 3570-7/2020/12, Vabilo stranke na ustno obravnavo 3.11.2020 na URSJV, October 26, 2020
- [17] URSJV dopis 3570-7/2020/13, Ustna obravnavo v upavnem postopku za odobritev programa tretjega obdobjnega varnostnega pregleda- PSR3, Novembar 11, 2020
- [18] URSJV praktične smernice, Uporaba splošne referenčne dokumentacije v upravnih postopkih, PS-1.05, izdaja 1, 2020

-
- [19] Dodatek k Praktičnim smernicam PS 1.05, Seznam prepoznane referenčne dokumentacije po področjih, 2020
- [20] International Atomic Energy Agency, IAEA Safety Standards, 'Periodic Safety Review for Nuclear Power Plants', Specific Safety Guide No. SSG-25, IAEA, Vienna, 2013
- [21] Western European Nuclear Regulators' Association, Reactor Harmonization Working Group, WENRA Reactor Safety Reference Levels, January 2008
- [22] NPP Krško Special Safety Review Final report, Rev. 0, October 2011
- [23] NEK ESD-TR-09/11, "NPP KRŠKO Analyses of Potential Safety Improvements", Rev.0, January 2012
- [24] Odločba (3570-9/2011/2) o izvedbi izrednega občasnega varnostnega pregleda NEK, May 30, 2011
- [25] Odločba (3570-11/2011/7) o izvedbi modernizacije varnostnih rešitev za preprečevanje težkih nesreč in blažitev njihovih posledic, September 1, 2011
- [26] Dopis (3570-11/2011/9) "Odobritev Programa nadgradnje varnosti NEK", February 6, 2012
- [27] Odločba 3570-11/2015/25 "Odobritev Programa nadgradnje varnosti NEK rev. 3 in podaljšanje roka za izvedbo" 20.1.2017
- [28] NPP Krško Decommissioning Plan Revision 3; 4520/CA/F0106405/01, Slempelkamp NIS Ingenieurgesellschaft mbH, Rev. 01, 2019
- [29] Dopis URSJV 3570-13/2018/10 - Odobritev Programa razgradnje NEK, tretja revizija, 12. 11. 2019
- [30] Beležka sestanka vodstev NEK in URSJV, 29. 5. 2020.
- [31] PSR2-NEK-7.2, Implementation Action Plan, Rev. 2, 2014
- [32] European Nuclear Safety Regulator's Group – ENSREG - 1st Topical Peer Review "Ageing Management" Country specific findings, October 2018
- [33] Slovenian Post-Fukushima National Action Plan, December 2012
- [34] Slovenian National Report on Nuclear Stress Tests, Final Report, December 2011
- [35] Report of the operational safety review team (OSART) mission to the Krško Nuclear Power Plant, Slovenia, 15 may to 1 june 2017 - IAEA-NSNI/OSART/194F/2017
- [36] EPREV – Peer Appraisal of Arrangements in the Republic of Slovenia For Preparedness and Response for a Nuclear or Radiological Emergency, IAEA, 2017
- [37] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Commissioning and Operation, IAEA Safety Standards Series No. SSR-2/2, rev. 1, Vienna (2016).
- [38] Odločba URSJV št. 3570-2/2011/97, 30. 5. 2014
- [39] Odločba URSJV št. 3570-5/2019/2, 20. 6. 2019
- [40] Program nadgradnje varnosti NEK, rev. 3 (DCM-RP-083), Januar 2017
- [41] NEK ESD-17/13, Counting Lines of Defense (CLOD) Ranking Approach, Rev 0, 2013

-
- [42] Odločba URSJV 3570-8/2012/5, "Sprememba dovoljenja za obratovanje NEK", 22. 4. 2013.
- [43] Updated Safety Analyses Report (USAR), Rev. 27
- [44] NEK – Technical Specifications, Rev. 181
- [45] Design Extension Conditions Technical Specifications (DEC TS), revision 5, 2019
- [46] Radiological Effluent Technical Specifications (RETS), revision 10, 2018
- [47] MD-1 Notranje usmeritve in cilji - petletni razvojni načrt, revision 15, 2020
- [48] MD-4 Program človeškega ravnanja, revizija 1, 2020
- [49] MD-7 Management of Dose Reduction Program, revision 0, 2009
- [50] ED-1 Design Modification Control Program, revision 5, 2020
- [51] Severe Accident Management Guidelines (SAMG-17), revision 0, 2019
- [52] Začetni odziv ob izgubi nadzora in upravljanja elektrarne, (EDMG-18.001), revizija 0, 2019
- [53] Odziv na večje požare ali eksplozije, (EDMG-18.002), revizija 0, 2019
- [54] AESP-3.11.002, Alternate AF Suction Source, rev. 0, 2018
- [55] AESP-3.11.006, Alternate Condensate Storage Tanks Makeup, rev. 0, 2018
- [56] EEOP-3.10, Evacuation Emergency Operating Procedures, rev. 2, 2020
- [57] Quality Assurance Plan (QD-1), revision 7, 2016
- [58] TD-6, Program požarne zaščite - požarni red, revizija 2, 2019
- [59] ESP-2.301; "Technical Specifications Changes and License Amendment, Rev.10, 2020
- [60] ESP-2.303; "Evaluation of changes in NEK", Rev. 7, 2020
- [61] ADP-1.1.205; "Navodilo o poročanju", Rev.12, 2019
- [62] ADP-1.2.013 "Inšpekcijski pregledi in komunikacija z upravnimi organi", Rev.6, 2020
- [63] ADP-1.1.200; "Operating Experience Assessment Program", Rev.7, 2019
- [64] ADP-1.1.230 "Planiranje in izpolnjevanje programa nadzornih zahtev", Rev.6, 2020
- [65] ADP-1.1.232 "Povezave med nadzornimi zahtevami in izvedbenimi postopki", Rev.12, 2020
- [66] ADP-1.2.010 "Technical Report", Rev.3, 2017
- [67] MD-2 Sistem vodenja – procesna organizacija, rev. 2, 2019
- [68] Annual PSA Reports (stored as records in NEK DCM system)
- [69] Dopis GEN energija d.o.o., 253/GEN/2020, Seznanitev glede statusa seizmoloških raziskav oz. projektov na GEN, 05. 05. 2020.
- [70] PSR2-NEK-7.1, Final Broad and Detailed Ranking of Safety Issues with Identification and Prioritization of Corrective Measures, Rev. 0, 2013
- [71] PSR2-NEK-8.0, Summary Report and Global Assessment of Plant Status, Rev.2, 2014

-
- [72] Final Independent Evaluation Report (FIER) of NEK Periodic Safety Review (PSR) 2 – Summary Report and Global Assessment, NUCCON /NEK PSR2 FIER, 2014, Rev.1
- [73] International Nuclear Safety Advisory Group, Defense in Depth in Nuclear Safety, INSAG-10, IAEA, Vienna, 1996
- [74] International Nuclear Safety Advisory Group, ‘Basic Safety principles for Nuclear Power Plants, 75-INSAG-3, Rev. 1, INSAG-12, IAEA, Vienna, 1999
- [75] U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results For Risk-Informed Activities, Revision 1, January 2007
- [76] U.S. NUCLEAR REGULATORY COMMISSION, Regulatory Analysis Technical Evaluation Handbook, NUREG/BR-0184, Draft for comments, Washington, D.C., 2017.
- [77] INTERNATIONAL ATOMIC ENERGY AGENCY, “Evaluation of the Safety of Operating Nuclear Power Plants Built to Earlier Standards” A common basis for judgement , Safety Reports Series No.12, IAEA, Vienna (1998).
- [78] Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG), Revision 3, March 31, 2014
- [79] The Operating Organization for Nuclear Power Plants, IAEA NS-G-2.4, 2001
- [80] Butcher, P., Holloway, N.J. »A Defence In Depth Approach to Safety Assessment of Existing Nuclear Power Plant«, Topical issues in nuclear, radiation and radioactive waste safety, Contributed papers, International conference held in Vienna, 30 August — 4 September 1998, IAEA, 1998
- [81] Holloway, N. J.: A Method for Pilot Risk Studies. IAEA-SR-111/44P. Seminar on Implications of Probabilistic Risk Assessment, Blackpool, U.K., 18 - 22 March 1985
- [82] UNITED STATES NUCLEAR REGULATORY COMMISSION, NRC Inspection Manual, Chapter 0308, Attachment 3 “Technical Basis for Significance Determination Process 09/10/04”, revised by Change Notice 16-103, 2016.
- [83] UNITED STATES NUCLEAR REGULATORY COMMISSION, NRC Inspection Manual, Chapter 0609, “Significance Determination Process 03/21/03”, revised by Change Notice 18-036, 2018
- [84] UNITED STATES NUCLEAR REGULATORY COMMISSION, “Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007)”, US NRC SECY-99-007A, 1999.

References for Safety Factors

SF1 Plant Design

- [85] INTERNATIONAL ATOMIC ENERGY AGENCY, The management system for nuclear installations, IAEA GS-G-3.5 (2009)
- [86] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

-
- [87] URSJV praktične smernice, Vsebina varnostnega poročila sevalnih ali jedrskih objektov, PS 1.04, Izdaja 3, 2019
- [88] Sandia Report 2013-2388, NPP Cable Materials: Review of Qualification and Currently Available Aging Data for Margin Assessments in Cable Performance (2013)
- [89] IGALL TLAA-201, Environmental Qualification of Electrical and I&C Equipment

SF2 Actual Condition of SSCs

- [90] RG 1.160, "Monitoring the Effectiveness of Maintenance Rule at NPPs, rev. 4, 2018

SF3 Equipment Qualification

- [91] ED-12 Environmental Qualification Program, revision 3, 2018
- [92] ED-17 Mechanical Equipment Environmental Qualification Program (MEQ), revision 1, 2018
- [93] ED-18 Seismic Qualification Program, revision 1, 2018
- [94] ED-19 Electromagnetic Compatibility Program, revision 0, 2018
- [95] IAEA NP-T-3.6, Assessing and Managing Cable Ageing in NPPs (2012)

SF4 Aging

- [96] MD-5 NEK Aging Management Program (Rev. 4), 2017.
- [97] ENSREG 1st Topical Peer Review National Action Plan on the Krško NPP Ageing Management Program, SNSA, September 2019
- [98] Dopis URSJV 3902-1/2004/140 z dne 13. 1. 2019 – Izvedba neodvisnega pregleda aktivnosti dolgoročnega obratovanja v NEK v okviru PSR3
- [99] 10 CFR 54 - Requirements for Renewal of Operating Licenses for Nuclear Power Plants, 2016
- [100] NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Revision 2, 2010
- [101] INTERNATIONAL ATOMIC ENERGY AGENCY, SALTO Guidelines Guidelines for peer review of long term operation and ageing management of nuclear power plants, Services Series 17, Vienna, December 2008
- [102] INTERNATIONAL ATOMIC ENERGY AGENCY, Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL), Safety Reports Series No. 82, Vienna, 2015
- [103] INTERNATIONAL ATOMIC ENERGY AGENCY, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants, Specific Safety Guide No. SSG-48, Vienna, 2018
- [104] IAEA NP-T-3.24, Handbook on Ageing Management for NPPs (2017)
- [105] NUCLEAR ENERGY INSTITUTE (NEI), "Industry Guideline for Monitoring the Effectiveness of Maintenance at NPPs", NUMARC 93-01, Rev. 4f (2018)

SF5 Deterministic Safety Analysis

- [106] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1, rev. 1, Vienna (2016).

-
- [107] INTERNATIONAL ATOMIC ENERGY AGENCY, Deterministic Safety analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2 (Rev.1), Vienna (2019)
 - [108] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment of Facilities and Activities, IAEA General Safety Requirements No. GSR Part 4, Rev. 1, Vienna, 2016

SF6 Probabilistic Safety analysis

- [109] Report of the Expert Mission on State-of-the-Art Approaches on Fire PSA, URSJV, Ljubljana, April 2020
- [110] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-3, Vienna, (2010)
- [111] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-4, Vienna, (2010)
- [112] Human Reliability Analysis in Probabilistic Safety Assessment for NPPs, Safety Series No. 50-P-10, IAEA (1996)

SF7 Hazard Analyses

- [113] NEK ESD-TR-18/16, Rev.1, Screening of External Hazards.
- [114] NEK ESD-TR-07/17, Rev.0, Screening of Internal Hazards.
- [115] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-9, IAEA, Vienna (2010).
- [116] INTERNATIONAL ATOMIC ENERGY AGENCY, External Human Induced Events in Site Evaluation for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-3.1, IAEA, Vienna (2002).
- [117] INTERNATIONAL ATOMIC ENERGY AGENCY, Meteorological and Hydrological Hazards in Site Evaluations for Nuclear Installations, IAEA Safety Standards Series No. SSG-18, IAEA, Vienna (2011).
- [118] INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.7, IAEA, Vienna (2004).
- [119] INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.11, IAEA, Vienna (2004).
- [120] INTERNATIONAL ATOMIC ENERGY AGENCY, External Events Excluding Earthquakes in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.5, IAEA, Vienna (2003).
- [121] INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of Seismic Safety for Existing Nuclear Installations, IAEA Safety Standards Series No. NS-G-2.13, IAEA, Vienna (2009).

-
- [122] IAEA SRS 86, Safety Aspects of NPPs in Human Induced External Events: General Considerations (2017)
 - [123] IAEA SRS 87, Safety Aspects of Nuclear Power Plants in Human Induced External Events: Assessment of Structures (2018)
 - [124] IAEA SRS 88, Safety Aspects of Nuclear Power Plants in Human Induced External Events: Margin Assessment (2018)
 - [125] WENRA Guidance Document Issue T: Natural Hazards Head Document (2015)

SF8 Safety Performance

- [126] MD-21 Plant Performance Monitoring Program, rev. 3, 2017
- [127] INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-2.6, IAEA, Vienna (2002).
- [128] INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Safety Performance Indicators for Nuclear Power Plants, IAEA-TECDOC-1141, IAEA, Vienna (2000).

SF9 Use of Experience from Other Plants and Research Findings

- [129] INTERNATIONAL ATOMIC ENERGY AGENCY, Operating Experience Feedback for Nuclear Installations, Specific Safety Guide No. SSG-50, IAEA, Vienna (2018)
- [130] INTERNATIONAL NUCLEAR SAFETY GROUP, Improving the International System for Operating Experience Feedback, INSAG-23, IAEA, Vienna (2008).

SF10 Organization and the Management System

- [131] INTERNATIONAL ATOMIC ENERGY AGENCY, The management system for nuclear installations, IAEA GS-G-3.5 (2009)

SF11 Safety Culture

- [132] Safety Culture in the Maintenance of NPPs, IAEA SRS 42 (2005)
- [133] Safety Culture in Nuclear Installations, Guidance for use in the enhancement of safety culture, IAEA-TECDOC-1329 (2002)
- [134] IAEA INSAG-13, Management of Operational Safety in NPPs, A Report by the International Nuclear Safety Advisory Group (1999)

SF12 Procedures

- [135] NUREG 0899: Guidelines for the Preparation of Emergency Operating Procedures, LWR Edition (1982)
- [136] Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG), Revision 3, March 31, 2014
- [137] INPO 11-003 »Guideline in excellence in procedure and work instructions use and adherence«, INPO (2011)
- [138] NEI 14-01 [Revision 1], Emergency Response Procedures and Guidelines for Beyond Design Basis Events and Severe Accidents (2016)

-
- [139] NEI 13-06 [Revision 1], Enhancements to Emergency Response Capabilities for Beyond Design Basis Events and Severe Accidents (2016)

SF13 Human Factors

- [140] IAEA INSAG-4, Safety Culture (1991)
- [141] Recruitment, Qualification and Training of Personnel for Nuclear Power Plants, IAEA NS-G-2.8 (2002)
- [142] The Management System for Facilities and Activities, IAEA GS-R-3, IAEA (2006)

SF14 Emergency planning

- [143] Načrt zaščite in reševanja ob izrednem dogodku (revizija 36), 2019
- [144] Državni načrt zaščite in reševanja ob jedrski ali radiološki nesreči, Verzija 3.0, Vlada Republike Slovenije, št. 84300-4/2010/3, 22. 7, 2010.
- [145] NPP Krško Development of Evacuation Time Estimates, KLD Engineering, P.C., 2019
- [146] INTERNATIONAL ATOMIC ENERGY AGENCY, Preparedness and Response for a Nuclear or Radiological Emergency, General Safety Requirements No. GSR Part 7, Vienna (2015).
- [147] IAEA, EPR-Method 2003, Method for Developing Arrangements for Response to Radiological and Nuclear Emergencies, Vienna, 2003.
- [148] Uredba o vsebini in izdelavi načrtov zaščite in reševanja (UVINZR)
- [149] Euratom Basic Safety Standards Directive, 2013/59/Euratom, EC (2013)

SF15 Radiological Impact on the Environment

- [150] Radiation Protection of the Public and the Environment, GSG-8, IAEA, 2018.
- [151] Programmes and Systems for Source and Environmental Radiation Monitoring, SRS-64, IAEA, 2010.

SF16 Radioactive waste and spent fuel

- [152] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. SSG-40, Vienna (2016)
- [153] RG 1.13, Spent Fuel Storage Facility Design Basis,
- [154] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for the Processing, Handling and Storage of Radioactive Waste, GS-G-3.3, IAEA, Vienna (2008).
- [155] INTERNATIONAL ATOMIC ENERGY AGENCY, Disposal of Radioactive Waste, GS-G-3.4, IAEA, Vienna (2008).
- [156] INTERNATIONAL ATOMIC ENERGY AGENCY, Core Management and Fuel Handling for Nuclear Power Plants Safety Guide Series No. NS-G-2.5, IAEA, Vienna (2002).
- [157] INTERNATIONAL ATOMIC ENERGY AGENCY, Storage of Spent Fuel, SSG-15, Vienna (2012).

-
- [158] INTERNATIONAL ATOMIC ENERGY AGENCY, Criticality Safety in the Handling of Fissile Material, SSG-27, IAEA, Vienna (2014).
- [159] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Fuel Handling and Storage Systems in Nuclear Power Plants, Safety Guide, NS-G-1.4, IAEA, Vienna (2003).

SF18 Radiation Protection

- [160] INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-2.2, IAEA, Vienna (2000).
- [161] INTERNATIONAL ATOMIC ENERGY AGENCY, Occupational Radiation Protection, IAEA Safety Standards Series No. GSG-7, Vienna (2018).
- [162] Ocena varstva izpostavljenih delavcev za Nuklearno elektrarno Krško, NEK-2015, rev. 0, januar 2015
- [163] Ocena varstva pred sevanji za razna nadzorna, projektantska, vzdrževalna, montažna in pomožna dela različnih izvajalcev v NEK, NEK- 2017, rev. 5, maj 2018
- [164] Poročilo o pregledu ocene varstva pred sevanji za Nuklearno elektrarno Krško za obdobje 2015-2019, (oznaka poročila NEK 1/2020), rev 1, April 2020
- [165] Mnenje o Pregledu Ocene varstva pred sevanji za Nuklearno elektrarno Krško, d.o.o. za obdobje 2015-2019, LDOZ-37/2020-GO, ZVD, 2020