

**ISLAMIC REPUBLIC OF IRAN**

**IRAN NUCLEAR REGULATORY AUTHORITY**

**NATIONAL NUCLEAR SAFETY DEPARTMENT**

***Report on Review and Assessment of***

***PSAR Chapter 5 (Primary circuit and related systems) of***

***"Bushehr-2 NPP Unit 2"***

***(Revision B01)***

**Doc. No.:** **NS-RT-053-21/03-0-Mar.2022**

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**فهرست دریافت کنندگان**

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| **2** | **دفتر ايمني هسته‌اي** |
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| **5** |  |

**Table of content**

 ***Subject Page No***

1. INTRODUCTION …………………….…………….…......…………………………….. 2

1.1 Background …………………………………………….………………………………...2

1.2 Basis for the review ………..…………………………......………...………….…….…. 2

1.3 Purpose and criteria of the review ………………………………......……………….…. 2

1.4. General description of topical issues ……………………………………………………2

LIST OF ABBREVIATIONS …………………………………......…………..…………… 4

PART 1 (NNSD Comments) ………………………………………………………………...6

2. COMMENTS ...……………………….…………….….....………………………….……7

2.1 General Comments …………………………………...…......……………………….….. 7

2.2 Detailed Comments Related to Each Individual Item ……............……...……..……...... 8

PART 2 (VO Safety Comments) …………………………………………………………….55

3. COMMENTS ON TOPICAL ISSUES OF THE REVIEW……………………………….56

4. CONCLUSIONS AND PROPOSALS…………………………………..……………….149

5. REFERENCES …………………………………………....……..………….………… 151

1. **INTRODUCTION**

## 1.1 Background

The present report contains the results of review and assessment of chapter 5 of PSAR of Bushehr-2 NPP unit 2, Revision B01, issued in 2017, which should have been elaborated in accordance with RG 1.70, the related Russian and IAEA requirements and recommendations. The present report has been prepared by the NNSD and JSC VO Safety experts.

**1.2. Basis for the review**

Chapter 5 of the Preliminary Safety Analysis Report of BUSHEHR-2 NPP Unit 2, has been reviewed on the basis of “US NRC R.G. 1.70, ‘Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants’ (LWR Edition), Rev.3, 2009” and the documents which are listed in REFERENCE section of this report.

**1.3. Purpose and criteria of the review**

Evaluation of format and contents of PSAR Chapter 5, revision B.01, and assessment of its compliance with the regulatory requirements and guides are the purposes of the review.

The review approaches described below were applied in the development of conclusions on certain topical issues: "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (NUREG-0800)” (hereinafter, NUREG-0800). When conducting the review, each expert independently determined the scope of the application of the methodology given in NUREG-0800 with due consideration of the topic of respective review issue.

The assessment criteria of the review were safety requirements and recommendations in the field of use of atomic energy as stipulated by the regulatory documents of the Russian Federation whose list is in Appendix M to Contract for the construction of the Bushehr nuclear power plant (Bushehr-2 NPP) (hereinafter, Appendix M to the Contract). The rules, regulations, and requirements of the Iranian Nuclear Regulatory Authority (INRA) as well as the IAEA safety standards listed in Appendix M to the Contract as applied to the review topics were considered during the review.

1.4. General description of topical issues

The approaches and methods set forth in the sections of Chapter 3 and Chapter 5 of NUREG-0800 were used during the review. Herewith, the peculiarities of the regulatory framework adopted for designing of Bushehr-2 NPP Unit 2 and the experience of the similar review for the Russian NPP units were taken into account.

Instead of the safety criteria given in the NRC regulatory documents, which are referred to in NUREG-0800, the criteria of the Russian regulatory documents specifying the general provisions on safety assurance, nuclear safety issues, the requirements for the applied structural materials and NPP component strength assurance are considered in this review.

During the review on some topical issues, the general approaches addressed in items 3.9.1, 3.9.2, 3.9.3, 5.2, 5.3, and 5.4 of NUREG-0800 were used, whereby the following aspects were assessed:

- information about the method of determination of loads and stresses in equipment components and pipelines under emergency conditions;

- information about design loads and design modes taken into account during strength analysis as well as information about permitted stresses required for strength assessment;

- information about analysis of equipment and pipelines’ seismic resistance;

- information about experimental research performed to substantiate the strength of equipment and pipelines;

- information about the structural materials of equipment and pipelines.

The provisions of section 5.2.2 of NUREG-0800 were used for reviewing topical issue 1, whereby the adequacy of the provided justification of the primary circuit overpressure protection system operability was assessed.

The provisions of NUREG-0800 section 5.2.4 as well as the requirements of the Russian regulatory documents specifying the conduct of in-service inspection and testing of the primary circuit were used for reviewing topical issue 2.

The provisions of NUREG-0800 section 5.2.5 were used for reviewing topical issue 3, whereby the extent of meeting the requirements for the primary coolant leak detection system was assessed.

The provisions of section 5.3.1 of NUREG-0800 as well as the requirements of the Russian regulatory documents and provisions of US NRC 10 CFR Part 50, Appendix H and ASTM E185-10.11 specifying the requirements for the RPV surveillance specimens were used for reviewing topical issue 5.

The provisions of the Russian regulatory documents specifying the requirements for the RPV destruction probability assessment as well as the provisions of IAEA INSAG-12 were used for reviewing topical issue 9.

The provisions of Regulatory Guide 1.70 were taken into account during the review.

The achieved science, technology and production level and previous operation experience related to Russian and foreign nuclear facilities were taken into account during the review; also, the available information about the occurrences at Russian and foreign nuclear facilities was used during the review.

**LIST OF ABBREVIATIONS**

|  |  |
| --- | --- |
| ACS | Automated control system  |
| ALMS | Acoustic leak monitoring system |
| AMCS | Access monitoring and control system |
| AOO | Anticipated operational occurrences |
| AOO | Abnormal operational occurrences |
| BRU-A | Atmospheric steam dump valve with discharge to atmosphere |
| BRU-K | Turbine bypass valve (condenser steam dump device) |
| CAD system | Computer-aided design system |
| CS | Containment system |
| DBA | Design basis accident |
| ECCS | Emergency core cooling system |
| ECR | Emergency control room |
| FA | Fuel assembly |
| HET | Heat exchange tube |
| HLMS | Humidity leak monitoring system |
| I&C | Instrumentation and controls |
| ICIS | In-core instrumentation system |
| LBB | Leak before break  |
| LCS | Leakage control system |
| MCR | Main control room |
| MSIV | Main steam isolation valve |
| MV | Main valve |
| NO | Normal operation |
| NOC | Normal operation conditions |
| NPI | Nuclear power installation |
| NRC | Nuclear Regulatory Commission |
| OBE | Operating basis earthquake |
| OKB | Russian abbreviation for R&D bureau |
| PCA | Pre-commissioning activities |
| PHRS | Passive heat removal system |
| PK | Russian abbreviation for regulations for inspection |
| PORV | Power-operated relief valve |
| PRZ | Pressurizer |
| PSAR | Preliminary Safety Analysis Report  |
| PTU | Protective tube unit |
| PV | Pilot-operated valve |
| RCP | Reactor coolant pump |
| RI | Reactor internals |
| RMS | Radiation monitoring system |
| RP | Reactor plant |
| SG | Steam generator |
| SHS | Software and hardware system |
| SSE | Safe shutdown earthquake |
| TEH | Tubular electric heater |
| ULCS | Unit-level control system |
| VVER | Water-moderated water-cooled power reactor  |

**PART 1**

**(NNSD Comments)**

**2. COMMENTS**

## 2.1 General Comments

* The following issues are not discussed in chapter 5: (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.
* The p-t limitation curve for Hydro test and operational condition is missed. There is only P-T curve for the planned of heating-up and cooling down.
* There is no reference or link for the working parameters. Also, there is no link to the calculations.
* The lifetime of equipment generally is 60 years, but there is no reference for the justification.
* In Some subchapters (e.g. 5.4.10 PRESSURIZER and 5.4.2.1 Steam generators) term “Destination” is used to classify systems and equipment. Clarification about the term and identification which consent is implied by such term shall be provided.
* Aging management program, list of critical SSCs, associated type and period of inspection and related monitoring program etc. shall be discussed and explained.
* The text of chapter 5 PSAR contains references to expired regulatory documents e.g. NP-001-97, NP-026-04, PNAE G-7-008-89, PNAE G-7-009-89, PNAE G-7-010-89; namely the classification of the reactor components provided in Chapter 5 was made in compliance with the mentioned regulatory documents.
* However, it should be noted that item 3.2 “Classification of reactor plant equipment and components” of chapter 3 of PSAR, contains classification of RP components made as per valid regulatory documents: NP-001-15, NP-089-15 (editorial note).

## 2.2 Detailed Comments Related to Each Individual Item

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| --- |
| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 1 |
| Section Number | 5.1.1 |
| Page | 2 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Description of coolant system |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.1.1, description of coolant system and its components is missing. Also, tabulation of important design and performance characteristics is missing. (e.g. pressure, temperature, flow rate, dimensions). Main design information and limits are not provided. |
| *2.2. Comments* |
| C1. The missing information shall be provided. Cross references to other chapters shall be provided as well. |
| *2.3. Recommendations* |
|  |
| *2.4. References* |
| RG 1.70  |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 2 |
| Section Number | 5.1.1 |
| Page | 3 & 4 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Description of reactor coolant pressure boundaryEquipment names |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.1.1.11, description of reactor coolant pressure boundary deals with the pipelines of the systems connected to the coolant system with 2nd valves closed during normal operation. Systems with valves which are opened during Normal Operation e.g. KBA are not treated. |
| *2.2. Comments* |
| C1. Definition of the pressure boundary connected with open valve shall be added. C2. In sub-chapter 5.1.1.11, the equipment names in details e.g. reactor with nozzles and seals shall be added. |
| *2.3. Recommendations* |
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| *2.4. References* |
| RG 1.70 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 3 |
| Section Number | 5.1.1.1 |
| Page | 2 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Lack of two parameters (*flow rates, coolant volume*) |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| According to Part 5.1.1 of RG 1.70, this section should provide a schematic flow diagram of the RCS denoting all major components, principal pressures, temperatures, flow rates, and coolant volume under normal steady-state full-power operating conditions. |
| *2.2. Comments* |
| C1. Information about flow rates, and coolant volume under normal steady-state full-power operating conditions shall be given.  |
| *2.3. Recommendation* |
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| *2.4. References* |
| RG 1.70 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 4 |
| Section Number | 5.1.2 |
| Page | 32 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Measurement unit of coolant level in reactor |
| **2. ISSUE IDENTIFICATION** |
| *2.1. Issue Description* |
| Parameters that are specified in table 5.1.2.1. |
| *2.2. Comments* |
| C1. In table 5.1.2.1, the measurement unit isn't specified for parameter "coolant level in reactor". It shall be specified. |
| *2.3. Recommendation* |
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| *2.4. References* |
| RG 1.70 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 5 |
| Section Number | 5.1.3 |
| Page | 48 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Surrounding concrete structures |
| **2. ISSUE IDENTIFICATION** |
| *2.1. Issue Description* |
| According to Part 5.1.3 of RG 1.70, in this section it shall be provided an elevation drawing showing principal dimensions of the RCS in relation to the supporting or "surrounding concrete structures" from which a measure of the protection afforded by the arrangement and the safety considerations incorporated in the layout can be gained. |
| *2.2. Comments* |
| C1. Surrounding concrete structures shall be shown in figure 5.1.3-1. |
| *2.3. Recommendation* |
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| *2.4. References* |
| RG 1.70 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 6 |
| Section Number | 5.2.2.2 |
| Page | 58 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Different conditions of actuation of the safety valves |
| **2. ISSUE IDENTIFICATION** |
| *2.1. Issue Description* |
| According to part 5.2.2.2 of RG 1.70, this section should provide an evaluation of the functional design of the overpressure protection system. This evaluation should include an analysis of the system’s capability to perform its function, describe the analytical model used in the analysis, and discuss the bases for its validity, ...According to sub-section 5.2.2.2.2 of PSAR, calculations of primary and secondary parameter variations under assumed conditions are provided using the procedure realized in the code "DYNAMIKA-97", which description is given in section 15.1.3. |
| *2.2. Comments* |
| C1. In sub-section 5.2.2.2.2 & also 15.1.3, different conditions whose running can result in actuation of the safety valves have not been considered. In this section, it shall be submitted Analysis results of different conditions of actuation of the safety valves. |
| *2.3. Recommendation* |
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| *2.4. References* |
| RG 1.70 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 7 |
| Section Number | 5.2.3 |
| Page | 65 - 80 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Missing information |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In section 5.2.3, some information is missing. |
| *2.2. Comments* |
| C1. In section 5.2.3, there is no any information on criteria to limit the occurrence of underclad cracking in low-alloy steel safety-related components clad with stainless steel. |
| *2.3. Recommendations* |
| R1. The above-mentioned information should be provided. It should be also demonstrated that prevention of underclad cracking is included.  |
| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 8 |
| Section Number | 5.2.3.4 |
| Page | 73 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Reference number |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.2.3.4.1, it is mentioned that the limiting concentrations of impurities and additives added in the primary coolant during power operation is maintained according to subsection 1.9.5.And also, in sub-section 5.2.3.7.1.5 it is mentioned that in order to ensure the design corrosion resistance of structural materials of equipment and pipelines during the entire service life the primary water chemistry is maintained as specified in Subsection 9.2.12. |
| *2.2. Comments* |
| C1. The cross-references are wrong. The cited subsections are not existing and they shall be corrected. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 9 |
| Section Number | 5.2.3.4 |
| Page | 73 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Material selection |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In table 5.2.3.8, it is mentioned that KCV for the steel 10ГН2МФА is 39 J/cm2.This material is used for SG vessel and PRZ vessel and main coolant line. |
| *2.2. Comments* |
| C1. Pressure and temperature for these applications are different. There is no justification for using this material with rather low KCV for SG vessel and PRZ vessel and main coolant line. This is regarding to the relevant concept of leak before break.  |
| *2.3. Recommendations* |
| R1. For justification, upper shelf energy should be given and be compared with the requirement. |
| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 10 |
| Section Number | 5.2.3.5 |
| Page | 75 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Lack of information about thermal insulation |
| **2. ISSUE IDENTIFICATION** |
| *2.1. Issue Description* |
| According to part 5.2.3.2 of RG 1.70, information about compatibility of the materials of construction and the "external insulation of the RCPB" with the reactor coolant should be provided. |
| *2.2. Comments* |
| C1. List of external insulation with their specifications shall be provided. |
| *2.3. Recommendation* |
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| *2.4. References* |
| RG 1.70 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 11 |
| Section Number | 5.2.3.7.1 |
| Page | 76 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Prevention of stressed corrosion cracking |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| “5.2.3.7.1.3 Water with concentration of chloride-ions not more than 0.05 mg/kg is used during manufacturing, hydraulic tests, circulation washing and hot running to prevent corrosion cracking of stainless steel.” |
| *2.2. Comments* |
| C1. Information about features or provisions which applied to control the chloride-ions concentration shall be provided. C2. What’s the base for the critical concentration of chloride-ions? Any referenced documentation/ requirement for such an amount shall be specified. C3. There is no any reference standard mentioned for control of any corrosive effects.  |
| *2.3. Recommendations* |
|  |
| *2.4. References* |
| NUREG 800 section 5.2.3 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 12 |
| Section Number | 5.2.3.7.1 |
| Page | 76 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Prevention of stressed corrosion cracking |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| “5.2.3.7.1.2 Unnstabilized chromium-nickel stainless steels are not used for manufacturing the primary circuit system components. Unstabilized welding materials used for manufacturing the indicated components, before their acceptance in manufacturing, are subject to ICC test by “go-no-go” principle, only materials that demonstrated resistance to ICC shall be used for manufacturing the primary circuit system components.” |
| *2.2. Comments* |
| C1. More information on basis of using unstabilized welding material for manufacturing indicated components shall be given. ICC test acceptance criteria shall be provided as well. |
| *2.3. Recommendations* |
|  |
| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 13 |
| Section Number | 5.2.4.7 |
| Page | 84 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | hydraulic pressure tests |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.2.4.7.2, it is mentioned: “According to /1/, /4/, in-service hydraulic pressure tests of strength of the equipment and pipelines of group A and B shall be performed at least once every four years. This period might be extended to 8 years.” |
| *2.2. Comments* |
| C1. It shall be explained that in which condition, in-service hydraulic pressure tests might be performed every 8 years. The proper reference for it shall be also provided. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 14 |
| Section Number | 5.2.5 |
| Pages  | 89 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Lack of information about floor drainage system |
| **2. ISSUE IDENTIFICATION** |
| *2.1. Issue Description* |
| According to part 5.2.5 of RG 1.70, the periodic testing of the floor drainage system that will check for blockage and ensure operability should be described. |
| *2.2. Comments* |
| C1. Information about floor drainage system & its periodic testing shall be provided. |
| *2.3. Recommendation* |
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| *2.4. References* |
| RG 1.70 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 15 |
| Section Number | 5.2.5.1.8 |
| Pages  | 95 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Frequency of testing and calibration of leakage detection systems |
| **2. ISSUE IDENTIFICATION** |
| *2.1. Issue Description* |
| According to part 5.2.5 of RG 1.70, this section should describe the provisions to test and calibrate all leakage detection systems and provide and justify the frequency of testing and calibration.  |
| *2.2. Comments* |
| C1. The frequency of testing and calibration shall be specified.  |
| *2.3. Recommendation* |
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| *2.4. References* |
| RG 1.70 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 16 |
| Section Number | 5.3.1.1.1 |
| Page | 105 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Missing reference |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.3.1.1.1 it is mentioned: “Steels 15Х2НМФА, 15Х2НМФА-А, 15Х2НМФА class 1 are made as per TU 08 93-013-00212179-2003.” |
| *2.2. Comments* |
| C1. The reference is missing and it shall be provided. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 17 |
| Section Number | 5.3.1.1.4 |
| Page | 107 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Missing reference |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.3.1.1.4 it is mentioned: “The mechanical properties of the other steels are provided in Topical Reports.” |
| *2.2. Comments* |
| C1. No reference is given. Referring to sub-chapter 5.2.3 with the mechanical properties of other materials shall be provided. (Mechanical properties of components like the fasteners are given in chapter 5.2.3)  |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 18 |
| Section Number | 5.3.1 |
| Page | 108 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | upper shelf energy |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| Table 5.3.1.7 contains the mechanical properties of reactor vessel steels. Information on upper shelf energy is missing. Also, the same situation exists for chapter 5.3.2. |
| *2.2. Comments* |
| C1. Information on upper shelf energy shall be provided. |
| *2.3. Recommendations* |
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| *2.4. References* |
| NUREG 800 SRP chap 5.3.1 & 5.3.2 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 19 |
| Section Number | 5.3.1.4.2 |
| Page | 110 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Wording  |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In section 5.3.1.4.2 it’s mentioned: “…and welding materials are specified in subsection 5.3.2.”  |
| *2.2. Comments* |
| C1. Title of sub-section 5.3.2 is “Permissible Pressure and Temperature” and there is no information about welding materials in that subsection. The above-mentioned statement shall be corrected as subsection 5.2.3 (primary circuit materials). |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 20 |
| Section Number | 5.3.1.6.7 |
| Page | 112 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Missing information |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.3.1.6.7, it is referred to reference No. 16.  |
| *2.2. Comments* |
| C1. No information about methods of surveillance and number of surveillance specimens is given in reference No.16. Also, time of withdrawal of surveillance specimens shall be provided.  |
| *2.3. Recommendations* |
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| *2.4. References* |
| PNEA G 7-008-89 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 21 |
| Section Number | 5.3.1.7.6 |
| Page | 115 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Inspection of fasteners |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| “5.3.1.7.6 RPV fasteners are subject to in-service inspection in accordance with /17/.During operation the fasteners are subject to the following inspection: - Studs М170х6: 1) 100% visual inspection;2) 100% magnetic powder or penetrant inspection;3) 100% ultrasonic inspection;4) 100% eddy-current inspection.- Nuts M170х6, spherical washers:1) 100% visual inspection;2) 100% penetrant test;3) Inspection intervals – each time when loss-of-sealing occurs.” |
| *2.2. Comments* |
| C1. Information on regular inspection of fasteners shall be provided.   |
| *2.3. Recommendations* |
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| *2.4. References* |
| SRP NUREG 0800 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 22 |
| Section Number | 5.3.1 |
| Page | 116 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Unclear reference  |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| Reference 1 is given as: “Regulations for design and safe operation of the nuclear power plant equipment and pipelines. PNAE G-7-008-89, Moscow, 2000”. |
| *2.2. Comments* |
| C1. The available document is dated in 1989 with English translation in 1999. The date of issuance of reference 1 shall be checked and corrected.   |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 23 |
| Section Number | 5.3.1 |
| Page |  |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Missing information |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| Information about surveillance specimens is missing in this sub-chapter.  |
| *2.2. Comments* |
| C1. Information on method of attachment and provision to ensure that capsules retained in position shall be provided.  |
| *2.3. Recommendations* |
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| *2.4. References* |
| RG 1.70 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 24 |
| Section Number | 5.3.2.2 |
| Page | 119 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | KI(thermal) is not considered in driving P-T limits |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| the following formula is required for heat up and cool down:Kapplied = 2KI(pressure) + KI(thermal) < KIcIn driving P-T limit curves, applicant uses the same approach as applied in NUREG 0800 which is basically: KI = [KI]iIt is not clear how the KI (thermal) for some other situations like heating and cooling down is considered. In fact, in US approach K applied include the stress intensity due to pressure (membrane) and Thermal gradient (bending) loads at the tip of the 1/4 T defect.  |
| *2.2. Comments* |
| C1. The formula for [KI]i shall be defined explicitly. Also, it shall be described that the factors n1, n2, n3 and delta T1, … T3 … how inserted in formula. More clarification is needed. C2. More information on calculation of KI shall be given (e.g. calculation report). |
| *2.3. Recommendations* |
|   |
| *2.4. References* |
| NUREG-0800 standard review plan chapter 5.3.2 |

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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 25 |
| Section Number | 5.3.2.1.1 |
| Page | 119 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | brittle failure resistance |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.3.2.1.1, it is mentioned that the conditions of assurance of brittle failure resistance can be written down as: KI = [KI]i |
| *2.2. Comments* |
| C1. According to PNAE G 7-002-89, the following condition is satisfied for the selected design defect in the form of the crack under the considered operating conditions: KI ≤ [KI]i  |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 26 |
| Section Number | 5.3.2.3.2 |
| Page | 120 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | brittle failure resistance |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In section 5.3.2.3.2, table 5.3.2.3.1 presents the values of temperature TК by the end of service life of 60 years for different components of the reactor vessel cylindrical part. |
| *2.2. Comments* |
| C1. Values of TK0 in the table are not equal to the given values in reference 1.C2. The difference of values for ΔTF for different locations cannot be understood from the available information. Especially, it is not clear that which fluence was taken into account at the different locations. The derivation of the temperatures shall be explained. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 27 |
| Section Number | 5.3.2.4.1 |
| Page | 120 & 121 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | permissible temperature of hydrotests |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In section 5.3.2.4.1, table 5.3.2.4.1 presents the values of minimum permissible temperature of hydrotests and checking the equipment for tightness. |
| *2.2. Comments* |
| C1. The values in the table cannot be validated with the available information. There is no reference on how the values are calculated. Standard 1 is not sufficient for this issue. Information about the calculation of the values of this table shall be provided. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 28 |
| Section Number | 5.3.2.5 |
| Page | 121 & 122 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Missing reference |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.3.2.5, it is mentioned that the permissible pressure in the reactor vessel for each the considered time moment of the conditions is determined by the formula obtained from the conditions of (5.3.2.1.1). And, figure 5.3.2.5.1 shows planned conditions of heating-up and cooling down. Results of permissible pressure calculations. |
| *2.2. Comments* |
| C1. Reference for the actual calculations and underlying report shall be provided.  |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 29 |
| Section Number | 5.3.3.1.2 |
| Page | 130 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | components of the vessel |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.3.3.1.2, it is mentioned the 121 nozzles semi-ellipsoid and the flange are welded between themselves.  |
| *2.2. Comments* |
| C1. From the point of operating experiences, these connections are important. So, information about all nozzle’s connections shall be provided.C2. It is required that all components of the vessel (e.g. nozzles, supports and saddles, instrument entry and connections) to be categorized/tabulated and each type connection geometry, dimensions, material and characteristic to be identified.  |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 30 |
| Section Number | 5.3.3.6 |
| Page | 134 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | operating condition of reactor vessel |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.3.3.6, the operating conditions of reactor vessel is mentioned. |
| *2.2. Comments* |
| C1. It is not clear how the operating conditions for heat up and cool down are derived. Maximum temperature gradients are given in chapter 4.4. However, the justification is missing and it shall be provided.  |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 31 |
| Section Number | 5.4.1.1.3 |
| Page | 142 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | criteria |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In section 5.4.1.1.3, it is mentioned: “The following criteria are assumed as a basis of designing:- structure of the reactor coolant pump set is designed proceeding from the condition of reliable fulfillment of its functions and strength keeping under operating, seismic and emergency loads, as well as their combinations occurring during operation of RCP set under the modes as stipulated in subitem 5.4.1.1.2 of the present subsection with regard for the number of their cycles and service life stipulated in the design for reactor plant, and in addition, non-exceeding the RCP set loads onto the systems connected to it and mentioned above in the design for RP;” |
| *2.2. Comments* |
| C1. The mentioned criteria are very general. The detail criteria can support the review of the design. Justification or reference for set of criteria shall be provided. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 32 |
| Section Number | 5.4.2.1.1 & 5.4.2.1.3 |
| Page | 162 & 163 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | differences between 2 types of SGs  |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.2.1.1, it is mentioned that the steam generator for BNPP-2 is PGV-1000MK.Also, in sub-section 5.4.2.1.3, it is mentioned: The basis for development of SG includes the following technical, design, process and operational requirements considering experience in designing and operation of steam generators PGV-1000 and PGV-1000MKP being in operation at NPPs with VVER in Russia and abroad” |
| *2.2. Comments* |
| C1. More information on differences between these SG types shall be provided and it shall be defined the similarity with PGV-1000MK. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 33 |
| Section Number | 5.4.2.1.2 |
| Page | 163 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Table 5.4.2.1 |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In table 5.4.2.1, it is mentioned that “Maximum total blowdown flow rate of each SG with the periodic blowdown connected is 35 t/h” |
| *2.2. Comments* |
| C1. The meaning of “total blowdown” shall be clarified.C2. There is no explanation of the meaning “periodic blowdown”.  |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 34 |
| Section Number | 5.4.2.1.4 |
| Page | 164 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Missing reference |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.2.1.4, it is mentioned: “Water treatment system shall provide for the required feedwater quality;” |
| *2.2. Comments* |
| C1. “required feedwater quality” is not defined or the reference is missing. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 35 |
| Section Number | 5.4.2.2.1 |
| Page | 171 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | blowdown of “salt cell” |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.2.2.1, the blowdown of “salt cell” is mentioned.  |
| *2.2. Comments* |
| C1. “Salt cell” is not described in the PSAR. A short description about “salt cell” shall be given. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 36 |
| Section Number | 5.4.2.2.2 |
| Page | 175 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title |  |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.2.2.2, it is mentioned: “Resistance of tubes in SG water medium, in the conditions of heat transfer and evaporation, considering presence of corrosion-active components, is checked by laboratory methods, bench experiments and full-scale operation experience at NPP with VVER.” |
| *2.2. Comments* |
| C1. No reference is given. The proof cannot be followed especially with regard to 60 years operation. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 37 |
| Section Number | 5.4.2.3.3 |
| Page | 177 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Limits and conditions of safe operation |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.2.3.3, information on Limits and conditions of safe operation is mentioned. |
| *2.2. Comments* |
| C1. Some of the technical information or cross reference to operational limits (e.g. level of the water, etc.) are missing. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 38 |
| Section Number | 5.4.2.5.1 |
| Page | 178 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Using unclear terms  |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.2.5.1, it is mentioned: “According to the classification by the parameters provided in GOST 27.003-90, the steam generators are the products of specific destination of kind 1, of continuous long-term application, aged and worn simultaneously, reparable by the repair with responsibility, maintainable during operation, inspectable before the beginning of application to the assigned destination, the product of a long-term storage.” |
| *2.2. Comments* |
| C1. The used terms are not clear. “specific destination of kind 1” shall be clarified. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 39 |
| Section Number | 5.4.2.5.1 |
| Page | 178 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | System reliability parameters |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.2.5.1, it is mentioned: “The longevity parameter of steam generators is assigned proceeding from the condition of meeting the requirements stated in subsection 5.4.2.1 during SG operation. In case of violation of limitations and requirements, indicated in subsection 5.4.2.1, the problem of the service life (operating life) of steam generators shall be solved by the Customer together with the steam generator design organization.”  |
| *2.2. Comments* |
| C1. The limitations and requirements are not clearly defined. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 40 |
| Section Number | 5.4.2.6 |
| Page | 179 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | residual stresses removal |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.2.6, it is mentioned that how to remove residual stresses in the outer layer of holes. Also, a set of procedures on tubes fixing ensures tightness of joint, absence of crevice (gap) in the tube-collector joint through the full thickness of the collector, fixing of tubes in the collector (against “tear-out”), maximum reduction of residual stresses in metal of the collector and tubes. |
| *2.2. Comments* |
| C1. The methods to remove residual stresses in the tubes are not described. This is a safety relevant issue because residual stresses could lead to stress corrosion cracking in operation. More information on residual stresses removal in the tubes shall be provided. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 41 |
| Section Number | 5.4.2.7 |
| Page | 180 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | maximum allowable thinning |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.2.7, it is mentioned: “Selection of the tube wall thickness, actually accepted for manufacturing, is made with regard for its thinning due to corrosion-erosion wear, thinning in the places of bends, as well as with regard for the available minus tolerance for the tube wall thickness as-delivered from the tube Manufacturers.” |
| *2.2. Comments* |
| C1. No value is given for the maximum allowable thinning duo to corrosion erosion wear during operation. That value is necessary for the in-service inspection program and it shall be explicitly given. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 42 |
| Section Number | 5.4.10.1 |
| Page | 218 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | TEH units |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| TEH unit groups and characteristics are not provided.  |
| *2.2. Comments* |
| C1. The number of TEH group shall be given in table 5.4.10.1. C2. The characteristics/specification and setpoints of each group shall be also provided. If such setpoints and specification are provided in other sub-chapters, it shall be cross referenced.  |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 43 |
| Section Number | 5.4.10.2.1 |
| Page | 219 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | system of emergency boron injection |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.10.2.1, it is mentioned: “…. The system of emergency boron injection provides for the injection of boric acid solution with concentration of 40 g/kg into PRZ to assure decrease in the primary pressure under the conditions of primary-to-secondary leak.”Also, in section 6.13 it is stated: “In case of design basis accidents (primary-to-secondary leaks), according to the reactor plant requirements for external systems, the JND50-80 (emergency boron injection) system injects boric acid solution (with a concentration of 16 g/kg, a temperature of 50 С, a flow rate of 29 m3/h ….” There is non-consistency of boron injection concentration in chapters 5 and 6. |
| *2.2. Comments* |
| C1. Those two above-mentioned statements are not consistence regarding to the boron injection concentration. It shall be clarified which one is correct.  |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 44 |
| Section Number | 5.4.10.2.8 |
| Page | 223 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Missing information |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| The information about PRZ strength calculation is missing. |
| *2.2. Comments* |
| C1. The PRZ strength calculation shall be provided in sub-section 5.4.10.2.8.  |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 45 |
| Section Number | 5.4.13.9.2 |
| Page | 269 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | PORV |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.13.9.2, it is mentioned: “PORV failure types are:- PORV failure to operate by demand;- PORV to close after actuation;- spurious actuation of PORV.- leaks through PORV valve gates exceeding permissible.” |
| *2.2. Comments* |
| C1. The permissible value for leaks through PORV valve gates during operation shall be given. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 46 |
| Section Number | 5.4.14.1.3 |
| Page | 274 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | Wrong cross-reference |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.14.1.3, it is mentioned: “Strength calculations of support structures consider the load combinations and strength criteria given in section 3.7.” |
| *2.2. Comments* |
| C1. Title of section 3.7 which is referred to is seismic design. There is no information for load combinations and strength criteria in that section. The cross-reference shall be corrected. |
| *2.3. Recommendations* |
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| *2.4. References* |
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| **ISSUE SHEET** |
| **1. ISSUE IDENTIFICATION** | Issue Number | 47 |
| Section Number | 5.4.14.1.3 |
| Page | 274 |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue Title | strength calculation |
| **2. ISSUE CLARIFICATION** |
| *2.1. Issue Description* |
| In sub-section 5.4.14.1.3, it is mentioned: “MCP instantaneous transverse break in the area of its welded joint to the reactor vessel nozzle with free double-ended coolant leak from MCP and the reactor is assumed as DBA.” |
| *2.2. Comments* |
| C1. The scope of DBA shall be defined. This DBA for the strength calculation of reactor vessel shall be explained and clarified. |
| *2.3. Recommendations* |
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| *2.4. References* |
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**PART 2**

**(VO Safety Comments)**

3. COMMENTS ON TOPICAL ISSUES OF THE REVIEW

3.1. Operability of the primary circuit overpressure protection system

According to the provisions of item 5.2.2 of Regulatory Guide 1.70, section 5.2.2 of PSAR, Chapter 5 for Bushehr NPP Unit 2 /1/ justifies the operability of the primary circuit overpressure protection system. Based on item 5.2.2.1.1 of the PSAR /1/, overpressure protection of the primary coolant system is provided by the pressurizer PORV jointly with the reactor trip system, protective interlockings and the RP systems and equipment actuating by these interlockings.

According to the provisions of item 5.4.12 of Regulatory Guide 1.70, the description of the PRZ PORV is provided in item 5.4.13 of PSAR Chapter 5 /1/, which states that three PRZ PORVs are envisaged for the RP (one control and two operating ones), which remove the medium from the PRZ to the bubbler upon actuation. Besides, the system description is provided in item 6.14 of Chapter 6 of the PSAR /D5/ (the description in PSAR Chapter 6 /D5/ mainly repeats the information given in PSAR Chapter 5 /1/ and contains references to sections 5.2.2, 5.4.13 of the PSAR /1/). The information about the PRZ is provided in item 5.4.10 of the PSAR /1/, the information about the steam relief pipelines is given in item 5.4.11 of the PSAR /1/ (when describing the pressure relief tank).

The process flow chart of the pressurizer system (specifying the monitoring points) is presented in item 5.4.11.2 of the PSAR /1/; the PRZ monitoring points are indicated in Fig. 5.4.10.3 of the PSAR /1/; the flow chart of the PRZ PORV is presented in Fig. 6.14.2.1 in item 6.14.2 of the PSAR /D5/ (compliance with provisions of item 5.2.2.3 of Regulatory Guide 1.70).

It is stated in item 5.2.2.1.2 of the PSAR /1/ that the requirement of item 6.2.2 of PNAE G-7-008-89 (as well as item 206 of NP-089-15) has been considered in the design, and namely the primary and secondary pressure under actuations of safety devices in design initiating events shall not exceed design pressure more than by 15% (considering dynamics of transients and actuation time of safety devices). The advanced response of the core has been taken into account, i.e. according to Table 5.2.2.2.2 of the PSAR /1/ the core actuates at the pressure of 17.6 MPa.

According to item 5.2.2 of PSAR Chapter 5 /1/, the pressure of 17.64 MPa was adopted as design pressure of the primary circuit; the following values were adopted as PRZ PORV actuation setpoints: opening pressure - 18.1 MPa (check pressure) and 18.6 MPa (operating pressure); closing pressure as per requirement of item 6.2.5 of PNAE G-7-008-89 - 17.2 MPa (check pressure) and 17.7 MPa (operating pressure).

It is stated in item 5.2.2.2.1 of the PSAR /1/ according to item 6.2.9 of PNAE G-7-008-89 that the following conditions are assumed in the calculation analysis of the primary and secondary overpressure protection systems capabilities: turbine stop valve closing and MSIV false closing (the calculations were made by means of the “Dynamika-97” code /D10/, which meets the requirement of item 2.1.15 of NP-082-07). The analysis of the calculation results is provided in items 15.3.2, 15.3.4 of PSAR Chapter 15 /D8/. According to item 15.3.2.3.3 /D8/, when assessing the LCS closing mode, “additionally (in order to decrease the number of calculation versions) a failure of one BRU-A of SG 2 steam line and a failure to open the control PRZ PORV are assumed, which results in the maximum increase rate and maximum absolute value of the primary circuit pressure. Such overlapping of failures makes it possible to check simultaneously the criterion on the maximum pressure of primary and secondary circuits.” According to item 15.3.2.7.2 /D8/, variants calculations were made, which demonstrated that the most conservative results in terms of the maximum pressure in the primary and secondary circuits are obtained when the following assumptions are made: availability of prohibition on operation of BRU-K; initial pressure at the reactor outlet is 16 MPa; injection to the pressurizer from the RCP head is not taken into account. As a result of calculations according to tables 15.3.2.6, 15.3.4.5 /D8/ the PRZ PORV fulfills its function in the stated modes: as a result of the calculation, the maximum value of the primary circuit value was obtained equal to 19.83 MPa (during LCS closing) and 17.42 MPa (during MSIV closing), which is lower than the acceptance criterion of 20.40 MPa (compliance with requirements of item 2.1.12 of NP-082-07 and provisions of items 3.45, 3.46 of NS-G-1.9).

According to item 5.2.2.2.7 of the PSAR /1/, to protect the primary circuit against over-pressurization during heating-up and cooling down of RP (at low temperature of the coolant) the following is applied:

- two safety valves of the system of emergency and planned cooling down of the primary circuit and cooling of the spent fuel pool (JNA) adjusted to actuation at pressure 2.2 MPA, with capacity of 360 m3/h each (the system is described in item 6.3.4 of PSAR Chapter 6 /D5/);

- interlocking on change-over of make-up pumps to recirculation and disconnection of all PRZ TEHs at pressure under RCP set pressure head more than 3.4 MPa and coolant temperature in cold legs of loops less than 100оС (the stated interlocking is indicated in Table 5.2.2.2.2 of the PSAR /1/);

- protection command for closure with prohibition against opening of quick-acting gate valves in the pipelines connecting ECCS accumulators with the reactor, at pressure at the reactor outlet more than 3.4 MPa and coolant temperature in hot legs of loops less than 100оС (availability of this interlocking is confirmed in Table 6.3.3.6 of the PSAR /D5/);

- RP operation is prohibited without steam (gas) blanket in the pressurizer (except the conditions of pressure increase during hydrotests).

According to item 5.2.2.2.8 of the PSAR /1/, during hydrotests for tightness of the primary circuit the function of the primary circuit overpressure protection is fulfilled by PRZ PORVs. During hydrotests for strength of the primary circuit the PRZ PORVs are mechanically stopped, the primary circuit overpressure protection function is fulfilled by safety valves of the pump of hydrotests that are adjusted to actuation pressure of 25 MPa corresponding to pressure above the core.

According to the information provided in subsection 5.4.10 of the PSAR /1/:

- item 5.4.10.1.4 of the PSAR [1]: the PRZ vessel is assigned to safety class 1 (classification designation 1H), group A as per PNAE G-7-008-89; the surge bottles, RI, TEH units, fasteners are assigned to safety class 2. All mentioned components are assigned to seismic category I as per NP-031-01 (compliance with the requirements of item 2.5 of NP-001-97, item 2.6.1 of NP-031-01, item 1.1.5 of PNAE G-7-008-89);

- item 5.4.10.3.3 of the PSAR [1]: under transients caused by equipment malfunctions, a part of the coolant through the surge line overflows from PRZ into the “hot” leg of loop No. 3 of the primary circuit, or from the “hot” leg of loop No. 3 of the primary circuit into PRZ. With this, the limitation of pressure deviations from the nominal value is reached due to change in volume of the steam blanket in PRZ (the nitrogen blanket is used in the heat-up/cooldown modes);

- the following monitoring is envisaged: level (5 monitoring points for different measurement ranges), pressure, temperature and chemical composition of the medium in PRZ (see Table 5.4.10.2 of the PSAR /1/ - compliance with the requirements of items 6.3.1, 6.3.6 of PNAE G-7-008-89);

- item 5.4.10.4 of the PSAR [1]: hydraulic testing for tightness and stress, metal inspection, internal and external examination are envisaged during pre-commissioning activities and operation (compliance with the requirements of item 8.2 of PNAE G-07-008-89).

According to item 5.4.11 of the PSAR /1/, the PRZ PORV steam relief pipelines are designed for steam relief to the relief tank upon actuation of the PRZ PORV. Two temperature sensors are installed on the stated pipelines.
It is stated in item 1.2.2.3 of Table 3.2.1.1.1 of the PSAR /D3/ that the steam relief pipeline as assembled is assigned to: Class 2 (from PRZ to PORV) and Class 3 (from PORV to the relief tank, fastenings) and has classification designation 2H (3H) as per NP-001-97; Groups B and C as per PNAE G-7-008-89, seismic category I as per NP-031-01 (compliance with the requirements of item 2.5 of NP-001-97, item 2.6.1 of NP-031-01, items 1.1.6, 1.1.7 of PNAE G-7-008-89).

It is stated in Table 5.2.3.2 of the PSAR /1/ that the PRZ is made of steel 10ГН2МФА, 38ХН3МФА. The mentioned materials are provided in Appendix 9 to PNAE G-7-008-89 (compliance with the requirements of item 3.1.2 of PNAE G-7-008-89).

According to the information provided in subsection 5.4.13 of the PSAR /1/:

- item 5.4.13.1.1 of the PSAR [1]: the number of the installed PRZ PORVs is determined according to the n+1 principle (where n is the number of PORV to ensure the discharged medium flowrate required for pressure reduction) assuring that the primary pressure will not exceed the design one by more than 15 % even in case of PORV failure with the least opening setpoint. According to item 5.2.2.2.3 of the PSAR /1/ and item 15.3.2.3.3 /D8/, a failure of one control PRZ PORV is assumed for making a design analysis (compliance with the requirements of item 6.2.7 of PNAE G-7-008-89, item 207 of NP-089-15);

- a correct reference to item 3.9.1.1 of PSAR Chapter 3 /D3/ is provided, where the number of allowable actuations of PRZ PORV is specified;

- item 5.4.13.1.2 of the PSAR [1]: PRZ PORV is assigned to class 2 (classification designation 2З) as per NP-001-97, group B as per PNAE G-7-008-89, seismic stability category I as per NP-031-01, classification designation 2BIIa as per NP-068-05 (compliance with the requirements of item 2.5 of NP-001-97, item 2.6.1 of NP-031-01, item 1.1.6 of PNAE G-7-008-89, item 2.1 of NP-068-05);

- item 5.4.13.2.2 of the PSAR [1]: Pilot valves operate as spring direct-acting valves in the event of electromagnetic drive failure (compliance with the requirements of item 6.2.17 of PNAE G-7-008-89);

- item 5.4.13.4 of the PSAR [1]: PROV main components (including the body) are made of steel 08Х18Н10Т-ВД, which is specified in the Appendix to the Summary List of Standardization Documents /D11/ (compliance with the requirements of item 85 of NP-089-15). Steel 08Х18Н10Т specified in Appendix 9 to PNAE G-7-008-89 is used for manufacturing of other PRZ PORV components as well as PRZ PORV medium supply and discharge pipelines (compliance with the requirements of item 3.1.2 of PNAE G-7-008-89).

- item 5.4.13.5.1 of the PSAR [1]: PORVs are actuated by the pressure transducer included into the set of the RP standard measurement facilities. The pressure is measured above the core, namely the unified current signals are generated by three pressure transducers above the core and processed by “2 out of 3” principle;

- position indication monitoring for main gate valve, pilot gate valve of PRZ PORV as per Table 5.4.13.2 of the PSAR /1/ is envisaged from the MCR and ECR (compliance with the requirements of item 4.4.3.2 of NP-001-97, item 3.4.2.7 of NP-001-15).

Item 5.4.13.7 of the PSAR /1/ provides the information about checks and tests of the system performed during pre-commissioning activities (preoperational work) including: hydraulic tests; check of the control system electric circuits; check of PORV serviceability with remote control of pilot valves; check of adjustment of pilot valve springs and check of PORV serviceability by real pressure increase to the actuation setpoint and determination of the main valve capacity (compliance with the requirements of items 6.2.27, 6.2.28 of PNAE G-7-008-89); check of leak tightness of gates of the main and pilot valves.

Item 5.4.13.8 of the PSAR [1] provides the information about in-service inspection and testing of the system performed once a year including the following checks: serviceability of PORV with the use of control keys from MCR; serviceability of pilot valves without MV actuation; serviceability of PORV MV in case of remote control from MCR; PORV control circuit test using a complex of the testing and control equipment for PORV control system; as well as check of adjustment of springs of pilot valves without MV actuation, in particular, when adjusting from the foreign pressure source (compliance with the requirements of item 6.2.28 of PNAE G-7-008-89, item 225 of NP-089-15).

Item 5.4.13.9 of the PSAR /1/ addresses analysis of the system failures:

- it is stated that the system reliability is provided among the other things by the presence of PORV control system self-diagnostic facilities and independent power supply of each of PORV three control channels;

- failure of PORV to close, or its spurious actuation by other causes is referred to loss-of-coolant accident. Damage of stop member, body items or break of pipelines connecting the PORV with the pressurizer results in SB LOCA. The mentioned accidents are addressed in items 15.7.1 “Inadvertent opening of the pressurizer safety valve followed by its failure to seat”, 15.7.2 “Small coolant leaks as a result of break in primary pipeline of equivalent diameter less than 100 mm” of PSAR Chapter 15 /D8/;

- unauthorized intervention protection is provided by the presence of a seal in the point of fastening the PV adjustment assembly; at the same time it is stated in item 5.4.13.2.2 of the PSAR /1/ that PV are equipped with bellows to ensure leak-tightness with respect to environment and spring protection from the direct effect of medium and overheating (compliance with the requirements of item 6.2.14 of PNAE G-7-008-89, item 215 of NP-089-15), in line with the requirements of the item operation manual as well as visual signaling of PV stopping;

- leaks through PORV gates are controlled upon change of temperature in the relief pipeline downstream the PORV during RP operation.

Strength analysis of the system components is evaluated under the topical issues “Strength of safety devices”, “Strength of the pressurizer”, “Strength of PRZ system pipelines” of the present Report.

The limits of safe operation for the primary circuit pressure including the pressure at low temperatures are established in items А 16.2.1.2 and В 16.2.1.2 of Chapter 16 of the PSAR /D9/.

It should be noted that items 3.1.11, 3.1.25, 3.1.26 of PSAR Chapter 3 /D3/ address implementation of the general design criteria, and namely: 15 “Reactor coolant system design”, 30 “Quality of reactor coolant pressure boundary”, 31 “Fracture prevention of reactor coolant pressure boundary” of document 10 CFR Part 50, Appendix A, which pertain to ensuring of design values of the primary circuit pressure.

The above mentioned technical decisions including the selected emergency modes for justification of the system serviceability have been tested and approved for those NPP units, which have been in operation in the territory of the Russian Federation for a long time period, as well as for those that have been recently commissioned (e.g. Unit No.1 of the Balakovo NPP, Unit No.1 of the Novovoronezh NPP-2) (compliance with the requirements of item 1.2.5 of NP-001-97, item 1.2.8 of NP-001-15). The justification provided in item 5.2.2 of the PSAR /1/ also confirms fulfillment of the provisions of items 3.39 ÷ 3.41, 3.45, 3.46 of NS-G-1.9.

The information provided in items 5.2.2, 5.4.10, 5.4.13 of the PSAR /1/ regarding the primary circuit overpressure protection system (taking into account the information provided in the other PSAR chapters) meets in the structure and content the provisions of items 5.2.2, 5.4.11 of Regulatory Guide 1.70.

According to the provisions of item 5.2.2 of Regulatory Guide 1.70, section 5.2.2 of the PSAR for Bushehr NPP Unit 2 /1/ justifies the operability of the secondary circuit overpressure protection system. According to item 5.2.2.1.1 of the PSAR /1/, secondary circuit overpressure protection is provided by steam generator PORVs and secondary steam dump devices (BRU-A). The design envisages two PORVs at each SG (item 5.4.2.2.3 of the PSAR /1/). According to items 5.2.2.3, 5.2.2.8 of the PSAR /1/, the description of the system including the diagrams and the information about inspections and tests is provided in section 6.15 of Chapter 6 of the PSAR /D5/.

Only the part of the information about the design analysis of the secondary circuit overpressure protection system, which is provided jointly with the information about the primary circuit overpressure protection system in PSAR Chapter 5 /1/, is evaluated in the framework of this topical issue. As for the evaluation of the secondary circuit overpressure protection system provided in the other chapters (e.g. Chapter 5 of the PSAR /D5/), it is not conducted in the framework of this section.

It is stated in item 5.2.2.1.2 of the PSAR /1/ that the following criterion is used for overpressure protection design: at design initiating events pressure in the secondary circuit under actuations of safety devices shall not exceed the design pressure more than by 15 % (considering dynamics of transients and actuation time of safety devices). Here, the design pressure should be understood in the PSAR /1/ as the maximum excessive pressure in the equipment or the pipelines used for calculation when choosing basic dimensions under which the manufacturing enterprise permits the operation of this equipment or piping under design temperature and normal operation conditions (this definition meets the definition given in Appendix No.1 to NP-089-15).

In item 5.2.2 of PSAR Chapter 5 /1/, the pressure of 7.84 MPa was adopted as design pressure of the secondary circuit; the following values were adopted as primary (secondary) SG PORV actuation setpoints: opening pressure – 8.23 (8.43) МPа; closing pressure – 6.89 (6.89) MPa. Considering the fact according to item 5.4.2.1.2 of the PSAR /1/ and Table 15.1.4.3.1.1 /D8/, the design pressure in the SG is -8.1 MPa; according to item 10.3.2.2 /D7/ and Table 15.1.4.3.2.1 /D8/, the design pressure of the live steam piping system is 7.84 MPa (compliance with the requirements of item 6.2.4 of PNAE G-7-008-89 (item 208 of NP-089-15) in terms of selection of the SG PORV setup for equipment designed for the lower pressure).

According to item 5.2.2.2 of the PSAR /1/, the conditions with primary and secondary pressure maximum increase are assumed in the calculation analysis of the primary and secondary overpressure protection systems capabilities to provide overpressure protection jointly with reactor emergency protection system and protective interlockings: turbine stop valve closings and MSIV false closing (calculations were made by means of certified code “DYNAMIKA-97 /D10/, which meets item 2.1.15 of NP-082-07). It is stated in item 5.2.2.2.3 of the PSAR /1/ that a failure of one PORV on one SG with the minimum actuation setpoint is assumed during calculations, which meets item 6.2.7 of PNAE G-7-008-89.

The analysis of the calculation results is provided in items 15.3.2, 15.3.4 of PSAR Chapter 15 /D8/.

It should be also noted that item 15.3.4.3.2 /D8/ specifies that the following assumptions and failures were considered during assessment of the MSIV false closure mode:

- the calculation was made with the assumption of the NPP blackout with consideration of the dependent failure of the pressurizer system (PRZ TEH), BRU-K and AFWP;

- a failure of the SG ECS channel is assumed, which results in generation of the signal for actuation for the relevant PHRS channel (this failure is more conservative as compared to the DG failure in the conditions of loss of the NPP auxiliary power supply since it causes the maximum actuation delay of the relevant PHRS channel);

- take-down for repairs (prior to occurrence of the initiating event) of the second ECS channel with connection to SG 2 is assumed.

As a result of calculations according to tables 15.3.2.6, 15.3.4.5 /D8/ the SG PORV fulfills its function in the stated modes: as a result of the calculation, the maximum value of the secondary circuit value was obtained equal to 8.46 MPa (during LCS closing) and 8.42 MPa (during MSIV closing), which is lower than the acceptance criterion of 9.43 MPa (compliance with requirements of item 2.1.12 of NP-082-07 and provisions of items 3.45, 3.46 of NS-G-1.9).

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 48 |
| Section No.  | 5.2.2 /1/ |
| Page | 6 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| It is stated in item 5.2.2.1.2 of PSAR Chapter 5 /1/: “The following criterion is used for overpressure protection design: primary and secondary pressure under actuations of safety devices in design initiating events shall not exceed the design pressure more than by 15 % ...” It is indicated that the design pressure is 17.64 МPа.It is stated in item 3.1.11 of PSAR Chapter 3 /D3/: “the design pressure means the maximum operating pressure in the operating modes of the reactor plant (17.64 MPa).”It is stated in item A 16.2.1.2.1 of PSAR Chapter 16 /D9/: “The safe operation limit for the primary circuit coolant system is the pressure not exceeding the value of the design pressure (17.64 MPa) by more than 15%, which makes up 20.29 MPa.”It is indicated in Table 15.3.2.1 of PSAR Chapter 15 /D8/: “the primary ...coolant system pressure shall not exceed 115 % of the design value (i.e. not exceed 20.40 MPa).” |
| *2.2. Comment* |
| The wording provided in items 5.2.2.1.2, 5.4.13.1.1 of the PSAR /1/ for the design criteria used in designing of the primary circuit overpressure protection system, whereby the “design pressure” is adopted as the primary circuit pressure, conflicts with the wording of item 6.2.2 of PNAE G-7-008-89 (as well as item 206 of NP-089-15). The values of the safe operation limit for the pressure in the primary circuit differ from those in item A 16.2.1.2.1 of the PSAR /D9/ and in Table 15.3.2.1 of the PSAR /D8/ although the method of their determination is the same. |
| *2.3. Recommendation* |
| Item 5.2.2 of the PSAR /1/ should formulate the design criteria for the primary circuit overpressure protection system in accordance with the terms used in NP-089-15, as well as specify the value of the primary circuit operating pressure. |
| *2.4. References* |
| PNAE G-7-008-89. Rules for design and safe operation of components and pipelines of nuclear power installations.NP-089-15. Rules for design and safe operation of components and pipelines of nuclear power installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 49 |
| Section No.  | 5.2.2 /1/ |
| Page | 6 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| According to item 5.2.2.2.7 of the PSAR/1/: “Overpressure protection system at low temperature includes the following equipment, devices and protections:- two safety valves of the system of emergency and planned cooling down of primary circuit, adjusted to actuation at pressure 2.2 MPa, with capacity of 360 m3/h each...”However, the information stated in item 5.2.2.2.7 of the PSAR /1/ is not confirmed in the EPCS justification provided in item 6.3.4 of the PSAR /D5/.According to PSAR item 5.2.2.2.8 /1/, during hydrotests for strength of the primary circuit the PRZ PORVs are mechanically stopped, the primary circuit overpressure protection function is fulfilled by safety valves of the pump of hydrotests that are adjusted to actuation pressure of 25 MPa corresponding to pressure above the core. |
| *2.2. Comment* |
| The information provided in item 5.2.2.2.7 of the PSAR /1/ about two safety valves of EPCS is not confirmed in the EPCS analysis presented in item 6.3.4 of the PSAR /D5/ since the following data on the EPCS SV is missing from the mentioned item of the PSAR /D5/: characteristics (including actuation pressure and capacity), information about fulfillment of primary circuit overpressure protection functions by the PV in the modes of heatup and cooldown (non-compliance with the provisions of items 6.3.2.2, 6.3.3 of Regulatory Guide 1.70).Item 5.2.2 of the PSAR /1/ does not provide information about the system of hydraulic testing and PV included therein.*(editorial comment)* |
| *2.3. Recommendation* |
| It is recommended that item 5.2.2.2.7 of the PSAR /1/ should give references to the sections of the other PSAR chapters containing detailed information about the equipment used to ensure overpressure protection of the primary circuit at lower temperatures. |
| *2.4. References* |
| US NRC Regulatory Guide 1.70. Standard Format and Content of Safety Analysis Report of Nuclear Power Plants. 1978. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 50 |
| Section No.  | 5.4.10, 5.4.13 /1/ |
| Page | 2 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| Items 5.4.10, 5.4.13 of the PSAR /1/ provide the classification of PRZ and PRZ PORV and justification of their design in accordance with the RD, which are no longer effective: NP-001-97 and PNAE G-7-008-89. Table 3.2.1.1.1 of PSAR Chapter 3 /D3/ provides the classification of PRZ, PRZ PORV as per NP-001-15, NP-089-15. |
| *2.2. Comment* |
| When determining the classification of the primary circuit overpressure protection system components (including the PRZ) in items 5.4.10, 5.4.13 of the PSAR /1/, obsolete documents NP-001-97 and PNAE G-7-008-89 were used, which conflicts with section 3.2 of the PSAR /D3/, which provides the classification of the system components in line with applicable NP-001-15, NP-089-15.*(editorial comment)* |
| *2.3. Recommendation* |
| It is recommended that items 5.4.10, 5.4.13 of the PSAR /1/ should provide the justification for the primary circuit overpressure protection system in line with NP-001-15 and NP-089-15. |
| *2.4. References* |
| NP-001-97. (PNAE G-01-011-97). General Provisions For Nuclear Power Plant Safety Assurance (OPB-88/97).NP-001-15. General Provisions on Nuclear Power Plant Safety Assurance.PNAE G-7-008-89. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations.NP-089-15. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 51 |
| Section No.  | 5.2.2 /1/ |
| Page | 5 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| Table 5.2.2.2.2 of the PSAR /1/ provides the main protections and interlockings of the primary circuit overpressure protection system, which are often duplicated in Table 6.14.3.2.1 of PSAR Chapter 6 /D5/.Item 5.2.2.2.3 of the PSAR [1] indicates: “Maximum time of signal generation from the moment of actual reaching the value of the setpoint by the parameter till output of discrete signal by a threshold device is meant to be signal sluggishness given in Table 5.2.2.2.2.”  |
| *2.2. Comment* |
| The values of temperature (100ºС) provided in Table 5.2.2.2.2 of the PSAR /1/ for actuation of PRZ PORV additional protection line algorithms differ from the values (130ºС) indicated in Table 6.14.3.2.1 of the PSAR /D5/. Similarly, the value of the temperature interlocking at the reactor outlet upon closing of the supporting control line valves indicated in Table 5.2.2.2.2 of the PSAR /1/ differs from the value specified in PSAR Chapter 6 /D5/.The values of the signal sluggishness provided in the text of item 5.2.2.2.3 of the PSAR /1/ are missing from Table 5.2.2.2.2 of the PSAR /1/.*(editorial comment)* |
| *2.3. Recommendation* |
| To exclude any duplication of the information it is recommended that the information about protections and interlocks of the overpressure protection system should be provided in PSAR Chapter 6 /D5/ along with justification on the necessity of using the mentioned interlocks; Chapter 5 of the PSAR /1/ should give a reference to this information. |
| *2.4. References* |
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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 52 |
| Section No.  | 5.2.2, 5.4.13 /1/ |
| Page | 2, 6 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| Item 5.2.2.1.4 of the PSAR [1] indicates: “The supporting control line is used for primary pressure decrease by remote control from MCR and ECR control boards or automatically under primary equipment protection against “cold” over-pressurization.” It is stated in item 6.14.1 of the PSAR /D5/: “PRZ PORV ensures fulfillment of the “feed and bleed” procedure intended for controlling the BDBA and mitigating consequences thereof. Fulfillment of the mentioned procedure is ensured by the PRZ PORV valves by reducing the pressure in the primary circuit under the remote control of the PORV using it to open the supporting control line...”In its turn, item 5.4.13 of the PSAR /1/ states: “PRZ PORV remote control has a priority over automatic control in actuating PORV by the operator for safety assurance of the NPP Unit”, whereas the information about the supporting line is not provided in this item. |
| *2.2. Comment* |
| Items 5.4.13 of the PSAR /1/ does not provide a description of the PRZ PORV supporting control line operation in different modes including the BDBA (when fulfilling the “feed and bleed” procedure); it does not specify either the criteria, whereby the necessity of the PRZ PORV remote control is determined in order to ensure safety of the NPP unit, or the measures preventing false opening of the PRZ PORV.*(editorial comment)* |
| *2.3. Recommendation* |
| It is recommended that item 5.4.13 of the PSAR /1/ should provide information about fulfillment of requirements of items 3.1.11, 3.1.12 of NP-001-15: justify permissibility of PRZ PORV actuation by the operator and specify how the false opening of PRZ PORV is prevented. |
| *2.4. References* |
| NP-001-15. General Provisions on Nuclear Power Plant Safety Assurance. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 53 |
| Section No.  | 5.2.2 /1/ |
| Page | 2 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| Item 5.4.13.2.3 of the PSAR [1] indicates: “The PORV design provides for the valves which allow isolating PV from MV for PV adjustment test without MV actuation.” |
| *2.2. Comment* |
| Item 5.4.13 of the PSAR /1/ does not demonstrate fulfillment of the requirements of items 6.2.11 (second paragraph), 6.2.15 of PNAE G-7-008-89 (item 214 of NP-089-15) taking into account the availability of isolation valves in the PRZ PORV design, and namely it is not demonstrated that overpressure protection is ensured in case of the PV disconnection by means of the isolating valve. |
| *2.3. Recommendation* |
| Item 4.3.13 of the PSAR /1/ should provide the analyses on failures of isolating valves envisaged in the PRZ PORV design as well as their impact on fulfillment of functions by the primary circuit overpressure protection system. |
| *2.4. References* |
| PNAE G-7-008-89. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations.NP-089-15. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 54 |
| Section No.  | 5.4.13 /1/ |
| Page | 5, 7 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| Item 5.4.13.3.1 of the PSAR /1/ specifies the reliability index for the PRZ PORV: “Operating durability (assigned service life) of the body items shall not be less than 30 years...” (the similar information is provided in item 5.4.13.8.1 of the PSAR /1/). At the same time, according to Table 1.2.2.2.1 of PSAR Chapter 1 /D2/, the NPP design service life is 60 years.Besides, it is stated in item 6.14.3.3.1 of PSAR Chapter 6 /D5/ that the assigned operating time of valve body items shall be no less than 60 years. |
| *2.2. Comment* |
| Items 5.4.13.3.1, 5.4.13.8.1 of the PSAR /1/ specify the assigned operating time of PRZ PORV body items as 30 years (non-compliance with the requirements of item 2.6.7 of NP-068-05, which states that the assigned service life of NPP valves shall be consistent with the assigned service life of the NPP unit and shall be at least 40 years). |
| *2.3. Recommendation* |
| Items 5.4.13.3.1, 5.4.13.8.1 of the PSAR /1/ should provide the information in accordance with the data specified in item 6.14.3.3.1 of PSAR Chapter 6 /D5/ stating that the assigned operating time of valve body items shall be no less than 60 years. |
| *2.4. References* |
| NP-068-05. Piping Valves for Nuclear Power Plants. General Technical Requirements. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 55 |
| Section No.  | 5.4.13 /1/ |
| Page | 8 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| Item 5.4.13.8.1 of the PSAR /1/ provides the following information about in-service inspections of the PRZ PORV: “...check of PORV MV serviceability by an actual pressure increase with MV actuation - only for those PORV which were subject to the following procedures:1) replacement of MV removable part;2) repair or replacement of the supply (or discharge) pipeline sections resulting in a change of the path pressure loss.” |
| *2.2. Comment* |
| The frequency specified in item 5.4.13.8.1 of the PSAR /1/ as well as the conditions of PRZ PORV serviceability check by an actual pressure increase with MV actuation, whereby this type of inspection is performed only for those PORV, on which the activities related to replacement or repair of specific components have been conducted, do not meet the requirements of item 6.2.27 of PNAE G-7-008-89 (item 224 of NP-089-15) and item 4.1.10 of NP-001-97. |
| *2.3. Recommendation* |
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| *2.4. References* |
| NP-001-97. (PNAE G-01-011-97). General Provisions for Nuclear Power Plant Safety Assurance (OPB-88/97).PNAE G-7-008-89. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations.NP-089-15. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 56 |
| Section No.  | 5.4.13 /1/ |
| Page | 8 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| Items 5.4.13.7, 5.4.13.8 of the PSAR /1/ when describing the PRZ PORV precommissioning activities, in-service inspection and tests, do not provide any information about technical examination (except for hydraulic testing in the course of precommissioning activities). |
| *2.2. Comment* |
| Items 5.4.13.7, 5.4.13.8 of the PSAR /1/ when describing PRZ PORV inspections and tests during precommissioning activities and in the course of operation, do not state that technical examination has been performed as per item 8.2 of PNAE G-7-008-89.*(editorial comment)* |
| *2.3. Recommendation* |
| It is recommended that items 5.4.13.7, 5.4.13.8 of the PSAR /1/ should provide the information about technical examination of the primary circuit overpressure protection system components as per applicable regulatory document NP-089-15. |
| *2.4. References* |
| PNAE G-7-008-89. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations.NP-089-15. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 57 |
| Section No.  | 5.2.2.1.3, 5.4.13.9 /1/ |
| Page | 2, 8 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| According to item 5.2.2.1.3 of the PSAR/1/: “Primary and secondary overpressure protection systems are designed considering a single failure principle.”According to item 5.4.13.9.1 /1/, PORV reliability is provided by: “independent power supply of each of PORV three control channels.”Item 5.4.13.2 of the PSAR /1/ describing the design of the primary circuit overpressure protection system does not provide any confirmation on fulfillment of the stated requirements aimed at ensuring protection against a common cause failure. |
| *2.2. Comment* |
| When describing the PRZ PORV, item 5.4.13.2 of the PSAR /1/ does not provide the following information confirming fulfillment of the requirement of item 4.1.6 of NP-001-97 (availability of measures on prevention or protection of the primary circuit overpressure protection system components against common cause failures):- power supply sources of the system components;- functional and (or) physical separation of the system components including power supply and control lines.*(editorial comment)* |
| *2.3. Recommendation* |
| Item 5.4.13 of the PSAR /1/ should demonstrate how the protection against common cause failures is ensured during designing of the primary circuit overpressure protection system, and namely how the diversity and independence principles specified in item 3.1.9 of NP-001-15 and item 3.42 of NS-G-1.9 were taken into account.  |
| *2.4. References* |
| NP-001-97. (PNAE G-01-011-97). General Provisions for Nuclear Power Plant Safety Assurance (OPB-88/97).NP-001-15. General Provisions on Nuclear Power Plant Safety Assurance.NS-G-1.9. Design of the Reactor Coolant System andAssociated Systems in Nuclear Power Plants, Safety Guide. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 58 |
| Section No.  | 5.4.13.3 /1/ |
| Page | 5 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
|  |
| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| Item 5.4.13.3 “Reliability indices” of the PSAR /1/ as well as similar item 6.14.3.3 /D5/ provides only the information about the reliability indices for valves (PRZ PORV); they contain no information about the reliability analysis for the primary circuit overpressure protection system. |
| *2.2. Comment* |
| Item 5.4.13.3 of the PSAR /1/ and item 6.14.3.3 /D5/ do not contain any information about the reliability analysis for the primary circuit overpressure protection system made with due account taken of common cause failures and personnel errors as well as about the system reliability indices (non-compliance with the requirements of item 4.1.12 of NP-001-97). |
| *2.3. Recommendation* |
| Item 5.4.13.3 of the PSAR /1/ should provide the information about the reliability analysis on fulfillment of functions by the primary circuit overpressure protection system with due account taken of common cause failures and personnel errors made in line with item 3.1.17 of NP-001-15. |
| *2.4. References* |
| NP-001-97. (PNAE G-01-011-97). General Provisions For Nuclear Power Plant Safety Assurance (OPB-88/97).NP-001-15. General Provisions on Nuclear Power Plant Safety Assurance. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 59 |
| Section No.  | 5.4 /1/ |
| Page | 8 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| Items 5.4.10, 5.4.11 of the PSAR /1/ when describing the primary circuit overpressure protection system have not provided information confirming fulfillment of the specific requirements of PNAE G-7-008-89. |
| *2.2. Comment* |
| Items 5.4.10, 5.4.11 of the PSAR /1/ do not provide the following information:- about availability of monitoring of displacements of the pressurizer system components including the components of the primary circuit overpressure protection system (item 6.3.1 of PNAE G-7-008-89);- about availability of acoustic and light alarms of upper and lower levels in the pressurizer and the location thereof (item 6.3.2 of PNAE G-7-008-89);- about the necessity of installing the drainage facilities at PRZ PORV relief pipelines (п. 6.2.26 ПНАЭ Г-7-008-89).*(editorial comment)* |
| *2.3. Recommendation* |
| - |
| *2.4. References* |
| PNAE G-7-008-89. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 60 |
| Section No.  | 5.2.2 /1/ |
| Page | 2 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
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| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| According to item 5.2.2.1.2 of the PSAR/1/: “Design pressures ... 7.84 MPa for the ... secondary circuits... are assumed for basic dimensioning on the basis of the results of thermohydraulic calculations provided within the frame of strength analysis”.According to item 10.3.2.2 of PSAR Chapter 10 /D7/: “The steam lines from steam generators to the main turbine valves are designed for the operating pressure of 6.27 MPa (abs.) … . The design pressure is 7.84 МPа (abs.).” |
| *2.2. Comment* |
| Item 5.2.2 of PSAR Chapter 5 /1/ justifies serviceability of the secondary circuit overpressure protection system basing on the design requirement reading as follows: “primary and secondary pressure under actuations of safety devices in design initiating events shall not exceed the design pressure more than by 15 % ”, where the value of 7.84 MPa is assumed as the design pressure (without specifying which pressure is meant: excessive or absolute), which does not comply with the requirements of item 6.2.2 of PNAE G-7-008-89 (item 206 of NP-089-15), whereby the “operating pressure” shall be considered. Here, according to item 10.3.2.2 of PSAR Chapter 10 /D7/, the value of the operating pressure (6.27 MPa) is lower than the value of the design pressure. |
| *2.3. Recommendation* |
| It is recommended that item 5.2.2 of PSAR Chapter 5 /1/ should provide the information about the secondary circuit pressure value selected for designing of the secondary circuit overpressure protection system in accordance with the wording of item 6.2.2 of PNAE G-7-008-89 (item 206 of NP-089-15), as well as should provide a reference to justification of the selected pressure value and specify what pressure is meant: absolute or excessive. |
| *2.4. References* |
| PNAE G-7-008-89. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations.NP-089-15. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 61 |
| Section No.  | 5.2.2 /1/ |
| Page | 17 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Operability of the primary circuit overpressure protection system  |
|  |
| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| According to item 5.4.2.2.3 of the PSAR/1/: “For SG overpressure protection for two pilot safety valves and BRU-А are provided.”According to item 5.3.4.7.1 of the PSAR /D8/: “To select a conservative scenario of the mode an analysis was made, which considered the impact of the initial pressure in the primary circuit, the possible failure of the damaged SG BRU-A operation...”. |
| *2.2. Comment* |
| When analyzing the overpressure protection in the secondary circuit, item 5.2.2 of the PSAR /1/ did not mentioned the operation of the BRU-A (it is not stated whether the BRU-A operation is considered when justifying serviceability of the SG PORV).*(editorial comment)* |
| *2.3. Recommendation* |
| Item 5.2.2 of the PSAR /1/ should provide the information about consideration of BRU-A failures when analyzing serviceability of the secondary circuit overpressure protection system, as well as give a reference to the section (chapter of the PSAR) describing the BRU-A. |
| *2.4. References* |
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3.2. In-service inspection and testing of the primary circuit

According to section 5.2.4 of Chapter 5 of Regulatory Guide 1.70 and Chapter VIII of GS-G-4.1, the information regarding in-service inspection and testing of the metal state of equipment and pipelines within the primary pressure boundary is provided in sections 5.2.4, 5.3.1, 5.3.3, 5.4.1, 5.4.2, 5.4.10, 5.4.11, 5.4.12, 5.4.13 of PSAR Chapter 5 /1/.

The approach for inspection and testing of the primary circuit equipment and pipelines set forth in the specified sections of PSAR Chapter 5 /1/ meets the following rules and regulations in the field of atomic energy use ; PNAE G-7-008-89, PNAE G-7-009-89, PNAE G-7-010-89, PNAE G-7-025-90, NP-068-05.

It is stated in item 5.2.4.1.1 of the PSAR /1/ that the RP equipment and pipelines of group A and B are subject to periodic in-process inspection in a scope established by the requirements of PNAE G-7-010-89 and technical documents specifying the metal state inspection procedure for NPP equipment and pipelines (compliance with the requirements of item 7.1.1 of PNAE G-7-008-89).

The boundaries of the primary circuit systems to be inspected are determined in item 5.2.4.2 /1/. Thus, the requirements of item 1, section 5.2.4 of Regulatory Guide 1.70 have been met.

Items 5.2.4.1.1, 5.2.4.4.1, 5.3.3.7.2, 5.4.2.14.3 of the PSAR /1/ specify in-process inspection of the state of the base metal and welded joints of equipment and pipelines by means of non-destructive methods (visual, liquid penetrant or magnetic particle, ultrasonic, radiographic, eddy-current, mechanical properties) and destructive methods (testing of surveillance specimens) (compliance with the requirements of items 7.3.1 – 7.3.5 of PNAE G-7-008-89, item 1.15 of PNAE G-7-010-89). The approach to the programs for the in-service inspection of the state of the metal of the reactor vessel and surveillance specimens is addressed in subsection 5.3.1.6 of the PSAR /1/. Thus, the requirement of item 3, section 5.2.4 of Regulatory Guide 1.70 has been met.

To perform in-service inspection of the state of metal and welded joints of equipment and pipelines, the in-service inspection standard and work programs are developed, which is stated in item 5.2.4.1.1 of the PSAR /1/ (compliance with the requirements of item 7.2.1 and sections 7.4, 7.5 of PNAE G-7-008-89). Thus, the requirement of item 5, section 5.2.4 of Regulatory Guide 1.70 has been met.

Item 5.2.4.1.3 of the PSAR /1/ provides the information on periodic and unplanned in-service inspections of the metal state (compliance with the requirements of items 7.1.3, 7.1.5, 7.1.6 of PNAE G-7-008-89). The intervals and the scope of the in-service inspection of the metal state are determined in items 5.2.4.5.2, 5.3.3.7.5, 5.4.1.5.4, 5.4.2.14.2 - 5.4.2.14.5, 5.4.13.8.1 of the PSAR /1/ (compliance with the requirements of item 7.1.1 and sections 7.6, 6.2 of PNAE G-7-008-89, section 9.11 of PNAE G-7-010-89). Thus, the requirements of item 4, section 5.2.4 of Regulatory Guide 1.70 have been met.

According to subsections 5.2.4.3.1, 5.4.1.4.1 of PSAR Chapter 5 /1/, the design of the reactor plant provides for ensuring accessibility for examination and inspection of the base metal, corrosion-resistant coatings and welded joints (complies with the requirements of item 8.2.4 of PNAE G-7-008-89). The RP design provides for protective means (unit removable thermal insulation is envisaged) and remote automated inspection means for the areas with no access due to radiation situation (compliance with the requirements of item 2.1.9 of PNAE G-7-008-89). Remote inspection means envisaged for periodic inspection are also used in the framework of the pre-service inspection program to ensure succession of their use. Due to inaccessibility of PRZ room during operation (item 5.4.10.3.2 of the PSAR /1/), appropriate instruments are installed to ensure supervision over its state. The principal diagram of the PRZ monitoring points is presented in Figure 5.4.10.3 of the PSAR /1/ (compliance with the requirements of item 6.3.1 of PNAE G-7-008-89). Thus, the design of the reactor plant for the primary circuit and related system has considered ensuring of the access for in-service inspection and availability of the instruments for remote inspection of such areas of the pressure boundary, which are not accessible for the personnel; thereby, the requirements of item 2 of section 5.2.4 of Regulatory Guide 1.70 are met.

Subsection 5.2.4.7 and item 5.4.2.14.2 of the PSAR /1/ set forth the intervals and the procedure of hydraulic testing with the purpose to check strength and tightness of pressure loaded equipment and pipelines, their components and assembly units as per PNAE G-7-008-89 (complies with the requirements of section 5 of PNAE G-7-008-89). Conducting of periodic inspections and pressure tests as per PNAE G-7-008-89 ensures that the signs of the structural failure or loss of leak tightness during operation will be timely detected in order to make corrective actions prior to deranging of the safety function. Thus, the requirements of item 7, section 5.2.4 of Regulatory Guide 1.70 have been met.

Quality control of welded joints during operation as well as areas of equipment and pipelines repaired by welding is conducted depending on the category of welded joints established by the design documentation as per requirements of PNAE G-7-010-89 (compliance with the requirements of section 11 of PNAE G-7-010-89). If any defects subject to repair are found, one should be guided by the procedure specified in PNAE G-7-008-89. Inspection of the base metal of equipment and pipelines shall be conducted in line with the standards for inspection results evaluation specified in the working inspection programs. The quality of parts made of casting (bodies of check valves) is evaluated as per PNAE G-07-025-90. It is stated in item 5.4.12.4.1 of the PSAR /1/ that the pressurized valves parts (body, cover, etc.) meet the requirements of PNAE G-07-008-89, whereas the welded joints and claddings of the valves are in line with the requirements of PNAE G-7-010-89. Thus, the requirements of item 7, section 5.2.4 of Regulatory Guide 1.70 have been met.

The approach to in-service inspection and testing of the metal state of the primary circuit equipment and pipelines provided in PSAR Chapter 5 /1/ has been developed in line with the documents recognized as inapplicable such as PNAE G-7-008-89 (Rostechnadzor Order No. 69 of 24.02.2016), PNAE G-7-009-89 (Rostechnadzor Order No. 665 of 29.12.2018), PNAE G-7-010-89 (Rostechnadzor Order No. 665 of 29.12.2018).

It should be noted that PSAR Chapter 5 /1/ has not considered the requirements of NP-084-15, NP-089-15, NP-104-18, NP-105-18.

Despite the fact that PSAR Chapter 5 /1/ has been developed to support the requirements of canceled regulatory documents, the provided approach to in-service inspection and testing of the primary circuit mainly does not conflict with the requirements of newly approved federal rules and regulations.

According to the scope and structure, the information provided in the PSAR /1/ meets the provisions of Regulatory Guide 1.70 and Chapter VIII of GS-G-4.1.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 62 |
| Section No.  | 5.2.4 /1/ |
| Page | 12 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | In-service inspection and testing of the primary circuit |
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| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| It is stated in item 5.2.4.6.1 of the PSAR /1/ that quality of deposited corrosion-resistant coating and base metal shall be assessed by the results of in-service inspection in compliance with the requirements of the design documentation for the product or regulatory documents. It should be noted that any reference without indication of the specific documents is incorrect since it is not clear which documents shall be used by the executor during inspection of deposited corrosion-resistant coating and base metal. |
| *2.2. Comment* |
| - |
| *2.3. Recommendation*  |
| The regulatory documents on quality assessment of deposited corrosion-resistant coating and based metal based on in-service inspection results should be made clear in item 5.2.4.6.1 of the PSAR /1/.  |
| *2.4. References* |
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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 63 |
| Section No.  | 5 /1/ |
| Page | 4.3-1, 2÷8, 10 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | In-service inspection and testing of the primary circuit |
|  |
| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| The approach to in-service inspection and testing of the metal state of the primary circuit equipment and pipelines provided in PSAR Chapter 5 /1/is based on the requirements of the regulatory documents recognized as inapplicable. *(Editorial comment)* |
| *2.3. Recommendation* |
| PSAR Chapter 5 /1/ should consider the provisions of NP-084-15, NP-089-15, NP-104-18, NP-105-18.  |
| *2.4. References* |
| NP-084-15. Regulations for control of base metal, welded joints and deposited surfaces during the operation of equipment, pipelines and other elements of NPPs.NP-089-15. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations.NP-104-18. Welding and Cladding of NPP Equipment and Pipelines.NP-105-18. Regulations for control of pipeline and equipment metal of nuclear power installations during manufacture and installation. |

3.3. Primary leaks detection system

Section 5.2.5 of the PSAR /1/ provides the description and analysis of the primary leaks detection system.

The description of the system provided in subsection 5.2.5.1 of the PSAR /1/ contains the following information:

- designation and functions of the system;

- design conditions and input data;

- design principles;

- description of the system components;

- arrangement of the equipment;

- conditions of safe operation;

- operator actions when detecting the primary coolant leak by the system facilities;

- inspections and tests.

The analysis of the system provided in subsection 5.2.5.1 of the PSAR /1/ contains the following information:

- system reliability parameters;

- analysis of system functioning under failures;

- safety analysis of the system design;

- comparison with similar designs.

It is stated in item 5.2.5.1.1.1 of the PSAR /1/ that three leak monitoring systems are provided as a part of RP in connection with the “leak-before-break” safety concept accepted in the technical design of V-528 RP for MCP, pressurizing system surge line and ECCS pipelines:

- acoustic leak monitoring system (ALMS);

- humidity leak monitoring system (HLMS);

- radiation monitoring system (RMS) (it is part of APCS; it ensures monitoring of leakages based on indirect (radiation) parameters);

- comprehensive diagnostics system (ensures comprehensive analysis of the information received from ALMS and HLMS with the purpose to clarify the location and the size of the leak and submit this information to ICIS and ULCS for presentation to the MCR operator),

which meets the requirements of item 4.4.4.6 of NP-001-97 (item 3.4.3.2 of NP-001-15), item 3.3.3 of NP-001-15, item 2.5.13 of NP-082-07.

It is stated in item 5.2.5.1.3.1 of the PSAR /1/ that ALMS and HLMS are the systems important to safety, and their components have classification designation 3Н, which meets their purpose (the requirement of item 2.2 of NP-001-97 (item 2.2 of NP-001-15) is fulfilled) and safety impact (the requirement of items 2.3, 2.5 of NP-001-97 (items 2.3, 2.5, 2.6 of NP-001-15) is fulfilled).

According to item 5.2.5.1.3.1 of the PSAR /1/, ALMS and HLMS are referred to seismic category II, which meets their criticality level to ensure safety in case of seismic effects and operability after an earthquake (the requirement of item 2.6.2 of NP-031-01 is fulfilled).

It is stated in item 5.2.5.1.3.1 of the PSAR /1/ that the accuracy and the time of leak detection, limits and accuracy of leak flowrate measurement have been justified in the detailed project report of MCDS, which comprises ALMS and HLMS. According to item 5.2.5.1.4.2 of the PSAR /1/, the ALMS provides for detection of leaks with the sensitivity no less than: 3.8 L/min (for MCP and monitored RP equipment), 1.9 L/min (for surge line), and 1.5 L/min (for ECCS pipelines). According to item 5.2.5.1.4.3 of the PSAR /1/, the HLMS ensures detection of leaks with the sensitivity at least 1.5 l/min for all monitored points. It is stated in item 5.2.5.1.4.5 of the PSAR /1/ that the ALMS and HLMS provide for detection of leak location in the pipelines, with a relative error of not more  50 % of the distance between the transducers (not more than 4 m) on the pipelines, to which the LBB concept is implemented. Thus, the requirements of item 2.5.13 of NP-082-07 pertaining to justification of the precise location and size of the monitored leak are fulfilled.

According to the requirements of item 4.1.10 of NP-001-97 (item 3.1.18 of NP-001-15),
item 5.2.5.1.6.1 of the PSAR /1/ provides the conditions of safe operation of the primary coolant leak detection system, which envisage availability of fully operable SHC-ALMS and SHC-HLMS as well as RMS during RP operation in the design modes.

Item 5.2.5.1.8 of the PSAR /1/ states that before the beginning of Unit operation, it is necessary to perform the preliminary tests (independent and comprehensive) of ALMS and HLMS according to the corresponding testing programs and procedures, pilot operation and acceptance testing of systems, as well as periodical check of operability of SHC-ALMS and SHC-HLMS measuring channels in the course of operation. This approach complies with the requirements of items 1.2.5, 4.1.10, 5.1.5 of NP-001-97 (items 1.2.7. 3.1.14, 4.1.6 of NP-001-15), items 1.4, 2.1.6, 2.4.15 of NP-082-07.

Item 5.2.5.2.1 of the PSAR /1/ describes the reliability parameters of ALMS and HLMS components, which meets the requirements of item 3.1.17 of NP-001-15.

Item 5.2.5.2.4 of the PSAR /1/ addresses comparison of ALMS and HLMS designs used as part of V-528 RP with the similar designs. According to the PSAR /1/, the engineering solutions implemented in ALMS and HLMS have been approved by the operating experience at the Russian and foreign NPP units, as well as by the results of tests on the measuring channels for the full-scale conditions of VVER-1000 RP operation (Unit 4 of the Balakovo NPP), which meets the requirements of item 1.2.5 of Np-001-97 (item 1.2.7 of NP-001-15).

The scope and structure of the information provided in PSAR section 5.2.5 /1/ meet the provisions of Regulatory Guide 1.70.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 64 |
| Section number  | 5.2.5 /1/ |
| Page | 101, 102 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Primary leaks detection system |
|  |
| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| It is stated in item 5.2.5.1.1.1 of the PSAR /1/ that the primary coolant leak monitoring systems, ALMS and HLMS, are included into a set of MCDS, and item 5.2.5.1.3.1 of the PSAR /1/ specifies that the requirements for ALMS and HLMS predetermine the status of these systems as the diagnostic (information) systems. |
| *2.2. Comment* |
| Section 5.2.5 of the PSAR /1/ does not reflect the attribution of the primary coolant leak detection facilities (ALMS and HLMS) to the safety-related control systems of normal operation; their compliance with the requirements for NPP safety-related control systems established in NP-026-16 (NP-026-16) is not confirmed either (non-compliance with the requirements of items 1.1.2, 4.4.4.6 of NP-001-97 (items 1.1.2, 2.11, 3.4.3.2 of NP-001-15). |
| *2.3. Recommendation* |
| - |
| *2.4. References* |
| NP-001-97. (PNAE G-01-011-97). General Provisions For Nuclear Power Plant Safety Assurance (OPB-88/97).NP-001-15. General Provisions on Nuclear Power Plant Safety Assurance.NP-026-04. Requirements for the Control Safety-Related Systems of Nuclear Power Plants.NP-026-16. Requirements for the Control Safety-Related Systems of Nuclear Power Plants. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 65 |
| Section number  | 5.2.5 /1/ |
| Page | 101 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Primary leaks detection system |
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| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| It is stated in item 5.2.5.1.3.1 of the PSAR /1/ that the accuracy and the time of leak detection, limits and accuracy of leak flowrate measurement have been justified in the detailed project report of MCDS, which comprises ALMS and HLMS. |
| *2.2. Comment* |
| The accuracy of the primary coolant location is not justified in PSAR Chapter 5 /1/, whereas the accuracy of the leak size is justified in the “detailed project report of MCDS” as per item 5.2.5.1.3.1 of the PSAR /1/ (non-compliance with the requirements of item 2.5.13 of NP-082-07). |
| *2.3. Recommendation* |
| - |
| *2.4. References* |
| NP-082-07. Nuclear Safety Regulations for NPP Reactor Installations |

3.4 Structural materials of the reactor pressure vessel and top head

According to the requirement of item 5.3.1 of Regulatory Guide 1.70, the information on the structural materials of the reactor pressure vessel and top head is provided in section 5.3.1 of the PSAR /1/.

It is stated in section 5.3.1 of the PSAR that the following steel grades were used for the reactor vessel components:

* 15Х2НМФА – for manufacturing of the top head, flanges and reactor bottom;
* 15Х2НМФА-А class 1 – for manufacturing of the vessel shells and nozzles;
* 38ХНЗМФА – for manufacturing of fasteners;

- 08Х18Н10Т of a standard type or as modification 08Х18Н10Т-У – for protective jackets of the nozzles and nozzles;

- 20 – for manufacturing of the top head nozzles;

- 22К и 22К-Ш – for manufacturing of top head nozzle flanges and a separating ring.

According to the PSAR /1/, the electrodes, wires or strips of different types (more than 20 items) were used as welding and cladding materials. The nickel-based alloy was used as a gasket between the reactor vessel and top head.

The PSAR /1/ provides the names of the standards for the base materials of the reactor vessel and top head.

It is stated in the PSAR /1/ that the requirements for the base structural, welding (cladding) materials of the reactor vessel and top head as well as regulatory documentation are given in the specifications for structural materials and topical reports for each steel grade.

According to the PSAR /1/, the main structural materials applied to manufacture the reactor vessel and top head are permitted by Rostechnadzor to manufacture the NPP equipment and pipelines according to PNAE G-7-008-89, and the welding (cladding) materials are permitted for welding and cladding by Rostechnadzor according to PNAE G-7-008-89.

It is stated in the PSAR /1/ that the structural materials applied for manufacturing of the reactor vessel and top head were selected on the following basis:

vessel materials were chosen out of the List of materials (semi-finished items) permitted for usage for manufacturing the NPP equipment and pipelines provided in PNAE G-7-008-89;

among the materials specified in the above list, the materials were chosen with the properties meet the requirements for the vessel operability specified in Terms of Reference for the reactor plant.

According to the PSAR /1/, the operability of the materials is corroborated by the experience of operating VVER-type reactor plants (the requirement of items 1.2.2 and 1.2.7 of NP-001-15 regarding operating experience and approval of technical decisions during design and operation are fulfilled).

The PSAR /1/ provides the chemical composition for all base materials of the reactor vessel.

It is also stated in the PSAR /1/ that the welding materials used for welding of the reactor vessel shells located opposite the core ensure that the weld metal contain: 1.0 – 1.3 % nickel, not more than 0.06 % copper, not more than 0.012 % sulphur, not more than 0.008 % phosphorus, not more than 0.010 % arsenic, not more than 0.001 % stannum, not more than 0.008 % antimony, the content of cobalt in corrosion-resistant cladding does not exceed 0.05%. Limitation of the content of the mentioned elements enables ensuring the reactor vessel service life up to 60 years.

The PSAR /1/ also contains the initial values of the critical brittle temperature, Тк0 for the base materials of the reactor vessel and top head, which are equal to:

- 15Х2НМФА class 1 (forging) – minus 45оС;

- 15Х2НМФА-А (forging) – minus 35оС;

- 15Х2НМФА (stamped billet) – 0оС;

- 15Х2НМФА (forging) – minus 20оС.

The values of the critical brittle temperature, Тк0 provided in the PSAR /1 meet the requirements of section 5.8 of PNAE G-7-002-86.

Thus, section 5.3.1 of the PSAR /1/ meets the requirements of item 5.3.1.1 of Regulatory Guide 1.70.

It is stated in section 5.3.1.2.1 of the PSAR /1/ that no non-standard or special processes are applied in manufacturing the reactor top head and vessel (the requirement of item 5.3.1.2 of Regulatory Guide 1.70 is fulfilled).

It is stated in section 5.3.1.2.2 of the PSAR that in the course of manufacturing the reactor vessel components in-shop testing of materials, semi-finished products, welding materials and welded joints (claddings) is performed at the Manufacturer’s in accordance with the requirements of PNAE G-7-009-89, PNAE G-7-010-89. According to the PSAR /1/, the following methods of non-destructive testing are used in the course of inspection: ultrasonic inspection, radiographic inspection, magnetic particle inspection, capillary test (fluorescent flaw detection or dye penetrant test). The ultrasonic and radiographic inspection methods detect internal flaws and the magnetic particle and capillary tests detect the surface flaws. Particular methods and scopes of non-destructive testing of base materials, welded joints and claddings are established in accordance with the Quality Control Tables.

According to the PSAR /1/, the assessment of the quality of base materials, welded joints and claddings with their non-destructive testing is conducted in accordance with the requirements of the specification of structural materials and Quality Control Program.

According to the PSAR /1/, every time before the beginning of work and periodically in line with the established schedule the devices and equipment for the non-destructive testing are subjected to the check in the testing specimens. In the course of the check the operability is established of the equipment in accordance with the certificate data, sensitivity is checked up against the reference specimens and the requirements of the state standards.

It is stated in the PSAR /1/ that there are methodological procedures to perform the non-destructive testing of sheets, forgings, tubes, welded joints and claddings in the plants as well as the branch and state standards that specify the testing conditions and methods.

The specialists who directly perform the non-destructive testing are specially trained and pass appropriate certification test in theory and practical skills in particular types of inspection. After they have successfully completed their training, they are granted certificates to perform the inspection.

The PSAR /1/ gives a list of regulatory documents specifying the metal state inspection procedures.

Thus, section 5.3.1 of the PSAR /1/ meets the requirements of items 5.3.1.3 and 5.3.1.4 of Regulatory Guide 1.70.

According to the scope and structure, the provided information meets the provisions of Regulatory Guide 1.70.

3.5 Surveillance specimen of the RPV material

Item 5.3.1.6 of the PSAR [1] addresses the information related to the program of reactor vessel metal control by surveillance specimens /D12/.

It is stated in item 5.3.1.6.4 of the PSAR /1/ that according to the requirements of item 29 of NP-089-15 and item 19 of NP-089-15, changing of brittle fracture resistance of the vessel materials caused by neutron irradiation and temperature is to be controlled based on the results of surveillance specimens’ testing.

Item 5.3.1.6.2 of the PSAR /1/ specifies that as per requirements of item 6 of Appendix 3 to NP-084-15, the radiation and thermal embrittlement of the vessel materials is controlled after the tests of specimens for eccentric tension of ST type with determination of the fracture mechanics parameters, as well as Charpy tests of specimens for impact bend with determination of the steel critical brittleness temperature. The program /D12/ also comprises tensile tests of specimens. This approach meets the requirements of item 5.3.1.6 of Regulatory Guide 1.70, section 3 of 10 CFR Part 50, Appendix H and section 5 of ASTM E185-10.

As stated in item 5.3.1.6.2 of the PSAR /1/, the surveillance specimens program comprises the specimens cut out of the base metal of irradiated vessel shells, metal of irradiated welds and heat affected zones, which meets the requirements of items 155-156 of NP-089-15.

According to the requirement of item 19 of NP-089-15, item 5.3.1.6.4 of the PSAR /1/ states that container assemblies with irradiated surveillance specimens are located on the reactor vessel inner wall (compliance with the requirement of section 3 of 10 CFR Part 50, Appendix H), whereas the temperature specimens’ assemblies are located in the upper part of the protective tube unit.

It is stated in item 5.3.1.6.11 of the PSAR /1/ that the neutron flux affecting on the surveillance specimens is twice as much as the flux on the reactor vessel, which meets the requirement of item 9 of Appendix 3 to NP-084-15 and item 5.5.2.1 of ASTM E185-10.

According to the requirement of item 8.2.4 of PNAE G-7-002-86, it is stated in item 5.3.1.6.11 of the PSAR /1/ that the irradiation temperature of surveillance specimens differs from the internal wall of the vessel by no more than 10C.

Item 5.3.1.6.8 of the PSAR /1/ specifies that according to the requirements of item 4 of Appendix 3 to NP-084-15, the parameters of neutron fields affecting the surveillance specimens are monitored by neutron flux monitors, and the specimen temperature is monitored by fusible thermal indicators. The neutron field monitors and thermal indicators are located directly in the surveillance specimens. This approach meets the provisions of item 5.5.3 and item 5.7 of ASTM E185-10.

Pursuant to the requirement of item 8 of Appendix 3 of NP-084-15 regarding the quantity of specimens necessary for monitoring of the reactor vessel state within the design service life , item 5.3.1.6.7 of the PSAR /1/ specifies in that the Program /D12/ comprises 12 sets of irradiated specimens, 12 sets of temperature specimens, and 4 sets of reference specimens. The indicated number is sufficient to meet the requirement of item 5.3.1.6 of Regulatory Guide 1.70 regarding the quantity of irradiated capsules.

It is stated in item 5.3.1.6.7 of the PSAR /1/ that the quantity of surveillance specimens in each set determined by the Program /D12/.

The values of the critical brittle temperature of materials in non-irradiated state are determined on the reference sets as per requirements of item 5.5 of Appendix 2 to PNAE G-7-002-86.

On the whole, the information provided in subsection 5.3.1.6 of the PSAR /1/ pertaining to the program of reactor vessel metal control by surveillance specimens meets the requirements of item 5.3.1.6 of Regulatory Guide 1.70 as well as provisions of 10 CFR Part 50, Appendix H and ASTM E185-10.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 66 |
| Section No.  | 5.3.1.6 /1/ |
| Page | 10 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Surveillance specimen of the RPV material |
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| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| It is stated in item 5.3.1.6.2 of the PSAR /1/ that the characteristics of steel small-cyclic fatigue are controlled based on the results of surveillance specimen testing. |
| *2.2. Comment* |
| It is stated in item 5.3.1.6.2 of the PSAR /1/ that the characteristics of steel small-cyclic fatigue are controlled based on the results of surveillance specimen testing. However, there are no samples for small-cyclic fatigue testing in the Surveillance Specimen Program /D12/ (see item 5.3.1.6.5 of the PSAR /1/).*(editorial comment)* |
| *2.3. Recommendation* |
| The phrase “...and the characteristics of small-cyclic fatigue” should be removed from item 5.3.1.6.2 of the PSAR /1/.  |
| *2.4. References* |
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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 67 |
| Section No.  | 5.3.1.6 /1/ |
| Page | 10 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Surveillance specimen of the RPV material |
|  |
| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| It is not stated in item 5.3.1.6 of the PSAR /1/ how the results of surveillance specimen testing shall be documented and how they are going to be used in the safety analysis of the reactor pressure vessel operation.*(editorial comment)* |
| *2.3. Recommendation* |
| It should be stated in item 5.3.1.6 of the PSAR /1/ how the results of surveillance specimen testing shall be documented and how they are going to be used in the safety analysis of the reactor pressure vessel operation. The direction for documentation of surveillance specimen test results and application thereof is given in item 10 of Appendix 3 of NP-084-15. Section 4A of 10 CFR Part 50, Appendix Н specifies the requirements for documentation of surveillance specimen test results. Section 6.1 of ASTM E853-13 gives recommendations for the use of surveillance specimen test results to analyze the integrity of the reactor vessel. |
| *2.4. References* |
| NP-084-15. Regulations for control of base metal, welded joints and deposited surfaces during the operation of equipment, pipelines and other elements of NPPs.10 CFR Part 50, Appendix Н. Reactor Pressure Vessel Surveillance Program Requirements.ASTM E853-13. Standard Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 68 |
| Section No.  | 5.3.1.6 /1/ |
| Page | 13 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Surveillance specimen of the RPV material |
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| **2. EXPLANATION OF THE COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| The information about the strength analysis on the container assemblies-to-reactor vessel fastening is missing from section 5.3.1.6 of the PSAR /1/ (the requirement of subitem 6 of item 5.3.1.6 of Regulatory Guide 1.70).*(Editorial comment)* |
| *2.3. Recommendation* |
| It is recommended that subsection 5.3.1.6 of the PSAR /1/ should give a reference to the calculation justifying the strength of surveillance specimen container assemblies-to-vessel fastening units.  |
| *2.4. References* |
| US NRC Regulatory Guide 1.70. Standard Format and Content of Safety Analysis Report of Nuclear Power Plants. 1978. |

3.6 Strength of the reactor pressure vessel

Section 5.3 of the PSAR for Bushehr NPP Unit 2 /1/ provides the information on analysis of the nuclear reactor vessel integrity as per requirement of item 5.3 of Regulatory Guide 1.70.

Table 5.3.1.1 of the PSAR /1/ specifies the structural materials used to manufacture the reactor vessel, the characteristics of which meet the requirements of Appendix 1 to PNAE G-07-002-86 (base metal) and PNAE G-07-009-89 (weld metal). The operability of the materials provided in Table 5.3.1.1 of the PSAR /1/ is corroborated by the experience of operating VVER-type reactor plants.

As stated in item 5.3.1.3.1 of the PSAR /1/, in the course of manufacturing the reactor vessel components the non-destructive testing of the base metal and welded joints is performed at the Manufacturer’s in accordance with the requirements of PNAE G-7-002-86, PNAE G-010-89, PNAE G-7-014-89 PNAE G-015-90, PNAE G-7-016-90, PNAE G-017-90, PNAE G-7-018-89, PNAE G-019-89, PNAE G-7-030-91, PNAE G-031-91, PNAE G-7-032-91. The requirements of the stated regulatory documents mainly meet the requirements of subsection 5.3.1 of Regulatory Guide 1.70.

According to the requirements of item 1.2.6 of NP-001-15, subsection 5.3.2 of the PSAR /1/ substantiates the permissible pressure and temperature for the reactor vessel for all design modes including hydraulic testing and emergency situations. The substantiating documents were prepared in compliance with the requirements of PNAE G-7-002-86, NP-001-97, PNAE G-07-008-89 and NP-031-01.

Item 5.3.3.1 of the PSAR /1/ describes the design of the reactor vessel and top head. According to the PSAR /1/, the Chief Designer of the water-cooled water-moderated power reactor vessel and top head for Bushehr-2 NPP, Unit 2, is OKB “GIDROPRESS”. It is stated in the PSAR /1/ that the rules and regulations for design are presented in subsection 5.2.1 of the PSAR /1/, whereby the design of the reactor vessel meets the requirements of PNAE G-7-002-86, NP-001-97 (PNAE G-01-011-97), NP-031-01, NP-090-11, NP 082-07, PNAE G-7-008-89, PNAE G-009-89, PNAE G-010-89. Thus, it is demonstrated in the PSAR /1/ that manufacturing and operation of the reactor pressure vessel comply with the highest requirements of federal rules and regulations in the field of atomic energy use (compliance with the requirements of item 1.2.12 of NP-001-97).

The information provided in section 5.3.3.1 of the PSAR /1/ meets the provisions of item 5.3.3.1 of Regulatory Guide 1.70.

Item 5.3.3.2 of the PSAR /1/ describes the main structural materials of the reactor vessel, which are presented in Table 5.3.3.1 of the PSAR /1/. According to the PSAR /1/, the supporting shell of the core is made of steel 15Х2НМФА class 1. The upper shell of the nozzles area, the lower shell of the nozzles area, ECCS nozzles, I&C nozzle are made of steel 15Х2НМФА-А. The ellipsoid of the top cover and the reactor bottom are made of steel 15Х2НМФА. It is stated in the PSAR /1/ that more detailed information about the materials used in manufacturing the reactor top head and vessel, criteria for material selection, as well as measures taken to improve their properties and quality (impurity limitation, melting features, etc.) are given in subsection 5.3.1 of the PSAR /1/.

The information provided in section 5.3.3.2 of the PSAR /1/ meets the provisions of item 5.3.3.2 of Regulatory Guide 1.70.

It is specified in item 5.3.3.3 of the PSAR /1/ that the manufacturing of the vessel shall be performed according to production and process documentation. The methods of manufacturing and the manufacturing process correspond to PNAE G-7-008-89. According to the PSAR /1/, the methods of manufacturing of the reactor vessel and top head are stated in subsection 5.3.1 of the PSAR /1/. The methods stated in subsection 5.3.1 of the PSAR /1/ have been used for manufacturing of 35 vessels and top heads for VVER-1000 reactors. Other methods for manufacturing of the reactor vessels and top heads were not used. Thus, the methods used during manufacturing of the reactor vessel have been tested and approved by previous experience at the other power units (compliance with the requirements of item 1.2.3 of NP-001-97).

The information provided in section 5.3.3.2 of the PSAR /1/ meets the provisions of item 5.3.3.2 of Regulatory Guide 1.70.

Item 5.3.3.4 of the PSAR [1] provides the main requirements for ensuring inspection of the reactor vessel. According to the PSAR /1/, the nondestructive and destructive testing is performed at the Manufacturers during manufacturing of the reactor vessel and top head components for the materials, semi-finished items, welding materials and welded joints (claddings). According to the PSAR /1/, the following methods of non-destructive testing are used in the course of inspection: visual examination, measurement check, ultrasonic test, radiographic test, magnetic powder test, penetrant test. It should be noted that the PSAR /1/ reflects the major inspection methods for the reactor vessel in the course of manufacturing of reactor vessel and top head components (compliance with the requirements of item 91 of NP-089-15).

The information provided in section 5.3.3.4 of the PSAR /1/ meets the provisions of item 5.3.3.4 of Regulatory Guide 1.70.

Item 5.3.3.6 of the PSAR [1] provides the main requirements for the operating conditions of the reactor vessel. It is stated in item 5.3.3.6.1 of the PSAR /1/ that information about design conditions and working conditions is stated in
item 3.9.1.1 of the PSAR /D3/ and in section 4.4 of the PSAR /D4/. The PSAR /1/ specifies that check calculations including calculation for brittle fracture resistance have been performed to justify operability and safe operation of the reactor vessel and top head in a set of the design according to PNAE G-7-002-86. Therefore, check (verification) calculations are envisaged for the reactor vessel (compliance with the requirements of item 5.1.1 of PNAE G-7-002-86).

Pursuant to the requirements of item 1.2.9 of NP-001-15, the calculations justifying the strength of the vessel have been made by means of the software tools, the certificates for which are listed in Table 3.9.1.2.1 of the PSAR /D3/. As stated in item 3.9.3.1.1.2 of the PSAR /D3/, justification of static and cyclic strength of the reactor vessel and components is made in compliance with the requirements of sections 5.4 and 5.6 of PNAE G-07-002-86. The calculations analyzed the static and cyclic strength of the following reactor vessel components:

- vessel with a bottom;

- nozzle Dn 850;

- emergency core cooling system nozzle;

- instrumentation and control nozzle;

- bracket;

- separating ring.

To analyze the static and cyclic strength it was necessary to select the modes characterized by the highest stresses in the reactor vessel components concerned.

The nominal permissible stresses are defined in the mentioned calculations as per requirements of section 3 of PNAE G-7-002-86 and are specified in Table 3.9.3.1.1 /D3/. The analysis of the static strength of the reactor vessel components is made in compliance with the requirements of section 5.4 of PNAE G-7-002-86 for the following design stress category groups: (σ)1, (σ)2, (σ)3W, (σ)4W, as well as for shear and bearing stresses.

The analysis of the cyclic strength of the reactor vessel components in question is made in line with the requirements of section 5.6 of PNAE G-7-002-86. The permissible number of cycles [N0]i was defined for each design cycle based on the stress range values and the cycle asymmetry factor. The total metal damage was defined as a total of ratios of the actual number of load cycles and the permissible number of cycles.

The analysis of the reactor vessel brittle fracture resistance was made as per section 5.8 of PNAE G-7-002-86 and is provided in section 3.9.3.1.1.5 /D3/ and section 5.3.2.1 of the PSAR /1/. It is stated in the PSAR /1/ that the justification of the reactor vessel integrity in terms of brittle fracture resistance was made with the due account taken of the change of material properties in the course of operation by introducing of critical temperature shifting caused by various factors.

Table 5.3.2.3.1 of the PSAR /1/ specifies the values of temperature Тк by the end of service life of 60 years for different components of the reactor vessel cylindrical part. According to the PSAR /1/, the maximum design fluence of fast neutrons with energy Е > 0.5 МeV onto the vessel in the core zone during the specified lifetime of 60 years is presented in PSAR section 4.3 /D4/. It is stated in item 4.3.2.8.1 of the PSAR /D4/ that the maximum value of fast neutron fluence with energy more than 0.5 MeV for the design operating period of 60 years on the reactor vessel will not exceed 4.9∙1019 cm-2. According to the PSAR /D4/, the first, transient and steady-state operation fuel loading of the basic fuel cycle were considered when calculating the fluence accumulated within the operation period.

Item 5.3.2.2 of the PSAR /1/ specifies the type of fracture viscosity temperature dependence KIC for the base metal and welded joints of the vessel corresponding to the requirements of section 5.8 of PNAE G-7-002-86. Permissible values of the stress intensity factor [К1С] were defined based on initial temperature dependence К1С considering the operating modes and safety margins.

The information about ensuring integrity of the reactor vessel provided in item 5.3 of the PSAR /1/ (considering the information provided in the other PSAR chapters) meets in structure and content the provisions of item 5.3 of Regulatory Guide 1.70.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 69 |
| Section No.  | 5.3.3 /1/ |
| Page | 9 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the reactor pressure vessel |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| It is stated in item 5.3.3.6.1 of the PSAR [1] that the brittle fracture resistance analysis of RPV materials is conducted using the procedure described in RD EO 1.1.2.99.0920-2014. However, the references to the mentioned procedure are not provided in section 5.3.2 of the PSAR /1/ when analyzing the vessel materials’ brittle fracture resistance.*(editorial comment)* |
| *2.3. Recommendation* |
| Section 5.3.2 and item 5.3.3.6.1 of the PSAR /1/ should be made consistent in terms of the approaches to analyzing the reactor vessel materials’ brittle fracture resistance. |
| *2.4. References* |
| RD EO 1.1.2.99.0920-2014. Brittle fracture resistance analysis of the VVER reactor vessels at the stage of design. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 70 |
| Section No.  | 5.3 /1/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the reactor pressure vessel |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| It is stated in item 5.3.3.6.1 of the PSAR /1/ that to justify operability and safe operation of the reactor vessel and top head as part of the design as per PNAE G-7-002-86, the strength analyses were performed, based on which a conclusion was made about the feasibility of the reactor vessel integrity up to 60 years. |
| *2.2. Comment* |
| It is not demonstrated in the PSAR /1/ that the strength analyses will be sufficient as per criteria established in PNAE G-7-002-86 to justify the strength of the reactor vessel for analysis period of operation up to 60 years since the criteria established in PNAE G-7-002-86 were selected assuming that the structural strength is ensured upon the 30-years’ service life justification. Thus it is not demonstrated in the PSAR /1/ that the selected approaches to the reactor vessel strength analysis make it possible to justify its safe operation for the period of the entire assigned service life and that the strength is ensured for the entire period of the NPP unit operation (non-compliance with the requirements of item 2.5.3 of NP-082-07).The PSAR /1/ does not provide sufficient justification for the fact that the reactor vessel strength will be ensured in case of the most severe postulated transients (non-compliance with the requirements of item 5.3.3.6 of Regulatory Guide 1.70). |
| *2.3. Recommendation* |
| The sufficiency of the reactor vessel strength analysis to ensure safe operation within the entire assigned NPP service life of 60 years made as per requirements of PNAE G-7-002-86 should be justified in the PSAR /1/. |
| *2.4. References* |
| NP-082-07. Nuclear Safety Regulations for NPP Reactor InstallationsRegulatory Guide 1.70 US NRC Regulatory Guide 1.70. Standard Format and Content of Safety Analysis Report of Nuclear Power Plants. 1978 |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 71 |
| Section No.  | 5.3 /1/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the reactor pressure vessel |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| Table 5.3.2.3.1 of the PSAR /1/ specifies the values of temperature Тк by the end of service life of 60 years for different components of the reactor vessel cylindrical part. |
| *2.2. Comment* |
| The PSAR /1/ does not provide the comparison of the calculated values of the reactor vessel critical brittle temperature with the limiting value of the brittle temperature estimated based on results of the brittle fracture resistance calculation. Thus, it is not demonstrated in the PSAR /1/ that the reactor vessel brittle fracture resistance criteria are met for the entire period of the NPP unit assigned service life (non-compliance with the requirements of item 2.5.3 of NP-082-07).The PSAR /1/ does not provide sufficient justification for the fact that the reactor vessel integrity (in terms of the brittle fracture resistance) will not be compromised in case of the most severe postulated transients (non-compliance with the requirements of item 5.3.3.6 of Regulatory Guide 1.70). |
| *2.3. Recommendation* |
| It is recommended that the PSAR /1/ should demonstrate that the calculated values of the reactor vessel critical brittle temperature do not exceed their limiting permissible values obtained based on results of the brittle fracture resistance calculation. |
| *2.4. References* |
| NP-082-07. Nuclear Safety Regulations for NPP Reactor InstallationsUS NRC Regulatory Guide 1.70. Standard Format and Content of Safety Analysis Report of Nuclear Power Plants. 1978. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 72 |
| Section No.  | 5.3 /1/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the reactor pressure vessel |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| According to the PSAR /1/, the maximum design fluence of fast neutrons with energy Е > 0.5 МeV onto the vessel in the core zone during the specified lifetime of 60 years is presented in PSAR section 4.3 /D4/. It is stated in item 4.3.2.8.1 of the PSAR /D4/ that the maximum value of fast neutron fluence with energy more than 0.5 MeV for the design operating period of 60 years on the reactor vessel will not exceed 4.9∙1019 cm-2. According to the PSAR /D4/, the first, transient and steady-state operation fuel loading of the basic fuel cycle were considered when calculating the fluence accumulated within the operation period. |
| *2.2. Comment* |
| The PSAR /1 does not provide the comparison of the calculated fast neutron fluence values with the energy of Е > 0.5 MeV on the reactor vessel with the limiting value of the fluence justified by the brittle fracture resistance calculation, which does not make it possible to conclude that the criteria for the reactor vessel burn-up life are met. Therefore, the PSAR /1/ does not demonstrate that the calculations of the fluence on the vessel were made with observance of the conservative approach principle (non-compliance with the requirements of item 1.2.3 of NP-001-97).The PSAR /1/ does not provide the neutron flux distribution and the core spectrum at the main core boundaries and in the high-pressure vessel wall (non-compliance with the requirements of item 4.3.2.8 of Regulatory Guide 1.70). |
| *2.3. Recommendation* |
| It is recommended that the PSAR /1/ should demonstrate that the calculated fast neutron fluence values with the energy of Е > 0.5 MeV on the reactor vessel do not exceed their limiting permissible values obtained based on results of the brittle fracture resistance calculation.It is recommended that the PSAR /1/ should provide the neutron flux distribution and the core spectrum at the main core boundaries and in the high-pressure vessel wall. |
| *2.4. References* |
| NP-001-97. (PNAE G-01-011-97). General Provisions for Nuclear Power Plant Safety Assurance (OPB-88/97).US NRC Regulatory Guide 1.70. Standard Format and Content of Safety Analysis Report of Nuclear Power Plants. 1978. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 73 |
| Section No.  | 5.3 /1/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the reactor pressure vessel |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| According to item 5.3.3.7.1 of the PSAR /1/, the reactor vessel and top head are subject to periodic examination and inspection PSAR /1 during operation according to PNAE G-008-89. |
| *2.2. Comment* |
| When substantiating the sufficiency of the reactor vessel inspection program in the PSAR /1/, the criteria and requirements established in PNAE G-07-008-89, which is canceled as of now, were used. However, at present the requirements for the reactor vessel inspection programs are established in accordance with effective federal rules and regulations NP-089-15 and NP-084-15. Thus, the criteria used in the PSAR /1/ are incorrect and do not meet the requirements of federal rules and regulations (non-compliance with the requirements of item 1.1.2 of NP-001-97).It is not confirmed in the PSAR /1/ that the program for inspection of radiation embrittlement and thermal aging of the reactor vessel is envisaged for the reactor vessel (non-compliance with the requirements of item 19 of NP-089-15).The PSAR did not justify the sufficiency of in-service inspection programs and reactor vessel metal inspection programs (non-compliance with the requirements of item 5.3.3.7 of Regulatory Guide 1.70). |
| *2.3. Recommendation* |
| It is recommended that the PSAR /1/ should provide the information about the reactor vessel metal inspection programs developed in accordance with effective federal rules and regulations NP-089-15 and NP-084-15.The PSAR /1/ should provide the information about the program for inspection of radiation embrittlement and thermal aging of the reactor vessel.The PSAR should justify the sufficiency of in-service inspection programs and reactor vessel metal inspection programs.  |
| *2.4. References* |
| NP-001-97. (PNAE G-01-011-97). General Provisions For Nuclear Power Plant Safety Assurance (OPB-88/97).PNAE G-7-008-89. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations.NP-084-15. Regulations for control of base metal, welded joints and deposited surfaces during the operation of equipment, pipelines and other elements of NPPs.NP-089-15. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations.US NRC Regulatory Guide 1.70. Standard Format and Content of Safety Analysis Report of Nuclear Power Plants. 1978. |

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# 3.7 Strength of the reactor upper unit

The brief information about the reactor upper unit components’ strength analysis is provided in the following sections of the PSAR for Bushehr NPP Unit 2 - 3.2.1 /D3/, 4.1 /D4/ and 5.3.1, 5.3.3 /1/.

The information about the reactor upper unit classification as per requirements of NP-001-15, NP-089-15 and NP-031-01 is provided in section 3.2.1 of the PSAR /D3/. Section 5.3.3 of the PSAR /1/ provides the information about compliance of the reactor upper unit and top head with the RF regulatory documentation requirements. Thus, it is demonstrated that the requirements of item А1.3 of INRA-MA-RE-000-00/02 are fulfilled.

The design of the reactor upper unit and top head is described in PSAR sections 4.1 /D4/ and 5.3.3.1 /1/.

The information about the reactor upper unit materials and mechanical properties necessary for strength analyses is provided in section 5.3.1 of the PSAR /1/. The technical specifications (TU) listed in section 5.3.1.1.1 of the PSAR /1/ for delivery of materials for manufacturing of the upper unit and top head are included into the Summary List of Standardization Documents /D11/, which meets the requirements of item 85 of NP-089-15.

According to PSAR item 4.1.13 /D4/, the upper unit is intended for sealing of the reactor main joint, arrangements of drives, sealing of outputs of in-core instrumentation detectors of in-core instrumentation system, sealing of the nozzle of outlet of air and gases from the reactor, holding of PTU, FA and core barrel against their lifting. The upper unit structure combines the elliptical top head with flange and nozzles, metalwork, cross-piece, leak indicator.

The general information on the approaches to carrying out strength analyses for RP equipment and pipelines is provided in Section 3.9 of the PSAR /D3/, which contains:

- a list of design modes of the RP operation (NOC, AOO, DBA);

- list of software tools used in strength analyses as well as information on certification thereof;

- information about experimental research performed to substantiate the strength of equipment and pipelines;

- information about the method of determination of loads and stresses in the components of equipment and pipelines.

According to the Section 3.9.3 of the PSAR /D3/ the strength of the reactor upper unit is substantiated in accordance with the requirements of PNAE G-7-002-86 and NP-031-01. The mentioned section also contains the information about the approaches to determination of permissible stresses when making calculations on the static strength, cyclic strength, seismic impact strength and brittle fracture resistance, which meet the requirements of PNAE G-7-002-86 and NP-031-01.

According to sections 3.9.3.1.2 and 3.9.3.1.3 /D3/, the strength of the reactor upper unit and top head components is substantiated for all design modes and all combinations of loads envisaged by the requirements of PNAE G-7-002-86 and NP-031-01 for equipment of seismic category I. Thus, it is demonstrated that the requirements of item E22 of INRA-MA-RE-000-00/02 in terms of seismic loads in the upper unit strength analysis are fulfilled.

According to the PSAR /D3/, a full section break of one of the pipelines connected to the RCP, for which the “leak before break” concept with maximum impact on the reviewed equipment or the rupture of CSS casing is not applicable, was considered as a DBA for the reactor upper unit and top head components.

According to item 5.3.3.3 of the PSAR /1/, the reactor top head integrity is assured by the approved manufacturing process, application of well proved steels with uniform properties and appropriate characteristics. Material properties are confirmed by certification studies performed both in the initial state and in the irradiated state at operational parameters. During operation the inspection programs are implemented as well as the studies of materials of the reactor vessel and top head to guarantee its further safe keeping. Thus, it has been demonstrated that the requirements of item 1.2.5 of NP-001-97 (item 1.2.7 of NP-001-15) and item А1.19, C9.5 of INRA-MA-RE-000-00/02 are fulfilled.

According to the PSAR /D3/, check (verification) calculations including the brittle fracture resistance calculation were made to justify the strength of the reactor top head as per PNAE G-7-002-86. Basing on the results of the calculation on the reactor top head brittle fracture resistance, the minimum values of hydraulic strength and density test temperature were determined.

Thus, the information provided in items 3.2.1 of PSAR Chapter 3 /D3/, 4.1 PSAR Chapter 4 /D4/ and items 5.3.1, 5.3.3 of PSAR Chapter 5 /1/ regarding the strength analysis of the reactor upper unit meet in its structure in content the provisions of items 3.2, 4.1, 5.3.1 and 5.3.3 of Regulatory Guide 1.70.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 74 |
| Section No.  | 5.3.1 /1/ |
| Page | 2-4 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the reactor upper unit |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| PSAR Table 5.3.1.1 /1/ lists the materials used for manufacturing of the reactor vessel and top head. |
| *2.2. Comment* |
| Out of the total list of materials used for manufacturing the reactor vessel and top head and presented in Table 5.3.1.1 of the PSAR /1/, Section 5.3.1 of the PSAR /1/ gives a reference to the technical specifications (TU) only for steels 15ХН2НМФА, 15ХН2НМФА-А, 15ХН2НМФА class 1. *(editorial comment)* |
| *2.3. Recommendation* |
| Section 5.3.1 of the PSAR /1/ should provide the information about the technical specifications (TU) for all materials applied for manufacturing of the reactor vessel and top head.  |
| *2.4. References* |
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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 75 |
| Section No.  | 5.3.1 /1/ |
| Page | 9 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the reactor upper unit |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| According to item 5.3.1.4.2 of the PSAR /1/, the requirements for the manufacturing, processing and inspection of the base austenitic stainless steels and welding materials are specified in subsection 5.3.2 of the PSAR /1/. |
| *2.2. Comment* |
| According to item 5.3.1.4.2 of the PSAR /1/, the requirements for the manufacturing, processing and inspection of the base austenitic stainless steels and welding materials are specified in subsection 5.3.2 of the PSAR /1/. However, the mentioned section of the PSAR /1/ does not contain any requirements for manufacturing, processing and inspection of the base austenitic stainless steels and welding materials.*(editorial comment)* |
| *2.3. Recommendation* |
| A reference in item 5.3.1.4.2 of the PSAR /1/ should be corrected. |
| *2.4. References* |
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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 76 |
| Section No.  | 5.3.3 /1/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the reactor upper unit |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| The PSAR /1/ should be supplemented with the information about the strength analyses made for the reactor upper unit envisaged by the requirements of item 1.2.2 of PNAE G-7-002-86 and comprising the calculation for selection of the main dimensions and verification strength calculations, as well as provide the main results of such calculations. |
| *2.4. References* |
| PNAE G-7-002-086. Regulations for strength analysis of equipment and pipelines of nuclear power installations. |

3.8 Strength of the reactor support structures

According to the requirement of sections 3.9.3 and 5.4.14 of Regulatory Guide 1.70, the information on the strength analysis of reactor support structures is provided in section 5.4.14 of the PSAR /1/ as well as in section 3.9.3.4 of the PSAR /D3/.

It is stated in section 5.4.14 of the PSAR /1/ that the reactor fastening structure consists of three independent units: lower supporting unit, upper thrust unit and wiring unit with a fixture for the upper unit fastening. The supporting unit includes a supporting truss and a supporting shell shoulder. The supporting truss is a welded metalwork consisting of radially-arranged beams. The beams are the boxes made of the vertical and horizontal ribs. The rib external ends are combined by plates. The beams in their lower part are combined by plate. The reactor vessel is fastened to the supporting unit through a supporting ring.

According to the PSAR /1/, the supporting truss takes the force from the reactor vessel supporting ring. The loads from the reactor are transferred through the supporting ring, wedges and keys to the truss beams, and from the truss beams through the lower plate act on the shoulder of supporting shell fastened in the concrete of civil structure. The supporting unit receives vertical loads and prevents the reactor vessel from turning round the longitudinal axis, tilting and horizontal displacement.

According to the PSAR /1/, the thrust unit includes a thrust truss and a thrust ring. Thrust truss is a welded metalwork consisting of support that is rigidly fastened in concrete, and the arms welded to the support. The arms get into the slots of the thrust ring slipped over the reactor vessel flange.

Support is a welded metalwork comprising a cylindrical shell, circumferential plates and stiffening ribs. Anchor rods embedded in concrete are welded to the ribs.

The upper thrust unit prevents the reactor vessel from tilting and horizontal displacement.

Fixture for upper unit seismic fastening is mounted on the wiring unit.

Free thermal expansion of the reactor is ensured by the structure of the lower and upper fastening units, and fixture for the upper unit fastening.

According to the PSAR /1/, the design of the reactor supports shall provide for functioning as per NP-001-15 and take up load combinations as per NP-031-01.

MCP instantaneous transverse break in the area of its welded joint to the reactor vessel nozzle with free double-ended coolant leak from MCP and the reactor is assumed as DBA in strength analyses of the reactor support structures.

The reactor supports’ welded joints shall be made and inspected as per rules of PNAE G-7-009-89 and PNAE G-7-010-89.

It is stated in section 3.9.3.4 of the PSAR /D3/ that the following load types have been considered in the verification strength analyses of the reactor support structures:

- the weight of the reactor vessel together with the core, reactor internals and the coolant;

- forces on the reactor vessel from the adjoining pipelines;

- thermal loads;

- seismic loads;

- emergency loads occurring in case of pipelines rupture.

Thus, section 3.9.3.4 of the PSAR /D3/ considered the requirements of section 3.9.3.1 of Regulatory Guide 1.70.

According to the PSAR /D3/, the verification calculation for the components’ supports contains calculations for static and cyclic strength as well as brittle fracture resistance.

The verification calculation for supports was made to determine the stresses therein at all values of loads and temperatures in the reactor plant operating modes specified in the design and comparison of the obtained stresses with the permissible values. The following design modes and load combinations were addressed:

- normal operating conditions;

- anticipated operational occurrences;

- imposition of the safe shutdown earthquake on normal operation conditions - NOC+SSE and on anticipated operational occurrences - AOO+SSE;

- imposition of the operation basis earthquake on normal operation conditions - NOC+OBE and on anticipated operational occurrences - AOO+OBE.

The mentioned load combinations meet the requirements of section 5 of NP-031-01; the requirements of section 3.9.3.1 of Regulatory Guide 1.70 have been considered as well.

The values of permissible stresses for various design load combinations were determined as per requirements of the Russian regulatory document, OTT 1.5.2.01.999.0157-2013.

According to the PSAR /D3/, the breaking strength and the yield strength of the material were used when determining the permissible stresses’ values; here the changes in design characteristics of materials of supports depending on the temperature were taken into account. Justification of strength of the reactor support structures is based on the analysis of the stressed state of support structures’ components. The values of permissible stresses for different calculation combinations were determined similarly to the calculations for equipment and pipelines.

Thus, section 3.9.3 of the PSAR /D3/ meets the requirements of section 3.9.3.1 of Regulatory Guide 1.70. Regarding providing of information on testing of supporting structures’ components. It is stated that the standard design of the damper consists of the cylindrical body filled with viscous actuation fluid, piston and internal components plunged into the actuation fluid. The dampers do not restrict thermal displacements of equipment and pipelines. The maximum force of the damper's resistance to thermal displacements of NPP components at positive ambient temperatures does not exceed 200 N. The assigned service life of viscoelastic dampers is 60 years.

According to the scope and structure, the information provided does not meet the provisions of Regulatory Guide 1.70, which is reflected in the comment provided in section 2.2.10.2 of the present Review Report.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 77 |
| Section No.  | 3.9.3.4 /D3/ |
| Page | 74-76 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the reactor support structures |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| Section 3.9.3.4 of the PSAR /D3/ addresses not all data, which are required by section 3.9.3.1 of Regulatory Guide 1.70 for the reactor support structures. |
| *2.2. Comment* |
| Section 3.9.3.4 of the PSAR /D3/:* does not contain the information about the applied methods for calculation and testing of the reactor supports in case of failures thereof;
* does not specify the maximum stresses, strains and cumulative coefficient for all operating modes for all the components;
* does not determine those values of stresses, which differ from the permissible limits by less than 10% (the requirements of section 3.9.3.1 of Regulatory Guide 1.70 are not fulfilled).
 |
| *2.3. Recommendation* |
| Section 3.9.3.4 of the PSAR /D3/ should be supplemented with the information about the applied methods for calculation and testing of the reactor supports in case of failures thereof, maximum stresses, strains and the cumulative coefficient for all operating conditions for all components, as well as about the values of stresses, which differ from the permissible limits by less than 10%. |
| *2.4. References* |
| US NRC Regulatory Guide 1.70. Standard Format and Content of Safety Analysis Report of Nuclear Power Plants. 1978. |

3.9. Assessment of the PRV destruction probability analysis

According to the requirement of item 1.2.13 of NP-001-15, the probability of destruction of the VVER-1000 reactor vessel for Bushehr-2 NPP within 1 year shall not exceed 10-7.

According to the recommendations of item 27 of IAEA guide INSAG-12, the frequency of occurrence of severe core damage for the NPPs being operated shall be below 10–4 events per plant operating year. At the same time, item 27 of INSAG-12 states that the target indicator for the probability of large off-site releases is non-exceeding of the frequency of such events being 10-5 per plant operating year.

The analysis on the probability of the Bushehr-2 NPP reactor vessel destruction shall be made for the design operating conditions within 60 years. The general probability of the vessel destruction shall be determined in the PSAR /1/ as a sum total of destruction probabilities for its cylindrical part, nozzle area, elliptical bottom and top head.

Pursuant to the requirement of item 2.1.15 of NP-082-07, determination of temperature fields and the stressed state of the vessel component, as well as calculation of the Bushehr-2 NPP reactor vessel destruction probability on their basis shall be made by means of the certified software tools.

The probabilistic analysis of the reactor vessel destruction was made in three main stages:

- statistical analysis of the input data for calculations;

- calculations of probability of reactor vessel component destruction;

- determination of the general probability of the vessel destruction;

- comparison of the general vessel destruction probability with the criterial value of 10-7 per reactor per year.

Based on the input information analysis, the following was determined: parameters of statistical laws of mechanical properties’ distribution (yield strength, ultimate strength, relative elongation and contraction ratio, critical brittle temperature), characteristics of the material brittle fracture resistance, size and quantity of design cracks in the vessel components, as well as the values of the neutron fluence for the unit service life of 60 years.

The modes related to the normal process of the unit operation and emergency modes caused by equipment failures, personnel errors and external effects were considered in the vessel destruction probability analysis. The design basis events comprising the modes important in terms of the brittle fracture and fatigue growth of defects were considered when determining the reactor vessel destruction probability.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 78 |
| Section No.  | 5 /1/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Assessment of the PRV destruction probability analysis |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| It is not demonstrated in the PSAR /1/ that pursuant to the requirement of item 1.2.13 of NP-001-15 the Bushehr-2 NPP reactor vessel destruction probability will not exceed 10-7 within 1 year.  |
| *2.3. Recommendation* |
| The PSAR /1/ should provide the information about the BNPP-2 reactor vessel destruction probability with the interval of 1 year and show compliance with the requirement of item 1.2.13 of NP-001-15. |
| *2.4. References* |
| NP-001-15. (PNAE G-01-011-97). General Provisions for Nuclear Power Plant Safety Assurance (OPB-88/97). |

3.10. RCP **Strength**

The brief information about the RCP strength analysis is provided in the following sections of the PSAR for Bushehr NPP Unit 2 - 3.2.1, 3.9.1-3.9.3 /D3/ and 5.2.1, 5.2.3, 5.4.1 /1/.

The information about the RCP classification as per requirements of NP-001-15, NP-089-15 and NP-031-01 is provided in section 3.2.1 of the PSAR /D3/. The information about the RCP classification as per requirements of NP-001-15, PNAE G-7-008-89 and NP-031-01 is provided in section 5.4.1.1 of the PSAR /1/. Sections 5.2.1, 5.4.1 of the PSAR /1/ provide the information about compliance of the RCP with the RF regulatory documentation requirements. Thus, it is demonstrated that the requirements of item А1.3 of INRA-MA-RE-000-00/02 are fulfilled.

The design of the RCP set is described in section 5.2.1.2 of the PSAR /1/.

The information about the RCP materials and mechanical properties necessary for strength analyses is provided in section 5.2.3 of the PSAR /1/. The regulatory documents listed in Table 5.2.3.2 of the PSAR /1/ for delivery of materials for RCP manufacturing are included into the Summary list of documents of standardization /D11/, which meets the requirements of item 85 of NP-089-15.

It is stated in section 5.4.1 of the PSAR /1/ that during development of RCP set design, its individual units were modified, which made it possible to avoid failures typical for GTsN-195M. Reactor coolant pump GCN-195М is assumed as a reference pump, which is now in operation in all VVER-1000 type NPPs and GCNA-1391 in the projects of the Kudankulam, Bushehr, Tianwan and AES-2006 NPPs. So far considerable experience of operating this pump sets has been gained which allows determining the trends in improving the RCP set design for the new reactor plants. The experience was widely used in the course of designing the reactor coolant pump set for Bushehr-2 NPP RP design. The base structural materials used in designing of some GCNA-1391 components were chosen with regard for the positive experience of operation of GCN-195М and GCNA-1391 (in the RP designs for Kudankulam, Bushehr, Tianwan and AES-2006 NPPs). Thus, it has been demonstrated that the requirements of item 1.2.5
of NP-001-97 (item 1.2.7 of NP-001-15) and item А1.19, C9.5 of INRA-MA-RE-000-00/02 are fulfilled.

The general information on the approaches to carrying out strength analyses for RP equipment and pipelines is provided in Section 3.9 of the PSAR /D3/, which contains:

- a list of design modes of the RP operation (NOC, AOO, DBA);

- list of software tools used in strength analyses as well as information on certification thereof;

- information about experimental research performed to substantiate the strength of equipment and pipelines;

- information about the method of determination of loads and stresses in the components of equipment and pipelines.

It is stated in section 3.9.2 of the PSAR /D3/ that the programs of precommissioning testing envisage monitoring of VVER RCP vibration and thermal displacement parameters to confirm the vibration strength of RP equipment and pipelines. It is stated in the PSAR /D3/ that precommissioning vibration measurements on the VVER prototype facility demonstrated that the level of vibration loading of inspected pipelines is not high.

According to the PSAR /D3/, the comprehensive approach to testing of the vibration condition of the pipeline systems performed by means of both stationary measuring facilities and portable vibration measurement equipment proved to be efficient in the course of precommissioning testing of a number of VVER RP units being commissioned. The programs of such dynamic testing providing for conducting of vibration testing of ECCS pipelines, PRZ, steam lines and feedwater pipelines simultaneously with the vibration measurements of the RCP set and reactor internals made it possible to identify quite rapidly the increased dynamic activity of the RCP, assess its impact on the life of piping systems as well as develop recommendations for ensuring the design levels of the equipment vibration loading.

According to the Section 3.9.3 of the PSAR /D3/ the strength of the RCP set is justified in accordance with the requirements of PNAE G-7-002-86 and NP-031-01. The mentioned section also contains the information about the approaches to determination of permissible stresses when making calculations on the static strength, cyclic strength, seismic impact strength and brittle fracture resistance, which meet the requirements of PNAE G-7-002-86 and NP-031-01.

According to section 5.4.1.3 of the PSAR /1/, the strength calculations for the RCP set have been justified in two stages as per requirements of PNAE G-7-002-86: calculation for basic dimensioning and check (verification) calculation. According to the PSAR /1/, the strength of pump casing components, auxiliary system equipment, supporting-connecting structures and rotor of RCP set shall be ensured under all the operation conditions with various combinations of loads (OBE, SSE, aircraft crash, DBA) throughout the service life equal to 60 years. The flywheel structure shall ensure its integrity and strength under all the modes.

Section 3.9.3.2.2 of the PSAR /D3/ provides the information about the performed strength analyses for RCP set components. According to /D3/, analysis of stresses and fatigue considering the temperature fields was made at the stage of the check calculation as per PNAE G-7-002-86. The calculations considered the loads on pump nozzles induced by pipelines under various operating conditions as well as seismic loads. Thus, it is demonstrated that the requirements of item E22 of INRA-MA-RE-000-00/02 in terms of seismic loads in RCP strength analysis are fulfilled.

Besides, according to the PSAR /D3/, the analysis of stresses and fatigue of the pump rotating parts and components was made. The lifetime was estimated based on linear summation of damages for all modes, including the emergency ones. The maximum accumulated fatigue damage in the vessel components and fastenings does not exceed the permissible values of PNAE G-7-002-86.

Thus, the information provided in items 3.2.1, 3.9.1, 3.9.3 of PSAR Chapter 3 /D3/, and items 5.2.1, 5.2.3, 5.4.1 of PSAR Chapter 5 /1/ regarding the strength analysis of the RCP set meet in its structure in content the provisions of items 3.2, 3.9.1, 3.9.3, 5.2.1, 5.2.3 of Regulatory Guide 1.70 and does not meet the provisions of section 5.4 of Regulatory Guide 1.70, which is stated in the comment in section 2.2.12.2 of the present Review Report.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 79 |
| Section No.  | 5.4.1/1/ |
| Page | 10 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the RCP set |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| It is stated in section 5.4.1 of the PSAR /1/ that the flywheel structure shall ensure its integrity and strength under all the modes. |
| *2.2. Comment* |
| It is not demonstrated in section 5.4.1 of the PSAR /1/ how the recommendations of Regulatory Guide 1.14 “Reactor Coolant Pump Flywheel Integrity” are observed (non-compliance with the requirements of item 5.4.1.1 of Regulatory Guide 1.70). |
| *2.3. Recommendation* |
| - |
| *2.4. References* |
| US NRC Regulatory Guide 1.70. Standard Format and Content of Safety Analysis Report of Nuclear Power Plants. 1978.US NRC Regulatory Guide 1.14. “Reactor Coolant Pump Flywheel Integrity”. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 80 |
| Section No.  | 5.4.1 /1/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the RCP set |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| The PSAR /1/ should be supplemented with the information about the strength analyses made for the RCP set envisaged by the requirements of item 1.2.2 of PNAE G-7-002-86 and comprising the calculation for selection of the main dimensions and verification strength calculations, as well as about the main results of such calculations. |
| *2.4. References* |
| PNAE G-7-002-086. Regulations for strength analysis of equipment and pipelines of nuclear power installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 81 |
| Section No.  | 5.2.3 /1/ |
| Page | 5 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the RCP set |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| Not all documents specifying supply of materials for manufacturing of the primary system components are listed in Table 5.2.3.2 of the PSAR /1/.  |
| *2.2. Comment* |
| Table 5.2.3.2 of the PSAR /1/ does not provide any information about the technical specifications (TU) for supply of RCP fastenings made of steel 38ХН3МФА as per Table 5.2.3.1 of the PSAR /1/.*(editorial comment)* |
| *2.3. Recommendation* |
| The PSAR /1/ should be supplemented with the information about the technical specifications (TU) for supply of RCP fastenings made of steel 38ХН3МФА. |
| *2.4. References* |
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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 82 |
| Section No.  | 3.9.3 /D3/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the RCP set |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| It is recommended that the section 3.9.3.1 PSAR /D3/ should be supplemented with the information on the main loads and combinations of loads adopted for the RCP strength analyses, as well as with the list of RCP components, for which such strength analyses were made. |
| *2.4. References* |
| - |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 83 |
| Section No.  | 5.4.1 /1/ |
| Page | 14 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the RCP set |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| It is stated in section 5.4.1.6 of the PSAR that “Normal functioning of RCP set system is based on the conditions of long-term parallel operation of four RCP sets in the circuit at nominal coolant parameters of RP at "Rooppur" NPP.” |
| *2.2. Comment* |
| It is stated in section 5.4.1.6 of the PSAR that “Normal functioning of RCP set system is based on the conditions of long-term parallel operation of four RCP sets in the circuit at nominal coolant parameters of RP at "Rooppur" NPP.” However, the PSAR /1/ is developed for Bushehr NPP Unit 2 and not for the Rooppur NPP.*(editorial comment)* |
| *2.3. Recommendation* |
| - |
| *2.4. References* |
| - |

3.11. Strength of the SG

The brief information about the SG components’ strength analysis is provided in the following sections of the PSAR for Bushehr NPP Unit 2 - 3.2.1, 3.9.1-3.9.3 /D3/ and 5.2.3, 5.4.2 of the PSAR /1/.

The information about the SG classification as per requirements of NP-001-15, NP-089-15 and NP-031-01 is provided in section 3.2.1 of the PSAR /D3/. The information about the SG classification as per requirements of NP-001-97 and NP-031-01 is provided in section 5.4.2.1 of the PSAR /1/. Section 5.2.1, of the PSAR /1/ provides the information about compliance of the SG with the RF regulatory documentation requirements. Thus, it is demonstrated that the requirements of item А1.3 of INRA-MA-RE-000-00/02 are fulfilled.

The design of the SG is described in section 5.2.2.2 of the PSAR /1/.

The information about the SG materials and mechanical properties necessary for strength analyses is provided in section 5.2.3 of the PSAR /1/. The regulatory documents listed in Table 5.2.3.2 of the PSAR /1/ for delivery of materials for SG manufacturing are included into the Summary list of documents of standardization /D11/, which meets the requirements of item 85 of NP-089-15.

The general information on the approaches to carrying out strength analyses for RP equipment and pipelines is provided in Section 3.9.1 of the PSAR /D3/, which contains:

- a list of design modes of the RP operation (NOC, AOO, DBA);

- a list of software tools used in strength analyses;

- information about experimental research performed to substantiate the strength of equipment and pipelines;

- information about the method of determination of loads and stresses in equipment components and pipelines under emergency conditions.

It is stated in section 3.9.2 of the PSAR /D3/ that the programs of precommissioning testing envisage monitoring of VVER SG vibration and thermal displacement parameters to confirm the vibration strength of RP equipment and pipelines.

According to the Section 3.9.3 of the PSAR /D3/ the strength of the SG is justified in accordance with the requirements of PNAE G-7-002-86 and NP-031-01. The mentioned section also contains the information about the approaches to determination of permissible stresses when making calculations on the static strength, cyclic strength, seismic impact strength and brittle fracture resistance, which meet the requirements of PNAE G-7-002-86 and NP-031-01.

According to section 3.9.3.1.6 of the PSAR /D3/, the strength of SG components is substantiated for all design modes and all combinations of loads envisaged by the requirements of PNAE G-7-002-86 and NP-031-01 for equipment of seismic category I.

Section 5.4.2.7 of the PSAR /1/ provides the information on the justification of strength and vibration stability of SG HET; section 5.4.2.12 of the PSAR /1/ provides the information about the calculational and experimental justification of SG strength. According to the PSAR /1/, the calculational and experimental justification has been made for the SG service life being at least 60 years. According to the PSAR /1/, strength analyses were made as envisaged by the requirements of PNAE G-7-002-86; it was concluded based on the results of these analyses that the strength of SG components including vibration strength for SG HET meets the requirements of PNAE G-7-002-86.

According to the PSAR /1/, the experience of designing and operation of steam generators PGV-1000M and PGV-1000MKP operating at Russian and foreign VVER NPPs was taken into account during SG designing. As a result, the design of some SG units was updated. Thus, it has been demonstrated that the requirements of item 1.2.5 of NP-001-97 (item 1.2.7 of NP-001-15) and item А1.19, C9.5 of INRA-MA-RE-000-00/02 are fulfilled.

Thus, the information provided in items 3.2.1, 3.9.1, 3.9.3 of PSAR Chapter 3 /D3/, and items 5.2.1, 5.2.3, 5.4.2 of PSAR Chapter 5 /1/ regarding the SG strength analysis meets in its structure in content the provisions of items 3.2, 3.9.1, 3.9.3, 5.2.1, 5.2.3, 5.4.2 of Regulatory Guide 1.70.

3.12. Strength of the MCP

The brief information about the MCP components’ strength analysis is provided in the following sections of the PSAR for Bushehr NPP Unit 2 - 3.2.1, 3.9.1, 3.9.3 /D3/ and 5.2.1, 5.2.3, 5.4.3 /1/.

The information about the MCP classification as per requirements of NP-001-15, NP-089-15 and NP-031-01 is provided in section 3.2.1 of the PSAR /D3/. The information about the MCP classification as per requirements of NP-001-15, PNAE G-7-008-89 and NP-031-01 is provided in section 5.4.3.1 of the PSAR /1/. Section 5.2.1, of the PSAR /1/ provides the information about compliance of the MCP with the RF regulatory documentation requirements. Thus, it is demonstrated that the requirements of item А1.3 of INRA-MA-RE-000-00/02 are fulfilled.

The design of the MCP is described in section 5.4.3.2 of the PSAR /1/.

The information about the MCP materials and mechanical properties necessary for strength analyses is provided in section 5.2.3 of the PSAR /1/. The regulatory documents listed in Table 5.2.3.2 of the PSAR /1/ for delivery of materials for MCP manufacturing are included into the Summary list of documents of standardization /D11/, which meets the requirements of item 85 of NP-089-15.

The general information on the approaches to carrying out strength analyses for RP equipment and pipelines is provided in Section 3.9 of the PSAR /D3/, which contains:

- a list of design modes of the RP operation (NOC, AOO, DBA);

- a list of software tools used in strength analyses as well as information on certification thereof;

- information about experimental research performed to substantiate the strength of equipment and pipelines;

- information about the method of determination of loads and stresses in the components of equipment and pipelines.

It is stated in section 3.9.2 of the PSAR /D3/ that the programs of precommissioning testing envisage monitoring of vibration and thermal displacement parameters of the VVER main pipelines also including the MCP to confirm the vibration strength of RP equipment and pipelines. It is stated in the PSAR /D3/ that precommissioning vibration measurements on the VVER prototype facility demonstrated that the level of vibration loading of inspected pipelines is not high.

According to the Section 3.9.3 of the PSAR /D3/ the strength of the MCP is justified in accordance with the requirements of PNAE G-7-002-86 and NP-031-01. The mentioned section also contains the information about the approaches to determination of permissible stresses when making calculations on the static strength, cyclic strength, seismic impact strength and brittle fracture resistance, which meet the requirements of PNAE G-7-002-86 and NP-031-01.

According to sections 3.9.3.1.9 of the PSAR /D3/ and 5.4.3.1 of the PSAR /1/, the strength of MCP components is justified for all design modes and all combinations of loads envisaged by the requirements of PNAE G-7-002-86 and NP-031-01 for pipelines of seismic category I. Thus, it is demonstrated that the requirements of item E22 of INRA-MA-RE-000-00/02 in terms of seismic loads in MCP strength analysis are fulfilled.

It is stated in item 5.4.3.1 of the PSAR that the LBB concept is applied in the design.

According to the PSAR /1/, the long-term experience in designing and operation of the MCP and the NPP was taken into account in development of the MCP. Thus, it has been demonstrated that the requirements of item 1.2.5 of NP-001-97 (item 1.2.7 of NP-001-15) and item А1.19, C9.5 of INRA-MA-RE-000-00/02 are fulfilled.

Section 5.4.3.1 of the PSAR /1/ provides the information about the measures on prevention of the MCP stress corrosion cracking in the course of designing, manufacturing and operation.

Thus, the information provided in items 3.2.1, 3.9.1, 3.9.3 of PSAR Chapter 3 /D3/, and items 5.2.1, 5.2.3, 5.4.3 of PSAR Chapter 5 /1/ regarding the MCP strength analysis meets in its structure in content the provisions of items 3.2, 3.9.1, 3.9.3, 5.2.1, 5.2.3, 5.4.3 of Regulatory Guide 1.70.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 84 |
| Section No.  | 5.4.3 /1/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the MCP |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| The PSAR /1/ should be supplemented with the information about the strength analyses made for the MCP envisaged by the requirements of item 1.2.2 of PNAE G-7-002-86 and comprising the calculation for selection of the main dimensions and verification strength calculations, as well as provide the main results of such calculations. |
| *2.4. References* |
| PNAE G-7-002-086. Regulations for strength analysis of equipment and pipelines of nuclear power installations. |

3.13. Strength of the PRZ

According to the requirements of sections 3.9.1, 3.9.2, 3.9.3 and 5.4 of Regulatory Guide 1.70, sections 3.9.1, 3.9.2, 3.9.3 of PSAR Chapter 3 /D3/ and 5.4.10 of PSAR Chapter 5 /1/ provide the information about the strength of the PRZ intended for creation of pressure in the primary circuit, maintaining the pressure within the assigned limits under steady state conditions and for limitation of pressure deviations under transient and accident conditions of the reactor plant.

The information about the PRZ materials and mechanical properties thereof necessary for strength analyses is provided in section 5.2.3 of the PSAR /1/. According to the data of item 5.4.10.2.6 of the PSAR /1/, the PRZ materials are resistant to impacts of the primary circuit medium, decontaminating solutions and the environment within the containment including sprinkling with the solution from the sprayers in case of accidents related to depressurization of the primary circuit (compliance with the requirements of item 2.1.4 of PNAE G-7-008-89).

The general information on the approaches to carrying out strength analyses for RP equipment and pipelines is provided in Section 3.9.1 of the PSAR /D3/, which contains:

- a list of RP design operating modes (NO, AOO, DBA);

- a list of software tools used in strength analyses;

- information about experimental research performed to substantiate the strength of equipment and pipelines;

- information about the method of determination of loads and stresses in equipment components and pipelines under emergency conditions.

Section 3.9.2.2 of the PSAR /D3/ provides the information about qualification seismic tests of the PRZ, whereby the PRS seismic impact calculation meets the requirements of section 5.11 of PNAE G-07-002-86 and the conducted tests confirm the calculated dynamic characteristics and capabilities of the PRZ to be operated during and after an earthquake.

Subsection 3.9.3.1 of the PSAR /D3/ provides the requirements for determination of permissible stresses as per requirements of sections 5.4, 5.11 of PNAE G-7-002-86 and section 5 of NP-031-01. This subsection of the PSAR /D3/ also contains the requirements for performing the static and cyclic strength analyses (compliance with the requirements of sections 5.4 and 5.6 of PNAE G-07-002-86), performing brittle fracture resistance calculations (compliance with the requirements of section 5.8 of PNAE G-07-00s-86) as well as for performing seismic impact and external dynamic effects’ calculations, which were evaluated based on the same criteria as the seismic impact calculation for the SSE level (compliance with the requirements of sections 5.11 of PNAE G-7-002-86 and 5 of NP-031-01).

In addition to the above information, subsection 3.9.3.1 of the PSAR /D3/ provides all design loads and all design modes. Thus, it is stated in the PSAR /D3/ that the design strength analysis was made with due account of the requirements of item 5.1.2 of PNAE G-7-002-86.

Therefore, according to the data of section 5.4.10 of the PSAR /1/, the PRZ is designed for operation in a set of the RP pressurizing system under NOC, AOO and DBA as per requirements of PNAE G-7-002-86; thus the PRZ design provides for normal functioning under the impact of the OBE, as well as for strength and leak tightness under the impact of the SSE (up to magnitude seven), aircraft crash and external shock wave with in the design service life equal to 60 years.

The information about the PRZ strength provided in sections 3.9.1, 3.9.2, 3.9.3 of PSAR Chapter 3 /D3/ and i5.4.10 of PSAR Chapter 5 /1/ meets the requirements of sections 3.9.1, 3.9.2, 3.9.3 and 5.4 of Regulatory Guide 1.70.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 85 |
| Section No.  | 5.4.10 /1/ |
| Page | 7 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the PRZ |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| Item 5.4.10.2.7 of the PSAR /1/ should be supplemented with the information about complying with the criteria of NP-031-01 when justifying the RPZ strength under seismic effects. |
| *2.4. References* |
| NP-031-01, “Design rules for aseismic nuclear power plants”.PNAE G-7-002-086. Regulations for strength analysis of equipment and pipelines of nuclear power installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 86 |
| Section No.  | 5.4.10 /1/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the PRZ |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| Section 5.4.10 of the PSAR /1/ should provide the results of the calculation on selection of the PRZ components’ main dimensions (it is stated in item 5.4.10.2.7 of the PSAR /1/ that such calculation is made according to PNAE G-7-002-86). |
| *2.4. References* |
| PNAE G-7-002-086. Regulations for strength analysis of equipment and pipelines of nuclear power installations. |

3.14. Strength of the PRZ system pipelines

The brief information about the reactor upper unit components’ strength analysis is provided in the following sections of the PSAR for Bushehr NPP Unit 2 - 3.2.1, 3.91-3.9.3 /D3/ and 5.2.1, 5.2.3, 5.4.11 /1/.

The information about the PRZ system pipelines’ classification as per requirements of NP-001-15, NP-089-15 and NP-031-01 is provided in section 3.2.1 of the PSAR /D3/. Section 5.2.1, of the PSAR /1/ provides the information about compliance of the MCP with the RF regulatory documentation requirements. Thus, it is demonstrated that the requirements of item А1.3 of INRA-MA-RE-000-00/02 are fulfilled.

The design of the PRZ system pipelines is described in section 5.4.11.2 of the PSAR /1/. According to the PSAR /1/, the PRZ system pipelines comprise:

- a connecting pipeline connecting the PRZ with the MCP hot leg;

- a PRZ injection pipeline;

- a PRZ discharge pipeline.

The information about the PRZ system pipeline materials and mechanical properties necessary for strength analyses is provided in section 5.2.3 of the PSAR /1/. The regulatory documents listed in Table 5.2.3.2 of the PSAR /1/ for delivery of materials for the relief tank manufacturing are included into the Summary list of documents of standardization /D11/, which meets the requirements of item 85 of NP-089-15.

The general information on the approaches to carrying out strength analyses for RP equipment and pipelines is provided in Section 3.9 of the PSAR /D3/, which contains:

- a list of design modes of the RP operation (NOC, AOO, DBA);

- a list of software tools used in strength analyses as well as information on certification thereof;

- information about experimental research performed to substantiate the strength of equipment and pipelines;

- information about the method of determination of loads and stresses in the components of equipment and pipelines.

It is stated in section 3.9.2 of the PSAR /D3/ that the programs of precommissioning testing envisage monitoring of vibration and thermal displacement parameters of the VVER main pipelines also including the PRZ system pipelines to confirm the vibration strength of RP equipment and pipelines. It is stated in the PSAR /D3/ that precommissioning vibration measurements on the VVER prototype facility demonstrated that the level of vibration loading of inspected pipelines is not high.

According to the Section 3.9.3 of the PSAR /D3/ the strength of the PRZ system pipelines is justified in accordance with the requirements of PNAE G-7-002-86 and NP-031-01. The mentioned section also contains the information about the approaches to determination of permissible stresses when making calculations on the static strength, cyclic strength, seismic impact strength and brittle fracture resistance, which meet the requirements of PNAE G-7-002-86 and NP-031-01.

According to section 3.9.3.1.10 of the PSAR /D3/, the strength of the PRZ system pipeline components is substantiated for all design modes and all combinations of loads envisaged by the requirements of PNAE G-7-002-86 and NP-031-01 for the pipelines of seismic categories I and II. Thus, it is demonstrated that the requirements of item E22 of INRA-MA-RE-000-00/02 in terms of seismic loads in PRZ system pipelines’ strength analysis are fulfilled.

It is also stated in section 3.9.3.1.10 of the PSAR /D3/ that the hydrodynamic loads during actuation of PRZ PORV were also considered in the strength analysis for the discharge pipelines.

It is stated in the PSAR /D3/ that a full section break of the pipelines with the maximum diameter adjoining the PRZ and MCP, which are not covered by the LBB concept and the rupture of which impacts the stressed-strained state of the pipeline reviewed, was assumed as a DBA for the pressurizer system pipelines. Only the strength of their supporting structures was justified for postulated ruptures of injection and discharge pipelines.

Thus, the information provided in items 3.2.1, 3.9.1, 3.9.3 of PSAR Chapter 3 /D3/, and items 5.2.1, 5.2.3, 5.4.11 of PSAR Chapter 5 /1/ regarding the PRA system pipelines strength analysis meets in its structure and content the provisions of items 3.2, 3.9.1, 3.9.3, 5.2.1, 5.2.3, 5.4.11 of Regulatory Guide 1.70.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 87 |
| Section No.  | 5.4.11 /1/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the PRZ system pipelines |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| The PSAR /1/ should provide the results of the strength analyses made for the PRZ system pipeline components envisaged by the requirements of item 1.2.2 of PNAE G-7-002-86 and comprising the calculation for selection of the main dimensions and verification strength calculations. |
| *2.4. References* |
| PNAE G-7-002-086. Regulations for strength analysis of equipment and pipelines of nuclear power installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 88 |
| Section No.  | 3.9.3.1.10 /D3/ |
| Page | 50 /D3/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the PRZ system pipelines |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| According to section 3.9.3.1.10 of the PSAR /D3/, the strength of the PRZ system pipeline components (including the supporting structures) is substantiated for all design modes and all combinations of loads envisaged by the requirements of PNAE G-7-002-86 and NP-031-01 for the pipelines of seismic categories I and II. |
| *2.2. Comment* |
| According to PSAR Chapter 3 /D3/, the strength of the PRZ system pipeline components (including the supporting structures) is justified in accordance with the requirements of PNAE G-7-002-86 and NP-031-01. However, the requirements of PNAE G-7-002-86, the application scope of which is determined by NP-089-15, don't apply to pipeline supporting structures (see subitem “h” of item 3 of NP-089-15). |
| *2.3. Recommendation* |
| The strength of supporting structures of the pressurizer system pipelines should be justified as per requirements of regulatory documents intended for the strength analysis of supporting structures. One of such documents is ОТТ 1.5.2.01.999.0157-2013 "Support Structures of Components of Nuclear Power Plants with Water-Cooled Water-Moderated Power Reactors. General Technical Requirements."  |
| *2.4. References* |
| NP-031-01. Design Standards for Aseismic Nuclear Power Plants.NP-089-15. Rules for Design and Safe Operation of Components and Pipelines of Nuclear Power Installations.PNAE G-7-002-086. Regulations for strength analysis of equipment and pipelines of nuclear power installations.OTT 1.5.2.01.999.0157-2013. Support Structures of Components of Nuclear Power Plants with Water-Cooled Water-Moderated Power Reactors. General Technical Requirements. |

3.15. Strength of the pressure relief tank

The brief information about the reactor upper unit components’ strength analysis is provided in the following sections of the PSAR for Bushehr NPP Unit 2 - 3.2.1, 3.9.1, 3.9.3 /D3/ and 5.2.3, 5.4.11 /1/.

The information about the pressure relief tank classification as per requirements of NP-001-15, NP-089-15 and NP-031-01 is provided in section 3.2.1 of the PSAR /D3/.

The design of the pressure relief tank is described in section 5.4.11.2 of the PSAR /1/.

The information about the relief tank materials and mechanical properties necessary for strength analyses is provided in section 5.2.3 of the PSAR /1/. The regulatory documents listed in Table 5.2.3.2 of the PSAR /1/ for delivery of materials for the relief tank manufacturing are included into the Summary list of documents of standardization /D11/, which meets the requirements of item 85 of NP-089-15.

The general information on the approaches to carrying out strength analyses for RP equipment and pipelines is provided in Section 3.9 of the PSAR /D3/, which contains:

- a list of design modes of the RP operation (NOC, AOO, DBA);

- a list of software tools used in strength analyses as well as information on certification thereof;

- information about experimental research performed to substantiate the strength of equipment and pipelines;

- information about the method of determination of loads and stresses in the components of equipment and pipelines.

According to Section 3.9.3 of the PSAR /D3/ the strength of the relief tank is justified in accordance with the requirements of PNAE G-7-002-86 and NP-031-01. The mentioned section also contains the information about the approaches to determination of permissible stresses when making calculations on the static strength, cyclic strength, seismic impact strength and brittle fracture resistance, which meet the requirements of PNAE G-7-002-86 and NP-031-01.

According to section 3.9.3.1.13 of the PSAR /D3/, the strength of the relief tank components is substantiated for all design modes and all combinations of loads envisaged by the requirements of PNAE G-7-002-86 and NP-031-01 for the pipelines of seismic categories I and II. Thus, it is demonstrated that the requirements of item E22 of INRA-MA-RE-000-00/02 in terms of seismic loads in the relief tank strength analysis are fulfilled.

Thus, the information provided in items 3.2.1, 3.9.1, 3.9.3 of PSAR Chapter 3 /D3/, and items 5.2.3, 5.4.11 of PSAR Chapter 5 /1/ regarding the relief tank strength analysis meets in its structure and content the provisions of items 3.2, 3.9.1, 3.9.3, 5.2.3, 5.4.3 of Regulatory Guide 1.70.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 89 |
| Section No.  | 5.4.11 /1/ |
| Page | - |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the pressure relief tank |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| The PSAR /1/ should be supplemented with the information about the strength analyses made for the relief tank envisaged by the requirements of item 1.2.2 of PNAE G-7-002-86 and comprising the calculation for selection of the main dimensions and verification strength calculations, as well as provide the main results of such calculations. |
| *2.4. References* |
| PNAE G-7-002-086. Regulations for strength analysis of equipment and pipelines of nuclear power installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 90 |
| Section No.  | 3.9.3.1.13 /D3/ |
| Page | 51-52 /D3/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the pressure relief tank |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
|  - |
| *2.3. Recommendation* |
| Section 3.9.3.1.13 of the PSAR /D3/ should provide the results of the strength analysis for the relief tank supporting structures.  |
| *2.4. References* |
| - |

3.16. Strength of the valves

According to the requirements of sections 3.9.1, 3.9.2, 3.9.3, and 5.4 of Regulatory Guide 1.70, sections 3.9.1, 3.9.2, 3.9.3 of PSAR Chapter 3 /D3/ and 5.4.12 of PSAR Chapter 5 /1/ provide the information about the strength analysis of valves installed within the V-528 RP primary circuit boundaries.

The information about the valve’s materials and mechanical properties necessary for strength analyses is provided in section 5.2.3 of the PSAR /1/. According to the data of subsection 5.4.12.3 of the PSAR /1/, the valves’ materials are compatible with the environmental and working media, and also are decontaminating solution-resistant, which meets the requirements of item 2.1.4 of PNAE G-7-008-89.

The general information on the approaches to carrying out strength analyses for RP equipment and pipelines is provided in Section 3.9.1 of the PSAR /D3/, which contains:

- a list of RP design operating modes (NO, AOO, DBA);

- a list of software tools used in strength analyses;

- information about experimental research performed to substantiate the strength of equipment and pipelines;

- information about the method of determination of loads and stresses in equipment components and pipelines under emergency conditions.

Subsection 3.9.2.2 of the PSAR /D3/ provides the information about the qualification seismic tests of valves, whereby the seismic tests of valves are performed as per requirements of section 2.5 of NP-068-05.

The general provisions and assessment criteria of valves’ strength are provided in subsection 3.9.3.1 of the PSAR /D3/ as per requirements of PNAE G-7-002-86.

Subsection 3.9.3.2 of the PSAR /D3/ provides the criteria for ensuring operability of valves as per requirements of section 3.5 of NP-068-05; it also provides the information about the strength analysis of valves including the following:

- all design modes and all design loads and combinations thereof were considered during designing of valves (compliance with the requirements of item 5.1.2 of PNAE G-7-002-86, item 5.4 of NP-031-01 and item 2.3.17 of NP-068-05);

- permissible stresses are assumed as per requirements of items 5.4.2, 5.11.2.11 ÷ 5.11.2.13 of PNAE G-7-002-86;

- the design analysis of the valves’ strength was made as per requirements of NP-068-05 and PNAE G-7-002-86.

- It is stated in subsection 5.4.12.1 of the PSAR /1/ that permissible stresses in strength calculations are assumed according to the requirements of items 5.4.2, 5.11.2.11 ÷ 5.11.2.13 of PNAE G-7-002-86.

according to the data provided in subsection 5.4.12.1 of the PSAR /1/, the valves (as per calculation results) are designed to withstand the effects of design transients (variations of pressure and temperature of the working medium) occurring under normal operating conditions, operational occurrences and design basis accidents specified in subsection 3.9.1.1 of the PSAR /D3/.

It is stated in subsection 5.4.12.1 of the PSAR /1/ that in calculating the valves as per requirements of item 2.3.17 of NP-068-05 the loads acting on the valve’s nozzles from the pipelines under NOC, operational occurrences, design basis accidents and loads from seismic impacts including SSE are taken into account. It is also stated in this section that the seismic stability of valves is proved by calculations or experimentally.

According to item 5.4.12.4.1 of the PSAR /1/, the strength of all valves within the reactor coolant system pressure boundary meets the requirements of PNAE G-7-002-86.

According to item 5.4.12.1.19 of the PSAR /1/, the service life of the body parts of the valves is 60 years.

Therefore, the information about the valves’ strength provided in sections 3.9.1, 3.9.2, 3.9.3 of PSAR Chapter 3 /D3/ and 5.4.12 of PSAR Chapter 5 /1/ meets the requirements of sections 3.9.1, 3.9.2, 3.9.3 and 5.4 of Regulatory Guide 1.70.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 91 |
| Section No.  | 5.4.12 /1/ |
| Page | 4 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the valves |
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| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| It is stated in section 5.4.12.1 of the PSAR /1/ that the seismic stability of valves is proved by calculations or experimentally. However, according to the requirements of item 2.5.2 of NP-068-05, seismic stability of the valves shall be confirmed on the basis of calculations, while seismic resistance is to be confirmed by calculations and/ or experiments. It addition to this, it is stated in item 2.5.1 of NP-068-05 that the valves assigned to seismic category I as per NP-031-01 shall be seismic resistant. The valves installed within the V-528 RP primary circuit boundaries is assigned to seismic category I as per NP-031-01. |
| *2.2. Comment* |
| It is incorrectly stated in section 5.4.12.1 of the PSAR /1/ that the seismic stability of valves is proved by calculations or experimentally.*(editorial comment)* |
| *2.3. Recommendation* |
| It is recommended that the term “seismic resistance” should be used in section 5.4.12.1 of the PSAR /1/ instead of “seismic stability”. |
| *2.4. References* |
| NP-031-01. Design Standards for Aseismic Nuclear Power Plants.NP-068-05. Piping Valves for Nuclear Power Plants. General Technical Requirements. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 92 |
| Section No.  | 5.4.12 /1/ |
| Page | 4 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the valves |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| Item 5.4.12.1.1 of the PSAR /1/ should provide the results of the valves’ cyclic strength analysis and calculation for selection of the valve components’ main dimensions (it is stated in item 5.4.12.1.1 of the PSAR /1/ that the design strength analyses of valves have been made in accordance with the requirements of PNAE G-7-002-86). |
| *2.4. References* |
| PNAE G-7-002-086. Regulations for strength analysis of equipment and pipelines of nuclear power installations. |

3.17. Strength of the safety devices

According to the requirements of sections 3.9.1, 3.9.2, 3.9.3, and 5.4 of Regulatory Guide 1.70, sections 3.9.1, 3.9.3.2 of PSAR Chapter 3 /D3/ and 5.4.13 of PSAR Chapter 5 /1/ provide the information about the strength analysis of safety devices designed for the primary circuit overpressure protection.

The information about the safety devices’ materials and mechanical properties necessary for strength analyses is provided in section 5.2.3 and item 5.4.13.4 of the PSAR /1/. It is stated in item 5.4.13.4.1 of the PSAR /1/ that the PORV materials are chosen such that they be compatible with the working and environmental media and decontaminating solutions-resistant (compliance with the requirements of item 2.1.4 of PNAE G-7-008-89).

The general information on the approaches to carrying out strength analyses for RP equipment and pipelines is provided in Section 3.9.1 of the PSAR /D3/, which contains:

- a list of design modes of the RP operation (NOC, AOO, DBA);

- a list of software tools used in strength analyses;

- information about experimental research performed to substantiate the strength of equipment and pipelines;

- information about the method of determination of loads and stresses in equipment components and pipelines under emergency conditions.

Subsection 3.9.2.2 of the PSAR /D3/ provides the information about the qualification seismic tests of valves, whereby the seismic tests of valves are performed as per requirements of section 2.5 of NP-068-05.

Subsection 3.9.3.2 of the PSAR /D3/ provides the criteria for ensuring operability of safety devices as per requirements of section 3.5 of NP-068-05; it also provides the information about the strength analysis of the PORV including the following:

- all design modes and all design loads and combinations thereof were considered during designing of PORV (compliance with the requirements of item 5.1.2 of PNAE G-7-002-86, item 5.4 of NP-031-01 and item 2.3.17 of NP-068-05);

- permissible stresses are assumed as per requirements of items 5.4.2, 5.11.2.11 ÷ 5.11.2.13 of PNAE G-7-002-86;

- the design analysis of the PORV strength was made as per requirements of NP-068-05 and PNAE G-7-002-86.

It is stated in subsection 5.4.13.1 of the PSAR /1/ that the PORV design was developed on condition of its reliable functioning and keeping strength under impacts of operational, seismic and accident loads as well as their combinations arising during PORV operation under the conditions listed in item 3.9.1.1 of the PSAR /D3/ with consideration of the number of their cycles and service life provided for by the reactor plant design (compliance with the requirements of item 5.1.2, sections 5.4 and 5.11 of PNAE-G-7-002-86).

As stated in item 5.4.13.3 of the PSAR /1/, the assigned service life of the body items shall not be less than 30 years.

Therefore, the information about the valves’ strength provided in sections 3.9.1, 3.9.2.2, 3.9.3.2 of PSAR Chapter 3 /D3/ and 5.4.13 of PSAR Chapter 5 /1/ meets the requirements of sections 3.9.1, 3.9.2, 3.9.3 and 5.4 of Regulatory Guide 1.70.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 93 |
| Section No.  | 3.9.2.2 /D3/ |
| Page | 28 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the safety devices |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| Section 3.9.2.2 of the PSAR /D3/ should provide the information about seismic testing of the PORV. |
| *2.4. References* |
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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 94 |
| Section No.  | 5.4.13 /1/ |
| Page | 2 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the safety devices |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| Subsection 5.4.13.1 of the PSAR /1/ should provide the results of the calculation on selection of the PORV components’ main dimensions made as per requirements of PNAE G-7-002-86.  |
| *2.4. References* |
| PNAE G-7-002-086. Regulations for strength analysis of equipment and pipelines of nuclear power installations. |

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 95 |
| Section No.  | 5.4.13 /1/ |
| Page | 2 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the safety devices |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| - |
| *2.2. Comment* |
| - |
| *2.3. Recommendation* |
| Item 5.4.13.1.1 of the PSAR /1/ should provide the information about compliance with the requirements of NP-031-01 and NP-068-05 when making a strength analysis for the PORV.  |
| *2.4. References* |
| NP-031-01. Design Standards for Aseismic Nuclear Power Plants.NP-068-05. Piping Valves for Nuclear Power Plants. General Technical Requirements.PNAE G-7-002-086. Regulations for strength analysis of equipment and pipelines of nuclear power installations. |

3.18. Strength of the primary circuit supporting components

According to the requirement of sections 3.9.3 and 5.4.14 of Regulatory Guide 1.70, the information on the strength analysis of the primary circuit support structures is provided in section 5.4.14 of the PSAR /1/ as well as in section 3.9.3.4 of the PSAR /D3/.

It is stated in section 5.4.14 of the PSAR /1/ that supports of the steam generator, reactor coolant pump set and pressurizer are referred to the supports of primary circuit main components. According to the PSAR /1/, the structure of supports of the equipment shall ensure: functioning as per NP-001-15 and taking up load combination as per NP-031-01.

The instantaneous transverse guillotine break of steamline within the containment or feedwater pipeline or one of the MCP-connected pipelines, to which the LBB concept is not applied, is assumed as DBA in the strength analyses of the steam generator support structures.

Strength calculations of RCPS supporting structures consider DBA as guillotine break of one of the MCP-connected pipelines to which the LBB concept with the maximum effect on RCPS is not applied.

MCP strength calculations consider DBA as guillotine break of one of the MCP-connected pipelines to which the LBB concept with the maximum effect on MCP is not applied.

Strength calculations of PRZ supporting structures consider DBA as guillotine break of PRZ-connected pipeline of maximum diameter to which the LBB concept is not applied.

Welded joints of supports shall be made and inspected according to regulations of PNAE G-7-009-89 and PNAE G-7-010-89.

It is stated in section 3.9.3.4 of the PSAR /D3/ that the following load types were considered in the verification strength analyses of the components’ supports:

- the weight of the installed equipment and pipelines including the weight of the coolant and heat insulation;

- forces on equipment from the adjoining pipelines;

- thermal loads;

- loads occurring upon actuation of safety devices;

- equipment;

- seismic loads;

- emergency loads occurring in case of pipelines rupture.

Thus, section 3.9.3.4 of the PSAR /D3/ considered the requirements of section 3.9.3.1 of Regulatory Guide 1.70.

According to the PSAR /D3/, the verification calculation of the components’ supports usually comprises the static and cyclic strength analyses. Stability calculations were additionally performed for supporting shells of the pressurizer.

The verification calculation for supports was made to determine the stresses therein at all values of loads and temperatures in the reactor plant operating modes specified in the design and comparison of the obtained stresses with the permissible values. The following design modes and load combinations were addressed:

- normal operating conditions;

- anticipated operational occurrences;

- for seismic resistance category I component supports: imposition of the safe shutdown earthquake on normal operation conditions - NOC+SSE and on anticipated operational occurrences - AOO+SSE;

- for seismic resistance category I and II component support: imposition of the operating basis earthquake on normal operation conditions - NOC+OBE and on anticipated operational occurrences - AOO+OBE.

Besides, the design situation with imposition of the design basis accident and operation basis earthquake on the normal operating conditions - NOC+DBA+OBE - was considered for the supports of equipment and pipelines, which do not fall under the LBB concept.

The mentioned load combinations meet the requirements of section 5 of NP-031-01; the requirements of section 3.9.3.1 of Regulatory Guide 1.70 have been considered as well.

The values of permissible stresses for various design load combinations were determined as per requirements of the Russian regulatory document, OTT 1.5.2.01.999.0157-2013.

According to the PSAR /D3/, the breaking strength and the yield strength of the material were used when determining the permissible stresses’ values; here the changes in design characteristics of materials of supports depending on the temperature were taken into account. The impact of the concrete considered here as Winkler foundation (a number of elastic springs not connected between each other and fixed on the absolutely rigid foundation) was taken into account as necessary.

The strength analysis for equipment and pipeline supports is based on the analysis of the stressed state of support structures’ components. The values of permissible stresses for different calculation combinations were determined similarly to the calculations for equipment and pipelines.

It is stated in the PSAR /D3/ that hydraulic shock absorbers and viscoelastic dampers were applied to fasten the reactor plant equipment and component against seismic and accident dynamic load impacts.

The PSAR /D3/ indicates that the hydraulic shock absorbers structurally consist of the cylinder with the lug for fastening to the fixed support, a piston with the stem for fastening to the equipment or pipelines, and a valve holder with two valves. The cylinder is filled with the special liquid, which flows freely through the valve holder from one-cylinder piston side separated by the piston to the other during slow displacements of equipment. Upon dynamic impact on the equipment, the valves close in turn depending on the fluid flow direction, fluid transfer stops and the shock absorber turns into the support with high rigidity. The value of the fluid transfer rate, at which the valves are locked, is selected in such a way so that to ensure free thermal expansion of pipelines and secure fastening in case of fast dynamic loads. The vibration loads are not perceived by hydraulic shock absorbers.

According to the PSAR /D3/, the characteristics of hydraulic shock absorbers to be considered in seismic analyses, shall be determined based on the special dynamic testing. The strain of the hydraulic shock absorber caused by the nominal load corresponding to the given type of the hydraulic shock absorber is the main parameter determined during testing. During development of hydraulic shock absorbers, tests were conducted for different positions of the piston relative to the cylinder and temperature of the operating liquid; here the liquid burnability, metal deformation, as well as available clearances in the structure were considered. The stiffness characteristics obtained by this method for every type of the hydraulic shock absorber were considered when making relevant seismic effects’ analyses.

It is stated in the PSAR /1/ that the service life of hydraulic shock absorbers is equal to the NPP service life; the service life of component parts (operating liquid and rubber sealings) is 12 years. Upon expiration of this period, the shock absorbers shall be removed from the equipment, dismantled, the fluid and sealings shall be replaced; after that the shock absorbers shall be reassembled, tested on test machines to check their certificate characteristics, as well as installed into their position. The operation of hydraulic shock absorbers shall be carried out in compliance with the operating instructions. The in-service inspection of hydraulic shock absorbers’ functioning is reduced to periodical checking of the liquid’s availability in the cylinder, inspection of the piston position and visual examination. The design of the shock absorbers ensures resistance to the environment (conditions under the containment), their installation on the equipment, dismantling, disassembly and repair, and transportation by any type of the transport vehicle. It is stated in the PSAR /D3/ that the reliability of the used design of hydraulic shock absorbers has been confirmed by the operating experience at the Balakovo NPP and a number of other NPPs.

According to the PSAR /1/, the viscoelastic dampers are designed to protect the NPP components in the wide range of dynamic load frequencies; they ensure damping of NPP components’ oscillations upon dynamic displacements in any directions. The most efficient damping of oscillations and the guaranteed experimentally proved dynamic characteristics of dampers (dependency of the dampers’ dynamic stiffness on the frequency of dynamic effects) are ensured upon dynamic loading within the frequency range from 0 to 40 Hz. The efficiency of damping is reduced in a higher frequency range upon an increase of the dynamic load frequency with a simultaneous increase of dampers’ stiffness. The nominal loads and dynamic characteristics for the dampers were obtained during testing of prototype samples based on the methodology of the manufacturer (GERB).

Thus, section 3.9.3 of the PSAR /D3/ meets the requirements of section 3.9.3.1 of Regulatory Guide 1.70. Regarding providing of information on testing of supporting structures’ components. The design of the damper consists of the cylindrical body filled with viscous actuation fluid, piston and internal components plunged into the actuation fluid. The dampers do not restrict thermal displacements of equipment and pipelines. The maximum force of the damper's resistance to thermal displacements of NPP components at positive ambient temperatures does not exceed 200 N. The assigned service life of viscoelastic dampers is 60 years.

According to the scope and structure, the information provided in the PSAR /1/, /D3/ does not meet the provisions of Regulatory Guide 1.70, which is reflected in the comment provided in section 2.2.20.2 of the present Review Report.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 96 |
| Section No.  | 3.9.3.4 /D3/ |
| Page | 74-76 /D3/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Strength of the primary circuit supporting components |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| Section 3.9.3.4 of the PSAR /D3/ addresses not all data, which are required by section 3.9.3.1 of Regulatory Guide 1.70 for the reactor support structures. |
| *2.2. Comment* |
| Section 3.9.3.4 of the PSAR /D3/:* does not contain the information about the applied methods for calculation and testing of the supports in case of failures thereof;
* does not specify the maximum stresses, strains and cumulative coefficient for all operating modes for all components;
* does not determine those values of stresses, which differ from the permissible limits by less than 10% (the requirements of section 3.9.3.1 of Regulatory Guide 1.70 are not fulfilled).
 |
| *2.3. Recommendation* |
| Section 3.9.3.4 of the PSAR /D3/ should be supplemented with the information about the applied methods for calculation and testing of the reactor supports in case of failures thereof, maximum stresses, strains and the cumulative coefficient for all operating conditions for all components, as well as about the values of stresses, which differ from the permissible limits by less than 10%. |
| *2.4. References* |
| US NRC Regulatory Guide 1.70. Standard Format and Content of Safety Analysis Report of Nuclear Power Plants. 1978. |

3.19. Reactor plant instrumentation equipment

The information about the RP instrumentation is provided in section 5.1.2 of the PSAR /1/.

The list of points for monitoring the coolant and equipment parameters in the main coolant pipeline is given in Table 5.1.2.1 of the PSAR [1].

Piping and instrumentation diagrams of the equipment involved in the reactor coolant system, as well as the list of monitoring points are presented:

- for reactor - in Figure 5.1.2.2, 5.1.2.3 and Table 5.1.2.2 of the PSAR /1/;

- for steam generator - in Figure 5.1.2.4 and Table 5.1.2.3, as well as in subsection 5.4.2.3 of the PSAR /1/;

- for pressurizer and relief tank – in Figure 5.1.2.5, Table 5.1.2.4, as well as in subsections 5.4.10.3, 5.4.11.3 and Figures 5.4.10.3, 5.4.11.2 of the PSAR /1/;

- for RCP set – in Figure 5.1.2.6, in Table 5.1.2.5, as well as in subsection 5.4.1.4 of the PSAR /1/.

It is stated in item 5.1.2.4 of the PSAR /1/ that the number of monitoring points over the reactor coolant system presented in Tables 5.1.2.1 ÷ 5.1.2.5 of the PSAR /1/ is sufficient for fulfillments of all the assigned functions of normal operation systems and safety systems and takes into account the principles of NPP I&C designing.

According to item 5.1.2.10 of the PSAR /1/, the instrumentation lines for sensors of coolant pressure and level measurement go outside the containment and are not isolated by the valves at its boundary. It should be noted that the analysis for the break of the I&C DN10 lines passing through the containment is provided in section 15.7.4 of the PSAR /D8/. According to item 15.7.5.1.3 of the PSAR /D8/, the I&C instrumentation lines (primary coolant pressure measurement lines) go beyond the containment. The break of these lines outside the containment results in the primary circuit coolant release into the environment. The analysis of the mentioned mode is evaluated in section 2.2.33 of Review Report DNP-4525/3-2019.

It is stated in item 5.1.2.11 of the PSAR /1/ that the detailed information about monitoring points of MCDS is provided in sections 5.2.5 of PSAR Chapter 5 /1/, 7.2.3 and 7.4 of PSAR Chapter 7 /D6/.

According to items 5.2.5.1.4.2, 5.2.5.1.4.3 of the PSAR /1/, the ALMS set includes piezoelectric acoustic transducers and the HLMS includes the remote probes for humidity monitoring. The arrangement of ALMS and HLMS I&C on the RP equipment is presented in Figures 5.2.5.1, 5.2.5.2 of the PSAR /1/. The analysis of the primary coolant leak detection system comprising the ALMS and HLMS is evaluated in Section 5 of the present Review Report.

The scope and structure of the information provided in PSAR section 5.1.2 /1/ meet the provisions of section 5.1.2 of Regulatory Guide 1.70.

It should be noted that the detailed analysis of the I&C operability is provided in PSAR Chapter 7 /D8/; it meets the provisions of section 7.5 “Safety-Related Display Instrumentation” of Regulatory Guide 1.70.

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| **ISSUE SHEET** |
| **1. IDENTIFICATION OF COMMENT** | Comment No. | 97 |
| Section number  | 5.1.2 /1/ |
| Page | 31 /1/ |
| Facility | BUSHEHR-2 NPP UNIT 2 |
| Issue title | Reactor plant instrumentation equipment |
|  |
| **2. EXPLANATION ON COMMENT** |
| *2.1. Description of the comment* |
| It is stated in item 5.1.2.11 of the PSAR /1/ that the detailed information about monitoring points of MCDS is provided in subsection 5.2.5 of PSAR Chapter 5 /1/, and sections 7.2.3 and 7.4 of the PSAR /D6/. |
| *2.2. Comment* |
| It is incorrectly stated in item 5.1.2.11 of the PSAR /1/ that the detailed information about the MCDS monitoring points is provided in subsection 5.2.5 of the PSAR /1/ and sections 7.2.3 and 7.4 of the PSAR /D6/, since section 7.2.3 is missing from PSAR Chapter 7 /D6/ and PSAR section 7.4 /D6/ provides the information pertaining to the systems insuring safe shutdown (NO I&C, the unit electrical equipment instrumentation and control system - PU I&C EE).*(editorial comment)* |
| *2.3. Recommendation* |
| - |
| *2.4. References* |
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4. CONCLUSIONS AND PROPOSALS

4.1. The analysis of the following components as part of the reactor of BNPP-2, Unit 2, provided in Chapter 5 of the PSAR /1/:

* structural materials of the reactor vessel and top head, surveillance specimens of the RPV material;
* strength of the reactor vessel, reactor upper unit, RCP set, SG, MCP, PRZ, PRZ system pipelines, relief tank, valves, safety devices and supporting components of the primary circuit including the reactor;
* operability of the primary circuit overpressure protection system;
* in-service inspection and testing of the primary circuit components;
* primary coolant leak detection system, RP I&C;

complies with requirements of regulatory documents specified in item 4.3 of the present Review Report, except for non-compliances with:

* provisions of items 3.9.3.1, 5.3.3.6, item 5.4.1 of Regulatory Guide 1.70 and the requirements of item 1.2.13 of NP-001-15, item 2.5.3 of NP-082-07, item 3 of NP-089-15 as regards insufficiency of the strength analysis for the reactor vessel, RCP set, PRZ system pipelines, primary circuit components’ supporting structures including the reactor provided in the PSAR /1/ (see the comments given in sections 2.2.6, 2.2.8, 2.2.10, 2.2.14, 2.2.18 of the present Review Report);
* requirements of item 1.2.3 of NP-001-97 in terms of the non-conservativeness of the approach used in the PSAR /1/ for the reactor vessel strength analysis (see the comments of section 2.2.6 of this Review Report);
* provisions of item 4.2.8 of Regulatory Guide 1.70 regarding the fact that the PSAR /1/ does not provide any information about the neutron and physical characteristics necessary for the reactor vessel strength analysis (see the comments of section 2.2.6 of this Review Report);
* provisions of item 5.3.3.7 of Regulatory Guide 1.70 and the requirements of items 1.1.2 of NP-097, 19 of NP-089-15 regarding the in-service inspection of the reactor vessel (see the comments of section 2.2.6 of this Review Report);
* requirements of items 6.2.2, 6.2.11, 6.2.12, 6.2.15, 6.2.27 of PNAE G-7-008-89 and items 4.1.10, 4.1.12 of NP-001-97 regarding insufficiency of the primary circuit overpressure protection system operability justification (see the comments of section 2.2.1 of this Review Report);
* requirements of item 2.6.7 of NP-068-05 regarding the assigned service life of the primary circuit overpressure protection system components, namely PRZ PORV (see the comments in section 2.2.1 of this Review Report);
* provisions of items 6.3.2.2, 6.3.3 of Regulatory Guide 1.70 regarding the fact that the PSAR /1/ does not contain the information about the primary circuit overpressure protection system (see the comments of section 2.2.1 of this Review Report);
* requirements of items 1.1.2, 4.4.4.6 of NP-001-97 regarding the fact that the PSAR /1/ does not provide the information about compliance of the primary circuit coolant leak detection facilities with the requirements of NP-026-04 (see the comments of section 2.2.3 of this Review Report);
* requirements of item 2.5.13 of NP-082-07 regarding the fact that the PSAR /1/ does not substantiate the accuracy of the primary circuit coolant leak detection (see the comments of section 2.2.3 of this Review Report).

4.2. The non-conformities found in the results of the review do not prevent the commencement of the unit construction as they are mainly the evidence of the non-complete information provided in the PSAR /1/.

4.3. The Developer should take into account the comments and consider the recommendations of this Review Report.

5. REFERENCES

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