Article 9, RESPONSIBILITY OF THE LICENCE HOLDER

*Each Contracting Party shall ensure that prime responsibility for the safety of a nuclear installation rests with the holder of the relevant license and shall take the appropriate steps to ensure that each such license holder meets its responsibility.*

9-6) Description on safe management on the all operational status (normal and accident conditions and mitigation of its repercussions);

بر اساس مدرک FSAR، مدارک طراحی و مدارک کارخانه ای تجهیزات اصلی نیروگاه، اصلی ترین مدرک بهره برداری نیروگاه تحت عنوان Technical specification of safe operation of Bushehr Nuclear power plant unit 1 (TSSO) تهیه شده است. این مدرک به تایید طراح اصلی نیروگاه، طراح مجموعه تاسیسات راکتور، مدیریت علمی قلب راکتور، سازمان بهره برداری (دارنده پروانه NPPD) و سازمان نظارتی کشور جمهوری اسلامی ایران INRA-NNSD رسیده است. در این مدرک حدود و شرایط بهره برداری ایمن نیروگاه مشخص شده است. طبق این مدرک در شرایط بهره برداری نیروگاه دارای مدهای (رژیم های) بهره برداری به شرح ذیل می باشد:

Based on the FSAR document, manufacturing and design documents of main equipment of Bushehr NPP, the main documents of plant have been prepared under the title of Technical specification of safe operation of Bushehr Nuclear power plant unit 1 (TSSO). This document has been approved by main designer of Bushehr NPP, designer of reactor plant, reactor core scientific management, Operating Organization (license holder: NPPD) and Iranian Nuclear Regulatory Authority ( INRA-NNSD). In this document, the limits and conditions of safe operation of the plant has been designated. According to this document, the plant has operating modes during operation as follows:

حالت اول: کار در قدرت؛

Mode 1: operating at power

حالت دوم: کار در حداقل قدرت قابل کنترل MCL؛

Mode 2: operating at MCL ( minimal controllable level )

حالت سوم: حالت گرم؛

Mode 3: hot mode

حالت چهارم: حالت سرد؛

Mode 4: cold mode

حالت پنجم: توقف برای تعمیرات؛

Mode 5: outage for repairs

حالت ششم: تعویض سوخت؛

Mode 6: refueling

در مدرک TSSO کلیه شرایط و حدود بهره برداری operational limit and conditionو شرایط و حدود بهره برداری ایمنOperational safety limit and condition برای 6 حالت فوق مشخص شده است و شرایط و حدود گذر از هر یک از رژیم های فوق الذکر به سایر رژیم ها مشخص شده است. برای اساس مدرک TSSO و مدارک طراحی و کاخانه ای تجهیزات نیروگاه ، سایر مدارک بهره برداری تهیه و اجرایی شده است.

In the document TSSO, all the operational limit and condition and Operational safety limit and condition for the 6 above-mentioned modes have been specified. Also limits and conditions for passing each one of these modes have been specified. Based on the document TSSO and designing and manufacturing documents of the plant equipment, other operational documents have also been prepared and put into effect.

در صورت بروز و نمایانگر شده هر سیگنال در اتاق کنترل اصلی نیروگاه یا بروز حادثه، دستورالعمل های ذیل تهیه و اجرایی شده است:

In case of occurrence and emergence of any signal in main control room) MCR) of the plant or in case of accident, the following instructions have been prepared and put into effect:

1. دستورالعمل واکنش به سیگنال های مدیریت راکتور؛

1- Instruction of reaction to reactor management signals

1. دستورالعمل واکنش به سیگنال های مدیریت توربین؛

2- Instruction of turbine management signals

1. دستورالعمل مدیریت حادثه در مجموعه تاسیسات راکتور (DBAحوادث و انحرافات در نظر گرفته شده در طرح نیروگاه) این مدرک جزی مدارک لایسنسی نیروگاه می باشد و به تایید طراح اصلی نیروگاه، طراح مجموعه تاسیسات راکتور و مدیریت علمی نیروگاه، سازمان بهره بردار و سازمان نظارتی کشور جمهوری اسلامی ایران رسیده است.؛

3- Instruction of DBA( design-based accidents) management in reactor plant. This document is one of license documents of plant and has been approved by main designer of Bushehr NPP, designer of reactor plant, plant scientific management, Operating Organization and Iranian Nuclear Regulatory Authority.

1. دستورالعمل مدیریت حوادث فراطراحی (BDBA) این مدرک جزی مدارک لایسنسی نیروگاه می باشد و به تایید طراح اصلی نیروگاه، طراح مجموعه تاسیسات راکتور و مدیریت علمی نیروگاه، سازمان بهره بردار و سازمان نظارتی کشور جمهوری اسلامی ایران رسیده است؛

4- Instruction of BDBA (beyond design-based accidents) management in reactor plant. This document is one of license documents of plant and has been approved by main designer of Bushehr NPP, designer of reactor plant, plant scientific management, Operating Organization and Iranian Nuclear Regulatory Authority.

1. دستورالعمل مدیریت حادثه در توربین؛

5- Instruction of accident management in turbine plant

1. دستورالعمل مدیریت حادثه در تاسیسات برقی؛

6- Instruction of accident management in power facilities

1. دستورالعمل مدیریت حادثه در تاسیسات تهویه؛

7- Instruction of accident management in ventilation facilities

1. دستورالعمل مدیریت حادثه در تاسیسات کنترل و ابزار دقیق؛

8- Instruction of accident management in I&C facilities

1. دستورالعمل مدیریت حوادث فرا طراحی در نیروگاه اتمی بوشهر؛

9- Instruction of BDBA management in Bushehr NPP

دستورالعمل حفاظت از کارکنان و مردم در شرایط حادثه. این مدرک جزی مدارک لایسنسی نیروگاه می باشد و به تایید سازمان بهره بردار و سازمان نظارتی کشور جمهوری اسلامی ایران رسیده است؛

Instruction of protecting the staff and the public in accidents. This document is one of license documents of the plant and has been approved by Operating Organization and the Iranian Nuclear Regulatory Authority.

Article 10, Priority to Safety

*Each Contracting Party shall take the appropriate steps to ensure that all organizations engaged in activities directly related to nuclear installations shall establish policies that give due priority to nuclear safety.*

10-2) Description on Plans, programs and policies regarding safety culture, management system, safety review and assessments, self- assessments, modifications and improvements;

1: گام نخست براي بهبود فرهنگ ايمني، مشخص نمودن وضعيت فرهنگ ايمني با استفاده از روش‌هاي گوناگون خودارزيابي مي‌باشد. به همين منظور و در ابتدا، كارگروهي در سطح نيروگاه اتمي بوشهر سازماندهي و تشكيل گرديد كه وظيفه اين كارگروه، سازماندهي انجام خودارزيابي در سطح نيروگاه براساس دستورالعملي مصوب و با استفاده از روش‌هاي استاندارد در تطابق با الزامات و معيارهاي بين‌المللي مي‌باشد. هدف كلي از انجام خودارزيابي فرهنگ ايمني، پشتيباني و بهبود سطح عملكردها و شاخص‌هاي ايمني از طريق:

1. First step for improving safety culture is to determine the status of safety culture by using diverse methods of self-assessment. In order to do this, a workgroup was primarily organized and set up in Bushehr NPP, which was assigned to organize the implementation of self-assessment in the plant according to the approved instruction and by using the standard methods in compliance with the international criteria and requirements. The general objective of safety culture self-assessment is to support and improve the level od performances and safety indicators via:

* ايجاد يك تصوير مشترك از فرهنگ ايمني؛
* Creating a common picture of safety culture
* بررسي و بهبود ارتباط متقابل افراد، تكنولوژي و سازمان؛
* Studying and improving the interaction of people, technology and organization
* برجسته‌ نمودن دلايل فرهنگي مشكلات و كمبودهاي ايمني؛
* Highlighting the cultural causes of safety deficiencies and problems
* شناسايي نقاط قوت و ضعف فرهنگ ايمني در مقايسه با شاخص‌هاي فرهنگ ايمني تعريف شده؛
* Identifying the strengths and weaknesses of safety culture in comparison with the defined safety culture indicators
* تعيين فرصت‌هايي براي بهبود فرهنگ ايمني مي‌باشد.
* Determining opportunities for safety culture improvement

از سال 2017 تاكنون، اين خودارزيابي دو مرتبه در سطح نيروگاه اتمي بوشهر انجام گرديده كه براساس نتايج و تحليل‌هاي انجام شده، نقاط قابل بهبود شناسايي و برنامه بهبود متناسب با آن تدوين و اجرايي گرديده است.

This self-assessment has been performed two times in Bushehr NPP Since 2017. Based on the results and analyses, the areas for improvement have been identified and improvement plan appropriate for them have been formulated and executed.

در راستاي استقرار سيستم مديريت يكپارچه در شركت بهره برداري واحد يكم نيروگاه اتمي بوشهر، خط مشي سيستم مديريت يكپارچه شركت بهره برداري به شرح زير تدوين و اجرايي گرديد.

In order to establish the integrated management system in Bushehr NPP Operating Company, the policy of the integrated management system in Bushehr NPP Operating Company was formulated and executed as follows:

شرکت بهره­برداری نیروگاه اتمی بوشهر وابسته به شرکت توليد و توسعه انرژي اتمي ايران، در زمینه بهره­برداری از واحدهاي نيروگاه اتمي برق بوشهر و تولید برق از اولین نیروگاه اتمی در ايران فعال بوده و با رويکرد تحقق اصول ايمني با اولويت ايمني هسته‌اي، توليد برق ايمن، مطمئن، اقتصادي و با کيفيت، تامين سلامت و بهداشت کارکنان، و محافظت از مردم و محيط زيست، در راستاي تبديل شدن به شرکت تراز اول در صنعت انرژي هسته­اي کشور با بهره­گيري از دانش مهندسي روز دنیا و سرمایه‌های انسانی متخصص و مجرب، سیستم مدیریت یکپارچه را براساس آخرین ویرایش استانداردهای:

The BNPP Operating Company, subsidiary of the Nuclear Power Production and Development (NPPD) Company, is active in the field of operating the nuclear power plants in the country and generates electricity from the first nuclear power plant in Iran. Adopting the approach of 1) fulfilling the safety principles with the nuclear safety as the priority, 2) generating safe, reliable, economical and high quality electricity, 3) ensuring the health and well-being of the personnel, and 4) protecting the people and the environment, and aiming at becoming a first-class company in the national nuclear industry by making use of the state-of-the-art engineering knowledge and experienced and skilled human capitals, this company has designed, established, implemented and maintained the Integrated Management System based on the latest revisions of the standards “ISO45001, ISO14001, ISO9001, GSR-Part2”, and has made the following the framework for its goals:

"ISO45001, ISO14001, ISO9001, GSR Part2"

طرح ریزی، ایجاد، پیاده­سازی و نگهداری نموده و به اين منظور محورهای زیر را چارچوبی براي اهداف خود قرار داده است.

توسعه فرهنگ ایمنی

توسعه دانش سازمانی

ارتقاي سطح عملکرد و بهره‌وری سازمانی

توسعه سطح کیفیت محصولات و خدمات

ارتقاي سطح اعتماد و رضایتمندی ذینفعان

تعهد به صیانت از محیط زیست، پیشگیری از آلودگی یا کاهش آن

توسعه رویکرد سیستمی، رفع ریشه‌ای مشکلات و تعهد به بهبود مستمر

توسعه رويکرد محافظه کارانه در تصميم سازي ها در زمان بهره برداري از نيروگاه

توسعه انطباق با الزامات و استانداردهای ملی، بین‌المللی و حرفه‌ای مرتبط

توسعه و ارتقاي سطح آگاهی، تخصص، مهارت و مشارکت سرمایه‌های انسانی

توسعه بکارگيري تجارب بهره برداري بدست آمده در نيروگاه اتمي بوشهر و ساير نيروگاه هاي خارج از کشور

توسعه و ارتقاي سطح امنیت و کاهش تهدیدات و آثار ناشی از آن بر زیرساخت‌ها، اطلاعات و سرمایه‌های انسانی

توسعه و افزایش سطح آمادگی و قابلیت های کلیدی جهت پاسخ به شرایط اضطراری هسته ای و بحران های محتمل

تعهد به حفظ و ارتقاي بهداشت و سلامت کارکنان، پیشگیری از حوادث و بیماری‌های شغلی یا کاهش آنها

|  |
| --- |
| * Development of safety culture * Development of organizational knowledge * Promotion of level of organizational performance and efficiency * Development of the quality level of products and services * Promotion of level of trust and satisfaction of the stakeholders |
| * Commitment to protecting the environment, preventing the contamination or reducing it * Development of the systematic approach, elimination of root causes of problems, and commitment to continuous improvement * Development of conservative approach in decision-makings during the BNPP operation * Development of compliance with the relevant national, international and professional requirements and standards * Development and improvement of level of knowledge, expertise, skill and involvement of the human capitals |
| * Development of applying the operating experiences obtained in the BNPP and other NPPs overseas * Development and promotion of security level and reduction of threats and their impacts on infrastructures, information and human capitals * Development and increase of level of preparedness and key capabilities for responding to the nuclear emergencies and possible crises * Commitment to maintaining and promoting the health and well-being of the personnel, preventing accidents and occupational diseases or reducing them |

2: خودارزيابي ايمني بهره برداري نيروگاه

1. Bushehr NPP safety self-assessment

رویکرد خود ارزیابی کلی در شركت بهره برداري شامل پیماش های تفصیلی شش ماهه با مشارکت گسترده ی کارکنان شركت، خود ارزیابی های واحدي با تیم های چند رشته ای، گزارش های اماری سالانه و ارزیابی ها می باشد.

The general approach of self-assessment in the Operating Company include six-month detailed surveys by the extensive participation of the Company staff, intra-departmental self-assessment with multi-discipline teams statistical annual report and assessments.

روند انجام خودارزيابي ايمني بهره برداري در نيروگاه به شرح زير مي باشد:

The process of performing operational safety self-assessment in the plant is as follows:

1. تعيين مديراني كه مسئول سازماندهي اجراي خودارزيابي هستند؛
2. تعيين واحدهايي كه مسئول اجراي خودارزيابي، كنترل اجراي خودارزيابي، تدوين اقدامات اصلاحي و آناليز نتايج حاصل از خودارزيابي در كليه سطوح شركت هستند؛
3. فهرست مديراني كه در خودارزيابي شركت مي نمايند؛
4. تعيين شخص هماهنگ كننده در سطح واحدها براي انجام خودارزيابي در سطح واحدها؛
5. تعيين كاركنان با صلاحيت براي اجراي خودارزيابي در سطح شركت و در سطح واحدها؛
6. اجراي دوره اي خودارزيابي؛
7. تحليل نتايج خودارزيابي؛
8. Assigning managers who are responsible for organizing implementation of self-assessment;
9. Assigning departments that are responsible for implementation of self-assessment, controlling implementation of self-assessment, developing corrective measures, and analyzing the results obtained from self-assessment in all levels of the company;
10. List of managers who participate in self-assessment;
11. Assigning coordinator in departments for implementation of self-assessment in level of departments;
12. Assigning competent staff for implementation of self-assessment in company level and level of departments;
13. Periodical implementation of self-assessment;
14. Analyzing self-assessment results.

خودارزيابي در نيروگاه شامل خودارزيابي دائمي و خودارزيابي هدفمند(برنامه ريزي شده و برنامه ريزي نشده) مي باشد.

Self-assessment at NPP includes permanent and just-in-time self-assessment (planned and unplanned).

خودارزيابي دائمي، ارزيابي است كه مديران در زمينه عملكرد كاركنان و تجهيزات خود و روند انجام فعاليت هايي كه بر ايمني بهره برداري تاثيرگذار است انجام مي دهند. بر اساس نتايج اين خودارزيابي ها، اقدامات اصلاحي و پيشگيرانه تعيين مي شود و يا براي تعيين علل ريشه اي آنها، اجراي خودارزيابي هدفمند سازماندهي مي گردد.

Managers implement permanent self-assessment in order to assess the performance of their staff and equipment and the process of carrying out activities that affect operation safety. Based on the results of these self-assessments, corrective and preventive actions are determined and/or the implementation of just-in-time self-assessment is organized for determining their root causes.

در خودارزيابي هدفمند وضعيت ايمني هسته اي 17 مديريت از مجموعه مديريت هاي شركت بهره برداري شركت مي نمايند. اين خودارزيابي با تشكيل كميته مركزي به رياست سرمهندس نيروگاه و كميته هاي داخلي هر مديريت با رياست مدير واحد انجام مي گيرد.

Seventeen managements of Bushehr NPP operating company managements participate in just-in-time self-assessment nuclear safety status. This self-assessment is implemented by forming central committee chaired by Plant chief engineer and internal committees of each management chaired by department manager.

3: اصلاحات و بهبودها

3. Modifications

كنترل تغيير، عبارت است از بررسي تمامي فعاليت‌هاي مرتبط با هر گونه تغيير، ويرايش و تصحيح مدارک داخلي و خارجي مي‌باشد. تمامي تغييرات طراحي توسط مديريت صاحب تجهيز مورد بازنگري كامل قرار گرفته و در صورت پذيرش تغيير، دليل اعمال آن به عنوان سابقه ثبت مي‌گردد. تغييرات اعمال ‌شده مستند گرديده و مدرك تغيير يافته همانند مدرك اصلي مورد ثبت و كنترل قرار مي‌گيرد.

Controlling the change is the review of all activities related to any kind of change, revision, and correction of external and internal documents. All design changes are revised completely by the equipment-owning management and if the change is accepted, reason for its implementation is registered in the form of record. Implemented changes are documented and the changed document is registered and controlled like the main document.

اصلاح و بهبود موقت

Temporary modification

اصلاح و بهبود موقت تجهيزات و سيستم‌ها براي رفع مشكلات برخاسته از عدم تطابق‌هاي كوتاه مدت از حالت طراحي يا رژيم‌هاي بهره‌برداري تجهيزات نيروگاه با درج در مدارك معتبر در يك دوره زماني تا اينكه به يكي از دو روش اعمال تغييرات در مدارك موجود و يا جايگزيني يا تعمير تجهيز بر طرف گردد كه طبق مدرك دستورالعمل اصلاح و بهبود موقت تجهيزات و سيستم‌ها مصوب در نيروگاه انجام مي‌شود.

Temporary modification of equipment and systems for removing the problems arising from short-term non-conformities with design mode or operating regimes of Plant equipment, and also recording them in valid documents for a time period so that they be removed by 1) making changes in the existing document or 2) replacing or repairing the related equipment, which is implemented according to the instruction for temporary modification and improvement of equipments and systems approved in the plant.

اصلاح و بهبود دايم

Permanent modification

الزامات اصلاح و بهبود دايمي در سيستم‌ها و تجهيزات نيروگاهي طبق بندهاي ذيل مورد توجه مي باشد:

The requirements of permanent modification in Plant systems and equipment are considered according to the following articles:

* قبل از اجراي اصلاح بهبود تاثير بر حدود ايمني نيروگاه توسط واحد متقاضي اصلاح و بهبود دايم با همكاري معاونت ايمني تهيه و توسط سرمهندس نيروگاه تاييد مي‌گردد. اين گزارش شامل موارد ذيل مي‌باشد:
* Before the implementation of modification, the report of impact on Plant safety limits is developed by the department requesting the permanent modification in collaboration with safety division and is approved by Plant chief engineer. This report includes the followings:
* محاسبات ميزان تغييرات حاشيه ايمني (Safety Margin) ناشی از تغییر و یا اصلاح براي تجهيزات و سيستم‌هاي ايمني و مهم براي ايمني، بایستی توسط ارگان صاحب صلاحیت شامل طراحان ، سازندگان و موسسات و مشاورین علمی و فنی به درخواست صاحبان تجهیز تعیین شود. میزان کفایت و استدلال ایمنی ارائه شده توسط معاونت ايمني ارزيابي مي‌گردد.
* The calculations of the amount of safety margin changes caused by modification of safety and important to safety systems and equipment shall be determined by competent organizations including designers, manufacturers, institutions, and technical and scientific consultants at the request of equipment owners. Safety division reviews the amount of sufficiency and safety justification presented.
* الزامات مندرج در مدرك گزارش نهايي ارزيابي ايمني (FSAR) نيروگاه در خصوص اصلاح مورد نظر مورد ارزيابي قرار گيرد.
* The requirements included in FSAR document of the Plant regarding intended modification shall be reviewed.
* الزامات مندرج در مشخصات فني مورد ارزيابي قرار گيرد.
* The requirements included in technical specifications shall be reviewed.
* كسب تاييديه از كميته راهبردي اصلاح و بهبود تجهيزات جهت اجراي اين اصلاحات الزامي است.
* Obtaining confirmation from strategic committee of equipment modification is necessary for implementing these modifications.
* تغييرات در طراحي بايد مطابق الزامات، كدها، استانداردها و قوانين کاربردي انجام پذيرند.
* Changes in the design shall be implemented according to practical regulations, standards, codes, and requirements.
* تغييرات در طراحي بايد با توجه به اهداف و الزامات طراحي اوليه صورت گرفته و توسط افرادي كه مجوزهاي لازم را دارند، طراحي، بازنگري و مورد تصويب قرار گيرند.
* Changes in design shall be implemented according to initial design requirements and goals and shall be designed, revised, and approved by the staff holding necessary licenses.
* تهيه تصميم فني جهت اجراي کليه اصلاح و بهبود دايمي الزامي است.
* Preparing technical decision for implementing all permanent modification is necessary.
* تغييرات در مراحل نصب و راه‌اندازي بايد مطابق الزامات، كدها، استانداردها و قوانين کاربردي انجام گيرند.
* Changes in the stages of installation and start up shall be implemented according to practical regulations, standards, codes, and requirements.

مستندات بهره‌برداري و آموزش، بايد به واسطه تغييرات در طراحي به‌روز گردند.

Training and operating documents shall be updated according to the design changes.

## Article 19, Operation

19-3) Description on safe operation (safety margin in operation and operation with operational limits and conditions (operation based on Technical Specification on Safe Operation);

در مدرک TSSO نیروگاه تمام حدود شرایط بهره برداری ایمن و تمام محدودیت ها جهت بهره برداری ایمنی در رژیم های مختلف کاری مشخص شده است. بر اساس مدرک فوق الذکر حدود به شرح ذیل می باشد:

In TSSO document of the Plant, all limits and conditions for safe operation and all limits for safe operation in different working regimes are specified. Based on the above mentioned document, limits are as follows:

DESIGN LIMITS FOR THE UNIT OPERATION

Safety criteria

#### The reactor core as well as the associated coolant system of the reactor is designed with the relevant margin to provide for non-exceedance of the defined permissible design limits at any operational conditions as well as under effect of the events expected during operation.

#### In general, depending on the operation sphere the design limits have the following classification:

* operational limits;
* safe operation limits;
* maximal design limits.

#### For normal operation conditions, the operational limit is established for FE damage due to formation of micro-cracks with the defects such as gaseous leakiness of the cladding, which shall not exceed 0,2 % of FRs and 0,02 % of FRs at the direct contact of the nuclear fuel with the coolant.

#### Non-exceedance of the operational limits in the core is provided by maintaining RP parameters within the design limits by means of control and monitoring systems of normal operation (APC, ROM, APP, PP, SGIC).

#### For normal operation conditions and at disturbances of normal operation condition, safety operation limit is established by quantity and size of FE - 1 % of FRs with the defects such as gaseous leakiness and 0,1 % of FRs for which the direct contact of the coolant and nuclear fuel is available in the modes of normal operation disturbance (taking into regard functioning of the protective systems):

* at malfunctioning of the reactor control and monitoring system;
* at RCPS power supply loss;
* at tripping the turbine-generators and heat consumers;
* at complete loss of the external power supply sources;
* at primary circuit leakages compensatory and non-compensatory by the primary circuit makeup systems.

#### During design basis accidents (taking into regard functioning of the emergency core cooling system) the maximum limit of FE damage is not exceeded:

* fuel cladding temperature is not more than 1200 °С;
* local oxidation depth of the fuel cladding is not more than 18 % of the initial wall thickness;
* fraction of the reacted zirconium is not more than 1 % of its mass in the fuel claddings.

#### The values of the reactivity coefficients by the coolant specific volume, by coolant temperature, by fuel temperature and by the reactor power are negative within all range of the reactor parameters variations at normal operation, disturbances in normal operation and design accidents.

#### The reactor core structure jointly with the non-interruptible power supply system, CPS, ECCS, interlocks eliminates the possibility of the core disruption and fuel melting in all design modes and allows to perform the reactor core unloading in the specified modes.

#### Maximal linear thermal flux from FRs in normal operation modes shall not exceed the limit values at the bottom half of the core height - 448 W/cm, in the top half of the core – linear decreasing with the intermediate value 360 W/cm at the height as 80 % of the core bottom taking into account errors of the defined parameters.

Basic setpoints of EP, PP-1, PP-2 for the linear energy release for the different number of operating RCPS at energy reactor power levels are specified below. In case of changing the number of operating RCPS, the time delay of 80 seconds gets actuated in SHWC-3 equipment to apply reduction factors to setpoints of the linear energy release.

#### Departure from nuclear boiling ratio (DNBR) in the core taking into account the error of its determination in modes with anticipated operational occurrences at confidence probability 95 % shall be not less than 1,0 taking into account the errors of the determined parameters.

ICIS software and application software related to departure from nuclear boiling ratio determination shall be verified according to ICIS operating instructions, including comparison with results of calculations by codes used for RP design justification. By verification results, DNBR setpoints for steady and unsteady modes shall be as follows:

- EP - 1.2;

- PP-1 - 1.3;

- PP-2 - 1.35;

#### The safety criteria defining the integrity of the reactor coolant system is non-exceedance of the pressure value to more than 15 % of the operating pressure value.

#### The specified safety criteria is the safety operation limit for the primary circuit coolant system.

#### All range of pressure increasing from the nominal one up to the safety operation limit has a number of setpoints on pressure upon reaching the values of which the process protections shall be activated to prevent pressure increasing over the operational limits established by the design, or the safety systems shall be activated in case of exceeding the operational limits by pressure, to mitigate the emergency condition consequences and prevent against reaching the safety operation limit.

#### Non-exceedance of the operational limits by pressure in all spectrum of NOC modes is provided by the pressurizing system.

RP design shows non-exceedance of the following safety criteria:

- power fluctuation rates shall not exceed the values specified in Table 2;

- maximal efficient time of FA operation in the core depends on fuel enrichment value;

- maximal calendar time of FA being in the core is 5 years;

- maximal deviation of offset current state from its steady value is 5 %. Steady offset value is the offset value established at the reactor operation at nominal power;

- reactivity coefficient by coolant specific volume, coolant and fuel temperature and the reactor power is negative within all range of the reactor parameters variation;

- minimal efficiency of the emergency protection at Nnom, at the beginning and the end of fuel loading operation, not less that the value defined in NPC album for this specific loading;

- re-criticality temperature is not more than 120 °С;

- maximal power of the reactor at RCPS switching – at three running ones it is 30 % Nnom, at two running ones it is 20 % Nnom.

Operational limits

Operational limits by process parameters

#### In Table 1 the process parameters are given for the unit condition “operation at power” for normal operation.

Table 1 – Process parameters in condition “operation at power ” for normal operation

|  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- |
| Parameters | Number of RCPS running | | | | Accuracy of determination |
| 4 | 3 | 2  opposite loops | 2  adjacent loops |
| 1 Maximally permissible thermal power of the reactor (taking into account accuracy of maintaining by the control system), MW/%Nnom | 3000+60  100+2 | 2010+60  67+2 | 1500+60  50+2 | 1200+60  40+2 | ±60/2 |
| 2 Defined (permissible) power, MW / % | 3000/100 | 2010/67 | 1500/50 | 1200/40 | ±60/2 |
| 3 Coolant heating in the reactor (in every circulation loop), max., °С | 32,5 | 25,5 | 26,5 | 21,5 | ±0,5 |
| 4 Pressure differential at the reactor without taking into consideration the nozzles, MPa | 0,381±0,06 (at 4 running RCPS and flow-rate 84800 m3/hr) | | | | |
| 5 Coolant temperature at the reactor top, °С, not more than | 325 | 316 | 317 | 310 | ±1 |
| 6 Maximal coolant temperature at FA outlet, °С not more than | 335 | 333 | 338 | 330 | ±1 |
| 7 Reactor coolant flow, m3/hr | 84800  (+4000  -4800) | 64000  (+1700  -2400) | 40700  (+1250  -1750) | 40700  (+1250  -1750) |  |
| 8 Coolant temperature at the reactor bottom, °С, not more than | 292 | 290 | 290 | 288 | ±1 |
| 9 Coolant pressure at the reactor top, MPa (kgf/cm2) | 15,7±0,2 (160±2) | | | | ±0,1 (1) |
| 10 Steam pressure in running SG at the steam header outlet not more than, MPa (kgf/cm2) | 6,27±0,05 (64±0,5) | | | | ±0,05 (0,5) |

#### APC shall be calibrated by weighted average value of thermal power calculated by ICIS taking into account the possibility to provide for the power control error to be not more than 2 % of the nominal one.

#### Nominal value of water level in SG shall amount (2400±50) mm as per double-chamber level-meters.

The nominal level is water level in the steam-generator as per the double-chamber level meter, at which with the margin at least 50 mm the steam moisture at the steam-generator outlet is provided (in the steam-line) not more than 0,2 % at the steam-generator steam capacity equal to 100 %.

#### The reactor coolant flow and SG steam flow-rate shall be defined by an indirect method. Coolant flow shall be defined by RCPS rating data and by heat balance; steam flow-rate shall be defined by feed water flow-rate taking into account the flow-rate for SG blow-down and by the heat balance as well.

#### If the parameter values exceed the established operational limits, the operative measures shall be immediately taken to provide for correspondence to the specified limits including also reducing the reactor power, with the maximally-permissible rate specified in Table 2. If it is impossible to restore the permissible parameter values, the reactor shall be shutdown at the operating rate.

#### Table 2 shows the permissible rates for the reactor power variation and the associated permissible number of FA loading cycles.

Table 2 - permissible rates for power variation and permissible number of FA loading cycles

|  |  |  |  |
| --- | --- | --- | --- |
| Permissible rates for power variation | | | permissible number of TA loading cycles |
| 1 POWER DOWN (except for EP, APP and PP1) | | | Number of cycles is unlimited |
| From 100% Nnom and less  Rate | up to  not more | MCL  3% Nnom /min |  |
| 2 POWER ASCENSION (excluding power ascension as per item 3) | | | 70 cycles per FA life-time |
| 2.1 From MCL  Rate | up to  not more | (40÷45)% Nnom  3% Nnom /min |  |
| 2.2 From (40÷45)% Nnom  Rate | up to  not more | (75÷85)% Nnom  1% Nnom /min |
| 2.3 At power level  Delay | not less | (75÷85)% Nnom  3 hours |
| 2.4 From (75÷85)% Nnom  Rate | up to  not more | 100% Nnom  1% Nnom /min |
| 3 POWER ASCENSION   1. after more than 12 days operation at any decreased power level 2. after refueling at more than 12 days of the reactor operation at the end of the previous cycle at the power effect of the reactivity 3. at connection of the idling loop | | | 23 cycles per FA life-time, without taking into account RCPS activation at MCL |
| 3.1 From MCL  Rate | up to  not more | 50% Nnom  3% Nnom /min |  |
| 3.2 From 50% Nnom  Rate | up to  not more | 80% Nnom  0.17% Nnom /min |
| 3.3 From 80% Nnom  Rate | up to  not more | 100% Nnom  0.017% Nnom /min |
| Average rate of power ascension within the range from 50% Nnom to 100% Nnom shall be provided by:   1. sequential power ascension to (2÷4) % Nnom 2. rate of sequential power ascension to 2% Nnom /min 3. delay between the ascension steps | | |  |
| *Note.* At idling loop cut-in - 23 cycles within FA lifetime per each RCPS, without taking into account RCPS connection at MCL | | | |
| 4 POWER SURGE TO 20% Ncur at load variations | | | 15 cycles per FA life-time |
| 4.1 From MCL | to | 50% Nnom |  |
| Implemented by:   1. one step 2. at rate provided by the reactor control system | | |  |
| 4.2 From 50% Nnom | to | 100% Nnom |  |
| Implemented by:   1. two steps 10% Ncur per each 2. at rate provided by the reactor control system 3. delay between the ascension steps at least 3 hours | | |  |
| Note:   1. Upon selection of the column (RP power ascension rates) the determining factor is the operation at a stable (considering measurement accuracy) low power level within 50%Nnom < Nlow < 98%Nnom 2. In case RP worked at Nlow more than 12 days, and then by some reason occurred power decrease to Ncur < Nlow and RP worked at Ncur less than 12 days, power ascension from Ncur to Nlow shall be performed according to item 2 of the table, and from Nlow to Nnom – according to item 3 of the table. 3. If in the a.m. case RP worked at the power level Ncur more than 12 days, the following variants are possible:   a) Ncur > 50%Nnom load increase from Ncur to Nnom shall be performed according to item 3 of the table;  b) Ncur < 50%Nnom load increase from Ncur to Nlow shall be performed according to item 2 o the table, and from Nlow to Nnom – according to item 3 of the table. | | | |

#### Prior to connection of the idling loop to three (two) operating ones, the reactor power shall be decreased up to 30 (20) % Nnom respectively. The rate of power ascension after RCPS activation shall be in compliance with the third item of Table 2 up to the value defined by the second item of Table 1.

#### Operational limits as per the power-flux variation coefficients.

#### The power flux variation coefficients by the core volume Kv(i,j), controlled by MCDS readings shall not exceed Kv max.(i,j) defined by the following ratio:

#### Kv max (i,j) = Kv nom(i,j) /(0.17+0.83Ncur./Nnom),

#### where i – a number of the core height layer with numbering from the core bottom (i =1,2,…,16);

#### j – FA number (j =1,2,…,163);

#### Kv nom(i,j) – maximally permissible Kv(i,j) value at the nominal level of the reactor power;

#### Ncur. – current reactor power;

#### Nperm – permissible reactor power,

The array of Kv nom (i,j) for charge core life shall be calculated within neutronic calculation period to substantiate safety of the current fuel column with the step for the charge core life not exceeding 40 effective days and specified in neutron physical characteristics album for each fuel element column.

Kv max maximal values shall be set in MCDS as alarm setpoint and measured automatically depending on the current value of the reactor power.

ICIS controlled value of power flux axial peaking factor – Kq at the nominal power level has a limit:

Kq ≤ Kq max,

where Kq max=1,35.

At any power level the following limit shall be fulfilled:

Kq ≤ Kq max×Ncur nom / Ncur,

where:

Ncur nom – permissible reactor power at the current value of running RCPS,

Ncur – current reactor power.

The algorithms of control for power and power-flux distribution in the core are specified in Appendix E.

The permissible values of the quality indices of the operating media in the primary and secondary circuits during the Unit operation are specified in Appendix F.

#### Limits on the equipment loading conditions

#### .1 In the process of the Unit operation the quantity of the equipment loading cycles shall be recorded at normal operation and at disturbances in normal operation, as well as the quantity of modes and quantity of cycles for the design basis accidents which are limited by the design

#### Permissible number of cycles and loading conditions, as well as the lifetime of auxiliary system pipelines directly joining the primary circuit pipelines up to the second isolation valves counting from the place of in-cut to the primary circuit, damage of which may be the initial emergency event, corresponds to the cycles, loading conditions and the lifetime of the relevant primary circuit equipment.

#### Number of FA loading cycles within the lifetime shall not exceed the values specified in TSSO and those specified in the catalogue description «Complex of the integral part of the WWER-1000 (type V-446) core 0401.16.00.000 DKO.

Limitations, relating to the core, of the number of cycles of each mode is not applied to cases of concurrent occurrence of two and more of these modes. In case two and more modes of this section occur concurrently, it may require to carry out an inspection of the core. While implementing mode, specified in TSSO, it is necessary to inspect the core to define the suitability of fuel assemblies, CPS control rods and BAR bundles to their further operation. Operability of fuel assemblies, CPS control rods and BAR bundles shall be defined by an experienced group of representatives of the Principal (Consumer), Supplier and Manufacturer of the core component parts.

#### At the beginning of commissioning activities, the continuous acquisition and storage of the data on recording the passed normal operation modes, disturbances in normal operation and the design basis accidents shall be provided in order to accumulate the statistics.

To control the remaining lifetime of RP elements (units), the design envisages the relevant CAS subsystem within MCDS, including the software of monitoring the residual cyclic resource of RP main equipment and performs functions of monitoring the residual cyclic resource of the following RP equipment elements:

* reactor vessel;
* pressurizer;
* steam generators;
* main coolant pipelines;
* pressurizing system pipelines.

#### The following limits are imposed to the primary circuit coolant temperature and the reactor plant elements during hydraulic testing by pressure 24,5 MPa (250 kgf/cm2) and tests for tightness by pressure 17,6 MPa (180 kgf/cm2) within the lifetime:

|  |
| --- |
| 1) Minimum temperature of primary circuit coolant, when (taking into account coolant and equipment metal cooling-down during hydraulic tests) increase of pressure is allowed in the primary circuit to the hydraulic tests pressure depending on the time of RP operation, is given in table 3 |

Table 3 - Dependability of coolant minimum temperature before pressure increase to the pressure of the primary circuit hydraulic tests on RP operation duration

|  |  |  |  |  |  |  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- | --- | --- | --- | --- | --- | --- |
| RP operation duration, years | 0 | 1 | 4 | 7 | 8 | 12 | 16 | 20 | 24 | 28 | 30 | 40 |
| Minimal temperature, °С | 100 | 130 | 130 | 130 | 107 | 113 | 118 | 122 | 125 | 129 | 130 | 137 |

2) Minimum temperatures of RP equipment external surfaces during hydraulic tests are given in table 4

Table 4 - Minimum temperatures of RP equipment external surfaces during hydraulic tests

|  |  |
| --- | --- |
| Name of equipment, pipelines | Minimum temperature of external surface, °С |
| 1 SG coolant header | 70 |
| 2 Pressurizer body | 75 |
| 3 RCPS primary circuit body and elements | 50 |
| 4 RCP | 50 |
| 5 Pressurizer system connecting pipeline | 45 |
| 6 ECCS passive part pipelines within the boundary of the primary circuit | 40 |
| 7 Emergency gas removal system pipelines within the boundary of the primary circuit | 20 |
| 8 SG vessel | 80 |
| 9 ECCS tank vessel | 30 |
| 10 ECCS pipelines from the tank to the first shutoff gate valve | 40 |
| 11 Bubbler tank vessel | 20 |
| *Note* – Minimum temperature of the surfaces of RCPS primary circuit vessel and elements, SG coolant header and bubbler tank vessel is identified by the coolant temperature. | |

3) Maximum temperature of the coolant during hydraulic tests is not more than 140 °С.

4) Hydraulic tests parameters of RP equipment not included into the primary circuit are given in table 5.

Table 5 - Hydraulic tests parameters

|  |  |  |  |  |
| --- | --- | --- | --- | --- |
| Description of system, part of system,  equipment or pipelines | Oper. pressure,  MPa | Hydraulic tests pressure, (permissible limits of fluctuation) MPa | Inspection pressure,  (permissible limits of fluctuation),  MPa | Hydraulic tests temeprature, °С |
| 1 Dead-end sections of ECCS pipelines | 17,66 | 24,5  (24,5-25,0) | 19,62  (19,62-20,6) | 40-120 |
| 2 SG for secondary circuit | 7,84 | 10,79  ( 10,79-11,3) | 8,62  (8,62-9,07) | 90-120 |
| 3 ECCS tank | 6,37 | 8,33  (8,1-8,5) | 6,64  (6,64-7,0) | 40-60 |
| 4 ECCS pipeline section from the tank to the first shutoff gate valve | 6,37 | 8,33  (8,1-8,5) | 6,64  (6,64-7,0) | 40-60 |
| 5 Bubbler tank vessel with the discharge pipeline section to pressurizer PORV | \*) | 0,98  (0,96-1,09) | 0,78  (0,77-0,87) | 20-60 |
| \*) Operating pressure for bubbler tank vessel is 0,69 MPa, for discharge pipeline section from the bubbler tank to pressurizer PORV is 11,3 MPa. | | | | |

Table 6 - Dependability of the reactor vessel surface temperature on the operation period when hydraulic tests and leaktigtness tests

|  |  |  |  |  |  |  |  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- | --- | --- | --- | --- | --- | --- | --- |
| Duration of RP operation, years | 0 | 1 | 4 | 7 | 8 | 12 | 16 | 20 | 24 | 28 | 32 | 36 | 40 |
| Minimal temperature at hydraulic tests for strength  (Р=24,5 MPa), °С | 75 | 105 | 105 | 105 | 82 | 88 | 93 | 97 | 100 | 104 | 106 | 109 | 112 |
| Minimal temperature at hydraulic tests for tightness (Р=17,7 MPa), °С | 48 | 78 | 78 | 78 | 62 | 68 | 73 | 77 | 80 | 83 | 86 | 89 | 91 |

#### Within the operation period at the heated-up SG, not more than 240 blow-downs of each pulse lines of the single-chamber surge tanks are allowed with water at temperature (164±4) °С.

#### Number of drastic feed water temperature variation cycles from 220 °С to 160 °С is not more than 400 within the whole lifetime.

#### In case of the emergency cool down, water with temperature from 5 °С to 40 °С may be supplied by an individual main pipe in SG within the whole cool down period. The number of cycles is 80 per each steam-generator within the whole service life.

Earthquake effects (according to MSK-64 scale):

Safe shutdown event (SSE), points: 8 (7.6)

- horizontal acceleration, g 0.40

- vertical acceleration, g 0.22

Design basis earthquake (DBE), points: 6 (6.2)

- horizontal acceleration, g 0.20

-vertical acceleration, g 0.13

#### Duration of RCPS operation in the design basis earthquake conditions is not more than one minute, the number of the design basis earthquakes for the whole lifetime of the bearings and shaft sealing is not more than two minutes. After SSE and after design basis earthquake the inspection of the pump is required.

#### Parameters characterizing serviceable condition of the systems important to safety

#### Parameters characterizing serviceable condition of the systems important to safety and controlled in the process of normal operation are specified in Appendix B.

#### Operational limits as per radiation parameters

Reference levels (RL) of radioactive gases and aerosols releases to the atmosphere for NPP per month are brought in the following table 7.

Table 7.

|  |  |
| --- | --- |
| Radionuclides | RL, Bq/month |
| IRG | 5,7×1013 |
| Iodine-131 | 1,5×109 |
| 60Co | 6,2×108 |
| 134Cs | 7,5×107 |
| 137Cs | 1,7×108 |
| Note: in individual months a release of radionuclides, which exceeds RL 3 times, is allowed on condition that the annual PR will not be exceeded. | |

#### The permissible discharge with radioactive drains within the defined time period in the process of operation for any reference radioactive nuclide or sum of the radioactive nuclides shall not exceed the value equal to a portion of the maximally permissible annual discharge value, proportional to the time passed from the beginning of the calendar year. Operational limit shall be calculated by formula:

PDop = PDannual × t/ 365, Ci

where PDop – operational limit for the permissible discharge;

PDannual – annual permissible discharge (safety operation limit);

t – number of days within the time period from the beginning of the calendar year up to the moment of sampling.

During calculation of the operational limits, PDannual values shall be taken Table 8.

If there are several radioactive nuclides in the drains, the sum of these individual radioactive nuclides activity ratios to their operational limit PDop shall not exceed one.

At the irregular discharge within a year it is allowed to exceed the operational limits during certain periods provided this exceeding is compensated during further periods of the calendar year.

#### The operational limits of discharge for individual radioactive nuclides, may vary within the large range, but their specific concentration in liquid wastes shall not exceed the values listed in Table 8. The specified values are the permissible concentration for drinking water (PL Apop.) as per P-2 NRB-96.

Table 8– Permissible concentration of radionuclides in liquid discharges outside the NPP

|  |  |  |  |  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- | --- | --- | --- | --- |
| Nuclide | Cr-51 | Mn-54 | Co-58 | Fe-59 | Co-60 | Sr-89 | Sr-90 | Zr-95 | Nb-95 | Mo-99 |
| Permissible concentration, Bq/kg | 3,3×104 | 1,8×103 | 1,7×103 | 6,9×102 | 3,7×102 | 4,8×102 | 4,5×101 | 1,3×103 | 2,1×103 | 2,1×103 |
| Nuclide | Ru-103 | Ru-106 | I-131 | Te-132 | Cs-134 | Cs-137 | Ba-140 | La-140 | Ce-141 | Ce-144 |
| Permissible concentration, Bq/kg | 1,7×103 | 1,8×102 | 5,7×101 | 3,4×102 | 6,6×101 | 9,6×101 | 4,8×102 | 6,3×102 | 1,8×103 | 2,4×102 |

#### The permissible value of the total specific activity of iodine 131-135 radioactive nuclides in the primary circuit coolant is not more than 3,7×107 Bq/kg (1×10-3 Cu/kg). The specified value corresponds to the operational limit of RP operation by number of leaking FE equal to 0,2% of gas-leaking FE and 0,02% of FE having direct contact of fuel with coolant (in recalculation to the design flow-rate of the primary circuit coolant blow-down to filters ТС – 30 t/hr and degassing flow-rate in ТА system - 5 t/hr).

#### The permissible value of the primary circuit coolant leakage by individual steam-generators is 4 kg/h.

#### The permissible value of the actually measured activity of isotope 131I in blow-down water from «salt» chamber of each SG is 370 Bq/kg.

Safe operational limits

The value of the total specific activity of iodine 131-135 radioactive nuclides in the primary circuit coolant shall not be over 1,85×108 Bq/kg (5×10-3 Cu/kg). The specified value corresponds to the safety operation limit by number of leaking FE, equal to 1 % of gas-leaking FE and 0,1 % of FE having direct contact of fuel with coolant (in recalculation to the design flow-rate of the primary circuit coolant blow-down to filters ТС – 30 t/hr and degassing flow-rate in ТА system - 5 t/hr).

The safety operation limit of the value of iodine-131 radionuclide specific activity in blow-down water of each SG is not more than 740 Bq/kg (2×10-8 Cu/kg).

The safety operation limit of the value of leakage from primary circuit to the secondary one by individual SG is not more than 5 kg/hour.

Safe operational limits by radiation parameters.

Annual permissible levels of releases of radioactive gases and aerosols into the atmosphere for the NPP brought in table 9.

### Table 9

|  |  |
| --- | --- |
| Radionuclides | Annual release, Bq/year |
| IRG | 6,9 ×1014 |
| I-131 | 1,8×1010 |
| 60Co | 7,4×109 |
| 134Cs | 9,0×108 |
| 137Cs | 2,0×109 |

Permissible discharge of radionuclides in the environment with liquid drains of the NPP (outside the NPP) per one calendar year shall not exceed the values brought in table 10.

Table 10

|  |  |
| --- | --- |
| Permissible discharge (PD), Bq/year | |
| Tritium | 4,4×1013 |
| Radionuclides\* | 2,1×1010 |
| *Note*: Radionuclides\* - total activity of the following radionuclides in liquid drains of the NPP: Sr-89, Sr-90, Mo-99, Ru-103, Ru-106, I-131, Te-132, Cs-134, Cs-137, Ba-140, La-140, Ce-141, Ce-144, Zr-95, Nb-95, Fe-59, Co-58, Cr-51, Mn-54, Co-60 | |

Values of the personnel exposure dose limits shall not exceed the values specified in Table 11.

### Table 11 – the main exposure dose limits for different categories of the personnel

|  |  |
| --- | --- |
| Personnel category | Individual effective dose |
| Group A | 20 mSv (2 Rem) per year in average for any successive 5 years, but not more than 50 mSv (5 Rem) per year |
| Group B | Radiation dose shall not exceed 1/4 of values for the personnel of group А |

#### Maximal design pressure (absolute) under the confinement during the design basis accident is 0,46 MPa (4,7 kgf/cm2).

#### The safe operation limits by the process parameters of the primary and secondary circuits are specified in Table 12.

Table 12 Safe operational limits by process parameters of the primary and secondary circuits

|  |  |
| --- | --- |
| Parameter | Parameter value |
| Period of thermal neutron flux variation | 10 sec |
| Maximum primary circuit pressure | 180/190 kgf/cm2 \*\* |
| Minimal primary circuit pressure at power not less than 75 % of Nnom, at power less than 75 % of Nnom | 148/Р(TsI+10°С) kgf/cm2  140/Р(TsI+10°С) kgf/cm2 |
| Maximal pressure in one of SG by the secondary circuit | 80/86 kgf/cm2 \*\* |
| Minimal pressure in one of SG by the secondary circuit | 52/45 kgf/cm2 \* |
| Maximal temperature in hot leg of the circulation loop | (Тnom+8)/(TsI-10)°С |
| Minimal level in one PG after protection activation «Нnom-900 – EFWP actuation moment» | (Нnom-900)/(Нnom-1100)mm\* |
| Minimal level in PRZ | 4600/4200 mm |
| Maximal temperature at the core top after cool down in the mode of residual heat removal (including the mode of maintenance cool down) | 80 °С |
| Notes: 1. In numerator, the value of the parameter is specified which is before the reactor emergency protection activation, in denominator – after the reactor emergency protection activation.  2. \*) The parameter value may be reduced in case of taking a decision on the Unit changeover to the mode in which reaching the energy level of power (e.g., cool-down) is not envisaged.  3. \*\*) Except for mode of hydraulic testing. | |

19-7) Description on responding to operational transient and accidents (DBA and DEC);

در رابطه با حوادث مبنای طراحی در مدرک FSAR آنالیز انجام شده است. بر اساس مدارک طراحی، FSAR و مدرک کارخانه ای تجهیزات مدرک مدیریت حوادث مبنای طراحی DBA تهیه شده است. این مدرک جزی مدارک لیسنسی نیروگاه می باشد که به تایید طراح اصلی نیروگاه، طراح مجموعه تاسیسات راکتور و سازمان بهره برداری و ارگان نظارتی جمهوری اسلامی ایران INRA-NNSDرسیده است. این مدرک در محل کاری اتاق کنترل اصلی نیروگاه موجود است و در شرایط بروز حوادث مبنای طراحی کارکنان بر اساس آن عمل می نمایند. کلیه کارکنان در رابطه با استفاده از مدرک مذکور آموزش دیده اند و همچنین در شبیه ساز تمام عیار نیروگاه، موارد را شبیه سازی و آموزش دیده اند. بصورت دوره ای مانور شرایط اضطراری در شبیه ساز برگزار می گردد و عملکرد کارکنان در شرایط اضطراری ارزیابی می شود.

As for the DBA, the FSAR analysis has been performed. Based on the design documents, the FSAR and manufacturing documents of equipment, the DBA document has been developed. This document is one of license documents of plant and has been approved by main designer of Bushehr NPP, designer of reactor plant, Operating Organization and Iranian Nuclear Regulatory Authority. This document is available in MCR of plant and the staff performs according to it in case of design basis accidents. All the staff has been trained how to use this document and they have simulated and received training regarding all cases. The periodical emergency drill is performed in the simulator and the performance of staff in emergencies is assessed.

In case of disturbances in normal operation and during emergencies the operative personnel headed by SSP shall take all possible measures to restore its normal operation.

If it is impossible to recover normal operation of the Unit, the operative personnel shall control and provide for activation of the emergency (preventive) protection, SS, protective units and functioning of interlocks, boric acid injection to the primary circuit, and take measures for stabilizing the parameters at the safe level.

At disturbance of normal Unit operation, the operative personnel under head of SSP shall:

notify by radio-searching and operative communication all operative personnel of the Unit as well as CEP, DCEO and the telephone operator for notification of all the personnel as per the list approved by NPP Director;

quickly and correctly find out the reason for disturbance in normal operation by readings of the devices and alarm, as well as by the messages from the operative personnel from their workplaces;

make sure in correctness of the Unit equipment protections and interlocks activation; if some of the protections or interlock failed to activate, make the switchovers envisaged by these protections (interlocks) remotely or using the manual drives;

take measures on stabilizing the Unit parameters at the safe level;

arrange continuous monitoring for radiation situation in the constantly attended premises of the Unit and monitoring for the radioactive isotope release to the environment;

arrange evacuation of the personnel from the premises of the Unit, where the conditions hazardous for their life and health occurred, and with help of the subordinate personnel arrange taking of measures preventing the personnel access to these premises.

After emergency it is required to accumulate compete information on the emergency development, check FE integrity by the primary circuit coolant activity, by the results of the accumulated information analysis, it is necessary to establish observation of the safe operation limits stipulated in section 3.

Regardless of the administrative-and technical personnel presence at the Unit, the operative personnel of the Unit is personally responsible for the emergency or disturbances in normal operation mitigation individually taking the decisions and measures on recovery of the normal Unit operation.

NPP Director, CEP DCEO are entitled to remove any person from the operative personnel of the Unit and entitle his responsibilities to another person or take upon himself liquidation of the emergency or normal operation disturbance consequences, preliminary making the relevant note in the operative log-book of the removed person.

It is forbidden to interfere into functioning of automatics, protections or interlocks, except for cases of their failure.

In the process of emergency (disturbance of normal operation) liquidation at the Unit, the operative personnel headed by SSP shall provide for:

1. non-admission of the uncontrolled reactor power increasing;
2. reliable cooling of the reactor core;
3. non-admission of running RCPS, FWP stoppage;
4. reliable operation of deaeration and feeding facilities of the Unit;
5. reliable operation of RP cool-down system;
6. pressure in the Unit vessels and pipelines to be not more than the permitted one;
7. non-admission of water spreading to the steam-lines and flow-through part of the turbine;
8. oil supply for turbine rotating rotors bearings lubrication;
9. non-exceeding of the limit values for the axial offsets and relative expansions of the turbine rotors;
10. preventing of blocking in the flow-part of the turbine;
11. preventing of non-admitted turbine rotation frequency increasing;
12. reliable Unit HL power supply;
13. taking the measures required to prevent water, steam, nitrogen leakage from the Unit equipment and pipelines and localizing of the occurred leakage as well;
14. preventing of ignition and fire fighting;
15. fulfillment of conditions on brittle and thermocyclic strength of RP equipment units;
16. operation of SS and supporting systems within the time sufficient for the emergency liquidation;
17. in case of any emergency the operative personnel shall provide for residual heat removal form the reactor core and eliminate possibility of coolant loss.

The specific actions of the operative personnel on the certain accident (disturbances in normal operation) liquidation are defined by the following documents:

1. Instruction on emergency liquidation at the BNPP-1 reactor plant
2. Manual on beyond-design basis accidents management;
3. Instruction on malfunctions liquidation in the turbine compartment
4. Instruction on liquidation of malfunctions in the electrical part of the Unit Instruction on liquidation of malfunctions in APCS operation
5. OI for the Unit systems and equipment;
6. BNPP-1 fire fighting plan

In case of radiation accident, actions of the operative personnel are additionally defined by «Emergency plan on the personnel protection at BNPP-1»

Sequence of RP processes control changeover from MCR to ECR and sequence of the personnel actions in such cases shall be defined by an individual instruction.

The Unit control from ECR is allowed only for its cool-down and shutdown in case if personnel attendance in MCR is not possible or in conditions endangering lives of the operative personnel at MCR.

At normal operation of the Unit, ECR compartment shall be closed and sealed by APCS seal.

Shift operating personnel shall immediate shut down the reactor by pressing appropriate buttons of emergency protection system in MCR or ECR in the following situations:

1 in case any of EP signals was generated and the protection system was not actuated (in the hot condition before the elimination of the defect);

2 in case there occur some malfunctions or a failure in two or more than two different EP channels in the modes «Power operation» and «Reactor at MCL»;

3 in case of malfunction in two different EP channels of one set in the mode «Reactor at MCL»;

4 in case of malfunction in two different EP channels of different in the mode «Reactor at MCL»;

5 in case of failure of two sets of EP actuating part in the modes «Power operation» and «Reactor at MCL»;

6 in case of failure of two and more channels of the working range of one NFME set in the modes «Power operation»

7 in case of failure of two and more channels of one NFME set in the mode «Reactor at MCL»;

8 in case the monitoring system of the reactor neutron power is not actuated in two out of three channels of any NFME sets (in the hot condition before the elimination of the defect);

9 in case the monitoring system of the reactor power increase is not actuated in two out of three channels of any NFME sets (in the hot condition before the elimination of the defect);

10 in case there is no monitoring of the following parameters:

1. pressure in the primary and the secondary circuits;
2. temperature of water at the output from the reactor active core;
3. water level in SG of any loop with the RCPS in operation;
4. pressure drop in any of RCPS during the operation of the Unit and two RCPS units;
5. pressure drop in the reactor active core;

(in the hot condition before the elimination of the defect);

11 in case there occurred an uncontrolled continuous movement of CPS CR reactor group upwards caused by failures in the control circuit (accompanied with transferring of RP in the hot condition);

12 in case there occurred an uncontrolled continuous movement upwards of any of CPS CR or a group of CPS CR (accompanied with transferring of RP in the hot condition);

13in case two or more CPS CR dropped to LLS (accompanied with transferring of RP to the hot condition);

14 in case there occurred a failure of design order of CPS CR group movement (accompanied with transferring of RP in the hot condition):

1. moving of several CPS CR upwards with the wrong group;
2. absence of automatic movement transferring between CPS CR groups;
3. simultaneous movement of two groups when they are higher than 50% from the active core bottom.

15 in case there is a leakage of the primary circuit coolant to the secondary one exceeding the rate of 5 kg/h (accompanied with transferring of RP in the hot condition);

16 in case there occurred a failure of one RCPS out of two operating units, which requires shutting it down according to the RCPS operating instruction (accompanied with transferring of RP in the hot condition);

17 in case the water feeding of the component cooling system was blocked to all RCPS units (accompanied with transferring of RP in the hot condition);

18 in case the three make-up pumping units failed (cold condition before the elimination of causes and consequences);

19 in case all feedwater pumps at RP power level exceeding 25 % (accompanied with transferring of RP in the hot condition);

20 in case of any signs of MCR personnel poisoning with gases having asphyxiating, poisoning or narcotic effect (cold condition before the elimination of causes and consequences);

21 in case of fire in one or more rooms

In case of fire at the Unit, the personnel actions are defined by «BNPP-1 fire fighting plan»»

In case of the accident related to the earthquake as an initial event, the operative personnel headed by SSP shall:

control and duplicate EP activation;

control for the state of systems, equipment and pipelines. Incase of any damages, the measures on localizing their consequences shall be taken;

act depending on the certain situation at NPP and changeover the Unit to «cold» condition.

At WC upsets, personnel actions shall be defined by the requirements stated in Appendix F of this specification.

In case of actuation of EP operating personnel shall double EP actuation with key.

* + - 1. At non-design development of an accident, the operative personnel shall act in compliance with the “Manual on beyond-design basis accidents management”.
      2. In case of an accident occurrence, the actions of the personnel are additionally defined by “Emergency plan on the personnel protection at BNPP-1”. NPP administration shall notify the relevant authorities about an incident at NPP as per the established procedure.
      3. In absence of the director or its deputy, the person directly responsible for initial evaluation of an accident, and taking a decision on bringing in force “Emergency plans on the personnel protection at BNPP-1 in case of the radiation accident” is SSP.
      4. RP operator is entitled and shall singly shutdown the reactor in cases, envisaged by the technical specification, and if further operation endangers NPP safety.
      5. If in some premises of the Unit the conditions hazardous for lives and health of people occurred, SSU shall arrange evacuation of personnel from these premises and with help of his subordinates provide for taking the measures preventing personnel access to these premises.
      6. SSP shall notify on all malfunctions in the Unit operation. The notifications shall be made as per the lists of organizations and officials depending on the malfunction type.
      7. At operation with deviations (with violation of the operational limits or conditions but without violating safe operation limits and conditions), the operative personnel shall restore the normal operation, and in case of impossibility to resort it, the reactor shall be shutdown.
      8. During operation, accumulation, processing, analysis, storage of the information on failure of system elements important to safety and the personnel mistakes shall be provided at NPP.
      9. Withdrawal of individual safety system channels from readiness condition, not related to elimination of the channel elements failure, for the time permitted by this technical specification shall not be considered as malfunction in the Unit operation.
      10. Occurred violations of the safe operation limits and conditions including accidents shall be thoroughly reviewed by the committees in compliance with the existing provisions.
      11. Prior to establishing a committee on investigation of malfunction in NPP operation, NPP administration shall take measures on keeping the situation at the place of malfunction such as it was during malfunctioning, terminate all activities at the systems (elements), where the malfunction happened, if it does not cause any danger for lives of people and any serious development of the malfunction.
      12. Prior to start the committee functioning, NPP administration shall:

1) define the nature and scope of malfunction;

2) arrange call of the representatives from relevant organizations, if required;

3) take measures on keeping the diagrams of the recording devices, oscillograms, printouts, tape-recording of the operative communications, operative log-books;

4) record the values of the reactor neutron-physics characteristics, position of switchboards, cut-off and control valves, blinkers, overlaps during malfunctioning;

5) immediately after shift handing over, collect explanatory notes of the personnel involved into malfunction liquidation, its eye-witnesses, managers of the workshops;

6) based on available initial information, draw up the diagrams (in the unified time scale) of parameters variations at the malfunction occurrence and development required to analyze operation of systems (elements), where the notes on switchovers, activation of process protections and interlocks are made;

7) prepare the required design documentation, protocols of tests inspections, checks, circuits operating instructions;

8) prepare maintenance and repair documentation, as well as the information on previously occurred similar malfunctions at NPP.

* + - 1. The internal and external emergency centers shall be established and equipped with the required equipment, devices and communication means, from which management by emergency plans implementation shall be performed in case of an accident.

19-9) Overview on procedures and guidance to manage accident situations at multi-unit (BNPP1 and 2) installations;

با توجه به اینکه در سایت نیروگاه اتمی بوشهر در زمان حاضر فقط یک واحد نیروگاه فعال می باشد و واحد دوم در مرحله بتون ریزی قرار دارد عملا این بند برای نیروگاه اتمی بوشهر کاربرد ندارد و در این زمینه دستورالعمل خاصی وجود ندارد.

Since just one Unit is operating currently in Bushehr NPP site and the second Unit is at the stage of concrete pouring, this item does not practically apply to Bushehr NPP and there is no special instruction in this regard.

19-16) Description on spent fuel management and radioactive waste during operation (on-sit/off-site storage, rad waste predisposal at site, transport and storage);

1: نگهداری و پایش سوخت مصرف­شده در استخر سوخت مصرف­شده

1. Storage and monitoring of the spent fuel in spent fuel pool

نگهداری موقت و پایش سوخت مصرف­شده در استخر سوخت مصرف ­شده در ساختمان راکتور، جهت خنک­سازی و کاهش سطح پرتودهی

آن، بر اساس الزامات فنی ایمنی هنگام نگهداری سوخت مصرف­شده مدرک

«Instruction. Nuclear safety assurance

during storage, transportation, and refuelling of fresh and spent nuclear fuel » به شماره 85.BU.1 0.0.NS.INS.FNSM13215 توسط شرکت بهره­برداری نیروگاه اتمی بوشهر صورت می­گیرد.

Temporary storage and monitoring of the spent fuel in spent fuel pool in reactor building in order to cool and reduce its radiation level based on the Instruction of Nuclear safety during storage, transportation, and refuelling of fresh and spent nuclear fuel assurance 85.BU.1 0.0.NS.INS.FNSM13215 is implemented by the Operating Company of Bushehr NPP.

ارسال سوخت مصرف­شده

Dispatching the spent fuel

شرکت بهره­برداری بر اساس مدرک «Regulation for Granting Permits during Operation of BNPP-1 » به شماره INRA-NS-RE-051-15/01-0-May.2013 جهت ارسال سوخت مصرف­شده از نیروگاه اقدام به اخذ مجوز ویژه از دفتر نمایندگی ایمنی هسته­ای مستقر در نیروگاه اتمی بوشهر می­نماید

Bushehr NPP Operating Company takes special permit No. INRA-NS-RE-051-15/01-0-May.2013 from the representative office of NNSD in Bushehr NPP for dispatching the spent fuel from plant according to the Regulation for Granting Permits during Operation of BNPP-1.

جابجایی سوخت مصرف­ شده بر اساس الزامات فنی ایمنی هنگام جابجایی سوخت مصرف­ شده مدرک «Instruction. Nuclear safety assurance during storage, transportation, and refueling of fresh and spent nuclear fuel» به شماره 85.BU.1 0.0.NS.INS.FNSM13215 توسط شرکت بهره­برداری نیروگاه اتمی بوشهر صورت می­گیرد. جهت ارسال سوخت مصرف­شده، محدودیت­های فنی و اقدامات ایمنی هنگام ارسال، الگوریتم انجام عملیات ارسال و معیارها و کنترل صحت اتمام انجام فعالیت در مدرک «Package of working programs for nuclear fuel handling in the nuclear power plant (fuel receipt, preparation for loading, reloading and preparation for shipment) Spent nuclear fuel shipment program» به شماره 53.BU.1 0.0.AB.SPR.FNSM14292 آورده شده است.

Bushehr NPP Operating Company displaces the spent fuel based on the technical safety requirements for displacement of spent fuel according to the Instruction of nuclear safety assurance during storage, transportation, and refuelling of fresh and spent nuclear fuel No. 85.BU.1 0.0.NS.INS.FNSM13215. The technical restrictions and safety measures while dispatching, algorithm of the process of dispatching and implementation of activity have been mentioned in the Package of working programs for nuclear fuel handling in the nuclear power plant (fuel receipt, preparation for loading, reloading and preparation for shipment) of spent nuclear fuel shipment program No. 53.BU.1 0.0.AB.SPR.FNSM14292.

2: پسمان‌هاي راديواكتيو جامد توليدي نيروگاه پس از جمع‌آوري و تفكيك اگر قابل پرس باشند در بشكه‌هاي مخصوص پرس مي‌شوند و پسمان‌هايي كه غير قابل پرس‌ باشند با توجه به ابعاد بشكه به قطعات كوچكتر تبديل شده سپس در داخل بشكه قرار داده مي‌شوند و پس از اندازه‌گيري توسط تجهيزات گاما اسكنر و تهيه شناسنامه فني آنها به انبار نگهداري موقت پسمان واقع در ساختمان كمكي راكتور(ZC) منتقل مي‌گردند

2- The generated solid radioactive wastes are collected and segregated, then pressed if possible in special drums and those radwastes which cannot be pressed are transformed into smaller parts regarding the dimensions of drum, and then are laid into drums. Then they are measured by the gamma scanner equipment and their technical certificate is prepared and transferred to temporary radwastes storehouse located in the reactor auxiliary building called ZC.

پسمان‌هاي راديواكتيو مايع پس از تغليظ در سيستم عمل‌آوري و آمايش پسمان در داخل بشكه با سيمان و ساير افزودني‌هاي فني مخلوط مي‌شوند. رزين‌هاي تبادل يوني مستعمل و لجن‌هاي راديواكتيو ناشي از فعاليت سيستم‌هاي تصفيه شيميايي نيز در داخل بشكه‌هاي مخصوص سيمانكاري شده و همانند بشكه‌هاي پسمان جامد به انبار نگهداري موقت منتقل مي‌شوند. پس از اخذ مجوز از نظام ايمني هسته‌اي جهت حمل و نقل بشكه‌هاي پسمان راديواكتيو و همچنين اخذ مجوز ويژه از دفتر نمايندگي نظام ايمني در نيروگاه براي هر محموله، بشكه‌هاي پسمان به شركت پسمانداري صنعت هسته‌اي تحويل داده مي‌شود و آن شركت نيز آنها را به سايت محل دفن انارك منتقل مي‌نمايد

Liquid radwastes are concentrated in the treatment and conditioning system of wastes inside drum and then are mixed with cement and other technical additives. The used ion exchange resins and radwaste sludge resulting from activity of chemical treatment systems are transferred to temporary storehouse in especially-cemented drums and like solid radwaste drums. After taking the permit from NNSD for transportation of radwaste drums and also special permit from NNSD for each consignment, the radwaste drums are delivered to the Nuclear Industry Radwaste Management Company.

از شروع راه‌اندازي نيروگاه تا كنون 2719 بشكه پسمان توليد شده و از اين تعداد 2400 بشكه به سايت محل دفن انارك منتقل گرديده است.

From the beginning of the commissioning of the plant up to now, 2719 drums of radwaste have been generated and 2400 drums have been transferred to the Anarak radwaste repository facility.