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Application of probabilistic safety and reliability analysis for a system of fusion facility

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Fusion or thermonuclear power is a promising, literally endless source of energy. Development of fusion power is still in investigation and experimentation phases and a number of fusion facilities are under construction in Europe. Since fusion energy is innovative and fusion facilities contain unique and expensive equipment an issue of their reliability is very important from their efficiency perspective.

A Reliability, Availability, Maintainability, Inspectability (RAMI) Analysis is being performed or is going to be performed in the nearest future for such fusion facilities as ITER and DEMO in order to ensure reliable and efficient operation for experiments (e. g. in ITER) or for energy production purposes (e. g. in DEMO). On the other hand, rich experience of the Reliability and Probabilistic Safety Analysis (PSA) exists in nuclear industry for fission power plants and other nuclear installations.

In this article, the Wendelstein 7-X (W7-X) facility is mainly considered. It is a stellarator type fusion facility under construction in the Max-Planck-Institut für Plasmaphysik, Greifswald, Germany (IPP). In the frame of cooperation between the IPP and the Lithuanian Energy Institute (LEI) under the European Fusion Development Agreement a pilot project of a reliability analysis of the W7-X systems was performed with a purpose to adopt NPP PSA experience for fusion facility systems. During the project a reliability analysis of a divertor target cooling circuit, which is an important system for permanent and reliable operation of in-vessel components of the W7-X, was performed.

Key words: fusion facility, probabilistic safety analysis, reliability of systems

INTRODUCTION

In general, the purpose of the reliability and risk analysis is to provide support in making correct management decisions by evaluating the reliability and risk associated with a set of decision alternatives. The classical definition of the risk of failure is as follows:

$$R = p_f C, \qquad (1)$$

where R is the risk of failure, p_f is the probability of failure and C is the cost caused by the failure. The measure of the failure cost may be different (depending on various consequences). For production plants (including a nuclear power plant) it is usually not only the cost of failure (and accident in the worst case) and repair but also the amount of lost production (e. g. electricity) and lost profit.

Risk can be reduced from a level R to a lower level R' either by reducing the loss given failure or by reducing the probability of failure, or even by reducing both parts [1]. On the other hand, such risk reduction requires some investments and should be taken into account during the analysis. The values Δp_f and ΔC should be selected in such a way that the risk reduction ΔR is achieved at minimal costs.

The purpose of this article is to demonstrate how the reliability, as the main ingredient of safety (antonym of the risk), could be analysed for systems of fusion devices and show the practical application and results of such analysis by the possibility to reduce the risk and the cost related to the risk. In the *Fusion power* section, we will review the approaches of the reliability analysis used for fusion facilities; the section *Reliability analysis of fusion facilities* is devoted to a short survey of methods and techniques used for the analysis; in the section *Overview of methods for analysis* we will demonstrate the results of these methods practical application as a case study for the Divertor Target Cooling Circuit ACK10 of the Wendelstein 7-X experimental fusion facility.

Fusion power

Fusion power development in Europe

There are several fusion experimental installations in Europe operating or being constructed, or yet planned to be constructed. The aim of the European fusion research programme is developing nuclear fusion as an energy source, i. e. developing the knowledge in physics, technology and engineering required to design and build fusion power plants [4]. The roadmap towards a fusion reactor relies on three facilities: Joint European Torus (JET), its successor, an International Thermonuclear Experimental Reactor (ITER), see Fig. 1 [5], and a demonstration reactor called DEMO.

JET represents a pure scientific experiment. The ITER project aims to make long-awaited transition from experimental studies of plasma physics to full-scale fusion power plants. Construction of the ITER began in 2007, and the first plasma is expected to be produced in 2020. The ITER fusion reactor itself has been designed to produce 500 megawatts of output power for 50 megawatts of input power, or ten times the amount of energy put in. The machine is expected to demonstrate the principle of producing more energy from the fusion process than that used to initiate it, something that has not yet been achieved with previous fusion reactors. But it will be only a scientific demonstration; the ITER will not generate any electricity.

The next foreseen facility, DEMO, is expected to be the first fusion plant to reliably provide electricity to the grid.

Wendelstein 7-X

The Wendelstein 7-X (W7-X), see Fig. 2, is an optimized stellarator experiment which shall demonstrate the pos-

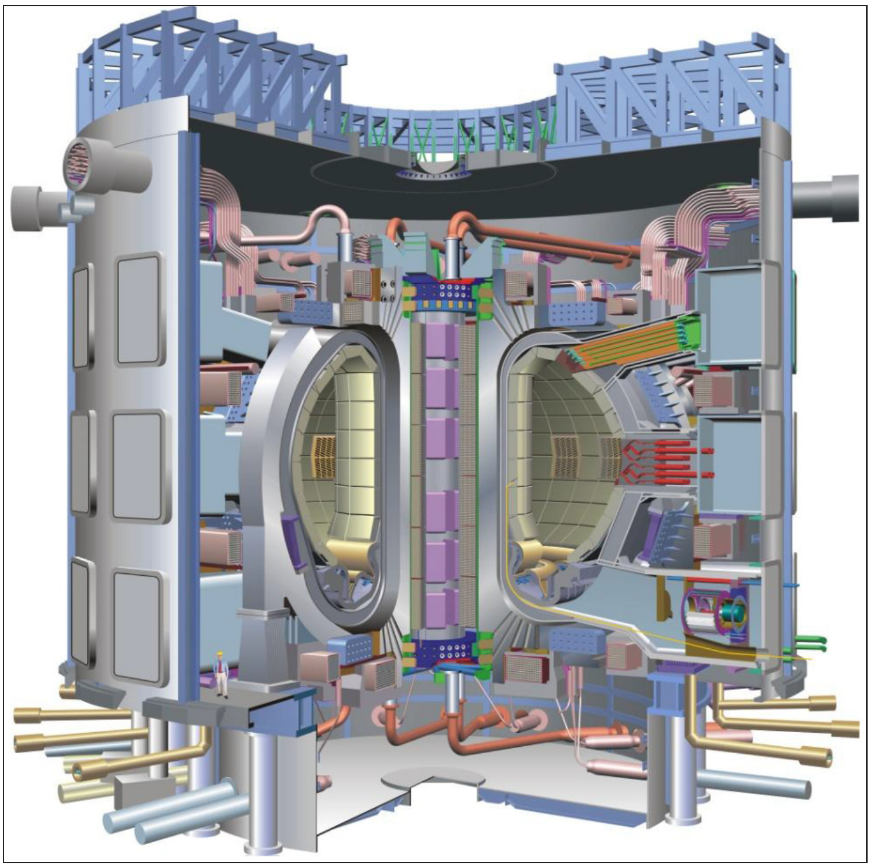

Fig. 1. ITER fusion device [5]

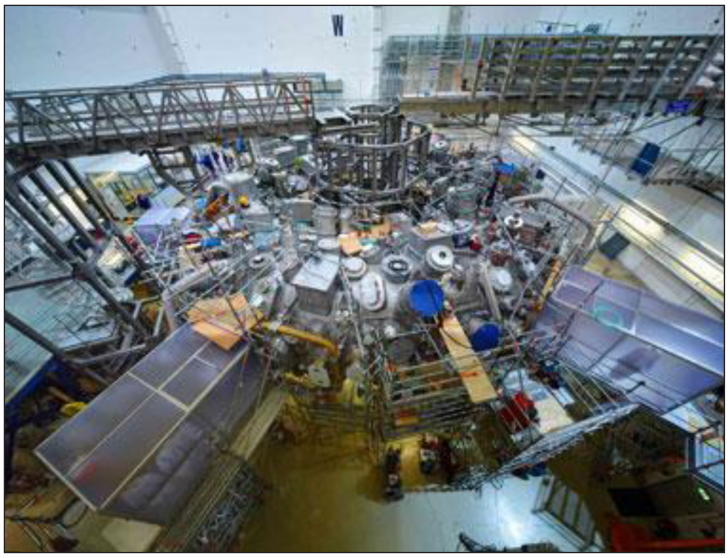

Fig. 2. Wendelstein 7-X fusion device [6]

sibility to use such a system as a nuclear fusion power plant [6]. The project is in the assembly and preparation for a commissioning phase at the Max-Planck-Institut für Plasmaphysik (IPP) in Greifswald, Germany. The Wendelstein 7-X will start operation step by step in 2014, the first plasma is expected in 2015.

The purpose of the Wendelstein 7-X is to evaluate the main components of a future fusion reactor built using stellarator technology, even if the Wendelstein 7-X itself is not an economical fusion power plant. The Wendelstein 7-X, when finished, will be the largest fusion facility created using the stellarator concept. It is planned to operate with up to 30 minutes of continuous plasma discharge, demonstrating an essential feature of a future power plant: continuous operation.

The Wendelstein 7-X is mainly a toroid, consisting of 50 non-planar and 20 planar superconducting magnetic coils, 3.5 m high, which induce a magnetic field that prevents the plasma from colliding with the reactor walls. The 50 non-planar coils are used for adjusting the magnetic field.

The main components are the magnetic coils, cryostat, plasma vessel, divertor and heating systems.

Reliability analysis of fusion facilities

Power plant availability is essential from the economical perspective as both fission and fusion power plants require very high initial investments. Returning of the investment and earning profit require the plant to generate the highest possible amount of electricity and this implies high availability requirement. High availability of experimental fusion facilities is required for the most efficient use of the facility for experiments.

A conceptual study of future commercial fusion power plants (FPPs) has been performed with a Power Plant Availability (PPA) study aimed at identifying the aspects affecting the availability and generating costs of FPPs [2, 3]. Among others, availability and reliability issues of FPPs were covered by the study. The study concludes that in order to be competitive, fusion plants starting from the first generation need to comply with the availability factor greater than 80%, similar to existing fission plants, with very few unplanned shutdowns. In order to guarantee continued safety of operation during fusion plant lifetime, in-service inspection and maintenance are needed and this aspect should be taken into consideration in the design of the systems [2, 4].

ITER RAMI approach

The availability objective for ITER is 60% inherent availability and 32% operational availability [3]. The inherent availability is the percentage of time during which the machine would be available if no delay due to the scheduled maintenance or supply was encountered. The operational availability reflects the inherent design including the effects of maintenance / upgrade delays taking into account the availability of maintenance personnel and spares and other non-design factors.

The ITER organization uses the RAMI (Reliability, Availability, Maintainability, Inspectability) approach to perform a technical risk assessment. The RAMI approach focuses on the operational functions required by the operation of the ITER rather than on physical components. It enables to define the requirements for the operational functions and provide the means to ensure that they could be met.

The RAMI process begins during the design phase of a system because corrective actions are still possible. The process is focused on the functions required to operate the ITER and their failure criticality. It is performed in 4 steps:

- Functional Analysis (FA);
- Failure Modes, Effects and Criticality Analysis (FMECA);

- Risk mitigation actions;
- RAMI requirements.

A functional analysis of the systems is performed with a functional breakdown (top-down description of the system as a hierarchy of functions) and an assessment of reliability and availability performance of the functions by using Reliability Block Diagrams (RBDs). The RBD approach uses the function blocks (FB) as a basis, but concentrates on the reliability-wise relationships between the function blocks. The input data, such as mean time between failures (MTBF) and mean time to repair (MTTR), are fed to the lowest level blocks.

A FMECA is performed in parallel to the RBDs to list the function failure modes and evaluate their risk level. A decision whether to accept or mitigate the failure mode is made based on the risk level. FB and RBD are input to FMECA.

Risk mitigation actions are initiated in order to reduce the risk level of the failure modes identified by the FMECA. After integrating RMA, new RBDs are prepared.

RAMI requirements are outputs of the ITER RAMI process. They are integrated in the system requirements:

- Availability and reliability targets for the system and main functions according to the project requirements.
- Required design changes that need to be integrated to improve the current design.
- Specific tests to be performed on the components or systems.
- Operation procedures and specific training to lower the risks when operating the machine.
- Maintenance requirements in terms of a list of spares, intervals of inspection and preventive maintenance, procedures and training.
- Proposals for standardization of common parts used in great numbers in the project, as ensuring inter-changeability of spares in the design of the systems shall then allow for shorter maintenance operation (replacement of consumables, repairs of failed components) and shall reduce the downtime of the systems and the severity ratings in the FMECA, reducing the risk level and allowing for more availability of the ITER for the experimental programme.

The process applied for the analysis of the plant systems defines failures of the functions, their criticality and provides risk mitigation actions. Up to 2010 RAMI was applied to 16 out of 21 main ITER systems. The analysis performed for the Tokamak Cooling Water System [5] identified initially 27 major risks, such as failure of the main pumps, leaks on the heat exchangers or associated valves leading to loss of cooling and possible damage for the plasma-facing components and failure of the coolant chemistry control leading to corrosion. For such major risks risk mitigation actions are considered which reduce either the likelihood (prevention) or the consequences (protection) of the failures. The analysis proves that after implementation of the identified actions the cooling system could be operated in higher reliability and availability at 97.7% as required by the project.

The RAMI analysis for the ITER fuel cycle system [8] identified several failure modes with high risks, majority of which were removed by implementing risk reducing means. However, some most critical risks remain, e. g. several critical components of the tritium plant, which are not easily replaced or repaired.

Up to date the ITER project is probably the one which achieved the biggest advance in systematic use of reliability and risk analysis methods for a fusion facility.

Approach used for W7-X

IPP has decided to use the RAMI approach for the W7-X. As the W7-X at that time was already in the manufacturing and assembly state, it was too late to make significant design changes. Therefore it was decided to perform a reliability analysis based on modelling of already existing systems and then provide recommendations for improvement of system reliability and availability. This approach is different from the one used for ITER where the overall ITER availability goal is "distributed" among the systems and is defined for the systems and components (top-down approach). For the W7-X, on the contrary, it was decided to use the "bottom-up" approach when the existing system availability is estimated and improved. Ideally, a full-scope analysis would enable to obtain the overall W7-X availability as a summary of all systems availabilities. Having such a complete model would enable seeing how improvements of each system design, operation, maintenance, inspections, etc. would improve both systems and overall W7-X availability.

In order to perform reliability / availability and risk analyses of the W7-X probabilistic safety assessment methods were used.

Overview of methods for analysis

Main methods for assessment

To estimate risk a Probabilistic Safety Assessment (PSA), which is typically used for nuclear power plants, can be applied for any hazardous systems, e. g. [9, 10]. PSA methodology integrates information about facility design, operating practices, operating histories, component reliabilities, humans' behaviour, thermal hydraulic facility response, accident phenomena and potential environmental and health effects. PSA is widely used for estimation of safety and reliability of energy generating complex systems.

A Fault Tree Analysis (FTA) together with an Event Tree Analysis (ETA) are two main tools in a system analysis. Both methods include a quantification part and visual representations of the Boolean logic for accident sequences [11]. FTA is an analytical technique, whereby an undesired state of the system is specified (usually a state that is critical from a safety or reliability standpoint), and the system is then analyzed in the context of its environment and operation to find all realistic ways in which the undesired event (top event) can occur. The fault tree is a graphic model of various parallel and sequential combinations of faults, caused by hardware failures, human errors, software errors, or any other pertinent events, that will result in the occurrence of the predefined undesired state. The FTA attempts to develop a deterministic description of the occurrence of an event, called the top event, in terms of the occurrence or non-occurrence of other (intermediate) events. Intermediate events are also described further until the lowest level of the detail, the basic events, is reached.

A fault tree analysis may be qualitative, quantitative, or both, depending on the objectives of the analysis. Possible results from the analysis could be the following:

- A listing of possible combinations of environmental factors, human errors (if included), normal operational events, and component failures that may result in a critical state of the system.
- The probability that the critical event will occur during a specified time interval.

As a result of the fault, tree initial qualitative analysis minimal cut sets (MCS) are generated. A cut set is a set of basic events which, if occurred, definitely lead to the top event. A minimal cut set is a cut set such that after removal of any basic events from it there is no more a cut set. When the fault trees are structured, the MCS generations and quantification for a quantitative analysis are made by the PSA software.

Importance and sensitivity

In order to better understand the influence of each component and each parameter on the total system reliability / unavailability and risk the importance and sensitivity analyses are performed. The importance measures are the following:

The Fussell-Vesely (FV) importance for a basic event is the ratio between the unavailability based only on all MCSs where the basic event i is included and the nominal top event unavailability is

$$I^{FV}{}_i = \frac{Q_{TOP}(MCS_{including}\, i)}{Q_{TOP}}, \tag{2}$$

where $I^{FV}{}_i$ is the FV importance; Q_{TOP} is the nominal top event unavailability; Q_{TOP} $(MCS_{including} i)$ is the unavailability based only on MCSs where the basic event i is included.

The risk decrease factor (RDF) is calculated as

$$I_i^D = \frac{Q_{TOP}}{Q_{TOP}(Q_i = 0)}, \tag{3}$$

where I_i^D is RDF; Q_{TOP} $(Q_i = 0)$ is the top event unavailability where the unavailability of the basic event i is set to zero (the basic event **does not** contribute to the top event unavailability).

The risk increase factor (RIF) is calculated as follows:

$$I_i^I = \frac{Q_{TOP}(Q_i = 1)}{Q_{TOP}}, \tag{4}$$

where I_i^I is RIF; Q_{TOP} $(Q_i = 1)$ is the top event unavailability where the unavailability of the basic event i is set to one (the basic event **does** contribute to the top event unavailability).

The fractional contribution (FC) is calculated as follows:

$$I_i^F = 1 - \frac{1}{I_i^D}. \tag{5}$$

The sensitivity S is calculated as a ratio between "sensitivity high" and "sensitivity low" indicators:

$$S = \frac{Q_{TOP,U}}{Q_{TOP,L}}, \tag{6}$$

where "sensitivity high" $Q_{TOP,\,U}$ is top event results where the unavailability of the basic event i is multiplied by a sensitivity factor (normally equal to 10); "sensitivity low" $Q_{TOP,\,L}$ is the top event results where the unavailability of the basic event i is divided by the sensitivity factor.

The importance calculations for parameters are calculated according to a similar procedure as for basic events. In some cases, importance measures cannot be defined, or would be meaningless. This is the case for time to the first test (TI) parameters, and for a risk increase factor for frequency (f) parameters.

- The parameter value is set to the "best theoretically possible", in all cases it is $X = 0$. This is made for all parameter types except for time to the first test (TF) parameters.
- A new top event result (unavailability or frequency, depending on the type of calculation in the MCS-analysis specification) is calculated. This new, lower, result is indicated with Q_{TOP} $(X = 0)$ in the following formula. The risk decrease factor can now be calculated as follows:

$$I_i^R = \frac{Q_{TOP}}{Q_{TOP}(X_i = 0)}. \tag{7}$$

The fractional contribution is the following:

$$I_i^F = 1 - \frac{1}{I_i^R}. \tag{8}$$

- The parameter value is set to the "worst theoretically possible". It is different depending on the type of the parameter, it is $q = 1$ for probability parameters and $X = \infty$ for all other parameters. This is not applicable for frequency parameters because an infinite frequency value would imply an infinite top event frequency.

- A new top event result (unavailability or frequency, depending on the type of calculation in the MCS-analysis specification) is calculated. This new, higher, result is indicated with Q_{TOP} ($q_i = 1$) below. The risk increase factor can now be calculated as follows.

  For probability parameters:

$$I_i^I = \frac{Q_{TOP}(q_i = 1)}{Q_{TOP}}. \quad (9)$$

For all other parameter types X (except frequency (f) and time to first test (TI) parameters for which no calculations are made):

$$I_i^I = \frac{Q_{TOP}(X_i = \infty)}{Q_{TOP}}. \quad (10)$$

Case study: Wendelstein 7-X divertor target cooling circuit

The divertor target cooling circuit is a part of the water cooling circuits for the W7-X. It provides cooling flow for the target modules during plasma operation and ensures water circulation during other operational modes. It also provides heating up of the divertor target modules up to 150 °C (the so-called baking mode) before starting operation campaign after outage for maintenance.

The cooling circuit consists of a primary part (ACK10 cooling circuit) and a secondary part (ECB10 water supply system). The circles are separated by two parallel heat exchangers. The secondary part cools the primary part during the experiment and holds its temperature constant.

The primary part includes a cooling circuit and a separate baking circuit with its own pump and provides water to 110 parallel target modules.

The pipes with diameters of 25–600, the valves and other components are stainless steel parts. Water for the primary part must be deionised. The total water content of the primary part is about 87 m³. The content of a lockable target module is about 25 litres.

A simplified flow diagram of the DTCC [12] is provided in the following Fig. 3.

Fault Tree Model

The top event of the fault tree is "ACK10 unavailable for experiment" (Fig. 4). This FT consists of five branches which model failures of five sections of the DTCC. The branches are connected by the OR-gate, which means that ANY of them may lead to the top event. Each of the five branches are further modelled by its own fault tree:

- Loss of modules cooling;
- Direct flow of water from the cold to the hot pipeline;
- No cooling water in supply pipeline;
- No water from pumps;
- No water from heat exchangers (HE) of ECB10 system.

Each fault tree models failure of different parts of the DTCC. The fault tree "No water from pumps AP002, AP003" is described below as an example.

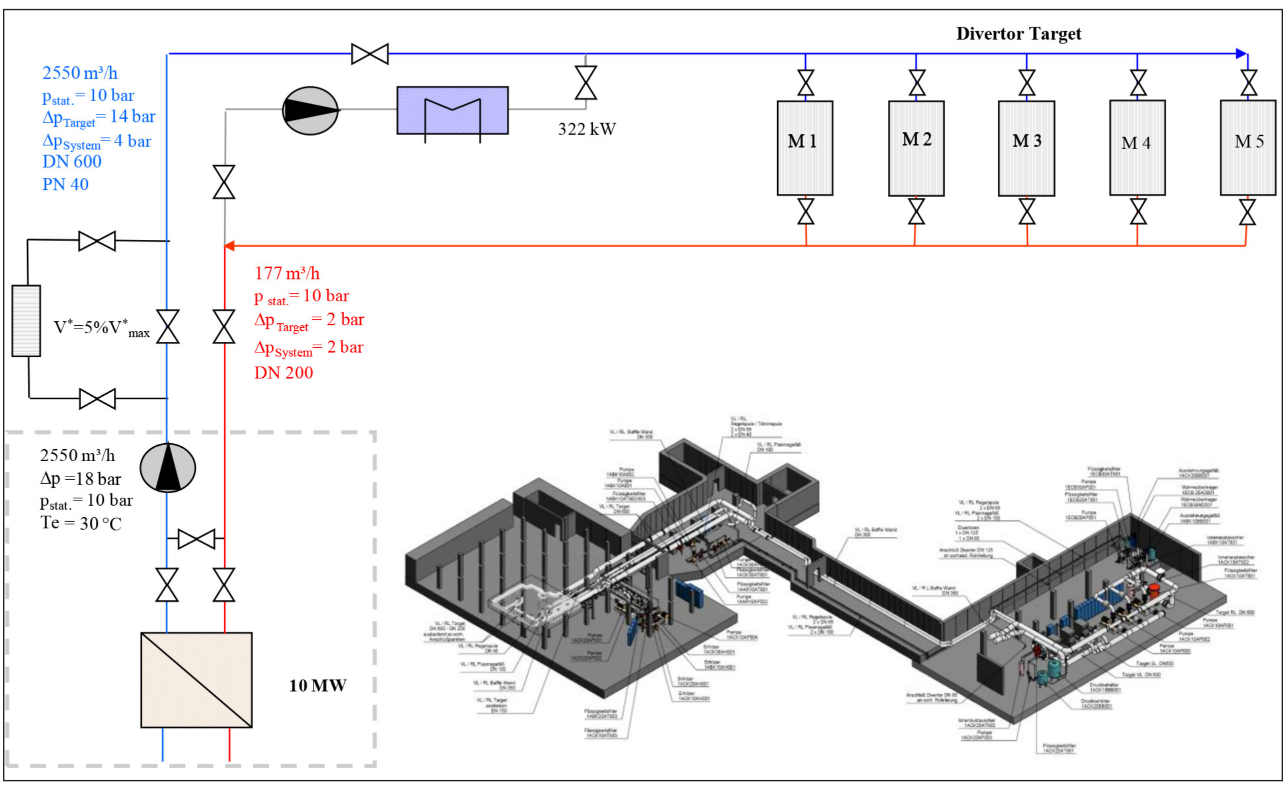

Fig. 3. Simplified flow diagram and 3D schema of W7-X DTCC [12]

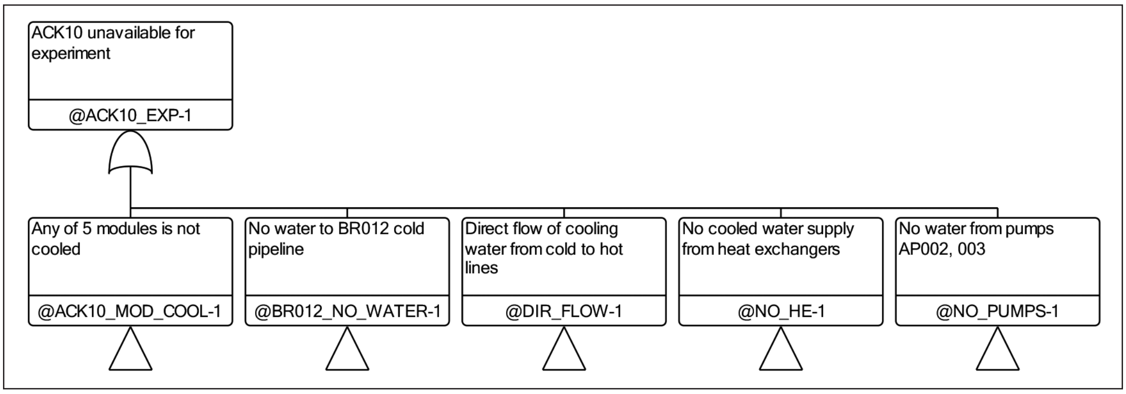

Fig. 4. Fault Tree "ACK10 unavailable for experiment"

Cooling water circulation in the ACK10 system [13] is ensured by pumps AP001, AP002, AP003 (Fig. 5).

Two pumps are required to provide sufficient cooling for Normal Load and Full Load operation modes. Only one pump AP003 is installed now and AP002 is planned to be installed. Installation of AP001 is under consideration and it is possible that this pump will not be installed and only two pumps will be in operation. Each pump line is equipped with the following:

- Manual gate valve KA505(504) is normally open and is closed only for pump maintenance;
- Pump AP003(002) may be in operation or in standby;
- Check valve KA508(507) is opened when the pump is running and closes due to the pressure difference when the pump stops;
- Pneumatic valve KA510(511) is opened when the pump is running and closes due to the automatic signal when the pump stops.

Most of the time the DTCC system will be in Part Load and Standby modes which correspond to pauses between W7-X experiments and non-working time, respectively. Cooling water flow rates for these modes are 425 m³/h and 177 m³/h, respectively. Such flow rate may be ensured by one pump which is assumed to be in operation all the time.

The second pump is started only when an additional flow up to 1 382 m³/h for Normal Load and 1 602 m³/h for Full Load is required. This determines the failure modes of the system. It is assumed that AP003 is the main running pump and AP002 is an auxiliary one.

Therefore the failure modes for AP003 are the following:

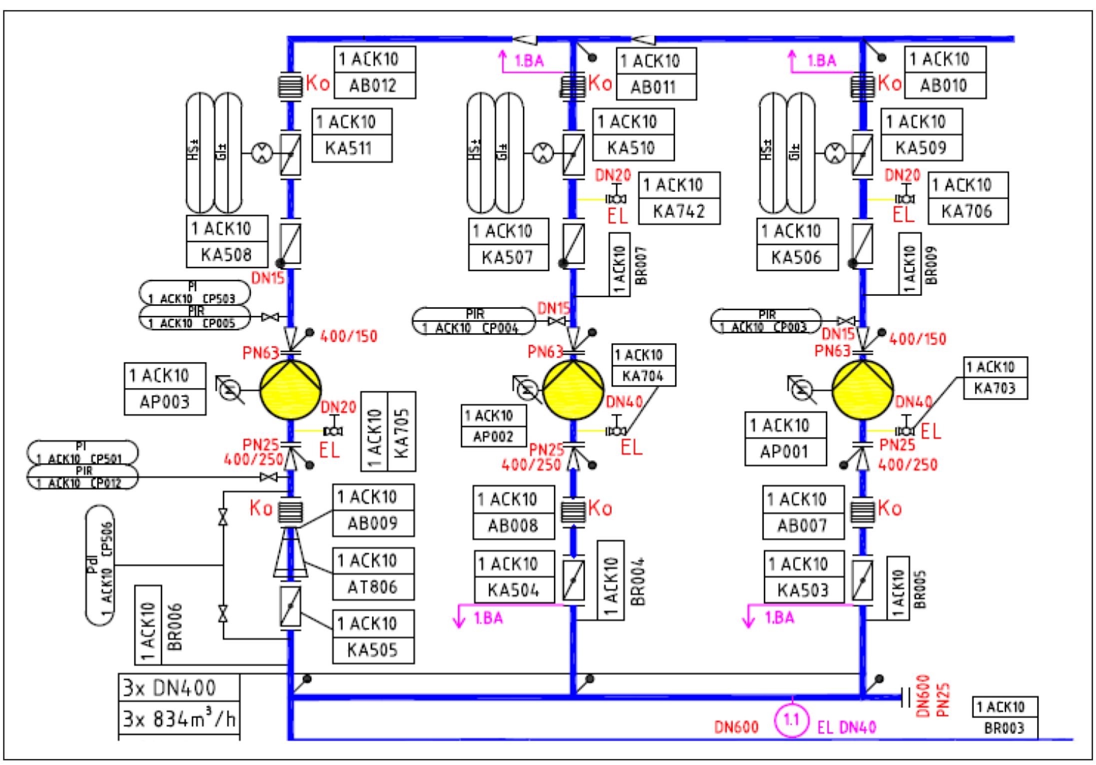

Fig. 5. Flow diagram of the cooling water pumps AP001, 002, 003 [13]

- Manual gate valve KA505 is erroneously closed or spuriously closes;
- Pump fails to run;
- Pneumatic valve KA511 is erroneously closed or spuriously closes.

The failure modes for AP002 are the following:

- Manual gate valve KA504 is erroneously closed or spuriously closes;
- Pump fails to start or fails to run;
- Check valve KA507 fails to open at pump startup;
- Pneumatic valve KA510 fails to open at pump startup or is erroneously closed or spuriously closes.

The fault tree is presented on Fig. 6.

Analysis results

A reliability analysis of the DTCC (ACK10) was performed using the FTA and RiskSpectrum PSA software. The developed FTA model quantification includes minimal cutsets generation and an uncertainty and sensitivity analysis. The calculation was performed for the time period of 6 526 hours, i. e. the total time of operation campaigns per one year. The MCS generations for the analysis are made by the PSA software.

The calculated total unavailability for the ACK10 operation period in a year is 0.188. This means that the system will be unavailable for 18.8% of the operation campaign. As a result of the ACK10 model analysis 56

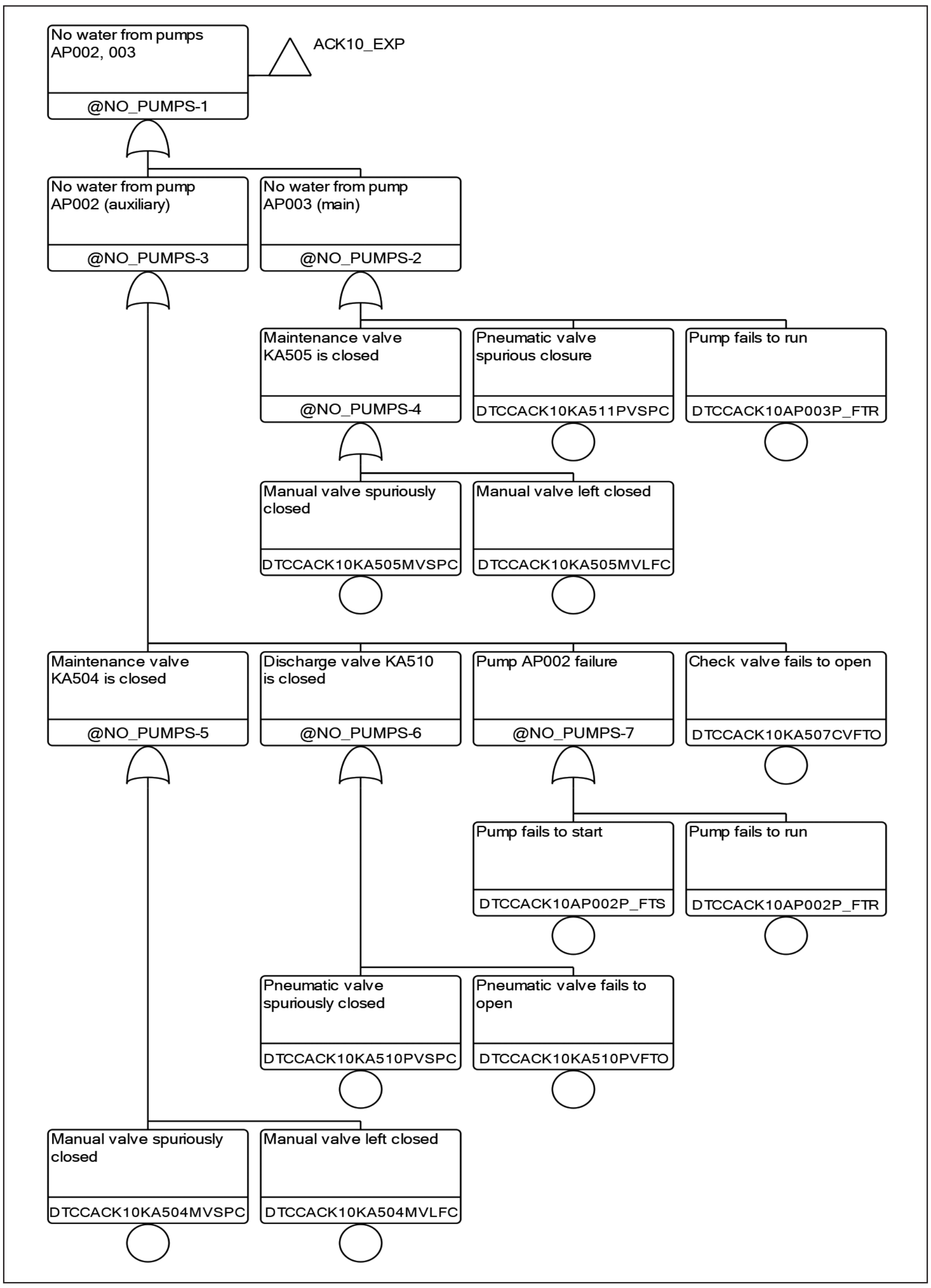

Fig. 6. Fault Tree "No water from pumps AP002, 003"

minimal cut sets were generated. The most important 7 MCS (which gives the highest unavailability) are presented in the following Table 1; the remaining MCS bring only 0.01% to the total unavailability.

The results show that due to low system redundancy the failure of a single component may lead to complete system unavailability. More than 50% influence on the system unavailability brings the failure of an auxiliary (secondary) cooling pump AP002 which has a high quantity of cyclic loads. About 35% bring the pneumatic valve KA510 located at the pressure line of the same pump. This valve is also subject to high cyclic loads.

In order to better understand the influence of each component and each parameter on the total unavailability the importance and sensitivity analyses were performed.

The results of the importance and sensitivity analyses for the most important 7 basic events are presented in the following Table 2.

It is obvious that the most important basic events are related to the minimal cut sets. The interesting outcome is that importance and sensitivity measures show how total unavailability would change if reliability of each component changes. For example, RDF shows that assuming "perfect" pumps with failure probability equal to 0 (1st basic event), this would decrease ACK10 unavailability 1.84 times and a "sensitivity low" indicator $Q_{TOP,\ L}$ shows that increasing pump's reliability 10 times, ACK10 unavailability would be 11% against current 18.8%.

The results of the importance and sensitivity analyses for parameters are presented in the following Table 3.

The final results show that the most important contributor to the system reliability is not equipment failure rates but one month time period for hardware repair or replacement which was assumed (parameter ONE_MONTH). The "Sensitivity low" indicator $Q_{TOP,\ L}$ shows that decreasing this time 10 times would result in ACK10 unavailability

Table 1. **ACK10 unavailability for each failure in ACK10**

No.	Unavailability	% total	Event name
1.	9.57E-02	51	Pump AP002 fails to start
2.	6.60E-02	35.1	Pneumatic valve KA510 fails to open
3.	2.11E-02	11.3	Pump AP003 fails to run
4.	6.44E-03	3.43	Heat exchanger AD002 fails
5.	6.44E-03	3.43	Heat exchanger AD001 fails
6.	3.52E-03	1.87	Check valve KA507 fails to open
7.	4.50E-04	0.24	Pump AP002 fails to run

Table 2. **Results of the basic events importance and sensitivity analysis**

No.	Normal value	FV	FC	RDF	RIF	S	$Q_{TOP,U}$	$Q_{TOP,L}$
1.	9.57E-02	5.10E-01	4.58E-01	1.84E+00	5.33E+00	8.71E+00	9.61E-01	1.10E-01
2.	6.60E-02	3.51E-01	3.06E-01	1.44E+00	5.33E+00	5.17E+00	7.04E-01	1.36E-01
3.	2.11E-02	1.13E-01	9.35E-02	1.10E+00	5.33E+00	2.01E+00	3.46E-01	1.72E-01
4.	6.44E-03	3.43E-02	2.80E-02	1.03E+00	5.33E+00	1.28E+00	2.35E-01	1.83E-01
5.	6.44E-03	3.43E-02	2.80E-02	1.03E+00	5.33E+00	1.28E+00	2.35E-01	1.83E-01
6.	3.52E-03	1.87E-02	1.53E-02	1.02E+00	5.33E+00	1.15E+00	2.14E-01	1.85E-01
7.	4.50E-04	2.40E-03	1.95E-03	1.00E+00	5.33E+00	1.02E+00	1.91E-01	1.87E-01

Table 3. **Results of the parameters importance and sensitivity analysis**

No.	ID	Type	Normal value	FC	RDF	RIF	S	$Q_{TOP,U}$	$Q_{TOP,L}$
1.	ONE MONTH	Tr	7.20E+02	9.96E-01	2.37E+02	5.33E+00	3.65E+01	8.02E-01	2.20E-02
2.	PUMP_STBY	R	1.47E-04	4.58E-01	1.84E+00	5.33E+00	5.07E+00	5.64E-01	1.11E-01
3.	PV_FTO	R	9.81E-05	3.06E-01	1.44E+00	5.33E+00	3.59E+00	4.90E-01	1.36E-01
4.	PUMP_FTR	R	3.00E-05	9.35E-02	1.10E+00	5.33E+00	1.85E+00	3.18E-01	1.72E-01
5.	HE_FAIL	R	9.00E-06	5.63E-02	1.06E+00	5.33E+00	1.54E+00	2.74E-01	1.78E-01
6.	CV_FTO	R	4.90E-06	1.53E-02	1.02E+00	5.33E+00	1.15E+00	2.13E-01	1.85E-01
7.	ONE_DAY	Tr	2.40E+01	3.44E-03	1.00E+00	5.33E+00	1.03E+00	1.94E-01	1.87E-01
8.	MV_SPC	R	9.19E-07	2.10E-03	1.00E+00	5.33E+00	1.02E+00	1.91E-01	1.87E-01
9.	PUMP_A_FTR	R	6.25E-07	1.95E-03	1.00E+00	5.33E+00	1.02E+00	1.91E-01	1.87E-01
10.	PV_SPC	R	9.19E-07	1.15E-03	1.00E+00	5.33E+00	1.01E+00	1.90E-01	1.88E-01
11.	PV_SPO	R	9.19E-07	1.91E-04	1.00E+00	5.33E+00	1.00E+00	1.88E-01	1.88E-01
12.	MV_FTO	Q	1.00E-04	6.60E-07	1.00E+00	1.01E+00	1.00E+00	1.88E-01	1.88E-01
13.	FILTER_FAIL	R	2.00E-06	6.22E-07	1.00E+00	1.00E+00	1.00E+00	1.88E-01	1.88E-01

only 2.2%. The next important parameter is pump standby failure rate (parameter PUMP_STBY) which is used for the pump AP002 and which improvement 10 times would change ACK10 unavailability from 18.8% to 11.1%. Other results for considered parameters can be interpreted in the same way.

CONCLUSIONS

A reliability and risk analysis of the Divertor Target Cooling Circuit ACK10 was performed applying PSA related methods. The analysis covered data collection, development of a fault tree model, failure modes and effects analysis, estimation of reliability parameters and unavailability calculations.

The most important results and conclusions are as follows:

1. Unavailability of the ACK10 is 18.8% of the operational campaign, i. e. about 1.5 months of 8-month operation in a year the system would be unavailable thus causing unavailability to use the W7-X for experiments.

2. The main impact on unavailability is an operational regime of the cooling pumps, where one pump is always running to provide cooling during all operational modes and the second one is started only to provide additional cooling for plasma experiments. This causes high cyclic load and corresponding high failure probability to the secondary pump (unavailability 95.7% which is almost certainly once per year) and its regulating valve (unavailability 66%, i. e. twice in three years). Unavailabilities of these components bring 51% and 35% to the total unavailability, respectively.

3. Another major reason for unavailability is long repair time which is assumed to be one month accounting for the time required to deliver and repair the equipment at the manufacturer's site or procure the spares required for the repair. Limited redundancy of the equipment does not enable to continue operation while the components are being repaired.

Comparison of the W7-X and ITER reliability and risk analysis shows that the W7-X analysis uses FTA and FMECA which are similar to the RBD-FMECA approach for the ITER. The W7-X has less possibilities to make design changes in comparison with ITER therefore it should concentrate on such risk prevention and mitigation measures which would require less intervention to already designed and installed systems, such as:

- Improvement of maintenance programme;
- Improvement of operating / maintenance procedures;
- Hardware or system configuration changes only for most safety important components.

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TIKIMYBINĖS SAUGOS IR PATIKIMUMO ANALIZĖS TAIKYMAS TERMOBRANDUOLINIO ĮRENGINIO SISTEMAI

Santrauka

Termobranduolinė sintezė yra perspektyvus ir sąlygine prasme begalinis energijos šaltinis. Vis dėlto pati termobranduolinė energetika tebėra tyrimų ir eksperimentavimo fazėje, Europoje konstruojama keletas termobranduolinių įrenginių. Kadangi termobranduolinė energetika yra inovatyvi, o termobranduoliniai įrenginiai susideda iš unikalios ir brangios įrangos, tai jos patikimumas yra labai svarbus ir dėl efektyvumo.

Patikimumo, pasirengimo, remonto ir inspekcijos analizė atliekama arba artimiausiu metu planuojama atlikti ITER ir DEMO įrenginiams siekiant užtikrinti patikimą ir efektyvų eksploatavimą bei eksperimentų atlikimą (pvz., ITER įrenginyje), pagaminti energijos daugiau nei suvartojama (pvz., DEMO įrenginyje). Iš kitos pusės, branduolinėje pramonėje yra sukaupta plati patikimumo analizės ir tikimybinės saugos analizės (PSA) patirtis, taikoma atominėms elektrinėms ir kitiems branduoliniams įrenginiams.

Šiame straipsnyje daugiausia nagrinėjamas Wendelstein 7-X (W7-X) įrenginys. Tai yra stelaratoriaus tipo termobranduolinis įrenginys, konstruojamas *Max-Planck-Institut für Plasmaphysik* (IPP) institute, Greifswalde (Vokietija). Siekiant pritaikyti atominėms elektrinėms skirtą PSA patirtį termobranduolinių įrenginių sistemoms pagal EFDA sutartį, bendradarbiaujant IPP ir Lietuvos energetikos institutui bei vykdant pilotinį projektą buvo atlikta W7-X sistemų patikimumo analizė. Projekto metu atliktas W7-X vidinių plazmos indo sistemų nepertraukiamam ir patikimam darbui svarbios sistemos, t. y. divertoriaus taikinio aušinimo kontūro sistemos, patikimumo analizė.

Raktažodžiai: termobranduolinis įrenginys, tikimybinė saugos analizė, sistemų patikimumas

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ПРИМЕНЕНИЕ ВЕРОЯТНОСТНОГО АНАЛИЗА БЕЗОПАСНОСТИ И НАДЕЖНОСТИ К СИСТЕМЕ ТЕРМОЯДЕРНОГО УСТРОЙСТВА

Резюме

Термоядерный синтез является перспективным, практически бесконечным источником энергии. Развитие термоядерной энергии находится все еще на стадии исследования и эксперимента, для этой цели в Европе строятся несколько экспериментальных термоядерных устройств. Так как энергия синтеза является инновационной областью, а экспериментальные устройства содержат уникальное и дорогое оборудование, вопрос их надежности очень важен с точки зрения их эффективности.

Анализ надежности, доступности, обслуживаемости и контролируемости выполняется или планируется в ближайшем будущем для таких устройств, как ITER и DEMO с целью обеспечения надежной и эффективной работы и с целью проведении экспериментов (например в ITER) или с целью выработки энергии (например в DEMO). В то же время, в атомной энергетике накоплен богатый опыт анализа надежности и вероятностного анализа безопасности (ВАБ) для атомных электростанций (АЭС) и других ядерных объектов.

В данной работе рассматривается, в основном, устройство Wendelstein 7-X (W7-X). Это термоядерное устройство, принадлежащее к типу стеллараторов, строительство которого ведется в Институте физики плазмы Макса Планка в г. Грайфсвальд, Германия (IPP). В рамках сотрудничества между IPP и Литовским энергетическим институтом по договору EFDA был выполнен пилотный проект по анализу надежности систем W7-X с целью адаптации опыта ВАБ АЭС к системам термоядерного устройства. В ходе проекта был выполнен анализ надежности контура охлаждения мишени дивертора, являющегося важной системой, обеспечивающей постоянную и надежную работу компонентов, находящихся внутри плазменного сосуда W7-X.

Ключевые слова: термоядерное устройство, вероятностный анализ безопасности, надежность систем