Kungliga Tekniska Högskolan

SH2705 Compact Reactor Simulator Exercises in Reactor Kinetics and Dynamics

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THE 1999 OSKARSHAMN-2

STABILITY EVENT

DRAFT

Group 3

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Abstract

This report constitutes a project work that corresponds to the theoretical part of the course SH2705 Compact Reactor Simulator Exercises in Reactor Kinetics and Dynamics. The report provides a summary of the Oskarshamn 1999 stability event along with an in-depth discussion of relevant nuclear safety concepts applied to the event. The report is divided into four sections. The first section provides the reader with a brief introduction to the event. In the second section, we describe in more detail the operational and accident conditions of the event. In the third section, we discuss the event on the basis of some relevant nuclear safety concepts, such as defense in depth, safety assessment and analysis, and acceptance criteria. In the last section, we discuss various types of safety analysis that could be applicable during the incident, e.g. DSA, PSA, BE, BEPU, Conservative, and the corresponding acceptance criteria.

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1 Introduction

The Oskarshamn-2 stability event was a nuclear incident that occurred on February 25, 1999, at the Oskarshamn Nuclear Power Plant in Sweden. The incident was initiated by a short (150 ms) power interruption when operating a breaker at the switchyard which led to severe reactor instabilities and subsequent shutdown.

The progression of the event was caused by a combination of factors, including malfunctions in the reactor control system, issues with filtered APRM signals, and other factors. This led to unstable reactor behavior, resulting in oscillations in reactor power output. These oscillations eventually culminated in divergent power instability, which led to an automatic shutdown of the reactor [1].

The incident has been studied in detail by nuclear safety organizations and researchers around the world, including the OECD/NEA [1]. The incident led to improvements in nuclear safety practices and the establishment of benchmarks for reactor stability analyses. The Oskarshamn-2 event is characterized as a significant nuclear power incident that has contributed to improving safety standards at nuclear power plants around the world.

This report constitutes a project work that corresponds to the theoretical part of the course SH2705 Compact Reactor Simulator Exercises in Reactor Kinetics and Dynamics. In the coming sections, we will provide a more detailed description of the Oskarshamn-2 1999 stability event and discuss it on the basis of relevant nuclear safety concepts.

1.1 Safety Objectives

1.1.1 Fundamental Safety Objectives

IAEA has formulated three fundamental safety objectives in nuclear power operation that applies for all facilities and activities, as well as for all lifetime stages of a facility or the radiation source. They are given as [2] [3]:

- Protecting people, society and the environment from harm by maintaining an effective defense against radiological accidents.
- To limit the harmful effects of ionizing radiation, as far as possible, during normal operation within the power plant, as a result of emissions of radioactivity from the power plant and from formed waste (ALARA).
- To prevent radiological accidents and mitigate the consequences of radiation damage in the case of accidents by taking all reasonable practical steps possible.

1.1.2 Safety Principles

In addition to IAEAs fundamental safety objectives, ten safety principles have been formulated. The safety principles act as a foundation in order to achieve the fundamental safety objectives. The ten safety principles are [3]:

- *Principle 1*: Responsibility for safety. The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks.
- *Principle 2*: Role of government. An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained.
- *Principle 3*: Leadership and management for safety. Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks.
- *Principle 4*: Justification of facilities and activities. Facilities and activities that give rise to radiation risks must yield an overall benefit.
- *Principle 5*: Optimization of protection. Protection must be optimized to provide the highest level of safety that can reasonably be achieved.
- *Principle 6*: Limitation of risks to individuals. Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm.
- *Principle 7*: Protection of present and future generations. People and the environment, present and future, must be protected against radiation risks.
- *Principle 8*: Prevention of accidents. All practical efforts must be made to prevent and mitigate nuclear or radiation accidents.
- *Principle 9*: Emergency preparedness and response. Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents.
- *Principle 10*: Protective actions to reduce existing or unregulated radiation risks must be justified and optimized.

2 The Oskarshamn-2 Stability Event

2.1 Basic features of Oskarshamn-2

The ABB ATOM-built Oskarshamn Unit 2 boiling water reactor is located in Oskarshamn, Sweden. In 1975, it was put into operation at 1,700 MW thermal power. Since then, it has been upgraded to 1,802 MW, or 106 % of its initial rating (all percentage power values relate to the initial rating) [4].

The reactor vessel is 5.2 m in diameter with an internal height of 20 m. The Oskarshamn-2 reactor has no jet pumps which implies that four recirculation loops receive the entire core flow. In addition, there are four feedwater pipelines and four steam lines in total [5]. The reactor vessel, along with the most important components, is displayed in Fig. (2).

2.2 Operational Conditions

At full reactor power (106 %) the nominal circulation flow of Unit 2 varies from 5,300 kg/s to 7,700 kg/s, according to Fig (1). The permissible reactor power decrease with lower flow rates. The lowest permitted flow is 2,500 kg/s. Due to the risk of pump cavitation, the maximum permitted recirculation flow is further constrained at low reactor power. Fig (1) shows the operating region for Oskarshamn 2 in terms of a power-flow map. It includes different trip lines that activate an action from the automatically controlled safety systems. The SS lines indicate scram or partial scram while the E lines indicate forced flow reduction. The SS9 and the SS10 conditions activate a reactor scram, respectively, when the APRM (Average Power Range Monitor) equals or exceeds 132 % of nominal power (1,700 MW), or when the APRM the allowed power in the power-flow dependent region ($\leq$ 3,800 kg/s). An important difference is that SS9 (at the time in Oskarshamn-2) was activated by the unfiltered APRM signal, while SS10 and SS29 use an unfiltered APRM signal. The E25 condition (filtered) initiates a reduction of the recirculation pump speed [4][6].

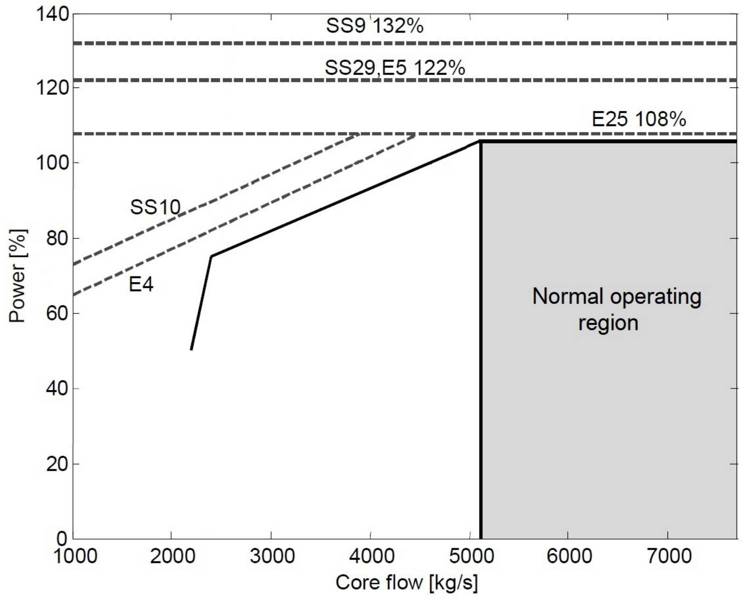

Figure 1: Power-flow map for Oskarshamn 2 [4].

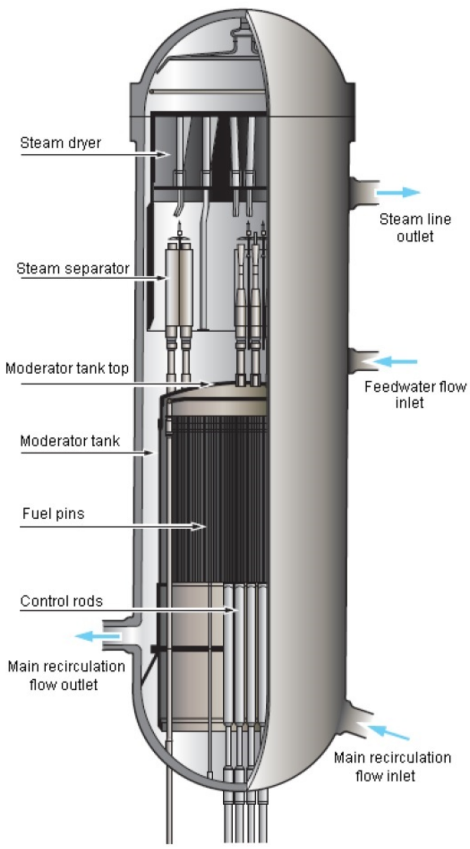

Figure 2: Oskarshamn-2 Reactor Pressure Vessel [5].

2.3 Accident Conditions

This section treats the accident conditions during the Oskarshamn-2 stability event. It follows the event description by Kozlowski et al. [1] and Galassi et al. [4].

The event at Oskarshamn Unit 2 occurred on February 25, 1999, during maintenance work on the switchyard near the power plant. The initiating event was caused by an unexpected 150 ms interruption in the power supply to a bus bar at the instant when the electric supply was restored. As a result, a load rejection signal was transferred to the turbine, causing a turbine trip, but due to a failure in the relay circuit, the reactor of Unit 2 received no load rejection signal. Thus, while the turbine underwent a turbine trip, the reactor continued in its normal operational state without the expected automatic insertion of control rods or any adjustment of the recirculation pump speed.

The turbine trip resulted in a power output decrease from 625 MWe to 585 MWe in the generator. In order to allow the excess steam (from the reactor at full power) into the main condenser, the steam line bypass valves opened. This measure, in turn, led to the deactivation of the feedwater pre-heater system, resulting in a fast feedwater temperature drop of 75 °C. The cold feedwater caused a positive reactivity feedback in the reactor vessel, increasing the reactor power level.

When the reactor power reached 2 % above the nominal power (i.e. 108 %), a pump controller, which controls the recirculation pumps rotation speed, automatically reduced the main recirculation flow in order to reduce the reactor power. The controller reduced the pump speed at a rate of 640 rpm/min until the power level dropped below 108 % (E25) [4]. However, this did not stop the cold feedwater to continue entering the vessel, which again raised the power level above 108 % and activated the pump controller. This loop was repeated three times.

In order to terminate this process, the operators partially scrammed the reactor by inserting 7 control rods as well as reducing flow to the minimum. This measure was taken approximately two minutes after the initiation of the event. The partial scram put the reactor power at 65 % and the flow to 3200 kg/s. Since the cold feedwater flow continued, the reactor power increased once more, placing the reactor in the unstable region of the power-flow map. During 20 seconds, the reactor power underwent diverging oscillations with successively increasing amplitudes.

3 minutes and 6 seconds after the initial load rejection event, the reactor automatically scrammed when the power exceeded 132 % at 2500 kg/s recirculation flow (SS9) [4]. The scram proceeded according to the design, placing the reactor into a hot shutdown state. Fig. (3) summarizes the key steps during the transient, while Figs. (4) and (5) displays recorded parameter data.

2.3.1 Why and where did the defense-in-depth breach?

The purpose of the defense-in-depth concept is to use multifunctional barriers to prevent single failures and avoid the release of radioactive substances. This coincides with using large design margins, high-quality components, and operation within design limits. Defense in depth works through five levels of defense, which can be summarized as: 1) Prevention of abnormal operation and failures; 2) Control of abnormal operations and detection of failures; 3) Control of accidents within the design basis; 4) Control of severe plant conditions, 5) Mitigation of radiological consequences [7].

The first level of defense regards sound (conservative) design, a high degree of freedom from faults and errors, high tolerance for malfunctions, and redundancy of instrumentation and control. The purpose of the first level of defense is to prevent incidents and equipment failures from occurring in the first place. The second level of defense regards the detection and control of failures or abnormal operations. This level of defense provides protection from further escalation, in the case of an incident, through redundant sources of electricity, sensitive detection systems, as well as systems for automatic shutdown [2][7].

In the case of the Oskarshamn-2 stability event, it is clear that the first level of defense breached through a malfunctioning relay circuit at the instant of the 150 ms power interruption. As a result, different signals were transmitted to the reactor and to the turbine. The turbine received a load rejection signal, while the reactor received no signal of the anomaly. Consequently, the reactor continued normal operation, while the turbine underwent a turbine trip. The immediate cause of the incident was the failure in the relay circuit, but the incident was also made possible by a disadvantageous control logic that allowed two independent signals to be transmitted to the reactor and the turbine.

In addition to the above comments, it can be argued that the defense in depth, to some degree, also failed at the second level. The aim of this level of defense is fast detection and control of the abnormality. Although the reactor underwent automatic shut down within 3 minutes and 6 seconds after the load rejection event, this was somewhat delayed by the use of filtered APRM signals, particularly at scram line SS10. It can be noted that after the Oskarshamn-2 event, the filtered APRM signal monitoring was replaced with an unfiltered equivalent. In addition to the static scram lines, a monitoring logic that can identify heavy oscillations in reactor power would be in place.

2.3.2 What safety objectives should be considered for the event?

IAEA has proposed a hierarchical approach to safety goals, divided for operational states and accident conditions, from low-level goals, via intermediate and upper-level goals, to top-level goals [6]. These goals run in parallel with the safety objectives given in Section 1.1, although, with a higher degree of detail. The low-level safety goals are technology-specific and are limited to the facility. The intermediate, upper-, and top-level safety goals are technology neutral and are related to the site and society. The safety goals can furthermore be characterized as either qualitative or quantitative (e.g. regulations regarding core damage frequency), where the latter can be deterministic or probabilistic. The overall objective of the low-level safety goal is to "Provide specific safety provisions for each facility and installation at the site to ensure adequate protection." Concretely this means that each nuclear facility at contributes to meeting the higher level safety goals by fulfilling a large number of specific quantitative/deterministic safety goals [6].

Based on this scheme, and the safety principles provided in Section 1.1.2, we assess that the low-level goals are the most relevant for the Oskarshamn-2 event along with safety principles 1, 3, and in particular, 8 and 9.

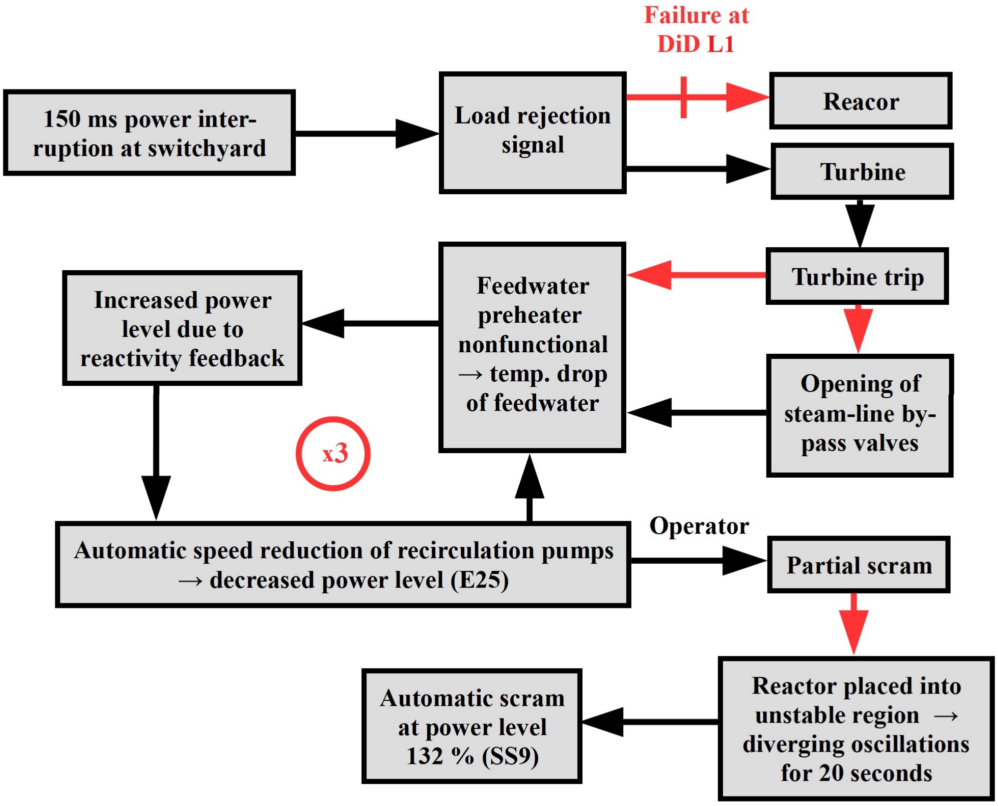

Figure 3: Progression of the Oskarshamn-2 stability event. Red symbols indicate actions that lead to further escalation of the event.

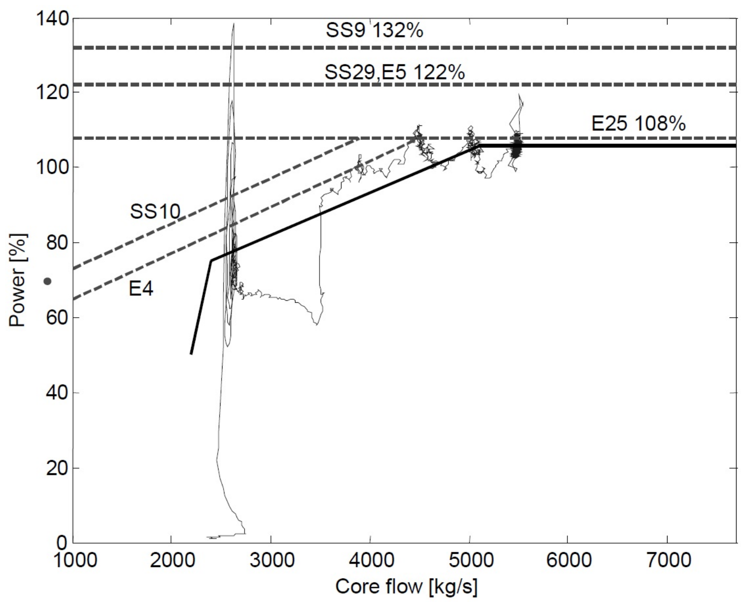

Figure 4: Power-flow map for Oskarshamn 2 showing the operational states during the transient [4].

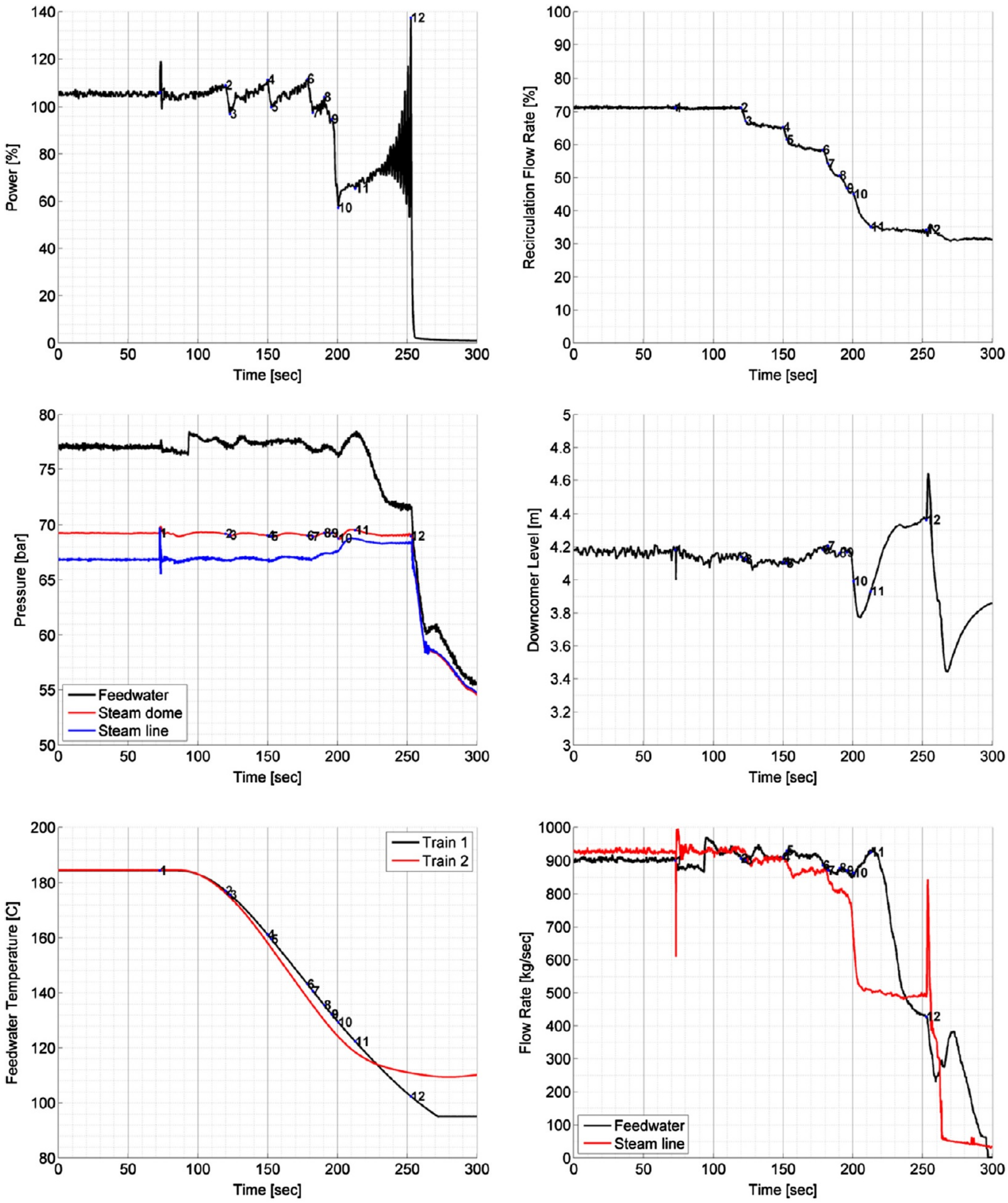

Figure 5: Oskarshamn-2 February 25, 1999 feedwater transient [1].

3 Discussion of Safety Concepts

In this section, we discuss different concepts of nuclear power safety treated in the course and explain how they are relevant to the event.

3.1 Defence in Depth

Defence in depth refers to the multiple layers of protection that are designed to prevent a nuclear accident or mitigate its consequences. This approach includes redundant safety systems, the use of physical barriers, and the implementation of procedures and guidelines to promote safety. The IAEA safety standards emphasize the importance of defence in depth as a means of ensuring nuclear safety.

Section not done. Read more in Section 2.2.1.

3.2 Safety Assessments

Section not done.

3.3 Safety Analysis

Section not done.

3.4 Deterministic Safety Analysis

Deterministic safety analysis is a tool used to ensure safety in nuclear power plants. It involves analyzing various scenarios and events that may occur during plant operation to determine potential risks and identify safety measures for mitigating them. Some applications of deterministic safety analysis include designing and licensing nuclear power plants, updating safety analyses during periodic safety reviews, developing emergency operating procedures, and supporting probabilistic safety assessments. It is also used to verify the performance of plant systems and assess the severity of potential consequences in the event of accidents or failures.

Section not done.

3.4.1 Conservative Analysis

Section not done.

3.4.2 Best Estimate Analysis

Best estimate analysis is a method used in deterministic safety analysis for nuclear power plants, which involves the use of computer codes with realistic input data along with the evaluation of uncertainties in the calculation results. This method allows for a more precise specification of safety margins and provides greater operational flexibility.

Section not done.

3.5 Probabilistic Safety Analysis

Section not done.

3.6 Acceptance Criteria

Acceptance criteria are a set of specific requirements that must be met in order to ensure an adequate level of defense in depth is maintained and unacceptable radiological releases are prevented during nuclear power plant operations. These criteria include measures to prevent the loss of protective function, such as ensuring redundancy and high functional reliability in the protection system, and provisions for diversification to reduce the possibility of systematic failures. Additionally, more stringent criteria may be applied for events with a higher frequency of occurrence to ensure safety margins are maintained. Overall, acceptance criteria are important in ensuring proper safety measures and procedures are in place during nuclear power plant operations.

Section not complete.

3.7 Computer Codes

Computer codes are complex software programs that are widely used in the safety assessment of nuclear power plants. These codes can be categorized into six groups including reactor physics codes, fuel behavior codes, thermal-hydraulic codes, containment analysis codes, atmospheric dispersion and dose codes, and structural codes. These codes have traditionally been developed independently but have, in recent years, been coupled together to reduce uncertainties or errors associated with interface data transfer and improve the accuracy of calculation. Coupling of computer codes is an efficient method of addressing the multidisciplinary nature of reactor accidents with complex interfaces between disciplines. The coupling of advanced, best estimate computer codes is an efficient method of addressing the multidisciplinary nature of reactor accidents with complex interfaces between disciplines. The use of coupled codes has become important for the licensing of new nuclear power plants, safety upgrading programs, periodic safety reviews, and the justification for lifetime extensions.

3.8 Verification and validation of Computer Codes

Section not done.

4 Application of Safety Analysis

Consider that you are in charge of the safety assessments of the unit, discuss types of safety analysis that you can do, DSA, PSA, BE, BEPU, Conservative and which acceptance criteria should be used for it.

The Oskarshamn-2 event had some features which were uncertain. Firstly, This includes a faulty circuit that prevented the signal from reaching the reactor, causing no automatic control rod insertions or reduction of re-circulation pump speed to occur. Secondly, The feed-water heater had also stopped working, causing a rapid decrease in the feed-water temperature, and finally, The power oscillations with a growing amplitude of 0.5 Hz.

These uncertain events point in the direction of using the Probabilistic Safety Assessment(PSA) for the event. A Probabilistic Safety Assessment is a comprehensive and systematic analysis that assesses the likelihood of various accident scenarios occurring in a nuclear power plant, as well as the potential consequences of those accidents. PSA considers a wide range of factors, such as the reliability of the plant's safety systems, the potential for human error, and the likelihood of external events.

In this particular case, the uncertain features of the event, such as the faulty circuit, the feed water heater failure, and the power oscillations, could all be considered in the PSA to assess the likelihood of a similar event occurring in the future.PSA would evaluate the probability of various accident scenarios, as well as the potential consequences.

Based on the results of the PSA, recommendations could be made to improve the safety of the plant and prevent similar events from occurring in the future. These recommendations might include modifications to the design of the plant, changes to operating procedures, or improvements to safety systems and equipment.

The conclusions drawn from the study on the butterfly effect in the dynamics of the Oskarshamn-2 Instability event look into the possibility of minor changes that could have led to drastic changes in the core dynamics. The study found that minor changes introduced in the Oskarshamn-2 core had a significant effect on the dynamic behavior, but not on the steady-state behavior. The study also concluded that the reason behind the drastic reduction in the power oscillation amplitudes in the event, following a minor core design change, was mainly due to the shift in the stability boundary. Finally, the study recommends a high-resolution bifurcation investigation to quantify the contribution of each fuel assembly to the stability response of the core. This study points in the direction of Best Estimate plus uncertainty analysis.

The acceptance criteria should include parameters such as departure from nuclear boiling ratio, maximum fuel pellet temperature, maximum pressure and temperature of the primary and secondary pressure boundary, and maximum cladding oxidation. To ensure safety requirements are met, the uncertainty method should be rigorously applied to all important parameters, with uncertainty bands produced for those closely related to the acceptance criteria [8]

Section not complete.

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