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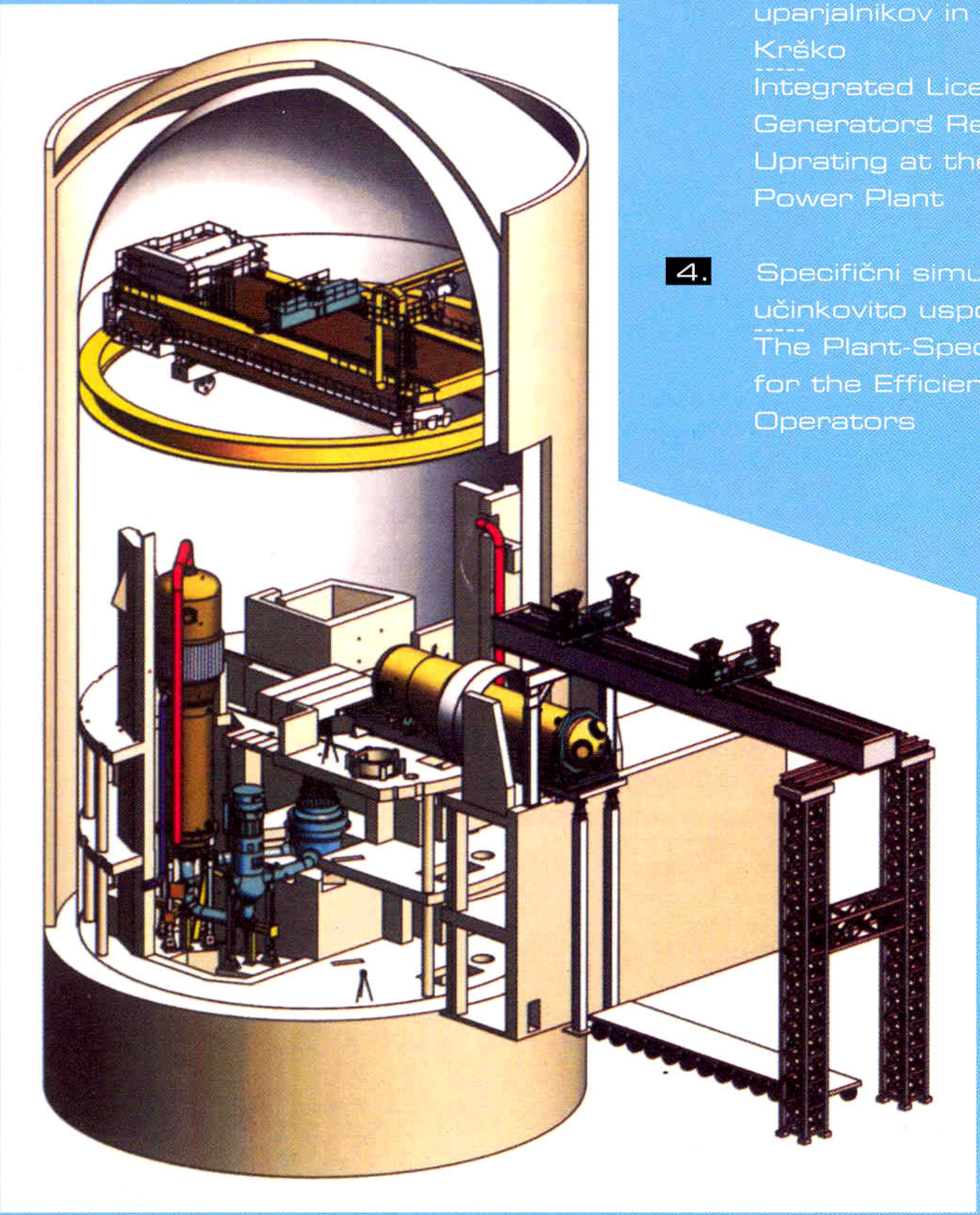

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Uvodnik Editorial

Uvodnik
Editorial

Tokrat nam je uspelo
We Did it This Time

Jedrske elektrarne zagotovo sodijo med sodobne globalne tehnologije. Globalne so predvsem zato, ker ne priznavajo tradicionalnih meja med narodi in med tehničnimi panogami, pa čeprav po prepričanju največjega poklicnega združenja strojnih inženirjev na svetu - The American Society of Mechanical Engineers – sodijo predvsem v domeno strojnega inženirstva[1] .

Osnovna značilnost vsake globalne tehnologije je vrhunska strokovnost in interdisciplinarnost ekip, ki jo ustvarjajo in upravljajo. Za te tehnologije praviloma velja, da jih preprosto ni mogoče varno in ekonomsko upravičeno izkoriščati samo na domačem trgu. Seveda pri tem ne gre prezreti, da je globalno mogoče delovati šele takrat, ko so lokalne razmere zelo dobro urejene. Še posebej je to pomembno pri tehnologijah, pri katerih želimo in moramo zagotoviti kar največjo varnost. In ker se vsaka tehnologija začne in konča pri inženirkah in inženirjih, je sklep jasen: naše uspešno sodelovanje pri snovanju, ustvarjanju in varnem upravljanju z globalnimi tehnologijami lahko zagotovijo le posebno usposobljeni domači strokovnjaki.

V Sloveniji se v skoraj desetletju samostojnosti praviloma nismo srečevali z večjimi projekti, ki bi bili povezani z globalnimi tehnologijami. To je med drugim tudi pričakovana posledica velikega krčenja domače strojne industrije in bistveno zmanjšanega splošnega zanimanja za študij tehnike, ki ga ne izkazujejo le študentje, ampak tudi država. Slovenska država kljub energetsko izredno pomembnemu jedrskemu objektu in velikih potrebah po strokovnjakih še danes nima za to usmerjenega dodiplomskega študijskega programa, za podiplomski interdisciplinarni študij jedrske tehnike pa npr. namenja le borih nekaj sto tisoč tolarjev na leto. Tako le NE Krško redno in sistematično usposablja svoje strokovno osebje v svojih izobraževalnih centrih in na simulatorju.

Vse to seveda ne pomeni, da projektov, povezanih z globalnimi tehnologijami, sploh ni bilo. Največji ali pa vsaj najodmevnejši izmed njih je zagotovo projekt posodobitve JE Krško, ki je začel teči v letu 1992 in se, po nekaj sto človekovih letih

[1] Strojni inženirji so ljudje, ki načrtujejo, razvijajo in izdelujejo stroje, ki proizvajajo, prenašajo ali uporabljajo energijo (David L. Belden)

Nuclear power plants (NPPs) belong to the modern global technology. The term global must be used because they cross the traditional borders between nations and between technical disciplines. Despite this nuclear power engineering is, according to the world's largest professional society of mechanical engineers – the American Society of Mechanical Engineers[1] - a part of the mechanical engineering domain.

The main feature of any global technology is a highly skilled and interdisciplinary workforce, which creates and uses this technology. Global technologies can in principle never be exploited in a safe and economically efficient manner in local markets. Additionally, any successful global action requires good solutions relating to local circumstances. This is especially important when working with technologies where a high level of safety is both desired and required. The conclusion is clear, since each technology starts and ends with engineers, our successful participation in design, creation and safe utilization of global technologies is only possible with highly skilled domestic experts.

Projects connected with global technologies were only rarely seen in independent Slovenia during the last decade. This was to be expected because of major reductions in the domestic mechanical engineering industry and a significantly reduced interest in university technical programs: demonstrated by both students and government. The Republic of Slovenia is not running any undergraduate program in nuclear engineering, although the importance of nuclear power to the domestic energy supply and the large demand for experts has been noted. A graduate program in nuclear engineering exists, but is poorly funded by the government with only a few hundred thousand Tolars per year. Only the Krško Nuclear Power Plant performs regular and systematic training of their own experts in their own training centers and simulator.

Nevertheless, projects connected with global technologies have taken place. The largest, or at least the best known to the public, is the modernization project of the Krško NPP, which started in 1992 and is, after a few hundred man-years

[1] Mechanical engineers are women and men, who design, develop and manufacture machines that produce, transmit or use power. (David L. Belden)

opravljenega dela, pravkar uspešno končuje z zamenjavo uparjalnikov in hkratnim 6,3 odstotnim povečanjem moči elektrarne.

Cilj tega projekta je predvsem zagotovitev varnega in stabilnega obratovanja JE Krško tudi po letu 2000. Seveda upošteva tudi projekcije razmer, ki jih pričakujemo na prostem trgu električne energije. Projekt posodobitve je načrtovala in izpeljala skupina vrhunskih strokovnjakov JE Krško: pod njihovo taktirko so sodelovali največji svetovni dobavitelji opreme in storitev na jedrskem področju (Westinghouse Electric Europe iz Belgije, Siemens-KWU iz Nemčije, Framatome iz Francije in CAE iz Kanade) z velikim številom partnerjev iz Evrope, Japonske in ZDA. Strokovnjaki JE Krško so si ob tem zagotovili tudi široko podporo domače jedrske stroke, ki je sodelovala predvsem s svetovanjem JE Krško in pri neodvisnih varnostnih ocenah opravljenega dela v upravnih postopkih za pridobivanje obratovalnega dovoljenja.

V zadnjih dveh zvezkih Strojniškega vestnika (2000/3 in 2000/4) so v osmih člankih predstavljeni razlogi za zamenjavo starih uparjalnikov in najpomembnejši poudarki projekta posodobitve JEK.

Posebna zahvala za prispevke gre avtorjem iz JE Krško, ki so dobršen del svojega izjemno skopo odmerjenega časa pri vodenju zahtevnega projekta posodobitve JEK prijazno namenili tudi obveščanju.

Projekt posodobitve jedrske elektrarne v Krškem je praktično končan, vrhunske domače strokovnjake s področja jedrske tehnologije pa bomo potrebovali še najmanj naslednjih 30 let. Naslednje desetletje bomo zato morali v prvi vrsti posvetiti tudi javno financiranemu izobraževanju jedrskih inženirjev, da ohranimo in še bolj razširimo domače znanje na tem področju.

Doc.dr. Leon Cizelj

of work, nearing its completion with the replacement of the steam generators and a 6.3 percent increase in the plant's nominal power.

The main goal of this project is to assure safe and stable operation of the Krško NPP after the year 2000, with the anticipated conditions in the free market for energy having already been incorporated. The modernization project was prepared and implemented by a team of experts from the Krško NPP: they worked with the world's largest suppliers of nuclear equipment and services (Westinghouse Electric Europe, Belgium; Siemens-KWU, Germany; Framatome, France; CAE, Canada) and with a large number of sub-contractors from Europe, Japan and the USA. The experts of the Krško NPP also took full advantage of the support offered from domestic nuclear professionals, who acted as consultants to the Krško NPP and performed independent safety reviews as part of the domestic licensing procedure.

The main reasons for the replacement of the old steam generators and the main topics of the modernization project are presented in a series of eight papers in the last two issues of the Journal of Mechanical Engineering (Vol. 46, No.3 and 4).

At this point it is important to acknowledge the contributions by the authors from the Krško NPP, who spent a significant part of their very precious time during the implementation of the modernization project to write these papers and to disseminate the information relating to the project.

The modernization in the Krško NPP is near to be completed, but domestic experts in nuclear engineering will still be needed for at least the next 30 years. One of the priorities over the next decade will be the establishment of a publicly funded education program for nuclear engineers, which would enable us to preserve and upgrade the domestic knowledge in this vital domain.

Doc.Dr. Leon Cizelj

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Mehanske in trdnostne analize v podporo zamenjavi uparjalnikov in povečanju moči JE Krško

Mechanical and Structural Analyses Supporting the Steam-Generator Replacement and Power Uprating at the Krško NPP

Božidar Krajnc - Janez Župec

JE Krško je ena zadnjih tlačnovodnih jedrskih elektrarn, zgrajenih po zahodni tehnologiji v Evropi, ki se je odločila za zamenjavo uparjalnikov s hkratnim povečanjem moči. V podporo temu je bil izdelan in neodvisno preverjen cel niz projektnih in varnostnih analiz, ki dokazujejo:
- da so novi uparjalniki skladni s sedanjo elektrarno in
- da lahko elektrarna deluje z ustreznimi varnostnimi rezervami tudi pri povečani moči.
V tem članku je predstavljen le del opravljenih analiz, mehanske in trdnostne analize. Opravljene analize so, kakor dokazujejo neodvisni pregledi, opravljene kakovostno in zagotavljajo, da bo Jedrska elektrarna Krško obratovala varno tudi po zamenjavi uparjalnikov in povečanju moči.
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(Ključne besede: celovitost, analize mehanske, analize trdnostne, puščanje pred zlomom (LBB))

Krško nuclear power plant (NPP) is one of the last pressurized water reactor NPPs of western design in Europe, which has decided to replace the existing steam generators and at the same time perform a power uprating. A comprehensive set of design calculations and safety analyses have been performed to demonstrate:
– that the new steam generators are compatible with the existing plant,
– that the plant can operate safely and with adequate margins at the uprated power.
In this paper only the mechanical and structural analyses are presented. These analyses meet, as verified by independent evaluations, high quality standards and ensure the safe operation of Krško NPP after the replacement of the steam generators and the power uprating.
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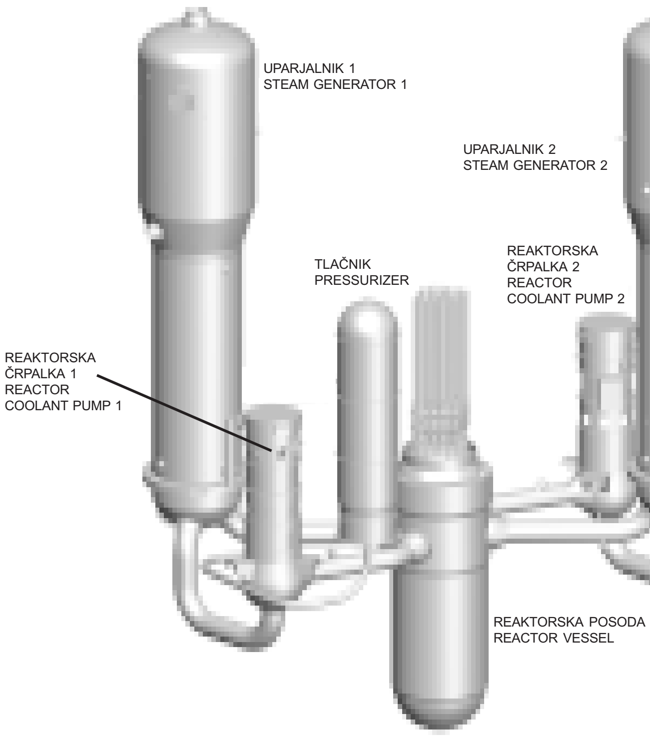
(Keywords: integrity, mechanical analysis, structural analysis, leak before break (LBB))

0 UVOD

Med pripravami za zamenjavo uparjalnikov v JE Krško je bila narejena študija izvedljivosti hkratnega povečanja moči elektrarne [1]. Ocenila je, da lahko moč brez večjih projektnih sprememb sistemov in komponent elektrarne povečamo za 6,3%. Sledila je analiza izvedljivosti uvedbe osnutka puščanja pred zlomom (LBB) za cevovod reaktorskega hladilnega sistema [2]. Poročilo ocenjuje, da cevovod reaktorskega hladilnega sistema izpolnjuje vse potrebne pogoje ([3] in [4]) za uvedbo osnutka LBB. Uporaba zamisli LBB dovoljuje izločitev dinamičnih obremenitev cevovoda reaktorskega hladilnega sistema in drugih komponent (sl. 1) med veliko izlivno nezgodo (LB LOCA) iz trdnostnih analiz oziroma projektnih zahtev in lahko pomembno prispeva k minimizaciji sprememb komponent in sistemov v elektrarni.

0 INTRODUCTION

A feasibility analysis of parallel power uprating was undertaken [1] in the framework of the preparatory works for the steam generator replacement. It estimated that apower increase of 6.3 % could be achieved without extensive modifications to the plant systems and components. It was followed by a feasibility analysis aiming at the implementation of the leak-before-break (LBB) concept for the reactor-coolant-system piping (RCL) [2]. The main conclusion was that the reactor coolant loop of Krško NPP fulfills all the necessary conditions ([3] and [4]) for the implementation of the LBB concept. The implementation of the LBB concept allows the elimination of dynamic effects associated with a large break (LB) loss-of-coolant accident (LOCA) from the design basis and the structural analyses of the reactor-coolant system (Fig. 1) and contributes significantly to the minimization of changes in the plant systems and components.



Sl. 1. *Reaktorski hladilni sistem*
Fig. 1. *Reactor coolant system*

Na temelju teh ugotovitev so se v letu 1997 pričele obsežne varnostne analize z namenom dokazati zmožnost varnega obratovanja ter preveriti mehansko celovitost in dobo trajanja sistemov in komponent. Te analize so v sklepni fazi in so dokumentirane v delovnih poročilih ter pregledane od neodvisnih recenzentov domačih in tujih pooblaščenih institucij.

V tem prispevku je poudarek le na delu analiz. To so mehanske in trdnostne analize, ki sta jih v okviru programa modernizacije JE Krško izvedli podjetje Westinghouse Electric Europe, ki je evropsko hčerinsko podjetje dobavitelja elektrarne, in Westinghouse Pittsburgh kot izvirni dobavitelj elektrarne. Analize je bilo treba znova izdelati zaradi naslednjih razlogov:

- Novi uparjalniki imajo nasproti prvotnim večjo maso, ki je tudi nekoliko drugače razporejena.
- V okviru projekta posodobitve JE Krško uvaja t.i. zamisel delovnega okna. To pomeni, da bo elektrarna delovala s polno močjo v okviru niza temperaturnih in tlačnih pogojev (trenutno sme delovati s polno močjo le v eni točki tlak-temperatura).
- Uvajanje novih analitičnih metod (seizmične analize, LBB itn.), ki so že bile uporabljene v drugih podobnih projektih in odobrene od upravnih organov tako v ZDA (United States Nuclear Regulatory Commission) kakor v Evropi.

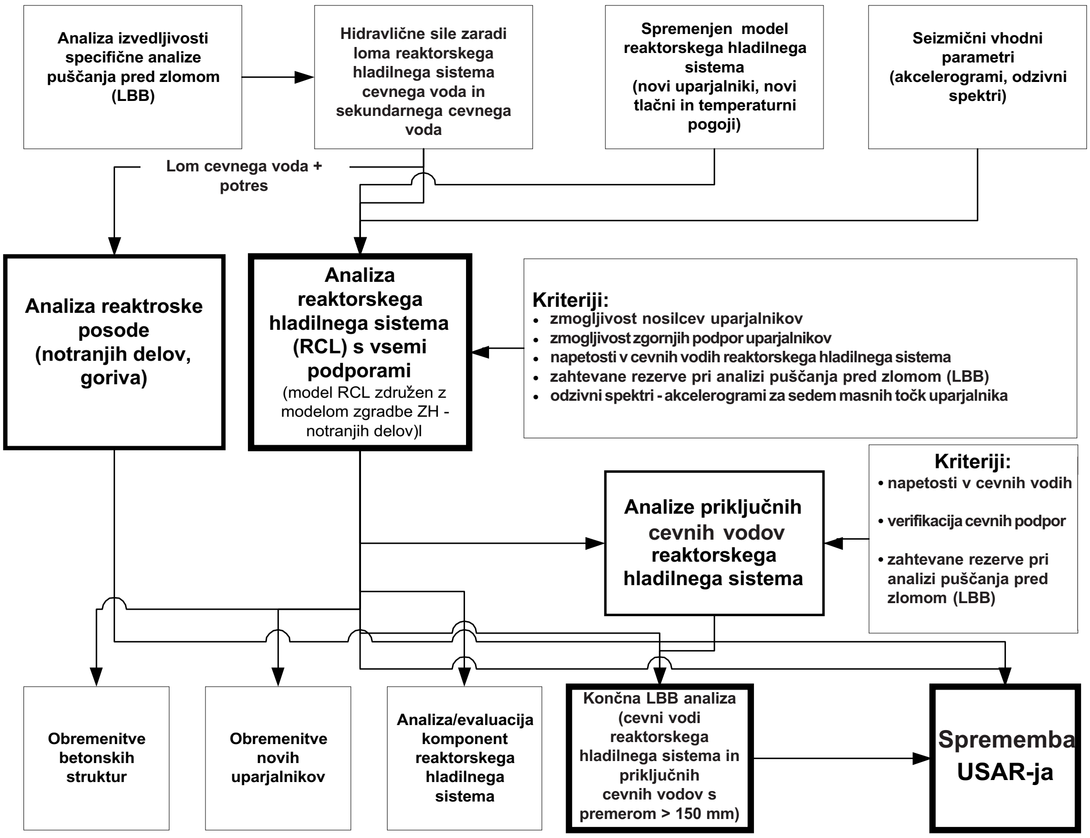
Zaradi naštetih razlogov je bilo treba pripraviti tudi nove vhodne podatke za mehanske in trdnostne analize:

Comprehensive safety analyses were started in 1997 to demonstrate plant safety performance and to confirm the mechanical integrity and lifetime of the systems and components. These analyses are about to be completed. Each of the analyses was independently reviewed and evaluated by domestic and foreign authorized institutions and documented in a work report.

This paper focuses on the mechanical and structural analyses. These were performed within the Krško NPP modernization project by Westinghouse, Pittsburgh, the original vendor of the plant and its Brussels-based daughter company Westinghouse Electric Europe. The original design analyses had to be repeated for the following main reasons:

- The new steam generators have a slightly higher mass and a different mass distribution to the original ones.
- The operating-window concept is to be introduced in the modernization project of Krško NPP. The plant will be therefore analyzed and licensed for full-power operation within a set of temperature and pressure conditions (the current license only allows for full-power operation at a single pressure-temperature operating point).
- Introduction of new analytical methods (seismic time-history analysis, LBB, etc.), which have already been implemented and approved in similar projects in the United States (United States Nuclear Regulatory Commission) and in Europe.

For the above reasons new inputs for the mechanical analyses had to be prepared. These can be listed as follows:



Sl. 2. *Potek mehanskih in trdnostnih analiz v projektu posodobitve JEK*

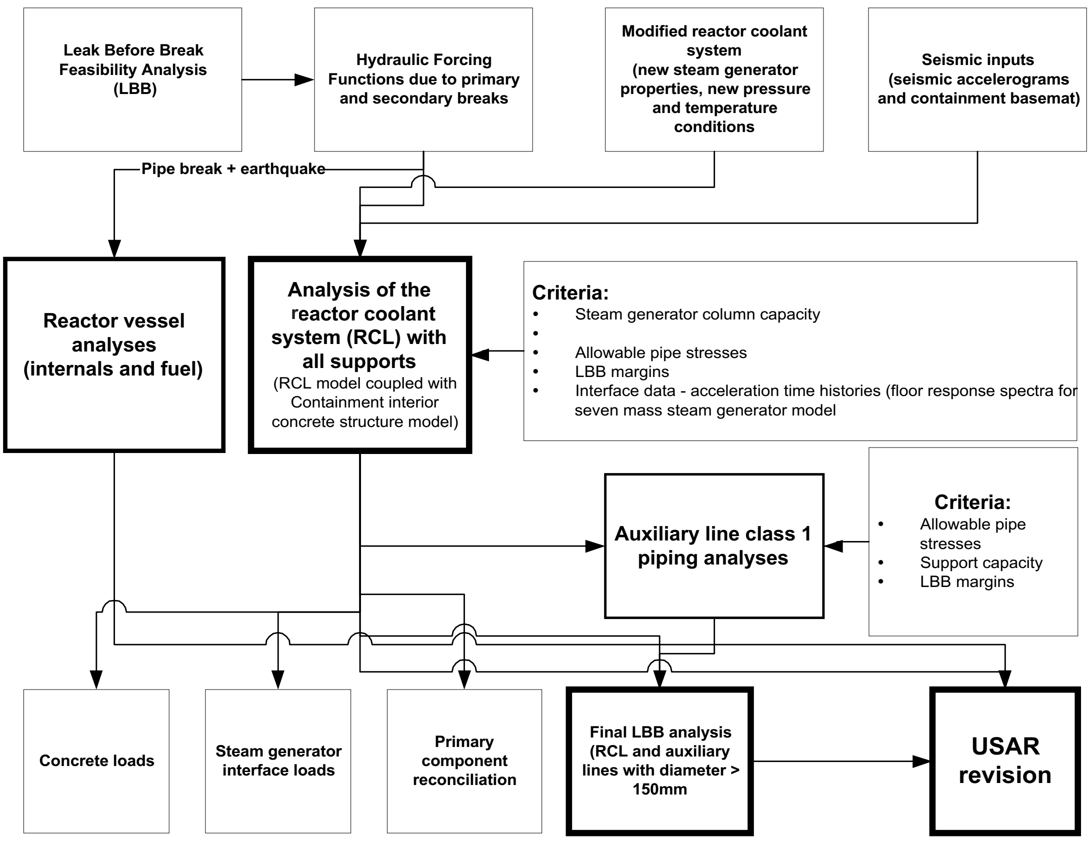


Fig. 2. *Structural and mechanical analyses flowchart for Krško NPP*

- Seizmični vhodni podatki (časovno odvisni pospeški – akcelerogrami) za temelje reaktorske stavbe. Potrebna je bila nova analiza interakcije zemljine in poslopja. Analiza je bila narejena s tremi statistično neodvisnimi akcelerogrami, ki delujejo hkrati v vseh treh prostorskih smereh. Akcelerogrami so bili pripravljeni na osnovi odzivnega spektra, kakor ga definira Regulatory Guide 1.60 [20].
- Specifikacija projektnih prehodnih pojavov.
- Nove hidravlične sile zaradi predpostavljenih zlomov cevovodov (izlivna nezgoda (LOCA), zlom glavnega parnega voda (MS), zlom cevnega voda glavne napajalne vode (FW)).

V nadaljevanju opisujemo štiri najpomembnejše sklope opravljenih analiz: (1) analize vhodnih podatkov, (2) trdnostne analize cevnih vodov reaktorskega hladilnega sistema, (3) analize, ki so dokazale ustreznost uporabe koncepta puščanja pred zlomom in (4) mehanske analize komponent reaktorskega sistema.

Kakovost vseh opravljenih analiz so neodvisno preverile in potrdile domače in tuje pooblaščene organizacije. Kakovostna neodvisna preverjanja je med drugim omogočilo tudi raziskovalno delo pooblaščenih organizacij v zadnjih desetletjih, ki ga povzemamo v posebnem poglavju.

Diagram poteka mehanskih in trdnostnih analiz, opravljenih v okviru projekta modernizacije JE Krško, je prikazan na sliki 2.

1 ANALIZE VHODNIH PODATKOV

1.1 Projektni prehodni pojavi za reaktorski hladilni sistem

Projektni prehodni pojavi so pričakovani in predpostavljeni dogodki, katerih pomembnost in pogostost sta bistveni za projektiranje komponent in preračune utrujenosti materiala. Praviloma so opisani s časovnimi poteki tlaka, temperature in pretoka hladiva. V skladu s projektnimi standardi ([21] in [22]) projektne prehodne pojave delimo v naslednja obremenitvena stanja:

- Normalno stanje so redni oziroma pogosti dogodki ob delovanju elektrarne, zamenjavi goriva in vzdrževanju elektrarne. Primeri takšnih prehodnih pojavov so zagon in zaustavitev elektrarne, povečevanje in zmanjševanje moči, zamenjava goriva ipd.
- Moteno stanje obsega zmerno pogoste dogodke, do katerih v posamezni elektrarni lahko pride med koledarskim letom. Ponovni zagon elektrarne po takšnem dogodku je mogoč takoj. Primera takšnega prehodnega pojava sta projektni potres in samodejna zaustavitev reaktorja.
- Nezgodno stanje obsega dogodke, do katerih v posamezni elektrarni lahko pride med celotnim obdobjem. Ponovni zagon elektrarne je mogoč po popravilu. Primer je zlom cevi v uparjalniku.

- Seismic inputs at the foundation level of the reactor building (acceleration time histories-accelerograms). This required new soil structure analysis based on three statistically independent accelerograms matching Regulatory Guide 1.60 [20] and applied simultaneously in all three spatial directions.
- Design transients specification.
- New hydraulic forcing functions due to the postulated pipe breaks (LOCA, MS/FW breaks).

The four main clusters of analyses performed are described in some detail below: (1) analyses of input data, (2) structural analyses of the reactor coolant-system piping, (3) analyses verifying the applicability of the LBB concept and (4) mechanical analyses of the RCS components.

The quality of all analyses performed was independently verified and confirmed by domestic and foreign authorized institutions. The quality of independent verification was, to some extent, supported by research work, performed by authorized institutions over the last two decades and is summarized in a separate section.

The flowchart of the mechanical and structural analyses performed as part of the Krško NPP modernization project is shown in Figure 2.

1 ANALYSES OF INPUT DATA

1.1 Reactor-Coolant-System Design Transients

The design transients are expected or postulated events with magnitude and frequency, which are significant in the component design and fatigue evaluation. They are usually described as time variations of pressure, temperature and flow of the coolant. The design transients in accordance with design codes ([21] and [22]) constitute the operating conditions as follows:

- Normal conditions encompass all regular or frequent events in the course of the plant normal operation, refueling and maintenance. Some examples of such transients are plant startup and shutdown, power increase and decrease, refueling etc.
- Upset conditions include events of moderate frequency that may happen within a calendar year. Immediate restart of the plant is possible after such events. Two examples are an operating basis earthquake and a reactor trip.
- Emergency condition events are expected within the plant lifetime. Restart of the plant is possible after some repair. An example is a steam-generator tube rupture.

- Stanje okvare obsega komaj verjetne napake, ki bi lahko povzročile izpuste velikih količin radioaktivnih snovi v okolje. Čeprav ne pričakujemo, da se bodo zgodili, je elektrarna za takšne dogodke projektirana in jih mora prenesti brez večjih posledic za okolje. Ponovni zagon elektrarne navadno ni več mogoč. Primera sta velika izlivna nezgoda (LOCA) in potres varne ustavitve.
- Testno stanje obsega dogodke ob tlačni preobremenitvi vključno s hidrostatičnim preskusom in preskusom puščanja.

Spremembe glede na sedanje projektne prehodne pojave za reaktorski hladilni sistem in sekundarni hladilni krog elektrarne so bile ocenjene oziroma ponovno analizirane z računalniškim programom LOFTRAN zaradi naslednjih razlogov:

1. Višja imenska moč elektrarne – moč jedrskega sistema za proizvodnjo pare se poveča za 6,3 %. Povečani najvišji tlak pri nekaterih prehodnih pojavih in nekoliko večja zaostala toplota v reaktorju imata lahko dolgoročen vpliv na posamezne prehodne pojave. S povečanjem moči se poveča tudi imenski pretok glavne napajalne vode za 6,3 %.
2. Zamenjava uparjalnikov – nova uparjalnika omogočata boljši prenos toplote z reaktorskega na sekundarno hladivo od prvotnega.
3. Spremenjena obratovalna temperatura reaktorskega hladiva pri imenski moči.

Novi projektni prehodni pojavi so dokumentirani v [13] in [14] in so bili kot vhodni podatki uporabljeni v vseh nadaljnjih analizah celovitosti cevnih vodov in komponent reaktorskega hladilnega sistema. Novi projektni prehodni pojavi [13] so tudi pomembni vhodni podatki za analize celovitosti novih uparjalnikov.

1.2 Hidravlične sile zaradi izlivne nezgode – LOCA

Predpostavljeni zlomi visoko energijskih cevnih vodov (izlivna nezgoda - LOCA) povzročijo velike dinamične obremenitve, ki so poleg razmer v cevnem vodu močno odvisne tudi od lokacije in velikosti zloma cevovoda. Ob upoštevanju koncepta puščanja pred zlomom (poglavje 3) so bile dinamične obremenitve analizirane za primer zloma cevnega voda premera 150 mm. Dinamične obremenitve zaradi izlivne nezgode glede na vrsto delimo na:

1. Mehanske obremenitve cevnega voda reaktorskega hladilnega sistema – trenutna sprostitev notranjih napetosti ob zlomu in potisna sila curka iztekajočega hladiva.
2. Notranje hidravlične obremenitve v reaktorski tlačni posodi so največje v primeru zloma hladne veje reaktorskega hladilnega sistema – razredčitveni val potuje skozi izstopno šobo reaktorja naravnost v zgornji del reaktorske posode, navzdol skozi sredico in nato spet navzgor proti izstopni šobi. Zlom vroče veje

- Faulted conditions consist of improbable faults, which could cause large releases of radioactive materials in to the environment. The plant is designed to withstand such events without undue consequences for the environment, although it is not expected that they will actually happen. Restart of the plant is usually not possible. Examples are a LOCA (break of reactor-coolant system piping) and a safe-shutdown earthquake.
- Test conditions include events with pressure overload including hydrostatic and leak tests.

The changes with respect to the original design transients for the reactor-coolant system and the secondary coolant system were re-evaluated or re-analyzed with the LOFTRAN code for the following reasons:

1. Higher nominal plant power – power of the nuclear steam supply system (NSSS) is increased by 6.3 %. Increased maximum pressure and decay heat can have a long-term effect on some transients. With uprating, nominal feedwater flow is increased by 6.3 %.
2. Replacement of the steam generators – The new steam generators have a better heat transfer capability then the existing ones.
3. Different operating temperature at nominal power.

The new design transients are documented in [13] and [14] and have been used as input data in all subsequent integrity analyses of the piping and components of the reactor-coolant system. The new design transients documented in [13] also represent one of the most important inputs for the design analyses of new steam generators.

1.2 Hydraulic Forcing Functions due to a LOCA

The postulated breaks of high-energy pipelines (LOCA) generate high dynamic loads, which depend heavily on the conditions in the pipe, the location and size of the break. The implementation of the leak-before-break concept (section 4) lead to analysis of the hydraulic loads caused by a break of a pipe with a diameter of 150 mm. LOCA loads may be characterized as:

1. Reactor-coolant-system-piping mechanical loads – immediate release of internal pipe stresses and jet forces caused by escaping coolant.
2. Reactor pressure vessel internal hydraulic loads, which are maximized at the break of the cold-leg of the reactor-coolant loop. In this case, the rarefaction wave passes through the outlet nozzle directly into the upper internals region, depressurizes the core and enters the downcomer annulus at the bottom of the vessel where it turns upward towards the inlet nozzle. The hot-leg

povzroči manjše vodoravne sile. Zaradi razredčitvenega vala, ki potuje neposredno v sredico, je tlačna razlika v notranjosti posode manjša kakor pri enako velikem zlomu hladne veje. Predpostavljeni čas odpiranja zloma je 1 ms, kar je zelo konzervativna predpostavka.

Časovno odvisni poteki tlaka, masnega pretoka in temperature reaktorskega hladiva med izlivno nezgodo so bili ocenjeni z računalniškim programom MULTIFLEX [8], ki upošteva medsebojno delovanje tekočine in trdnine. V vsakem časovnem koraku oceni ravnotežje med tlakom reaktorskega hladiva in deformacijami posameznih delov reaktorske tlačne posode. Rezultati MULTIFLEX-a so uporabljeni kot vhodni podatki v računalniška programa LATFORCE in FORCE2, ki izračunata hidravlične sile na reaktorsko tlačno posodo, sredico, termični ščit in preostale komponente v reaktorski posodi. Rezultati MULTIFLEX-a so tudi vhodni podatki za računalniški program THRUST, ki izračuna hidravlične sile v različnih točkah cevnega voda vzdolž zlomljene in nedotaknjene veje reaktorskega hladilnega sistema.

Časovni potek hidravličnih sil v notranjosti reaktorske tlačne posode in reaktorskem hladilnem sistemu je predstavljen v [8]. Hidravlične sile prenehajo delovati približno pol sekunde po začetku izlivne nezgode.

1.3 Hidravlične sile ob zlomu glavnega parnega voda in cevovoda napajalne vode

Najneugodnejša lokacija za zlom glavnega parnega voda je v bližini parne šobe uparjalnika. Predpostavljeni zlom neposredno ob parni šobi obremeni uparjalnik z navpično potisno silo. Če je zlom predpostavljen takoj za pravokotnim kolenom nad uparjalnikom, bo uparjalnik obremenjen z vodoravno potisno silo. V prvotnem projektu elektrarne sta bila upoštevana oba omenjena zloma. Razvoj tehničnih predpisov v zadnjih letih dovoljuje, da zlom na vodoravnem delu zanemarimo [5]. Glavni parni vod je namreč projektiran tako, da so napetosti med normalnim delovajem v vodoravnem delu znatno pod 80 % dovoljenih napetosti predpisanih v [21].

Za cevovod glavne napajalne vode je najneugodnejši zlom neposredno ob napajalni šobi uparjalnika.

1.4 Seizmični vhodni podatki – Analiza interakcije zemljine in stavbe

V izvirnem projektu elektrarne so bile seizmične obremenitve analizirane v frekvenčem prostoru z odzivnimi spektri. Razvoj računalnikov v zadnjih letih omogoča bistveno bolj podrobne analize in je v precejšnji meri botroval sodobni časovno odvisni seizmični analizi reaktorskega hladilnega sistema, ki je bila izvedena v dveh bistvenih korakih:

break produces smaller horizontal forces. In this case, the rarefaction wave travels directly in the core, causing smaller internal pressures than a cold leg break of the same size. The assumed break opening time is 1 millisecond, which is a very conservative assumption.

The MULTIFLEX computer code [8] is used to estimate the transient pressures, mass velocities, and thermal properties of the reactor coolant. The MULTIFLEX code evaluates the fluid structure interaction by finding the equilibrium between the coolant pressure and the deformed state of the reactor internals within each time step. The MULTIFLEX output is used as aninput to LATFORCE and FORCE2 codes. These codes calculate the hydraulic forces acting on the reactor pressure vessel, core barrel, thermal shield and other internal components of the vessel. The MULTIFLEX output is used as an input to the THRUST code to calculate the hydraulic forces at various locations along the reactor-coolant piping in the broken and intact loop.

The time histories of hydraulic forces at different locations within the rector-coolant loop and components are documented in [8]. The transient with LOCA hydraulic forces is finished after about half a second.

1.3 Hydraulic Forcing due to Main Steam and Main Feedwater Line Breaks

The most unfavorable main steam line break location is postulated in the immediate vicinity of the steam-generator steam nozzle, where a vertical thrust force is generated. A break that is postulated immediately after the 90-degree elbow above the steam generator would cause a horizontal thrust force. Both locations were postulated in the original plant design. However, the recent developments in the design codes allow the elimination of the break in the horizontal part of the pipe. The main steam line is designed with less than 80% of the allowable stress [21] in the horizontal part.

The most unfavorable location for the main feedwater line break is postulated in the immediate vicinity of the main feedwater nozzle on the steam generator.

1.4 Seismic Inputs – Soil Structure Analysis

The seismic loading in the original design was analyzed in the frequency domain using response spectra. The developments in computers in recent years has resulted in a much more accurate time history of the seismic analysis of the reactor coolant loop. This analysis was performed in two major steps:

– V prvem koraku so bili izdelani trije statistično neodvisni akcelerogrami na prostem površju. Akcelerogrami natančno opisujejo izvirni odzivni spekter, definiran v skladu z Regulatory Guide 1.60 [20], s pospeškom ničelne periode 0,3 g.
– V drugem koraku je sledila seizmična analiza interakcije zemljine in stavbe ob hkratnem delovanju vseh treh komponent potresnih pospeškov. Analiza interakcije zemljine in stavbe je nadgradnja prvotne analize z upoštevanjem novejših zahtev iz [17] in [18].

Rezultat te analize so časovno odvisni akcelerogrami na vrhu temelja zadrževalnega hrama (tri komponente) [9]. Ti so bili uporabljeni kot vhodni podatek za analizo seizmičnih napetosti v reaktorskem hladilnem sistemu.

2 TRDNOSTNA ANALIZA

Normalno delovanje in varna zaustavitev jedrske elektrarne pri vseh načrtovanih delovnih stanjih temeljita na ustrezni projektni zasnovi in celovitosti reaktorskega hladilnega sistema. Le-to dokazujemo z nizom analiz, ki zajemajo vse predvidene delovne pogoje, kar seveda vključuje zlom cevovoda in potres varne zaustavitve. Pri tem je treba upoštevati, da reaktorski hladilni sistem in vsi nanj priključeni cevni vodi predstavljajo tlačno mejo primarnega hladiva in zato v skladu z standardi ASME za tlačne posode [21] in [22] sodijo v najvišji varnostni razred 1. V grobem lahko trdnostne analize razdelimo v naslednje skupine:
- statična analiza reaktorskega hladilnega sistema (notranji tlak, lastna teža, sprememba gibalne količine hladiva v cevnih kolenih itn.),
- seizmična časovno odvisna analiza reaktorskega hladilnega sistema,
- dinamična analiza odziva reaktorskega hladilnega sistema na izlivno nezgodo,
- utrujenostna analiza reaktorskega hladilnega kroga in
- analiza pomožnih priključenih cevnih vodov.

2.1 Statična analiza reaktorskega hladilnega sistema

Statične napetosti v cevnih vodih reaktorskega hladilnega kroga nastanejo zaradi lastne teže, toplotnih obremenitev in preostalih splošnih obremenitvenih stanj in so ocenjene z uporabo računalniškega programa WESTDYN. Program deluje po postopku prenosnih matrik, ki je nekakšen predhodnik sodobnejše metode končnih elementov.

Cevne vode obravnavamo kot prostorske sisteme nosilcev, ki so podprti z elastičnimi vzmetmi. Komponente (npr. črpalka) so podane kot masne točke. Pri takem načinu analize ni bistvenih razlik med metodo prenosnih matrik in metodo končnih elementov, saj obe dajeta izjemno natančne rezultate.

– In the first step, three statistically independent accelerograms on the free field have been generated. They match the original Regulatory Guide 1.60 [20] based on the free-field response spectrum with a zero period acceleration (ZPA) value of 0.3 g.
– In the second step, soil structure analysis followed, applying simultaneously all three spatial earthquake components. The original analysis has therefore been upgraded to meet up to date requirements [17] and [18].

This analysis resulted in acceleration time histories at the containment basement (three components) [9]. These were then used as inputs for the seismic analysis of the reactor coolant system.

2 STRUCTURAL ANALYSIS

Normal operation and safe shutdown of a nuclear power plant depend on the adequate design and structural integrity of the reactor-coolant system. This is demonstrated by a set of analyses performed for loads under all postulated operating conditions, which includes pipe breaks and a safety-shutdown earthquake. It should be pointed out that the reactor-coolant system, together with the attached auxiliary piping, represents the reactor-coolant pressure boundary. According to the ASME Boiler and Pressure Vessel Code [21] and [22] it is classified as safety class 1 equipment. Basically, the structural analyses performed could be grouped as:
- static analysis of the reactor-coolant piping (internal pressure, dead weight, change of the coolant momentum in the elbows etc),
- seismic time history analysis of the reactor-coolant piping,
- dynamic analysis of the of the reactor coolant loop response to a LOCA,
- fatigue analysis of the coolant loop, and
- analysis of the attached auxiliary pipelines.

2.1 Static analysis of the reactor-coolant piping

The static stresses in the reactor-coolant-system piping, which develop because of the dead weight, thermal loads and other sustained loads, were estimated by the WESTDYN computer code. The code utilizes transfer matrices, which are one of the predecessors of the finite-element method.

The piping is modeled as a spatial system of beams, supported by elastic springs. The components (e.g., pump) are modeled as lumped masses. Such a choice of the analytical model does not differentiate between the finite-element and transfer-matrices results, as both methods yield very accurate results.

2.2 Seizmične analize reaktorskega hladilnega sistema

Za seizmično analizo je treba model z nosilci, vzmetmi in masnimi točkami, ki je bil uporabljen pri statični analizi, nekoliko dopolniti. Predvsem gre za natančnejši opis masnih karakteristik cevnega voda in opreme (uparjalnik, reaktorska črpalka, reaktorska tlačna posoda z notranjimi deli). Dodati je treba tudi model betonske stavbe in izboljšati modele podpor, ki povezujejo cevni vod in betonsko stavbo.

Zadovoljivo natančen dinamični odziv novih uparjalnikov je mogoče doseči že s sedmimi ločenimi masami (4 mase v spodnjem delu pod koničnim prehodom in 3 nad njim), povezanimi s cevastimi nosilci. Cevasti nosilci kažejo elastičnost lupine in notranjih delov uparjalnika. Najnižja masa je postavljena v točko, kjer se sekata središčnici vstopne in izstopne šobe reaktorskega hladilnega sistema na uparjalniku. Najvišja masna točka je na vrhu uparjalnika.

Glavni primarni črpalki sta modelirani z dvema ločenima masnima točkama. Vpetje črpalk je predstavljeno s sistemom togostnih matrik in posameznih vzmetnih karakteristik.

Lupina reaktorske tlačne posode je predstavljena s štirimi ločenimi masami, ki so povezane s cevastimi nosilci. Masne točke in cevasti nosilci modela so izbrani tako, da masa in masni vztrajnostni moment ustrezata reaktorski posodi. Model zajema tudi vztrajnost mehanizma krmilnih palic. Snop gorivnih elementov je predstavljen s prostorskim nosilcem, ki je izbran tako, da ustreza najnižji naravni frekvenci, določeni s preskusom.

Vhodni podatki v obliki akcelerogramov (gl. 1.4) so bili uporabljeni na nivoju temelja zadrževalnega hrama. Pospeški hkrati delujejo vzdolž obeh vodoravnih osi (sever-jug, vzhod-zahod) in vzdolž navpične osi. Časovno odvisna seizmična analiza temelji na načelu modalne superpozicije in je izvedena v več korakih:

1. Določitev lastnih frekvenc in lastnega vektorja pomikov.
2. Določitev sestavljenega modalnega koeficienta dušenja, ki temelji na razmerjih deformacijske energije absorbirane pri vsakem izmed načinov dušenega vibriranja.
3. Dinamična analiza odziva z modalno superpozicijo. Dinamično gibanje je izračunano z upoštevanjem odpiranja in zapiranje rež v nekaterih podporah. Reže v podporah so namreč v hladnem stanju elektrarne nastavljene tako, da se zaprejo šele pri delovni temperaturi. Optimalna nastavitev rež je izjemno pomembna, saj zagotavlja primerno statično in dinamično podprtje cevnega voda. Zapiranje rež pri temperaturah, ki so nižje od delovne tempera-

2.2 Seismic time history analysis of the Reactor Cooling System

The model composed of beams, lumped masses and springs, used in the static analysis, has to be modified: essentially by including detailed mass characteristics of the piping and equipment (steam generators, reactor-coolant pump, reactor pressure vessel with internals). Further, the modified model is coupled to the structural model representing the containment interior concrete structure. This also requires very detailed modeling of the piping support structures.

A reasonably accurate dynamic response of the new steam generators could be achieved using 7 lumped masses (4 in the lower part below the conical transition and 3 above it), connected with beams assuming the shape of a pipe. The pipe-shaped beams represent the elastic properties of the shell and steam-generator internals. The lowermost lumped mass is placed at the intersection of the symmetry lines of both reactor-coolant nozzles. The uppermost lumped mass is placed at the top of the steam generator.

The reactor-coolant pumps are represented by two lumped masses. Support structures are represented by stiffness matrices and/or individual spring members.

The reactor-pressure-vessel (RPV) shell is represented by four lumped masses connected by hollow cylindrical beams. The lumped masses and beams are tuned to represent the inertial properties of the vessel. The model also includes the inertial properties of the control-rod drive mechanism. The properties of the beam representing the fuel assemblies are adjusted to simulate the fundamental frequency which has been determined by experiment.

The time-dependent accelerations (see section 1.4) are applied simultaneously at the containment basement level along two horizontal axes (north-south, east-west) and the vertical axis. The time-dependent analytical method was based on the principle of modal superposition and was performed in several steps:

1. Determination of the natural frequencies and mode shapes.
2. Determination of the composite modal damping coefficients, which is based on the proportion of strain energy absorbed in each element for each mode.
3. Dynamic response analysis using modal superposition. The dynamic motion is calculated by taking into account the opening and closing of supports with gaps. The gaps in the supports are shimmed to be open in cold condition and to close when the plant approaches the operating temperature. The optimal gap shimming is extremely important, as it provides appropriate static and dynamic support to the piping. Closing of the gaps at temperatures below the oper-

ture, pa lahko povzroči nezaželene visoke napetosti zaradi zavrtega toplotnega raztezanja [6]. V analizah je upoštevana linearizirana togost podpor z režami.

Rezultate analize predstavljajo največje obremenitve, pomiki in napetosti v analiziranem časovnem obdobju ter odzivni spektri pospeškov v ločenih masnih točkah uparjalnika, ki so bile uporabljene pri seizmični overitvi novih uparjalnikov ([10] in [11]).

Izračunane seizmične napetosti v cevnih vodih so zelo nizke, kar pomeni, da so komponente reaktorskega hladilnega sistema primerno podprte in protipotresno varne [12].

2.3 Dinamična analiza odziva na izlivno nezgodo

Model, ki je bil uporabljen pri seizmični analizi, je prilagojen za analizo zloma cevnega voda z vključitvijo mas pomožnih cevnih vodov na mestih, kjer so priključeni na reaktorski hladilni sistem. Vhodni podatki so opisani v poglavjih 1.2 in 1.3. Diagram potrebnih korakov za to analizo prikazuje slika 3. Rezultati, ki so izraženi z dinamičnimi pomiki vseh masnih točk, se uporabijo za napetostno analizo cevnega voda in verifikacijo nosilnosti podpor.

2.4 Analiza utrujanja

Delovanje jedrske elektrarne povzroča nihanja temperature in tlaka reaktorskega hladiva in s tem tudi obremenitev cevnih vodov in opreme. Takšna nihanja obremenitev lahko povzročijo utrujenost materiala. Z analizo utrujanja dokažemo, da bodo vgrajeni materiali prenesli nihanja obremenitev zaradi vseh predpostavljenih normalnih in motenih prehodnih pojavov v celotni dobi trajanja elektrarne.

Vsak prehodni pojav je opisan z vsaj dvema obremenitvama, ki predstavljata območje med največjo in najmanjšo napetostjo v komponenti med izbranim prehodnim pojavom. Največjo in najmanjšo napetost določimo iz analize časovno odvisnih toplotnih napetosti, ki jim dodamo še membranske napetosti v ceveh zaradi notranjega tlaka.

Toplotni prehodni pojavi povzročajo časovno spremenljive temperaturne porazdelitve skozi steno cevi. To povzroča toplotne napetosti, ki jih delimo na tri dele: na stalni, linearni in nelinearni del. Stalni del povzroča splošno toplotno raztezanje cevovoda in s tem povezane obremenitve, linearni del povzroča upogibni moment prek stene in nelinearni del povzroča izrazito napetost tik ob notranji površini cevi (v STS = površinski učinek).

Za izračun toplotnega prehodnega pojava je uporabljen računalniški program THERST. Program deluje z uporabo metode končnih razlik in predpostavlja toplotni tok le v radialni smeri cevi. Predpostavljeno je, da je zunanja površina adiabatna,

ating temperatures might cause significant unwanted stresses due to the restrained thermal flexibility [6]. Linearized stiffness of the supports with gaps was used in the analysis.

The results are maximum loads in terms of displacements and stresses during the analyzed period and response spectra for all lumped masses in the steam-generator model. The response spectra were used for seismic qualification of the new steam generators ([10] and [11]).

The conclusion of the performed analysis is that all piping stresses remain very low, which means that the reactor-coolant piping is well supported and safe against earthquake [12].

2.3 Dynamic LOCA Response Analysis

The model used in the seismic analysis was further modified to be used in the pipe rupture analysis by including the mass of the auxiliary lines connected to the reactor-coolant-system piping. The input data is described in sections 1.2 and 1.3 above. A flowchart of the required steps in this analysis is shown in Figure 3. The results are expressed as dynamic displacements for each of the mass points and are used in the stress evaluation of piping and verification of the supports.

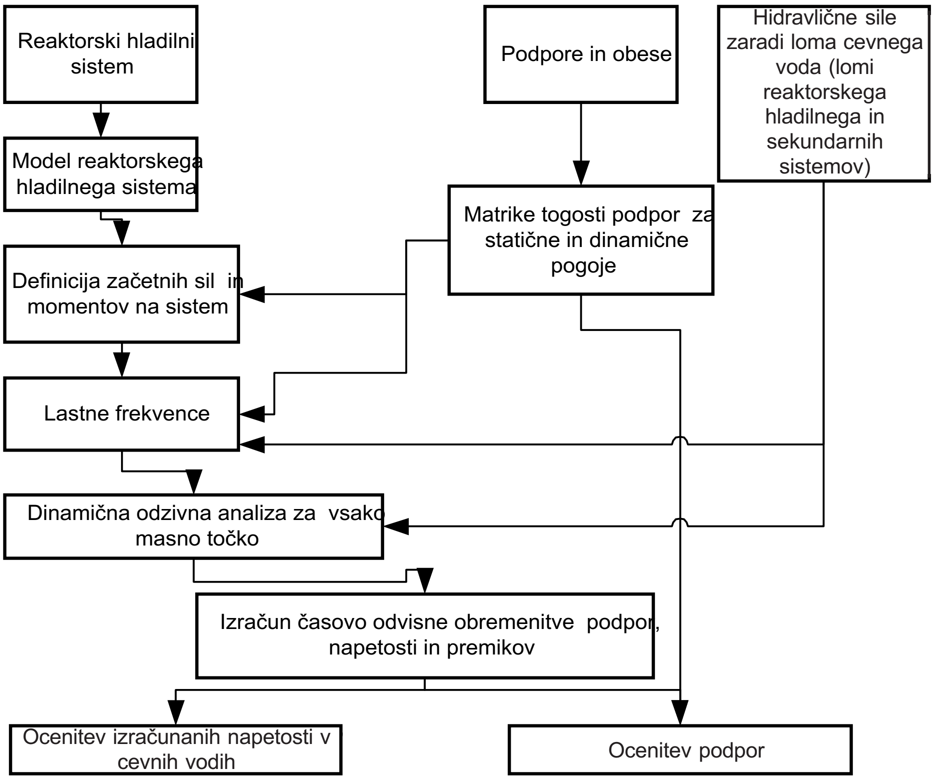
2.4 Fatigue Analysis

Operation of a nuclear power plant causes fluctuations in the reactor-coolant temperature and pressure and therefore also in the loads on the piping and equipment. Such fluctuations in load may cause fatigue. The fatigue analysis is performed to verify that the implemented materials are able to bear the load fluctuations due to all postulated normal and upset transients during the life time of the plant.

At least two loads representing the cycle between the lowest and the highest load of the component, during a given transient, describe each transient. The highest and lowest loads were determined by a time-history analysis of the thermal stresses, combined by membrane stresses due to the internal pressure.

The thermal transients cause time-varying temperature distributions across the pipe wall. These result in pipe-wall stresses that may be subdivided into three parts: uniform, linear and nonlinear portions. The uniform portion results in general expansion loads, the linear portion causes a bending moment across the wall, and the nonlinear portion causes a skin stress.

The THERST computer code is used to solve the thermal transient problem. It utilizes the finite-difference method and assumes heat flux in the radial direction only. The outer surface is assumed to be adiabatic, while the inner surface closely fol-



Sl. 3. *Potek oziroma zahtevani koraki pri izvedbi dinamične odzivne analize na izlivno nezgodo na primarni in sekundarni strani*

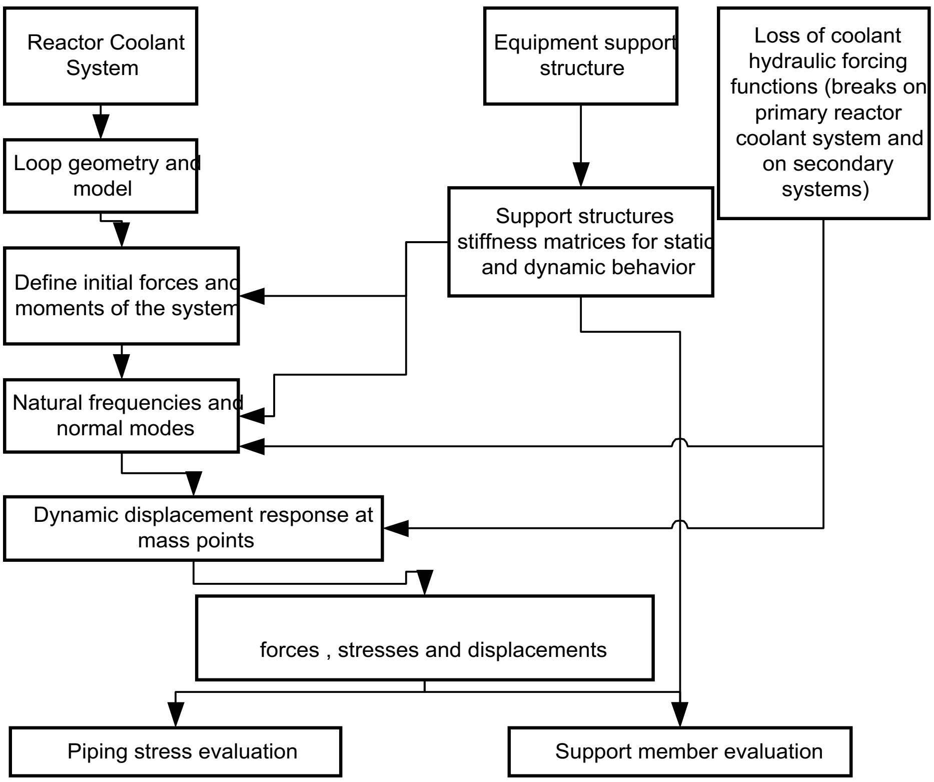


Fig. 3. *Flowchart of the required steps in a loss-of-coolant accident (including FW and MS break) RCL dynamic response analysis*

temperatura notranje površine cevi pa je kar enaka temperaturi hladiva. Največja toplotna napetost v steni cevi je običajno na notranji površini.

2.5 Analiza pomožnih cevovodov varnostnega razreda 1

Vsi pomožni cevni vodi varnostnega razreda 1 (varnostni razred 1 [21] označuje opremo, ki pomeni tlačno mejo primarnega hladiva, skupaj jih je 18) so bili ponovno analizirani z upoštevanjem trenutne razporeditve podpor in novih obremenitev. To vključuje utrujanje zaradi toplotne razplastitve (stratifikacije) tekočine v cevnih vodih, kjer je to potrebno (npr. sesalni vodi sistema za odvajanje zaostale toplote, prelivni vod tlačnika). Seizmične analize pomožnih cevnih vodov so izvedene z uporabo metode odzivnih spektrov (dvodimenzionalno vzbujanje). Analizirane so bile vse podpore. V nekaterih primerih, v katerih so obremenitve v primerjavi s prvotnimi bistveno narasle ali se zmanjšale, so bile predlagane spremembe podpor.

3 ANALIZA PUŠČANJA PRED ZLOMOM (LBB)

3.1 Zamisel LBB za reaktorski hladilni sistem

Reaktorski hladilni sistem JE Krško je v celoti projektiran za dinamične obremenitve, ki so posledica giljotinskega zloma cevi reaktorskega hladilnega sistema. V neposredni bližini cevnega voda so nameščene tudi posebne podpore, ki preprečujejo opletanje prelomljenega cevovoda zaradi sile iztekajočega hladiva.

V zadnjih dvajsetih letih so obširne analitične in empirične raziskave po svetu pokazale, da je giljotinski zlom cevnega voda reaktorskega hladilnega sistema zelo malo verjeten. To potrjujejo tudi dolgoletne delovne izkušnje. Zaradi tega je giljotinski zlom cevnega voda pri trdnostnih analizah Westinghousovih tlačnovodnih reaktorjev mogoče izločiti [3]. Pred tem je treba dokazati, da bo vsak giljotinski zlom cevnega voda pravočasno najavilo tolikšno puščanje reaktorskega hladiva, kakor ga sistemi elektrarne brez težav zaznajo. Dokazati je torej treba, da cevni vod izpolnjuje vse pogoje, ki jih za uporabo zamisli LBB predpisuje [4]:

1. Zadostni varnostni faktor med kritično velikostjo razpoke in velikostjo predpostavljene razpoke (predpostavljena razpoka povzroči puščanje, ki ga sistem za odkrivanje puščanja zagotovo zazna).
2. Zadostni varnostni faktor med izdatnostjo puščanja skozi predpostavljeno razpoko in zmožnostjo sistema za odkrivanje puščanja.
3. Zadostni varnostni faktor med predpostavljenimi obremenitvami in obremenitvami, ki bi povzročile nenadzorovano napredovanje kritične razpoke.
4. Večanje razpoke zaradi raznih mehanizmov (npr. lezenje, erozija, korozija, utrujanje, napetostna korozija itn.) je zanemarljiva oziroma zanesljivo predvidljiva.

lows the temperature of the coolant. The highest thermal stress usually occurs on the inside surface of the tube.

2.5 Analysis of auxiliary piping

All auxiliary lines falling into the safety class 1 (in total 18 lines, safety class 1 [21] is designation for equipment representing the reactor-coolant pressure boundary) have been re-nalyzed taking into account current support configurations. This includes fatigue due to thermal stratification of the coolant inside the piping where applicable (e.g., residual-heat-removal-system suction lines, pressurizer surge line). Seismic analysis of these piping systems has been performed using the response-spectrum method (2-D shock). All the supports have been verified, and in some cases where the loads have been increased or decreased compared to the original design, support modifications have been proposed.

3 LEAK-BEFORE-BREAK ANALYSIS (LBB)

3.1 LBB concept for the reactor-coolant system

The reactor-coolant system of Krško NPP is designed to withstand all dynamic loads resulting from a guillotine break of the reactor-coolant pipe. Special pipe whip restraints are mounted in the immediate vicinity of the pipe to prevent the jet driven motion of a broken pipe.

Operating experience combined with extensive experimental and analytical research performed over the last two decades has shown that a guillotine break of the reactor-coolant-system piping is highly unlikely. Consequently, the guillotine pipe break does generally not need to be included in the structural design basis of Westinghouse pressurized-water reactors [3]. However, it must be demonstrated that a guillotine break will always be preceded by a reliably detectable leak. In particular, the implementation of the LBB concept is possible if the compliance with requirements of [4] is demonstrated:

1. Sufficient safety margin exists between the critical crack size and a postulated crack that yields a reliably detectable leak rate.
2. There is a sufficient safety margin between the leak rate through a postulated crack and the leak detection capability.
3. Sufficient safety margin between postulated applied loads and loads leading to unstable crack growth exists.
4. Crack growth due to various mechanisms (fatigue, stress corrosion etc.) is negligible, or alternatively, reliably predictable.

Ustrezne analize, ki dokazujejo izpolnjevanje zgoraj navedenih pogojev, so bile izvedene v naslednjih korakih:

- Kritična lokacija razpoke: Analiza puščanja pred zlomom (LBB) mora biti izvedena za celotni cevni vod. Dovoljena poenostavitev je analiza v tisti točki, v kateri se pojavi kombinacija največjih napetosti in najslabših snovnih lastnosti. Ker predpostavljamo obodno razpoko, so najpomembnejše upogibne napetosti v cevnem vodu.
- Določitev velikosti razpoke: predpostavimo tako veliko razpoko, da bo izdatnost puščanja skoznjo najmanj 10-krat večja kakor je zmožen zaznati sistem za odkrivanje puščanja. Izdatnost puščanja je v posebni analizi za JE Krško ocenjena z metodami, priporočenimi v NUREG/CR-3464 [9], [19].
- Določitev obremenitev: Obremenitve za oceno stabilnosti predpostavljene razpoke dobimo neposredno iz trdnostne analize (poglavje 2.1 in 2.2), in sicer:
  - normalne obremenitve, ki obsegajo termične obremenitve, lastno težo, notranji tlak in
  - seizmične obremenitve.
- Analiza mehanike loma stabilnosti predpostavljene razpoke: Ta analiza dokazuje, da predpostavljena razpoka s primernim varnostnim faktorjem ostane stabilna tudi pri absolutni vsoti vseh zgornjih obremenitev. Pri lokalni analizi stabilnosti razpoke uporabljamo integrala J, pri analizi stabilnosti celotnega cevnega voda pa koncept mejne obremenitve.
- Dokazano je tudi, da je večanje razpoke zaradi utrujanja zanemarljivo.

Na temelju opravljenih analiz lahko sklenemo, da je cevni vod reaktorskega hladilnega sistema v JE Krško dovolj žilav, da ne bo prišlo do njegovega zloma brez poprejšnjega zaznavnega puščanja. Tako smemo glede na analize LBB v [7], ki dokazujejo, da JE Krško z zadostno zalogo izpolnjuje vse zahteve US NRC [3] in [4], dinamične obremenitve zaradi zloma cevnega voda reaktorskega hladilnega sistema v trdnostnih analizah zanemariti oziroma zamenjati z obremenitvami zaradi zloma največjega pomožnega cevnega voda, ki ne izpolnjuje pogojev za uvedbo koncepta LBB.

3.2 LBB za pomožne cevne vode varnostnega razreda 1 s premerom, večjim od 150 mm

Puščanje pred zlomom cevnega voda je bilo preverjeno tudi za pomožne cevne vode s premerom, večjim od 150 mm. Postopek je bil enak kakor pri cevnemu vodu reaktorskega hladilnega sistema. Za cevne vode s premerom pod 150 mm navadno ni mogoče zagotoviti stabilnosti razpoke z 10-krat večjim puščanjem, kot ga lahko zaznamo.

Appropriate analyses were performed to demonstrate compliance with above requirementa in the following major steps:

- Critical location of the crack. The LBB analysis has to be performed for the entire piping system. An acceptable simplification is to perform the analysis in the part of the tube with the most unfavorable combination of loads and material properties. Since a circumferential crack is postulated, the bending loads in the piping are of the primary importance.
- Crack size determination. The postulated crack size should yield a leak rate equal to the capability of the leak detection system multiplied by a safety factor of 10. The leak rates in the Krško specific LBB analysis were obtained using the methods suggested in NUREG/CR-3464 [9], [19].
- Loads determination. The loads to be used in the fracture mechanics stability analysis of the crack are derived from the structural analyses (section 2.1 and 2.2):
  - normal loads including thermal, deadweight and pressure loading,
  - seismic loads.
- Fracture mechanics analysis of the stability of the postulated crack. This analysis verifies that the postulated crack remains stable under the absolute sum of all the above listed postulated loads, multiplied by appropriate safety margin. In particular, local stability analysis is uses the J-integral concept, while the global stability analysis relies on the limit load
- Growth of the postulated crack due to fatigue-crack growth was demonstrated to be negligible.

The conclusion, based on the above analysis, is that the ductile reactor-coolant-system piping at Krško NPP will always show a detectable leak before the risk of a large break. According to the analysis in [7] that demonstrates the compliance of Krško NPP with all the US NRC requirements [3] and [4] with ample margins, the dynamic loads following the break of the reactor coolant piping may be neglected in the structural analysis and replaced by the dynamic loads caused by the break of the largest diameter auxiliary piping that does not qualify for the LBB concept.

3.2 LBB for safety class 1 Auxiliary piping with a diameter greater than 150 mm

The leak-before-break behavior was also verified for auxiliry piping with a diameter larger than 150 mm. The procedure used was the same as for the reactor-coolant piping. The cracks with 10 times larger than detectable leakage tend to have insufficient margin against critical crack size in piping with a diameter below 6” (150 mm).

Pomožni cevni vodi z uveljavljeno zasnovo puščanja pred zlomom (LBB) so:
- prelivni vod tlačnika (premera 305 mm (12”)),
- oba cevna voda akumulatorjev, vključno s priključkom premera 203 mm (8”) do prvega normalno zaprtega ločilnega ventila,
- oba odjemna cevna voda sistema za odvajanje zaostale toplote premera 203 mm (8”) od primarnega kroga do prvega normalno zaprtega gnanega ventila.

Obremenitve, uporabljene v analizi stabilnosti razpoke, so iz analize opisane pod 2.5.

Auxiliary lines, which qualified for LBB, are:
- The 12” (203 mm) surge line,
- both 12” (305 mm) accumulator lines, including an 8” (203 mm) branch connection up to a first normally closed isolation valve,
- both 8” (203 mm) reactor-heat removal-letdown lines from the reactor-coolant-system connection to the first normally closed motor-operated valves.

Loads taken into account in the crack stability analysis have been taken from the analysis described in section 2.5 above.

4 MEHANSKE ANALIZE KOMPONENT REAKTORSKEGA HLADILNEGA SISTEMA

V nadaljevanju povzemamo analitično preverjanje strukturne celovitosti komponent reaktorskega hladilnega sistema (reaktorska tlačna posoda, reaktorska črpalka in tlačnik). Preverjanje je bilo potrebno zaradi spremenjenih delovnih parametrov in projektnih prehodnih pojavov, povezanih s povečanjem moči in zamenjavo uparjalnikov.

4 MECHANICAL ANALYSIS OF THE REACTOR-COOLANT-SYSTEM COMPONENTS

This section summarizes the analytical verification of the structural integrity of the reactor-coolant-system components (reactor pressure vessel, reactor-coolant pump and pressurizer). The verification was necessary because of the revised operating parameters and design transients, which stem from the steam-generator replacement and power uprating.

4.1 Metoda za analize napetosti

Prvotne projektne analize jeklenih komponent reaktorskega hladilnega sistema so bile v skladu z [21] in [22] opravljene s predpostavko linearne elastičnosti. Morebitne lokalne plastične cone je v takih primerih v analizah mogoče zanemariti na račun primerne kompenzacije v dovoljenih napetostih. Velja tudi, da so spremembe delovnih parametrov in projektnih prehodnih pojavov, povezanih s povečanjem moči in zamenjavo uparjalnikov v velikostnem razredu 5 do 10%, torej relativno majhne. Zato je bilo mogoče napetosti iz prvotnih projektnih analiz spremeniti sorazmerno spremembam obremenitev. Zato v ta namen ni bila opravljena nobena nova analiza z metodo končnih elementov. Izjema so le notranji deli reaktorske posode.

4.1 Stress-Analysis Method

The original design analyses of all the steel components of the reactor coolant systems were in compliance with [21] and [22], performed under the assumption of linear elasticity. Potential local plastic zones could be neglected in the analysis and compensated within the suitably chosen allowable (word(s) missing here). The changes in the revised operating parameters and design transients, which stem from the steam-generator replacement and power uprating, are generally in the order of 5 to 10%, which is considered as relatively small. It was therefore feasible to extrapolate the stresses calculated in the original stress reports proportionally to the change in loads. No new finite-element analyses were therefore performed, with the exception of the reactor-pressure-vessel internals.

4.2 Komponente reaktorskega hladilnega sistema

Dokler so komponente obremenjene v območju linearne elastičnosti in so spremembe obremenitev majhne, daje načelo sorazmernosti natančno ekstrapolacijo rezultatov, dobljenih z metodo končnih elementov. Na podlagi te predpostavke so bili pripravljeni dodatki k projektnim specifikacijam in dodatki k poročilom o analizah napetosti za naslednje primarne komponente:
– tlačna posoda reaktorja (notranji deli so bili v celoti ponovno analizirani),
– reaktorska črpalka in
– tlačnik.

4.2 Reactor-Coolant-System Components

As long as the components are loaded in the linear-elastic range, the principle of proportionality provides for accurate extrapolation of analytical results obtained by finite-element analysis for slightly different, but comparable loading conditions. Based on the above reasoning, addenda to the design specifications as well as addenda to the component stress reports were prepared for the following primary components:
– reactor pressure vessel (the internals were re-analyzed completely),
– reactor-coolant pump,
– pressurizer.

Zaradi nekoliko večje moči in nižje temperature hladne veje reaktorskega hladilnega sistema, ki sta posledici načrtovanega delovnega okna, je bila posebna pozornost namenjena reaktorski posodi in njeni notranjosti. Nižja temperatura reaktorskega hladiva namreč pomeni večjo gostoto hladiva, kar posledično pomeni večje dinamične sile zaradi predpostavljenih zlomov cevi. Povečanje moči reaktorja pa ima še dve pomembnejši posledici, ki sta bili seveda primerno ovrednoteni:

– Notranji deli reaktorske posode so obremenjeni z notranjimi viri toplote, ki so posledica sevanja v reaktorju. Analiza toplotnih napetosti je potrdila, da so primerno projektirani tudi za nove delovne razmere.
– Nevtronsko obsevanje sten reaktorske posode na dolgi rok povečuje krhkost jekla, iz katerega je posoda izdelana. Dokazano je, da je odpornost proti krhkemu lomu zagotovljena že z izvirnimi projektnimi analizami in veljavnimi delovnimi postopki (predvsem ohlajanje in ogrevanje elektrarne).

Analize so dokumentirane v [16]. Povzamemo lahko, da bodo napetosti in utrujanje vseh primarnih komponent tudi po zamenjavi uparjalnikov in povečanju moči elektrarne še vedno znotraj vseh predpisanih omejitev [21] in [22], za nezgodna delovna stanja, kakor tudi vseh zahtev ustreznih projektnih specifikacij za opremo.

Special attention was given to the reactor vessel and the internals, mainly because of the slightly higher reactor power and the lower cold-leg temperature, which stem from the operating window. Lower temperature means a higher density of the reactor coolant, which in turn causes larger dynamic forces due to postulated pipe breaks. The power uprate has two additional important consequences, which were also evaluated:

– The reactor internals are loaded by internal heat generation rates caused by the radiation of the reactor. The stress analyses confirmed that the internals are suitably designed for the new operating conditions.
– Neutron fluxes in the reactor pressure vessel wall may cause long term embrittlement of the pressure-vessel steel. It was verified that the existing operational procedures (mainly plant cooldown and heatup) warrant adequate margins against brittle failure.

The analysis is documented in [16]. It is concluded that for all primary components the stress intensity and fatigue usage factor limits [21] and [22] for emergency conditions will also be satisfied after the steam-generator replacement and the power uprating. Also, all the requirements of the applicable original equipment specifications and the addenda specifications will be met.

5 PREGLED DOMAČIH RAZISKAV

JE Krško je za neodvisna preverjanja vseh opisanih analiz pridobila domače in tuje pooblaščene organizacije. V nadaljevanju podajamo kratek pregled domačih raziskav na področju trdnostih in mehanskih analiz. Rezultati domačih raziskav so namreč pomembno podprli neodvisna preverjanja in s tem izboljšali kakovost v tem prispevku opisanih analiz.

5 REVIEW OF DOMESTIC RESEARCH

Independent verification of all analyses described above was performed by domestic and foreign authorized institutions, engaged by Krško NPP. This section summarizes briefly the domestic research efforts in the field of structural and mechanical analyses. The results of domestic research have significantly improved the independent review and consequently improved the quality of the analyses described in this paper.

5.1 Analize vhodnih podatkov

Domači napori so se v preteklosti osredotočili v analize tistih obremenitev, ki so se glede na izkušnje v svetu ali pa na mnenje strokovne javnosti doma pokazale kot najpomembnejše. Sem nedvomno sodijo potresne obremenitve ([33] in [34]), obremenitve zaradi velike izlivne nezgode [23] in toplotne obremenitve notranjih delov reaktorja zaradi gama sevanja ([30] do [32]).

Primerna pozornost je bila posvečena tudi spremljanju morebitne krhkosti reaktorske tlačne posode zaradi obsevanja z nevtroni [29].

5.1 Analyses of Input Data

In the past domestic efforts were mainly focused on the analyses of loads, which were judged to be the most important from international experience or by domestic experts. This includes seismic loading ([33] and [34]), loads due to a large loss-of-coolant accident [23] and thermal loading of reactor internals due to gamma radiation ([30] to [32]).

Due attention was also given to the surveillance of potential embrittelment of the reactor pressure vessel because of irradiation by neutrons [29].

5.2 Trdnostne in mehanske analize

Trdnostne in mehanske analize so raziskovale obremenitve in varnostne zaloge

5.2 Structural and Mechanical analyses

Structural and mechanical analyses were devoted to providing information about the loads and najpomembnejših komponent reaktorskega hladilnega sistema (RCS): cevni vod RCS pri obremenitvah zaradi velike izlivne nezgode ([23], [28]) in potresa varne ustavitve ([24], [28]) in časovno odvisne obremenitve reaktorske tlačne posode [25] ter okrova reaktorske črpalke ([26], [27]).

safety margins of the most important components of the reactor-coolant system (RCS): the reactor-coolant-system piping loaded by the dynamic forces of a LOCA ([23], [28]), a safe-shutdown earthquake ([24], [28]), time-history analyses of the reactor pressure vessel [25] and the casing of the reactor-coolant pump ([26], [27]).

6 SKLEPI

JE Krško se je odločila izvesti obsežen niz mehanskih in trdnostnih analiz po priznanih in preverjenih postopkih v podporo zamenjavi uparjalnikov in povečanju moči. Vse uporabljene analitične metode so že bile odobrene v ZDA kakor tudi v številnih evropskih državah, ki imajo jedrske elektrarne, izdelane po tehnologiji, primerljivi z JE Krško. Niz trdnostnih in mehanskih analiz, opisanih v tem prispevku, dokazuje strukturno celovitost tlačne meje reaktorskega hladilnega sistema po zamenjavi uparjalnikov in povečanju moči za 6,3 %. Kakovost opravljenih analiz dokazujejo tudi neodvisna preverjanja domačih pooblaščenih organizacij.

6 CONCLUSIONS

Krško NPP decided to perform a comprehensive set of mechanical and structural analyses with well-established and verified methods to support the steam-generator replacement and power uprating. All implemented methods were approved in the United States as well as in a number of EU countries with reactors of similar design to Krško NPP. The set of structural analyses described above, verifies the structural integrity of the reactor-coolant pressure boundary after the replacement of steam generators and a power uprating of 6.3% at Krško NPP. The quality of the analyses performed was proven by independent verification of the domestic and foreign authorized institutions.

7 OKRAJŠAVE

FW – glavna napajalna voda, LB – velik zlom, LBB – puščanje pred zlomom, LOCA – izlivna nezgoda, MS – glavni parni vod, JEK – nuklearna elektrarna Krško, US NRC – jedrski upravni organ v ZDA, RCS – reaktorski hladilni sistem.

7 ABBREVIATIONS

FW – Feed Water, LB - Large Break, LBB – Leak Before Break, MS – Main Steam, NPP – Nuclear Power Plant, LOCA – Loss-of-Coolant Accident, US – United States, NRC- Nuclear Regulatory Commission, RCS – Reactor Coolant System

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Termohidravlične varnostne analize v podporo zamenjavi uparjalnikov in povečanju moči v jedrski elektrarni Krško

Thermal-Hydraulic Safety Analyses Supporting the Steam Generator Replacement and Uprating at Krško Nuclear Power Plant

Borut Mavko - Andrej Prošek

Jedrska elektrarna Krško izvaja pomemben projekt modernizacije. Glavi nameni projekta so: stabiliziranje dolgoročnega delovanja elektrarne, povečanje električne moči ter večja razpoložljivost in varnost elektrarne. Modernizacija je zahtevala ponovno temeljito ovrednotenje varnosti in s tem nove termohidravlične, mehanske in trdnostne analize. Narejene so bile vse termohidravlične varnostne analize, potrebne za zamenjavo uparjalnikov in povečanje moči. Analize so bile tudi neodvisno preverjene kakor to zahtevajo domači in mednarodni predpisi in standardi (ameriške uprave za jedrsko varnost). Analize in njihovo neodvisno preverjanje so se posebej osredotočile na hipotetične razmere med delovanjem elektrarne, ki bi bili najbolj kritični za delovno okno. Mejni prehodni pojavi in nezgode so bili analizirani in so opisani v posodobljenem varnostnem poročilu. Rezultati varnostnih analiz in neodvisnega varnostnega pregleda kažejo, da bodo glavni cilji projekta modernizacije lahko izpolnjeni in da bodo prispevali k zanesljivemu in bolj varnemu dolgoročnemu delovanju elektrarne.
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(Ključne besede: varnost jedrska, analize varnosti, kriteriji varnosti, modeli za vrednotenje)

The Krško Nuclear Power Plant has undertaken a major modernization project. The objectives of the project are: long-term stabilization of the plant's operation, uprating of the net electrical power output, higher availability and enhanced safety of the plant. The modernization also requires a thorough safety re-evaluation and therefore new thermal hydraulic, mechanical and structural analyses. The thermal-hydraulic part of the safety analyses necessary for the steam generator replacement and the power uprating were performed and independently reviewed according to Slovenian and US NRC (United States Nuclear Regulatory Commission) requirements. The analysis and the review focused on the plant conditions assumed to be the most critical for the operating window. The limiting transient and accident cases were identified and described in an updated safety analysis report. Results of the safety analysis and independent safety review indicate that the objectives of the Krško modernization project can be met and will contribute to the plant's safe and reliable long-term operation.
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(Keywords: nuclear safety, safety analysis, safety criteria, evaluation models)

0 UVOD

Najpomembnejša dela projekta modernizacije jedrske elektrarne v Krškem (JEK) sta zamenjava obeh uparjalnikov in povečanje toplotne moči s 1882 MW na 2000 MW. Zamenjava uparjalnikov in povečanje moči spreminjata osnovne obratovalne parametre elektrarne, tj. elektrarna bo delovala znotraj novega delovnega okna. V primerjavi s prvotnimi, imajo novi uparjalniki drugačno geometrijsko obliko, snovne lastnosti in hidravlične značilnosti. Ker vse spremembe in prilagoditve vplivajo na prvotno in sedaj veljavno

0 INTRODUCTION

The Krško Nuclear Power Plant (NPP) modernization project stipulates a replacement of both the plant's steam generators and an uprating of its thermal power from 1882 MW to 2000 MW. The replacement of the steam generators and the power uprating affect the current primary operating parameters, i.e. the plant will operate inside a new operating window. In addition the new steam generators have a different geometry, material properties and different hydraulic characteristics. All the changes and modifications impact on the original and current licenčno in projektno dokumentacijo, so bile potrebne nove varnostne analize in ocene. Te analize so morale potrditi, da bo elektrarna tudi po izvedeni modernizaciji delovala varno. Ponovno ocenjevanje varnosti je obsegalo termohidravlične, mehanske in trdnostne vidike sprememb, vpeljanih zaradi projekta posodobitve.

Ta prispevek obravnava: termohidravlične varnostne analize, ki so jih opravili jedrska elektrarna Krško in njeni dobavitelji; varnostne kriterije, ki jih je bilo treba izpolniti s projektom posodobitve in domače analize, ki so bile narejene v podporo neodvisnemu preverjanju.

Varnostne analize za elektrarno z novimi uparjalniki in s povečano močjo morajo potrditi, da med prehodnimi pojavi in nezgodami vsi pogoji ostanejo znotraj meja in kriterijev sprejemljivosti za delovno okno. Izvirne analize so bile narejene za eno obratovalno stanje, medtem ko nove analize pokrivajo delovno okno. Novi obratovalni podatki upoštevajo tudi 0 in 5% začepitev cevi uparjalnika [1]. Delovno okno, ki definira najnižjo in najvišjo povprečno temperaturo in pretok hladiva, bo prispevalo k delovni prilagodljivosti elektrarne, potrebni pri spremembi parametrov med zamenjavo goriva.

V skladu s slovenskimi predpisi mora biti status elektrarne glede varnosti dokumentiran v končnem varnostnem poročilu ([2] in [3]). Potrebno vsebino in obliko varnostnega poročila določa domača zakonodaja [4] (v ZDA je to v 10 CFR 50.34 [5]). V končnem varnostnem poročilu za jedrsko elektrarno Krško so zato zbrane informacije o napravi, predstavljene so projektne osnove in delovne omejitve ter analize struktur, sistemov in komponent ter analize obnašanja elektrarne med predpostavljenimi prehodnimi pojavi ali nezgodami. Najpomembnejše informacije o odzivu elektrarne na prehodne pojave in hipotetične nezgode so zbrane v 6. in 15. poglavju končnega varnostnega poročila.

V tem prispevku smo se osredotočili na kritične prehodne pojave in predpostavljene projektne nezgode. Domnevni kritični primeri so izbrani na podlagi delovnega okna in mejnih primerov, opisanih v sedaj veljavnem posodobljenem varnostnem poročilu JEK (USAR) [6]. Termohidravlične varnostne analize, ki so narejene v skladu z navodili ameriške Uprave za jedrsko varnost [7], morajo izpolniti varnostne kriterije, ki so opisani v nadaljevanju. Na koncu so predstavljena tudi domača prizadevanja za izboljševanje jedrske varnosti, ki strokovno podpirajo neodvisno preverjanje opravljenih analiz.

1 VARNOSTNE ANALIZE V SKLADU Z USNRC R.G. 1.70

Program analiz za projekt posodobitve jedrske elektrarne Krško vsebuje tudi izbrane spremembe metodologij glede na sedanje licenčne

licensing and design basis documentation; therefore new safety analyses and assessments are required to prove that the plant will be able to operate safely. The safety reassessment and analyses cover thermal-hydraulic (TH), mechanical and structural aspects of the modifications introduced by the modernization project.

This paper deals predominately with: the thermal-hydraulic safety analyses performed by Krško NPP and its contractors, the safety criteria that had to be met by the modernization project and the national analyses to support the independent review.

The analyses performed for the plant with new steam generators and at uprated power need to prove that all transient and accident conditions remain within the limits and acceptance criteria for the operating window. The original analyses were performed for one operating condition only, while the new analyses covers an operating window. The new operating parameters were determined taking into account 0 and 5% plugging of the steam generator tubes [1]. The operating window hence defines the minimum and the maximum average coolant temperature and reactor coolant system flow. It adds to the plant the flexibility need to change the operating parameters during each refueling.

According to Slovenian regulations ([2] and [3]) the nuclear power plant safety status must be documented in a Final Safety Analysis Report (FSAR). The minimum information required to be included and the format is established by regulatory guidelines [4] (for US in 10 CFR 50.34 [5]). The Krško NPP FSAR therefore contains information that describes the facility, presents the design basis and the limits on its operation, and presents analyses of the structures, systems, and components and postulated accident analysis of the facility as a whole.

The relevant information related to the thermal hydraulic transient and accident analysis is included predominately in Chapters 6 and 15 of the safety analysis report. This overview focuses on critical transients and postulated accidents. The selection of the cases which are assumed to be critical is based on the selected operating window and on the limiting cases described in the current Updated Safety Analysis Report (USAR) [6]. The thermal-hydraulic safety analyses, which are performed according to USNRC R.G. 1.70 [7], must meet safety criteria, which will also be described. Finally, national efforts to improve nuclear safety and which support the assessment are also presented.

1 ACCIDENT ANALYSES ACCORDING TO THE USNRC R.G. 1.70

The analyses program for the Krško NPP modernization project includes selective methodology changes to the existing licensing basis. The acosnove. Varnostne analize sledijo priporočilom ustreznega upravnega navodila [7]. To navodilo vsak začetni dogodek, prehodni pojav ali projektno nezgodo, razvršča v eno izmed naslednjih skupin:

1. povečano odvajanje toplote s sekundarnim sistemom,
2. zmanjšano odvajanje toplote s sekundarnim sistemom,
3. zmanjšanje pretoka v reaktorskem hladilnem sistemu,
4. nenormalne spremembe reaktivnosti in porazdelitve moči,
5. povečanje količine hladiva v reaktorskem hladilnem krogu,
6. zmanjšanje količine hladiva v reaktorskem hladilnem krogu,
7. sproščanje radioaktivnih snovi,
8. pričakovani prehodni pojavi brez zaustavitve reaktorja.

Za vsak začetni dogodek je treba najprej ugotoviti, kako pogosto bi se lahko pojavil, ali kakšna je verjetnost, da bi do njega prišlo. Začetni dogodki so glede na njihovo statistično verjetnost razvrščeni v štiri skupine. Po razvrstitvi Ameriškega društva jedrskih strokovnjakov (ANS) [8] so projektna stanja elektrarne razdeljena v štiri skupine v skladu s pričakovano pogostostjo dogodkov in njihovimi potencialnimi radiološkimi posledicami za prebivalstvo. Ta štiri stanja so naslednja:

I: normalno obratovanje in prehodni pojavi,
II: zmerno pogoste okvare,
III: redke okvare,
IV: mejne nezgode.

Osnovna projektna načela zahtevajo, da morajo stanja z najbolj verjetnimi dogodki pomeniti najmanjše mogoče radiološko tveganje za prebivalstvo in naj imajo hipotetični dogodki s potencialno največjim tveganjem za prebivalstvo najmanjšo mogočo verjetnost, da bi se pojavili.

Radioaktivna snov v sredici po daljšem delovanju reaktorja je namreč glavni vir potencialnega radioaktivnega izpusta. Sama zasnova sredice, reaktorskega hladilnega sistema in zadrževalnega hrama pa zagotavlja varovanje pred radioaktivnimi izpusti, da ne bi le ti med nezgodami presegli dovoljene meje. Prav varnostne analize so tiste, ki zagotovijo in hkrati potrdijo varne projektne zasnove vsake jedrske elektrarne. Zagotovijo torej, da je zasnova elektrarne takšna, da izpolnjuje vse predpisane meje glede radioaktivnih izpustov in doz v vsakem od obratovalnih stanj.

Dogodki stanja I so tisti, ki so pričakovani pogosto ali redno v času obratovanja, menjave goriva, vzdrževanja ali med spreminjanjem moči. Za dogodke stanja I obstaja zadostna rezerva med obratovalno vrednostjo poljubnega parametra elektrarne in med tisto vrednostjo tega parametra, ki terja avtomatski ali ročni poseg.

Zmerno pogoste okvare stanja II v najslabšem primeru povzročijo zaustavitev reaktorja. Po tem je elektrarno mogoče ponovno zagnati na

cident analyses follow the recommendations of the relevant regulatory guide [7]. According to this guide each postulated initiating event, transient or accident, is assigned to one of the following categories:

1. Increase in heat removal by the secondary system,
2. Decrease in heat removal by the secondary system,
3. Decrease in reactor coolant system flowrate,
4. Reactivity and power distribution anomalies,
5. Increase in reactor coolant inventory,
6. Decrease in reactor coolant inventory,
7. Radioactive release from a subsystem or component,
8. Anticipated transients without scram.

For each initiating event its frequency of occurrence is discussed. Each initiating event is then classified into one of the frequency groups. The American Nuclear Society (ANS) classification of plant conditions [8] divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four conditions are as follows:

I: Normal operation and operational transients,
II: Faults of moderate frequency,
III: Infrequent faults,
IV: Limiting faults.

The basic principle applied for design requires that the conditions with the most probable occurrence should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur.

The radioactive material present in the core after any long period of reactor operation is the main source of potential radioactive release. The design of the core, the reactor coolant system and the containment provide protection against radioactive release above acceptable limits under accident conditions. The safety analysis is required to establish and confirm the design basis of a nuclear power plant and to ensure that the plant design is capable of meeting the prescribed limits on radiation releases and doses for each plant operating condition.

Condition I occurrences are those which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with a margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action.

The faults of moderate frequency, Condition II, at worst, result in a reactor trip with the plant being capable of returning to power. By definition, these faults (or events) do not lead to a more serious polno moč. Po definiciji se te nezgode (dogodki) ne razvijejo v tako smer, da bi povzročile še hujše nezgode (dogodke stanja III ali IV). Pri okvarah, uvrščenih v stanje II, se ne pričakuje, da bi povzročile poškodbe goriva ali previsok tlak v reaktorskem hladilnem krogu (RCS) ali sekundarnem sistemu.

Dogodki stanja III so po definiciji tiste okvare, ki se lahko le zelo redko zgodijo v življenjski dobi obstojnosti elektrarne. Pri redkih okvarah bo poškodovan samo majhen del gorivnih palic. Radioaktivni izpust bo tako majhen, da ne bo motena ali prepovedana javna uporaba področij zunaj ožjega nadzorovanega področja. Nezgode stanja III same po sebi ne bodo povzročile dogodka stanja IV. Prav tako tak dogodek ne bo onesposobil pregrade reaktorskega hladilnega sistema in zadrževalnega hrama.

Mejne nezgode stanja IV so hipotetične nezgode, ki v življenjski dobi niso pričakovane, jih pa pri projektiranju predpostavimo, ker bi potencialno lahko povzročile izpust znatnih količin radioaktivnih snovi. To so najbolj resni dogodki, proti katerim je zavarovana elektrarna in pomenijo mejne projektne primere. Ena sama nezgoda stanja IV tudi ne bi povzročila izgube zasilnega hlajenja sredice in zadrževalnega hrama.

Pri varnostnem vrednotenju vsakega dogodka je poleg vzrokov in razvrstitve po pogostosti treba analizirati tudi začetne pogoje, zaporedje dogodkov in delovanje sistemov. Informacije o sredici in delovanju sistemov, o učinku pregrad in o radioloških posledicah so seveda različne in so odvisne od obravnavanega začetnega dogodka.

2 NAJPOMEMBNEJŠI VARNOSTNI KRITERIJI

Ocenjevanje in preverjanje analiz prehodnih pojavov in potekov nezgod ter njihovih posledic mora slediti načrtu za standardni pregled (SRP) [9]. Vsak od prehodnih pojavov, ki jih obravnava 15. poglavje SRP, mora biti opisan tudi v ustreznem poglavju končnega varnostnega poročila, kakor to zahteva USNRC R.G. 1.70. Najpomembnejše varnostne zahteve – kriteriji sprejemljivosti, povzeti iz ameriškega zveznega zakona (CFR) in načrta pregleda SRP [9], so zbrani v preglednici 1 (glej tudi [10]). Za izlivne nezgode (LOCA) morajo biti ti kriteriji izpolnjeni, ne glede na to, ali pri analizi uporabimo konzervativni model (EM) [11] ali pa model za najboljšo oceno (BE) [12]. Pri uporabi modela BE, je treba upoštevati tudi negotovost izračunanih rezultatov, ko delovanje sistema za hlajenje sredice primerjamo s kriteriji sprejemljivosti.

Kriteriji sprejemljivosti temeljijo na izpolnjevanju predpisanih zahtev, zbranih večinoma v prilogah A in K k zakoniku 10 CFR 50, v 10 CFR 50.46 [12] in 10 CFR 100 [13]. V prilogi A k zakoniku 10 CFR 50 [14] so določeni splošni projektni kriteriji za jedrske elektrarne. Da bi se

fault (Condition III or IV events) and are not expected to result in fuel rod or reactor coolant system (RCS) failures or secondary system overpressurization.

Condition III occurrences are faults which may occur very infrequently during the life of the plant. These infrequent faults will be accommodated with the failure of only a small fraction of the fuel rods. The release of radioactivity will not be sufficient to interrupt or restrict public use of areas beyond the exclusion area boundary. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant or the containment barriers.

Limiting faults are faults which are not expected to take place, but are postulated because their consequences would include the potential of the release of significant amounts of radioactive material. They are the most drastic occurrences which must be designed against and represent limiting design cases. A single Condition IV fault must not cause a consequential loss of the required functions of systems needed to mitigate the consequences of the fault including those of the Emergency Core Cooling System (ECCS) and the containment.

For each event evaluation, besides the identification of causes and frequency classification, initial conditions, sequence of events and system operation should be discussed. The information on core and system performance, barrier performance and radiological consequences will differ for the various initiating events.

2 SAFETY CRITERIA OF PRIMARY INTEREST

The assessment and review of transients and accident progression and their consequences should follow the Standard Review Plan (SRP) [9]. Each transient covered by Chapter 15 in the SRP should be discussed in a separate section of the safety analysis report, as required by the USNRC R.G. 1.70. The safety requirements of primary interest, according to the US Code of Federal Regulations (CFR) and the Standard Review Plan [9], are summarized in Table 1 (see also ref. [10]). For loss-of-coolant accidents (LOCA) and transients these criteria are stated for both, i.e. for the evaluation model (EM) [11] and the best estimate (BE) [12] analysis. When the BE approach is used, the estimated uncertainty in calculated results must be accounted for when the ECCS performance is compared to the acceptance criteria and App. K requirements are not applicable.

Acceptance criteria are based on meeting the relevant requirements of regulations, mostly Appendix A and K to 10 CFR 50, 10 CFR 50.46 [12] and 10 CFR 100 [13]. In the Appendix A to 10 CFR 50 [14] general design criteria for nuclear power plants izpolnili ti splošni kriteriji, je treba spoštovati specifične kriterije, naštete v preglednici 1.

Tudi kriteriji sprejemljivosti so odvisni od pogostosti nezgod. Izpolnjevanje predpisov se analitično dokaže z različnimi kakovostnimi (dolgoročno hlajenje in varna ugasnitev) in kolikerostnimi merili (npr.: razmerje do krize vrenja - DNBR > meje za DNBR, najvišja temperatura srajčke < 1478 K, največji tlak primarnega sistema ali zadrževalnega hrama < projektnega tlaka itn). Npr.: kriterij za DNBR velja za zmerno pogoste okvare. Če je razmerje do krize vrenja dovolj veliko, je prenos toplote med gorivno srajčko in reaktorskim hladivom dovolj velik, da prepreči poškodbo srajčke, ki bi nastala zaradi neustreznega hlajenja.

are set. To meet these design criteria, specific criteria, listed in Table 1 need to be met.

The compliance standards are related to accident frequency. Compliance is analytically demonstrated through various qualitative (long-term cooling and safe shutdown) and quantitative measures (departure from nucleate boiling ratio (DNBR) > DNBR limit, peak cladding temperature < 1478 K, peak primary system or containment pressure < design pressure). For example, the DNBR criterion is applicable for faults of moderate frequency. By preventing departure from nucleate boiling, adequate heat transfer is assured between the fuel cladding and the reactor coolant, preventing cladding damage as a result of inadequate cooling.

Preglednica 1. *Pomembni varnostni kriteriji za termohidravlične varnostne analize*
Table 1. *Important safety criteria for thermal-hydraulic safety analyses*

Vir **Source**	**Kriteriji** **Criteria**			
10 CFR 1.11	varovanje človekovega življenja in zdravja ter varstvo okolja itn. protect public health and safety, the environment etc.			
10 CFR 100	omejiti sproščanje radioaktivnih snovi* limit fission product release*			
Priloga A k 10 Appendix A to 10 CFR 50	omejiti poškodbe goriva limit fuel failure		omejiti puščanje iz reaktorskega hladilnega kroga limit RCS breach	omejiti puščanje zadrževalnega hrama limit containment breach
SRP 6.2				omejiti tlak in temperaturo zadrževalnega hrama, puščanje, vodik itn. limit containment pressure and temperature, leakage, hydrogen etc.
SRP 15.1.1 do/to 15.6.1 (ne-LOCA) (non-LOCA)		DNBR, temperatura goriva, napetost srajčke, raztezek srajčke DNBR, fuel temperature, cladding stress, cladding strain	prekomerni tlak, napetostne omejitve overpressure criteria, primary stress limit	
10 CFR 50.46 in/and SRP 15.6.5 (LOCA)	ECCS kriteriji ECCS criteria			
SRP 15.8 (ATWS)		DNBR	napetostne omejitve primary stress limit	

* Velja za mejne projektne nezgode in dogodke iz sedme skupine. Radiološke posledice se ocenjujejo v skladu z zakonikom 10 CFR 100.

* Applicable for design basis accidents and Category 7 events. The radiological consequences are evaluated according to 10 CFR 100.

Definicije kriterijev, uporabljenih v preglednici 1:
Kriterij za čezmerni tlak – Tlak v reaktorskem hladilnem sistemu in sistemu glavne pare mora biti vzdrževan pod 110% vrednosti projektnega tlaka [15].
DNBR – gorivna srajčka naj se ne poškoduje, če zagotovimo, da je najnižja meja za DNBR nad vrednostjo 95/95 DNBR meje za reaktorje PWR.
Radiološki – 10 CFR 100 kriteriji.

Definitions of criteria used in Table 1:
Overpressure criteria – Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (ref. [15]).
DNBR – Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs.
Radiological – 10 CFR 100 criteria.

ECCS kriteriji [12] – jih je pet: najvišja temperatura srajčke (< 1478 K), največja stopnja oksidacije srajčke (< 17%), največja tvorba vodika (< 1%), geometrijska oblika, ki zagotavlja hlajenje in dolgoročno ohlajanje.
Napetostne omejitve – meja za tlak stopnje C je najvišji tlak. Med vsakim delom zvišanja tlaka mora biti tlak nižji od vrednosti tlaka, ki bi povzročil napetost, ki bi presegla mejo stopnje C, kakor je definirana v standardih ASME [16].

3 PREGLED KRITIČNIH NEZGOD

Nezgode, za katere domnevamo, da so kritične, so bile izbrane na podlagi delovnega okna in mejnih primerov, opisanih v zdaj veljavnem posodobljenem varnostnem poročilu [6]. Analize so bile narejene za najbolj konzervativne začetne pogoje delovnega okna in z upoštevanjem različnih začepljenosti uparjalnika (0 in 5%). V skladu z ameriškimi navodili [7] morajo biti parametri in začetni pogoji primerno konzervativni za analizirani dogodek. Za analizo večine neizlivnih prehodnih pojavov je bil uporabljen računalniški program LOFTRAN [17], namenjen za simuliranje reaktorske kinetike, reaktorskega hladilnega kroga, tlačnika, uparjalnikov in sistema napajalne vode. Za preračune DNBR sta bila uporabljena računalniška programa THINC-3 in FACTRAN. Metodologija, uporabljena za izvirno analizo velike izlivne nezgode, je bila uporabljena tudi pri analizi te nezgode za elektrarno s povečano močjo. Metodologija vključuje več računalniških programov, med njimi tudi programa BASH in LOCBART. Za malo izlivno nezgodo (SB LOCA) je bil uporabljen računalniški program NOTRUMP. Obe analizi izlivnih nezgod sta utemeljeni na konzervativnem modelu priloge K [11].
Kot kritične so bile izbrane naslednje nezgode:
– rezerva do krize vrenja (DNB),
– rezerva za izlivne nezgode (LOCA),
– dogodki s segrevanjem (odvajanje zakasnele toplote),
– varovanje pred čezmernim tlakom,
– odziv sredice na zlom parovoda,
– celovitost zadrževalnega hrama.

Rezultati analiz vseh teh nezgod so na kratko opisani tudi v posebnem povzetku projektne dokumentacije JEK [18].

3.1 Rezerva do krize vrenja (DNB)

Analiza DNB pokriva nastavitve sistema za zasilno zaustavitev reaktorja zaradi prekoračitev temperature (OTΔT) in prekoračitve moči (OPΔT). Zaustavitev reaktorja na ΔT prekoračitev temperature varuje elektrarno pred krizo vrenja, ki močno zmanjša toplotno prestopnost med gorivnimi palicami in reaktorskim hladivom, kar povzroči visoke temperature srajčke ali celo njih prežig. Zaustavitev reaktorja na ΔT prekoračitev moči varuje gorivo pred preveliko vzdolžno proizvodnjo moči (preveč

ECCS criteria [12] – there are five criteria: peak cladding temperature (< 1478 K), maximum cladding oxidation (< 17%), maximum hydrogen generation (< 1%), coolable geometry and long-term cooling.
Primary stress limit - pressure "service limit C" is the maximum pressure. During any portion of the assumed excursion the pressure should be less than the value that will cause stress to exceed the "service Limit C" as defined in the ASME code [16].

3 OVERVIEW OF THE CRITICAL ACCIDENTS

The selection of cases, assumed to be critical, was based on the operating window and the limiting cases from the current Updated Safety Analysis Report [6]. The analyses were performed for the most conservative initial conditions from the operating window and steam generator tube plugging (0 and 5%). According to USNRC R.G. 1.70 [7] the parameters and initial conditions used in the analyses should be suitably conservative for the event being evaluated. For most of the non-LOCA transients the computer code LOFTRAN was used [17], which is intended to simulate the plant thermal kinetics, RCS, pressurizer, steam generators, and the feedwater system. The DNBR calculations were done by the computer codes THINC-3 and FACTRAN. The methodology used in the original large-break (LB) LOCA analysis was also used for the uprated design. It was performed with a combination of codes, including the codes BASH and LOCBART. For small-break (SB) LOCA the computer code NOTRUMP was used. Both LOCA analyses were based on the Appendix K [11] evaluation model.
The accidents selected to be critical accidents were:
– margin to DNB,
– LOCA margins,
– heatup events (decay heat removal),
– overpressure protection,
– steam line break core response,
– containment integrity.

These accidents are briefly reviewed in the following subsections [18].

3.1 Margins to DNB

The DNB analysis covers the overtemperature ΔT (OTΔT) and over power ΔT (OPΔT) setpoints. The over temperature ΔT is designed to protect against departure from nucleate boiling which causes a large decrease in the heat transfer coefficient between fuel rods and the reactor coolant, resulting in high fuel clad temperatures. Overpower ΔT is designed to protect against a high fuel rod power density (excessive kW/m) and subsequent kW/m) in s tem poškodbo srajčke in taljenje goriva. Z analizo sta bila potrjena oba kriterija sprejemljivosti, meja za DNBR in temperatura goriva v osi, ki ne sme preseči temperature tališča.

Analiza izvleka krmilnega svežnja (RCCA) mora potrditi, da je sredica primerno varovana z zasilno zaustavitvijo reaktorja na OTΔT. V analizi je treba upoštevati vse nivoje moči. Ugotovljeno je bilo, da se mejni primer pojavi pri 100% moči. Analiza nezgode z izgubo pretoka hladiva (CLOF), ki je mejni primer za DNB, mora dokazati, da DNBR ostane nad predpisano varno mejo. Izpolnjen mora biti tudi kriterij za čezmerni tlak. Vse analize v zvezi z mejo za DNB so pokazale, da so kriteriji sprejemljivosti izpolnjeni.

3.2 Rezerva za izlivne nezgode (LOCA)

Analize velike in male izlivne nezgode se naredijo zato, da dokažejo zadostno zmogljivost sistema za zasilno hlajenje sredice, ki mora učinkovito ohladiti sredico in po izlivni nezgodi vzdrževati reaktor v varnem stanju hladne ugasnitve. Rezultati analiz morajo izpolniti kriterije sprejemljivosti za zasilno hlajenje sredice [12] in [19]. Delovanje sistema za zasilno hlajenje sredice mora biti potrjeno za številne izlivne nezgode različnih velikosti zlomov, mest in drugih lastnosti. Dodatno mora biti zagotovljeno, da so bili analizirani tudi najbolj kritični primeri velikih izlivnih nezgod. Predpisi tudi zahtevajo, da mora biti uporabljen konzervativni model, narejen v skladu s prilogo K [11].

Za analize najpomembnejše lastnosti konzervativnega modela so:

- začetna moč je 102% dovoljene moči;
- največji faktor temenske moči, ki ga dovoljujejo tehnične specifikacije;
- zakasnela toplota ustreza 120% razpadu cepitvenih delcev, ki ga predpisuje standard ANS za neskončen čas obratovanja [20];
- kemijske reakcije med vodo in kovino se računajo z Baker-Justovo enačbo;
- za dvofazni tok se uporabi Moodyjev model z vsaj tremi iztočnimi koeficienti.

Velika izlivna nezgoda (LB LOCA) je bila analizirana za primer z iztočnim koeficientom C_D=0,4 (40% giljotinski zlom), za katerega se predpostavlja, da je najbolj kritičen za različne moči in povprečne temperature reaktorskega hladilnega kroga. Pri mali izlivni nezgodi (SB LOCA) so bili analizirani zlomi velikosti 3,81 cm (1,5 palca), 5,08 cm (2 palca), 7,62 cm (3 palce), 10,16 cm (4 palce) in 15,24 cm (6 palcev). Ko se je zaustavil reaktor, se je predpostavilo izpad zunanjega električnega napajanja in črpalke reaktorskega hladiva (RCP). Izračuni s programom NOTRUMP in konzervativnim modelom, kakršen je predpisan v prilogi K, so potrdili, da so kriteriji sprejemljivosti izpolnjeni.

fuel rod cladding failure and fuel melt. Both acceptance criteria, DNBR limit and fuel centerline temperature not exceeding the melting point, were checked by the analysis.

The analysis of the Rod Cluster Control Assembly (RCCA) withdrawal at power is performed to prove that the core is adequately protected by the OTΔT protection. The analysis must consider all power levels. The limiting case found was 100% power. The analysis of complete loss of flow (CLOF) that is limiting for DNB must confirm that DNBR remains above the limit. The over pressure criteria must also be fulfilled. All analyses related to the margin to DNB fulfilled the acceptance criteria.

3.2 LOCA margins

The large- and small-break LOCA analyses are performed to demonstrate the capability of the ECCS to effectively recover the core and maintain the reactor in a safe shutdown condition following a LOCA. Results of the analyses must meet the ECCS design criteria [12] and [19]. ECCS cooling performance must be proven for a number of LOCAs of different break sizes, locations and other properties. In addition, a sufficient assurance should be provided that the most severe postulated LOCAs are considered by the analysis. It is also required that an ECCS evaluation model is developed in conformance with the required and acceptable features of the Appendix K [11] ECCS evaluation model.

In the current safety analyses the original LOCA methodology was used, with the evaluation model based on Appendix K requirements [11]. The features from Appendix K that are considered the most significant for the analysis are:

- initial power at 102% of licensed power,
- maximum peaking factor allowed by the technical specifications,
- decay heat based on 120% of fission product decay rate specified by the ANS Standard for infinite operating time [20],
- the Baker-Just equation shall be used to calculate the metal-water reaction rate,
- the Moody model, with at least three discharge coefficients, shall be used for two-phase break flow.

For a LB LOCA the case with C_D=0.4 (40% guillotine break) supposed to be critical was analyzed for different power profiles and RCS average temperatures. For a SB LOCA the break sizes 3.81 cm (1.5 inch), 5.08 cm (2 inch), 7.62 cm (3 inch), 10.16 cm (4 inch) and 15.24 cm (6 inch) were analyzed. Loss of offsite power and reactor coolant pump (RCP) trip at the time of reactor trip were assumed. The calculation with NOTRUMP code and the Appendix K evaluation model for SB LOCA confirmed that the acceptance criteria were met.

3.3 Dogodki s segrevanjem

Kot dogodka, ki povzročita neželeno segrevanje, sta bila obravnavana izguba pretoka glavne napajalne vode (LONF) in zlom cevi napajalne vode (FLB). Razloga za izgubo pretoka glavne napajalne vode sta dva: okvara črpalke za napajalno vodo ali okvara ventila. Izguba pretoka glavne napajalne vode povzroči povišanje temperature in tlaka reaktorskega hladiva, kar na koncu pripelje do zaustavitve reaktorja, da bi se preprečila poškodba goriva.

Analizirana je bila tudi nezgoda, ki jo povzroči zlom cevi napajalne vode (FLB). Odvisno od velikosti zloma in obratovalnega stanja elektrarne v času zloma lahko pride do prevelikega ohlajanja reaktorskega hladilnega kroga zaradi čezmernega iztoka energije skozi zlom ali pa do segrevanja reaktorskega hladilnega kroga.

Analiza nezgode, ko elektrarna izgubi dovod glavne napajalne vode, je pokazala, da sta izpolnjena kriterija sprejemljivosti za čezmerni tlak in DNBR in dodatni Westinghouseov kriterij, ki omejuje prenapolnjenje tlačnika. Analiza zloma glavne napajalne vode je pokazala, da so izpolnjeni tudi dodatni radiološki kriteriji.

3.4 Varovanje pred čezmernim tlakom

Prehodni pojav velike izgube bremena, ki ga lahko povzročita ali izguba zunanjega bremena ali zaustavitev turbine, pripelje do hitrega zvišanja temperature in tlaka primarnega sistema. Analiza tega pojava mora pokazati, da so varnostni ventili tlačnika in uparjalnika dovolj veliki, da preprečijo pojav čezmernega tlaka v reaktorskem hladilnem krogu in v uparjalnikih.

Analizirana sta bila primera z nizko in visoko začetno povprečno temperaturo hladila. Rezultati analiz so pokazali, da je zagotovljeno varovanje pred čezmernim tlakom zadostno. Tlak v reaktorskem hladilnem krogu in tlak v sistemu glavne pare med prehodnim pojavom ostaneta pod zakonsko določeno mejo 110% projektnega tlaka v teh sistemih. Ugotovitev velja za celotno območje začetnih nivojev tlačnika, kar omogoča prožnost pri izbiri programa za nivo tlačnika v prihodnosti.

3.5 Odziv sredice na zlom parovoda

Vodni pari, ki izteka skozi zlom glavnega parovoda, se v začetku poveča pretok, pozneje pa se z nižajočim tlakom tudi pretok manjša. Odvajanje energije iz reaktorskega hladilnega kroga posledično zniža temperaturo hladiva in tudi tlak. Ko je temperaturni koeficient reaktivnosti moderatorja negativen, to ohlajanje povzroči zmanjšanje rezerve ugasnitve. Če predpostavimo, da bi se hkrati zataknil najbolj reaktivni krmilni sveženj v popolnoma izvlečenem položaju, bi obstajala povečana verjetnost, da bi sredica ponovno postala kritična, kar bi povzročilo ponovni dvig moči. Ponovna vrnitev

3.3 Heatup events

The considered critical heatup events were loss of normal feedwater (LONF) flow and feedline break (FLB). Causes for LONF flow are feedwater pump failure and valve failure. Loss of feedwater flow results in an increase in reactor coolant temperature and pressure, which eventually requires a reactor trip to prevent fuel damage. It is assumed that the auxiliary feedwater starts automatically.

The cause of a FLB transient is a break in the feedwater line. Depending upon the size of the break and the plant operating conditions at the time of the break, the feedline break could cause either a RCS cooldown due to excessive energy discharge through the break, or a RCS heatup.

The analysis of LONF flow showed that the acceptance criteria, namely the overpressure and DNBR criteria and the Westinghouse criterion requiring that the pressurizer does not fill up, were met. For FLB additional radiological criteria were also met.

3.4 Overpressure protection

Following a major loss of load, resulting either from a loss of external electrical load or from a turbine trip, the primary temperature and pressure may increase rapidly. This analysis must confirm that the pressurizer and steam-generator safety valves are adequately sized to prevent overpressurization of the RCS and the steam generators. The cases with the low and high average temperature are analyzed.

The results of these analyses showed that the provided overpressure protection is sufficient to maintain peak RCS pressure and the main steam system pressure below the code limit of 110% of the respective system design pressure. This is demonstrated over the full range of initial pressurizer levels allowing for future flexibility in choosing an appropriate pressurizer level program.

3.5 Steam line break core response

The steam release arising from a rupture of the main steam line would result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of the coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cool-down results in a reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position after a reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a na moč po zlomu parovoda je potencialni problem predvsem zaradi visokih temenskih faktorjev moči, ki so opazni pri popolnoma izvlečenem najbolj reaktivnem krmilnem svežnju. V takem primeru mora sredico ugasniti borova kislina, ki jo vbrizga sistem za varnostno vbrizgavanje.

Analiza zloma glavnega parovoda je pokazala, da tak zlom ne bi povzročil nobenih posledic v primarnem sistemu in tudi ne v sredici. Tudi doze sevanja ostanejo v okviru predpisov. Analiza je bila izvedena pri moči nič v stanju vroče pripravljenosti.

3.6 Celovitost zadrževalnega hrama

Sproščanje energije in snovi iz reaktorskega hladilnega kroga v zadrževalni hram je analizirano za primer izlivne nezgode in dvostranskega giljotinskega zloma glavnega parovoda (SLB). Začetni pogoji so konzervativni. Za izlivno nezgodo so bili energijski in snovni izpusti izračunani z računalniškim programom SATAN. Analiza SLB za zadrževalni hram vključuje določitev mejne velikosti zloma, pri katerem bi prišlo do odnašanja vode iz uparjalnika, ter izračun količine odnesene vode za tak primer. Analize SLB so bile izvedene s programom LOFTRAN za primere od moči nič do polne moči. Na temelju snovnih in energijskih izpustov sta bila izračunana tlak in temperatura zadrževalnega hrama. Na ta način so bile potrjene tudi projektne zasnove zadrževalnega hrama.

3.7 Druge nezgode

Poleg že omenjenih so bile opravljene tudi mnoge druge termohidravlične varnostne analize. Pomembna skupina analiz so t.i. pričakovani prehodni pojavi brez ustavitve reaktorja (ATWS), ki se analizirajo z realističnimi začetnimi pogoji. Analize so pokazale, da pri povečani moči in novih obratovalnih pogojih uparjalnikov tlak v reaktorskem hladilnem krogu med prehodnimi pojavi brez ugasnitve ATWS ostane pod dovoljeno mejo. Ker najmanjši DNBR ostane nad mejo DNBR pri ATWS-RWAP (izvlek palic pri moči) prehodnem pojavu, je potrjeno tudi delovanje AMSAC (sistem za blaženje ATWS) pri povečani moči. AMSAC je dodatno sredstvo za zaustavitev turbine in sprožitev pomožne napajalne vode (AFW), ki sta dela reaktorskega varovalnega sistema in varujeta reaktorski hladilni krog pred čezmernim tlakom.

4 DOMAČE ANALIZE V PODPORO NEODVISNEMU PREGLEDU

Ocenjevanje in preverjanje kritičnih ter drugih nezgod, ki so bile analizirane pred zamenjavo uparjalnikov in povečanjem moči v jedrski elektrarni Krško, je bilo dodatno podprto tudi z rezultati

steam line rupture is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the Safety Injection System (SIS).

The analysis of a main steam line rupture demonstrated that there is no consequential damage to the primary system and the core remains in place. Radiation doses do not exceed the requirements of 10 CFR 100. The analysis was performed at hot zero power.

3.6 Containment integrity

Mass and energy releases are calculated for the loss-of-coolant accidents and the double-ended guillotine break in the main steam line. The initial conditions are conservative. The LOCA mass and energy releases were determined using the SATAN code. The SLB containment analyses included the determination by the replacement steam generator supplier of the threshold break for which entrainment is predicted to occur, and calculation of the amount of water entrainment in such a case. The SLB analyses are performed for cases from zero to full power using LOFTRAN code. Based on mass and energy release the containment pressure and temperature response were calculated and the containment design was verified.

3.7 Other accidents

Other thermal-hydraulic safety analyses were performed to cover initiating events belonging to one of eight categories. An important category is Anticipated Transients Without Scram (ATWS) which should be analysed using realistic initial conditions. The analyses demonstrated that at the uprated power and new SG operating conditions, the RCS pressure during ATWS transients stays below the acceptable limit and the minimum DNBR remains above the DNBR limit during the ATWS-RWAP (rod withdrawal at power) transient, thereby supporting the plant AMSAC (ATWS Mitigation System Actuation Circuitry) operation under the plant's uprated conditions. AMSAC is an alternative means of tripping the turbine and activating the Auxiliary Feedwater System (AFW) part from the Reactor Protection System, providing assurance against RCS overpressure.

4 NATIONAL ANALYSES SUPPORTING THE INDEPENDENT REVIEW

The assessment and review of critical and other accidents, analyzed for the Krško NPP steam-generator replacement and power uprating were further supported by the results of national independent domačih neodvisnih raziskav in analiz. Velika pozornost je bila posvečena predvsem termo-hidravličnim varnostnim analizam s programi družine RELAP in z drugimi zapletenimi računalniškimi programi. Neodvisne raziskave in analize so bile izvedene z namenom, da bi napovedali obnašanje jedrske elektrarne Krško pred zamenjavo in po zamenjavi uparjalnikov in povečanju moči. Za veliko izlivno nezgodo je bila narejena obširna študija, ki je ovrednotila vpliv čepljenja cevi uparjalnika in velikosti zloma na najvišjo temperaturo srajčke (PCT) [21]. Ugotovljeno je bilo, da iztočni koeficient pri najbolj neugodnem zlomu leži med vrednostima 0,3 in 0,4. Glavni sklep študije je bil, da so rezultati velike izlivne nezgode precej odvisni od uporabljene metodologije. Ker analize z realističnim modelom, ki ocenijo tudi negotovost rezultatov, omogočajo vpogled v potek nezgode za določeno elektrarno, je bila opravljena tudi realistična analiza za veliko izlivno nezgodo [22]. Analiza je sledila kar se je dalo dosledno metodologiji CSAU [23]. Dodatno preverjanje rezultatov računalniških izračunov je bilo narejeno tudi analitično. Rezultati se niso bistveno razlikovali od tistih, dobljenih z računalniškim programom RELAP5/MOD2.

Neodvisne analize male izlivne nezgode so se intenzivno začele s temeljitim pregledom problematike male izlivne nezgode [24]. Sledil je razvoj orodja za določanje zveznega poteka negotovosti za napovedi nezgod [25]. Negotovost male izlivne nezgode je bila ocenjena z uporabo metodologije CSAU in orodja za določanje negotovosti [26].

Mejna projektna nezgoda, ki smo jo temeljito proučevali, zlom cevi v uparjalniku (SGTR) je bila narejena za potrditev nezgodnih delovnih navodil za jedrsko elektrarno Krško [27] med zlomom cevi uparjalnika. Za primera z zlomom dveh ali petih cevi je bilo ugotovljeno, da bi se skozi zlom elektrarna lahko učinkovito hladila tudi več ur.

Poleg omenjenih analiz, narejenih v podporo varnemu obratovanju in projektu posodobitve, so bili analizirani številni drugi hipotetični projektni prehodni pojavi in nezgode, npr. zagozditev rotorja črpalke in nezgode ATWS, pričete z izgubo napajalne vode.

5 SKLEPI

Članek daje pregled termo-hidravličnih varnostnih analiz, ki so bile narejene, da bi bila zagotovljena varna zamenjava uparjalnikov in povečanja moči JEK. Izdelava varnostnih analiz je sledila domači in ameriški zakonodaji. Prehodni pojavi in nezgode so bili razvrščeni v štiri stanja elektrarne v odvisnosti od pogostosti dogodka in njegovimi potencialnimi radiološkimi posledicami za prebivalstvo. Neodvisno preverjanje analiz nezgod je bilo osredotočeno na predpostavljene kritične

research and analyses. On the national level, significant efforts were devoted to the thermal-hydraulic safety analyses using the RELAP series and other complex computer codes. For the LB LOCA, a study was made to evaluate the impact of steam generator tube plugging and break size on the Peak Cladding Temperature (PCT) [21]. It was found that for the worst break size the discharge coefficient lies between 0.3 and 0.4. The main conclusion of the study was that the results of the LB LOCA analysis largely depends on the methodology used and that the best estimate analyses approach with uncertainty quantification offers a plant-specific insight into the accident progression. Therefore, the best estimate analysis for the LB LOCA was performed [22], as closely as possible, following the Code Scaling, Applicability and Uncertainty (CSAU) methodology [23]. The analysis demonstrated that the CSAU methodology can be applied to an individual plant. Additional verification of the computational efforts was made by the analytical solution and was not far from those calculated by the RELAP5/MOD2 computer code.

The independent analyses of the SB LOCA started with a thorough overview of the problems related to the SB LOCA [24]. Subsequently, a tool for continuous uncertainty evaluation of transient predictions was developed [25]. The uncertainty for the SB LOCA analysis was quantified following CSAU methodology and using the tool for uncertainty evaluation [26].

The next design basis accident analysis which was thoroughly studied was the steam generator tube rupture (SGTR) to validate the Krško NPP emergency operating procedures during SGTR [27]. It was concluded that for the case of the rupturing of two or five SG tubes the plant could be efficiently cooled down through the rupture for several hours with no need to initiate the primary feed and bleed procedure.

In addition, a number of other hypothetical design transient and accidents, like the locked rotor and the loss of feed flow initiated ATWS accident, had previously been studied to support safe plant operation and the modernization project.

5 CONCLUSIONS

The paper gives an overview of thermal-hydraulic safety analyses necessary to ensure safe steam generator replacement and power increase. For accident analyses, USNRC R.G. 1.70 Rev.3 is followed where the ANS classification of the plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public is used. The overview of accidents focused on the assumed critical accidents and the fulfillment of the acceptance nezgode in na izpolnjevanje kriterijev sprejemljivosti. Ugotovitve so bile podprte z neodvisnimi domačimi raziskavami in analizami.

Povzamemo lahko, da so vse analize potrdile, da bodo vsa stanja med prehodnimi pojavi in nezgodami ostala znotraj mej in kriterijev sprejemljivosti in da bo jedrska elektrarna Krško z novimi uparjalniki in povečano močjo v novem delovnem oknu lahko obratovala varno.

criteria. The findings were further supported by national research and analysis results that formed the basis for the independent review.

It can be concluded that the analyses performed for the Krško NPP plant with the new steam generators and uprated power has proved that all transient and accident conditions would remain within the limits and acceptance criteria for the newly designed operating window.

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Pridobivanje dovoljenj za zamenjavo uparjalnikov in povečanje moči JE Krško

Integrated Licensing of the Steam Generators' Replacement and Uprating at the Krško Nuclear Power Plant

Jože Špiler - Tadej Plestenjak

Namen prispevka je predstaviti poglavitne značilnosti postopka pridobivanja dovoljenj za posege, ki potekajo zadnja tri leta in vplivajo na jedrsko varnost. Sedanja slovenska zakonodaja temelji predvsem na zakonih in pravilnikih, ki jih je v osemdesetih letih izdala nekdanja Jugoslavija in ki so po osamosvojitvi leta 1991 ostali veljavni v Republiki Sloveniji. V prispevku je pregled zakonov in pravilnikov, ki zadevajo postopke pridobivanja dovoljenj za JE Krško. Celoten postopek pridobivanja dovoljenj za projekt zamenjave uparjalnikov in povečanje moči lahko razdelimo na dve skupini:

– *postopki pridobivanja dovoljenj za posege, ki vplivajo na jedrsko varnost ter*
– *postopki pridobivanja dovoljenj za posege, ki ne vplivajo na jedrsko varnost.*

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(Ključne besede: pridobivanje dovoljenj, specifikacije tehnične, poročila varnostna, okna obratovalna)

The aim of this paper is to outline the most important parts of the nuclear safety licensing process, which has been pursued continuously during the last three years. The present Slovenian regulatory system is essentially based on the laws and regulations which were passed in the 1980s by the former Yugoslavia and have remained in force in the Republic of Slovenia following its independence in 1991. The paper reviews the laws and regulations applicable to the licensing of Krško nuclear power plant (NPP). The integrated licensing process of the steam generators' replacement and the uprating at Krško NPP can be divided into two groups:

– *nuclear-safety-related licensing process,*
– *non-nuclear licensing process.*

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(Keywords: licensing procedure, technical specifications, safety analysis reports, operating windows)

0 UVOD

JE Krško je ena izmed zadnjih tlačnovodnih jedrskih elektrarn v Evropi, zgrajenih po zahodni tehnologiji, ki se je odločila za zamenjavo sedanjih uparjalnikov ter dvig povečanje moči hkrati. Elektrarna ima dovoljenje za delovanje pri polni moči z največ 18% začepljenih cevi v obeh uparjalnikih. Vendar pa se je v zadnjih desetih letih pokazalo, da je vzdrževanje začepljenosti cevi znotraj navedenih omejitev drag ter tudi časovno potraten postopek. Zaradi tega se je JEK odločila za zamenjavo sedanjih uparjalnikov. Novi, tehnično izboljšani uparjalniki, omogočajo dvig povečanje moči za 6,3%: od 1882 MW$_t$ do 2000 MW$_t$. Najpomembnejši parametri sistema NSSS (jedrski uparjalni sistem) pred povečanjem in po povečanju moči so prikazani v preglednici 1.

0 INTRODUCTION

Krško NPP is one of the last European pressurized-water reactor NPPs of western design that has decided to replace the existing SGs and perform the power uprating at the same time. Currently, the plant is licensed to operate at nominal power with up to 18% of the tubes plugged in both steam generators. However, maintaining the plugging level below this limit during the last ten years has proved to be a very expensive and time-consuming process. As a result, it was decided to replace the present steam generators. Better performance of the new steam generators enables an increase in the plant's nominal power of 6.3%: from 1882 MW$_t$ to 2000 MW$_t$. The most important parameters of the NSSS, before and after the power uprating are summarized in Table 1.

Projektni izračuni ter varnostne analize, ki so bile izvedene, dokazujejo da:

- so novi uparjalniki združljivi s sedanjo opremo elektrarne in
- da elektrarna lahko varno deluje pri večji moči z enokovrednimi delovnimi ter varnostnimi omejitvami.

Omenjene analize prav tako razpoznavajo vse potrebne spremembe v elektrarni in/ali delovne omejitve, ki bi se lahko pokazale pri delovanju na večji moči. Analize so zbrane v obsežnem paketu poročil, ki je bil uporabljen kot temelj za spremembo licenčne dokumentacije; ta je bila posredovana URSJV (Upravi za jedrsko varnost R Slovenije) z namenom, da bi pridobili obratovalno dovoljenje elektrarne z novimi uparjalniki ter pri večji moči.

Namen prispevka je predstaviti poglavitne značilnosti postopka pridobivanja dovoljenj za posege, ki potekajo zadnja tri leta in vplivajo na jedrsko varnost. Glede na posebnost zamenjave uparjalnikov in povečanja moči, je opisan upravni

Appropriate design and safety analyses were performed to demonstrate that:

- the new steam generators are compatible with the existing plant,
- that the plant can safely operate at higher power with adequate operating and safety margins.

These analyses also identified all the required modifications of the plant and/or operating limitations that might be imposed at uprated power conditions. They are documented in a comprehensive set of reports which were used as the basis for the changes in the licensing documentation and were submitted to the SNSA in order to obtain an operational permit for the plant with he new steam generators and uprated power.

The aim of this paper is to outline the most important parts of the nuclear safety licensing process, which has been pursued continuously during the last three years. The regulatory framework and practice for nuclear-safety-related objects in Slovenia

Preglednica 1. *Obratovalni parametri NSSS pri večji moči in po zamenjavi uparjalnikov*
Table 1. *NSSS operating parameters related to the power uprating and SGs' replacement*

Parametri Parameters	Parametri 16. gorivnega obdobja Original Design Cycle 16	Parametri 17. gorivnega obdobja (po zamenjavi SG ter povečanju moči) SGs' Replacement and Power Uprating Cycle 17
Skupna moč NSSS v MW_t NSSS Total Power in MW_t	1882	2000
Reaktorska moč v MW_t Reactor Power in MW_t	1876	1994
Pretok za toplotni projekt sredice v m^3/s Thermal Design Flow in m^3/s	12,011	12,013
Tlak reaktorskega hladila v MPa Reactor-Coolant Pressure in MPa	15,5	15,5
Temperatura reaktorskega hladila v °C Reactor-Coolant Temperature in °C		
Temp. na ničelni moči v °C Zero-Load Temperature in °C	291,7	291,7
Temp. izstopa iz sredice v °C Core Outlet Temperature in °C	325,9	327,4
Temp. izstopa iz posode v °C Vessel Outlet Temperature in °C	324,3	325,1
Povprečna temp. sredice v °C Core Average Temperature in °C	307,8	308,1
Povprečna temp. posode v °C Vessel Average Temperature in °C	305.9	305,7
Temp. vstopa v posodo/sredico v °C Vessel/Core Inlet Temperature in °C	287,5	286,2
Temperatura pare v °C Steam Temperature in °C	279,2	279,3
Tlak pare v MPa Steam Pressure in MPa	6,34	6,48
Pretok pare v t/s Steam Flow in t/s	1,029	1,0896
Temperatura napajalne vode v °C Feedwater Temperature in °C	221,1	219,4
Skupna električna moč v MW_e Gross Electrical Output in MW_e	664,4	706,8

okvir in praksa v Sloveniji pri objektih, ki so pomembni za jedrsko varnost.

1 PREGLED ZAKONODAJE PRIDOBIVANJA DOVOLJENJ ZA POSEGE, KI VPLIVAJO NA JEDRSKO VARNOST

Sedanja slovenska zakonodaja temelji predvsem na zakonih in pravilnikih, ki jih je v osemdesetih letih izdala nekdanja Jugoslavija in ki so po osamosvojitvi leta 1991 ostali veljavni v Republiki Sloveniji. Naslednji zakoni in pravilniki so uporabljeni pri procesu pridobivanja dovoljenj za delovanje, izvajanje projektnih sprememb in zato tudi sprememb SAR:

1. *Zakon o varstvu pred ionizirajočimi sevanji in o posebnih varnostnih ukrepih pri uporabi jedrske energije (Uradni list SFRJ 62/84*)
   Zakon zahteva izdelavo preliminarnega varnostnega poročila SAR za pridobitev gradbenega dovoljenja objekta ter izdelavo končnega SAR za pridobitev obratovalnega dovoljenja. Omenjeni zakon podaja temeljna načela varnega obratovanja jedrskih elektrarn. Med drugim zakon opredeljuje uporabo domače zakonodaje ter tehničnih standardov. V primeru, da domača zakonodaja ni sprejeta, se sme uporabiti tudi zakonodaja ter tehnični standardi države, od koder izvirajo, če so potrjeni od domačega upravnega organa (URSJV).
2. *Pravilnik o izdelavi in vsebini varnostnega poročila in druge dokumentacije, potrebne za ugotavljanje varnosti jedrskih objektov (Uradni list SFRJ 68/88)*
   Pravilnik podaja razlago vsebine SAR kot osnovne dokumentacije, potrebne za ugotavljanje varnosti jedrskega objekta. Varnostno poročilo mora biti med obratovanjem dopolnjevano s podatki in analizami o vseh spremembah, ki so nastale na jedrskem objektu. Pravilnik načrtuje tri kategorije sprememb in dopolnitev končnega SAR. Prva kategorija predvideva obveščanje URSJV o izvedenih spremembah po njihovi izvedbi. Druga kategorija sprememb predvideva obveščanje URSJV pred njihovo izvedbo. V tretjo kategorijo pa spadajo spremembe, za katere je treba pri URSJV vložiti prošnjo za dovoljenje za izvedbo. Natančnejši kriteriji ter navodila glede kategorizacije sprememb v elektrarni in SAR sprememb pa niso podani.
3. *Pravilnik o pogojih za lokacijo, graditev, poskusno obratovanje, zagon in uporabo jedrskih objektov (Uradni list SFRJ 52/88)*
   Pravilnik zahteva, da mora med obratovanjem jedrskega objekta njegov uporabnik stalno spremljati in analizirati stanje varnosti jedrskega objekta, pri čemer mora upoštevati izkušnje drugih jedrskih objektov in tehnološkega razvoja. Pravilnik med drugim zahteva, da je pri izvedbah sprememb tehničnih specifikacij TS, obvezna izdelava

are outlined, followed by the specifics of the steam generator replacement and the plant's uprating.

1 LEGISLATION FRAMEWORK FOR NUCLEAR SAFETY LICENSING

The present Slovenian regulatory system is essentially based on the laws and regulations which were issued in the 1980s by the former Yugoslavia and have remained in force in the Republic of Slovenia following its independence in 1991. The following laws and regulations are applicable to the licensing of the NPP's operation, modifications of its design and consequently the changes of the SAR:

1. *The law on Radiation Protection and the Safe Use of Nuclear Energy (Off. Gaz. SFRJ, 62/84)*
   The law requires submission and acceptance of the Preliminary SAR before the construction permit is issued and submission and acceptance of the final SAR to obtain an operating permit. The law provides the general framework for the safe operation of nuclear power plants. Among other points, it states that national regulations and technical standards shall be applied and, when not available, the regulations and technical standards of the country of origin can be applied, if approved by the domestic regulatory organization (SNSA).
2. *Regulation on Safety Analysis Reports (Off. Gaz. SFRJ, 68/88)*
   This regulation defines the SAR as the basic licensing document for nuclear installation with respect to nuclear safety. The SAR shall be continually supplemented to appropriately address all changes which have been implemented in the nuclear power plant. Three categories of plant changes, resulting in changes to the final SAR, are established. The first category allows immediate implementation of the change, followed by notification to the SNSA. The second category requires a notification to the SNSA before the implementation of the change. The third category requires an approval by the SNSA before the change is implemented. However, detailed criteria or guidance about the categorization of plant and SAR changes are not given.
3. *Regulation on Siting and Construction and Operation of Nuclear Facilities (Off. Gaz. SFRJ, 52/88)*
   This regulation requires that the licensee continuously monitors and analyzes the level of nuclear safety, with the experience of other nuclear facilities and new technological developments to be taken into account. This regulation requires, among other things, a mandatory independent third-party evaluation of the proposed changes to the TS. The TS is the most important chapter

neodvisnega vrednotenja pooblaščene organizacije. TS so najpomembnejši del SAR, saj definirajo dovoljena območja vseh parametrov, pomembnih za varnost elektrarne. Neodvisno vrednotenje lahko izvede organizacija, ki je pooblaščena od URSJV. Za organizacijo izvedbe neodvisnega vrednotenja je skladno z dogovorom z URSJV hkrati odgovoren upravljalec jedrskega objekta.

2 POSTOPKI PRIDOBIVANJA DOVOLJENJ ZA POSEGE, KI VPLIVAJO NA JEDRSKO VARNOST TER NJIHOVO IZVAJANJE V SLOVENIJI

Podrobnejša navodila glede postopkov pridobivanja dovoljenj za izvajanje sprememb SAR ter sprememb na jedrskem objektu (npr. glede kriterija razvrstitve, vsebine dokumentacije, neodvisnega pregleda) še niso izdelane niti pri upravnem organu niti pri upravljalcu. Zaradi tega postopki pridobivanja dovoljenj v NEK temeljijo na prej omenjeni zakonodaji, hkrati pa so skladni z zakonodajo države prodajalca (ZDA, npr. 10 CFR 50.59), JEK pa o vseh projektnih spremembah obvešča tudi URSJV. Takšen postopek je v skladu z Zakonom o varstvu pred ionizirajočimi sevanji in o posebnih ukrepih pri uporabi jedrske energije, v praksi pa se kaže v dnevnem sporočanju med URSJV ter JE Krško. V nadaljevanju je opisan pomembnejši del običajnega postopka.

Tehnične specifikacije določajo, da je elektrarna dolžna podati poročilo URSJV o načrtovanih spremembah, testih in poskusih vsaj 45 dni pred njihovo izvedbo.

JEK podaja URSJV v odobritev vsako načrtovano spremembo Tehničnih specifikacij. Elektrarna je prav tako dolžna sporočiti neodvisno strokovno oceno spremembe, ki jo izdelajo neodvisne organizacije. Neodvisne organizacije so pooblaščene od URSJV. Tako je v odobritveni postopek URSJV vedno vključena tretja, neodvisna oseba.

Pooblaščene neodvisne strokovne organizacije imajo izdelane QA postopke zagotovitve kakovosti za izdelavo strokovnih ocen za področja, za katera so pooblaščena. Preglede postopkov QA opravita skupaj URSJV ter JEK približno vsaki dve leti.

Podane načrtovane spremembe, ki imajo vpliv na SAR, so vedno podprte z varnostnim vrednotenjem, ki ga izdela osebje JEK. Če URSJV izrazi zahtevo, pa ji JEK sporoča tudi varnostno presojanje sprememb, ki kažejo, da ni vpliva na SAR. Omeniti je treba, da so varnostna vrednotenja načrtovanih sprememb namenjena predvsem ocenitvi možnega vpliva na varnost. Preden se načrtovana sprememba izvede ali pa poda v pregled URSJV, mora biti varnostno vrednotenje potrjeno od KOC (komisija, sestavljena iz osebja - strokovnjakov JE Krško) ter KSC (komisija zunanjih strokovnih sodelavcev).

of the SAR and defines the allowable range of all the parameters which are important for the safe operation of the plant. The independent evaluation shall be performed by organizations authorized by the SNSA. The arrangements to perform this independent evaluation are an implicit responsibility of the licensee, which is in practice accomplished in an agreement with the SNSA.

2 THE NUCLEAR SAFETY LICENSING PROCESS AND PRACTICE IN SLOVENIA

Detailed requirements or guidance for the licensing process involving modifications to the SAR and the plant are presently not established in either the Slovene regulations or in the operating license. Such guidance could, for example, include criteria to categorize the SAR changes and plant modifications, the content of the application document and the content of the independent third-party review. Krško NPP is currently using the criteria and guidance established in the relevant legislation of the vendor country (USA, Code of Federal Regulations 10 CFR 50.59) and notifies the SNSA about all design modifications. Such an approach is consistent with the applicable Law on Radiation Protection and the Safe Use of Nuclear Energy and resulted in practices established in daily communication between the SNSA and Krško NPP. The most important practices are described below.

The Technical specifications require that a report on planned modifications, tests and experiments on the plant is to be submitted 45 days before the license is expected to be issued by the SNSA.

Further, Krško NPP submits for approval every proposed change to the plant's Technical Specifications. Every submission is supported by independent expert opinion, prepared by the so-called Technical Support Organization (TSO). TSOs are authorized by the SNSA. The decision making process within the SNSA is therefore always supported by an independent third-party assessment.

All TSOs have developed QA procedures for quality assurance for their area of expertise for which they are authorized. Joint (SNSA and Krško NPP) QA audits are carried out approximately every two years.

Submittalls of proposed plant changes, which have an impact on the SAR, are also always accompanied by the safety evaluations prepared by the staff of Krško NPP. On request, Krško NPP also submits to the SNSA the safety evaluations of plant changes, which have no impact on the SAR. It should be mentioned here that the safety evaluations are aimed primarily at the assessment of potential safety consequences of the proposed modification. Before the proposed modification is implemented or submitted for approval to the SNSA, the safety evaluation has to be approved by the KOC (board of experts from the Krško NPP personnel) and the KSC (board of external experts).

V vseh primerih, ko se podaja načrtovana sprememba URSJV, se uporablja formalni postopek, predpisan z *Zakonom o splošnem upravnem postopku*. Še posebno je postopek primeren v primerih:
- pomembnejših sprememb,
- ko varnostno vrednotenje pokaže nerešeno varnostno vprašanje,
- sprememb v organizacijski strukturi (organizacijska shema, definicije odgovornosti, medsebojno sporočanje itn.),
- spremembah TS,
- spremembah pomembnejših programov ter postopkov (npr. program nadzora med uporabo (ISI), program požarne zaščite, program radiološkega nadzora itn.), ki so omenjeni v SAR, ne pomenijo pa pomembnejšega dela tega poročila.

Na koncu lahko navedemo opis tipične dokumentacije, posredovane URSJV za odobritev vloge za spremembo SAR:
- predlagano spremembo SAR (izdela jo JE Krško ali pogodbeni izvajalci),
- varnostno ovrednotenje (izdela ga JE Krško),
- varnostno analizo ali poročilo o upravičenosti (če je to treba, izdela jo JE Krško ali pogodbeni izvajalci),
- tehnično delovno poročilo (izdela ga JE Krško ali pogodbeni izvajalci),
- pozitivno neodvisno strokovno mnenje o spremembi, ki ga pripravi neodvisna pooblaščena strokovna organizacija - TSO.

Sam postopek pregleda upravnega organa ter varnostno vrednotenje uradno steče po dostavi potrebne dokumentacije URSJV. Med upravnim postopkom se opravi vsaj eno zaslišanje stranke, pri katerem sta navzoči obe strani, zastopniki elektrarne ter URSJV, in na katerem se po potrebi rešujejo odprta vprašanja. Postopek se konča s formalno odločitvijo URSJV o spremembah SAR in drugih obratovalnih razmerah.

Na koncu postopka pridobivanja dovoljenj izda URSJV odločbo.

3 PRIDOBIVANJE DOVOLJENJ ZA PROJEKT ZAMENJAVE UPARJALNIKOV IN POVEČANJE MOČI JE KRŠKO

Celoten postopek pridobivanja dovoljenj lahko razdelimo na dve skupini:
- postopki pridobivanja dovoljenj za posege, ki vplivajo na jedrsko varnost ter
- postopki pridobivanja dovoljenj za posege, ki ne vplivajo na jedrsko varnost.

Obe skupini postopkov sta na kratko predstavljeni v nadaljevanju.

3.1 Postopki pridobivanja dovoljenj za posege, ki vplivajo na jedrsko varnost

In all cases involving the submission of a modification to the SNSA, the formal administrative procedures prescribed in the *Law on Administrative Procedure* are followed. This is particularly necessary in the following cases:
- major modifications,
- when a safety assessment shows the elements of an unreviewed safety question,
- changes in the organizational structure of the plant (organizational chart, definition of responsibilities, communication lines, etc.),
- changes to the TS,
- changes of important programs and procedures (for example, the in-service inspection (ISI) Program, the Fire Protection Program, the Radiation Monitoring Program, etc.) which are referred to in the SAR, but are not a constitutive part of the SAR.

Let us summarize the above discussion with a description of typical documentation submitted to the SNSA to approve a change in the SAR:
- proposed change to the SAR (prepared by Krško NPP or its contractors),
- safety evaluation (prepared by Krško NPP),
- safety analysis or safety analysis justification report (if necessary, prepared by Krško NPP or its contractors),
- technical report packages (prepared by Krško NPP or its contractors),
- positive independent expert opinion about the proposed change prepared by an independent reviewer - TSO.

After the documentation package is submitted, the process of regulatory review, safety evaluation and decision making is officially started by the SNSA. During this process, at least one hearing takes place to give Krško NPP and the SNSA an opportunity to consider and discuss the application and open issues, if any. The process formally concludes with the SNSA's formal decision about the proposed change to the SAR and other licensing conditions.

At the end of licensing process, the SNSA issues the Licensing Amendment.

3 INTEGRATED LICENSING OF THE STEAM GENERATORS' REPLACEMENT AND UPRATING AT THE KRŠKO NPP

The whole licensing process can be divided into two groups:
- nuclear-safety-related licensing process,
- non-nuclear licensing process.

Both are discussed in some detail below.

3.1 Nuclear-safety-related licensing process

The manpower involved in the preparation of the steam generator replacement and power uprating AT Krško NPP exceeded 80 man-years. Therefore, the SNSA suggested that the licensing

Celotna priprava zamenjave uparjalnikov in dviga povečanja moči je zahtevala prek 80 človeških let. Zaradi tega je URSJV predlagala, da se postopek pridobivanja dovoljenj razdeli na naslednje vzporedne prostopke pridobivanja dovoljenj:

- projektiranje, izdelava in testiranje uparjalnikov JE Krško,
- analize ob zamenjavi uparjalnikov in analize za povečanje moči,
- skladiščenje starih uparjalnikov in radioaktivnih odpadkov, nastalih zaradi zamenjave.

JEK je tako na predlog URSJV dne 31. julija 1997 podala vlogo za projektiranje in izdelavo uparjalnikov JE Krško in dne 8. septembra 1997 še vlogo za vgradnjo novih uparjalnikov ter varnostne analize za povečanje moči ob zamenjavi uparjalnikov. Postopek pridobivanja dovoljenj stavbe za skladiščenje starih uparjalnikov in radioaktivnih odpadkov, nastalih zaradi zamenjave pa je že končan.

Projektiranje, izdelava in testiranje uparjalnikov JE Krško

Dela projektiranja, izdelave in zamenjave uparjalnikov JE Krško so zaupana konzorciju Siemens-Framatome in se izvajajo v skladu s standardom ASME vrelnih in tlačnih posod. Kot neodvisni ocenjevalci so pri nadzoru, zagotovitvi kakovosti in nadzoru kakovosti sodelovale naslednje pooblaščene organizacije iz Ljubljane:

- Fakulteta za strojništvo,
- Inštitut za varilstvo,
- Inštitut za metalne konstrukcije,
- Inštitut za kovinske materiale in tehnologije.

Njihova opažanja potrjujejo visoko tehnološko raven in kakovost novih uparjalnikov. Ugotovitve so zbrane v strokovnih ocenah, ki jih izdela vsaka izmed pooblaščenih organizacij. Strokovne ocene se skupaj z dokumentacijo projektanta oziroma izvajalca del konzorcija Siemens-Framatome podajo URSJV v odobritev.

Analize ob zamenjavi uparjalnikov JE Krško in analize za povečanje moči

Leta 1991 je bila izvedena študija izvedljivosti povečanja moči WENX 91-42. Omenjena študija je pokazala, da je mogoče ***povečanje moči za 6,3%*** (do 2000 MW_t) brez izvedbe pomembnejših posegov. Sama zamenjava uparjalnikov ter povečanje moči elektrarne je tako podprto z ugotovitvami več analiz.

Rezultati analiz dokazujejo zmožnost delovanja elektrarne v danem ***delovnem oknu*** ter zmožnost varnega delovanja z novimi uparjalniki pri večji moči. Osnutek delovnega okna zagotavlja večjo prožnost delovanja elektrarne kakor pa sedanja licencirana delovna točka (npr. ni treba izvajati

process should be systematic and divided into the following parallel licensing processes:

- design, manufacturing and testing of the Krško NPP steam generators,
- steam-generator replacement and power uprating safety analyses,
- storage of old steam generators and radioactive waste as a result of the replacement.

Following the above suggestion of the SNSA, the Krško NPP filed the application for the commencement of the licensing process for the steam generators' design, manufacture and testing on July 31, 1997. This was followed by the application for the commencement of the licensing process for the replacement of both steam generators and power uprating safety analyses on September 8, 1997. The licensing process for the storage of the old steam generators and the radioactive waste resulting from the replacement has been completed.

Design, manufacture and testing of the new steam generators

The design, manufacture and testing of the new steam generators was performed by the consortium Siemens-Framatome in accordance with the ASME Boiler and Pressure Vessel Code. The TSOs from Ljubljana listed below were involved as independent reviewers supervision, quality assurance and quality control of the manufacturing process:

- The Faculty of Mechanical Engineering,
- The Welding Institute,
- The Institute for Metal Structures,
- The Institute of Metals and Technology.

Their findings support the high technological level and the quality of the new steam generators. As documented in independent evaluation reports issued independently by each TSO, they were submitted to the SNSA for approval together with a comprehensive set of documentation prepared by the designer and manufacturer - consortium Siemens-Framatome.

Steam generator replacement and power-uprating safety analyses

The plant uprating feasibility study WENX 91-42 was performed in 1991. The main conclusion of this study was that a ***power increase of 6,3%*** (to 2000 MW_t) is feasible without major modifications to the plant. The steam generator replacement and power uprating are now supported by a substantial set of adequate analyses.

The analyses verified the plant's maneuverability for a selected ***operating window*** and safe operation with a new steam generators at an uprated nominal power. The concept of the operating window provides more flexibility in the plant operation than the currently licensed operating point (e.g., no dodatnih analiz po čepljenju cevi uparjalnika, če niso prekoračene omejitve delovnega okna). Izvajanje analiz z odobritvijo delovnega okna zagovarja tudi praksa v Evropi.

Drugi del analiz pa obravnava in potrjuje ***osnutek LBB*** za cevovode reaktorskega hladilnega sistema. Osnutek LBB dovoljuje, da se dinamične obremenitve, ki so posledica velikega zloma, pri izračunih ne upoštevajo. Uporaba osnutka LBB seveda temelji na analizah, ki kažejo, da je trdnost materiala cevovodov reaktorskega hladilnega sistema na taki stopnji, da se opazi puščanje skozi poškodbo pred zlomom cevovoda. Ena izmed mogočih prednosti takega postopka je tudi odstranitev za vzdrževanje zapletene cevne opreme (npr. cevne podpore).

Izvajalec vseh omenjenih analiz je Westinghouse Electric Systems Europe, ki je podružnica projektanta JE Krško s sedežem v Bruslju. Vse analize so zajete v delovnih poročilih (eno poročilo za posamezno analizo), zbirni oceni (zbirka vseh izvedenih analiz) in spremenjenem SAR vključno z spremenjenimi TS. Vsa ta dokumentacija je tudi poslana URSJV v odobritev.

Vsako delovno poročilo je bilo hkrati pregledano od JE Krško, pooblaščenih organizacij ter URSJV. Njihove ugotovitve lahko kategoriziramo kot komentarje, priporočila ter zahtevane spremembe (sl. 1). Po pregledu in/ali razrešitvi vseh komentarjev je pripravila pooblaščena organizacija neodvisno strokovno oceno, ki se je skupaj z drugo potrebno dokumentacijo poslala URSJ v odobritev. Z omenjenim postopkom (sl. 1) so se skupaj z URSJV pregledala in reševala vsa odprta vprašanja, ki zadevajo pridobivanje dovoljenj, tako da bodo za ponovni zagon elektrarne na večjo moč ter z zamenjanimi uparjalniki pridobljena vsa potrebna dovoljenja.

Naslednje pooblaščene organizacije so bile izbrane od JE Krško in z odobritvijo URSJV kot neodvisni ocenjevalci analiz, ki jih izvaja Westinghouse:

- Inštitut Jožef Stefan, Ljubljana,
- Fakulteta elektrotehnike in računalništva Zagreb,
- Fakulteta za gradbeništvo in geodezijo Ljubljana,
- Enconet, Dunaj.

Njihove ugotovitve dokazujejo primerno varnostno raven JE Krško po dvigu moči ter zamenjavi uparjalnikov.

Dodaten pregled dejavnosti zamenjave uparjalnikov ter preostalih remontih del bo izvajal ter nadzoroval URSJV ter pooblaščene organizacije v skladu z dosedanjo prakso pregleda remontnih dejavnosti.

re-analysis of the plant operation is needed after plugging of some tubes in the steam generators, if the limits of the operating window are not violated) The analyses supporting the operating window were consistent with European practice.

Another part of the analyses also verified the applicability of the ***LBB concept*** for the reactor-coolant-loop piping. The LBB concept allows that the dynamic loads resulting from a large break in the piping is not taken into account. Application of the LBB concept is of course based on analyses showing that the material of the reactor-coolant piping is tough enough to allow for reliable detection of leaks through defects before the risk of a break takes place. One of the possible advantages of such an approach is the removal of the complex and difficult-to-maintain piping support hardware (e.g., snubbers).

All of the above analyses were performed by Westinghouse Electric Systems Europe, a Brussels-based subsidiary of the designer of the Krško NPP. All analyses are documented in Work reports (one per analysis), Summary reports (summarizing all analyzes performed) and a revised SAR including a revised TS. These documents represent the documentation submitted to the SNSA for approval.

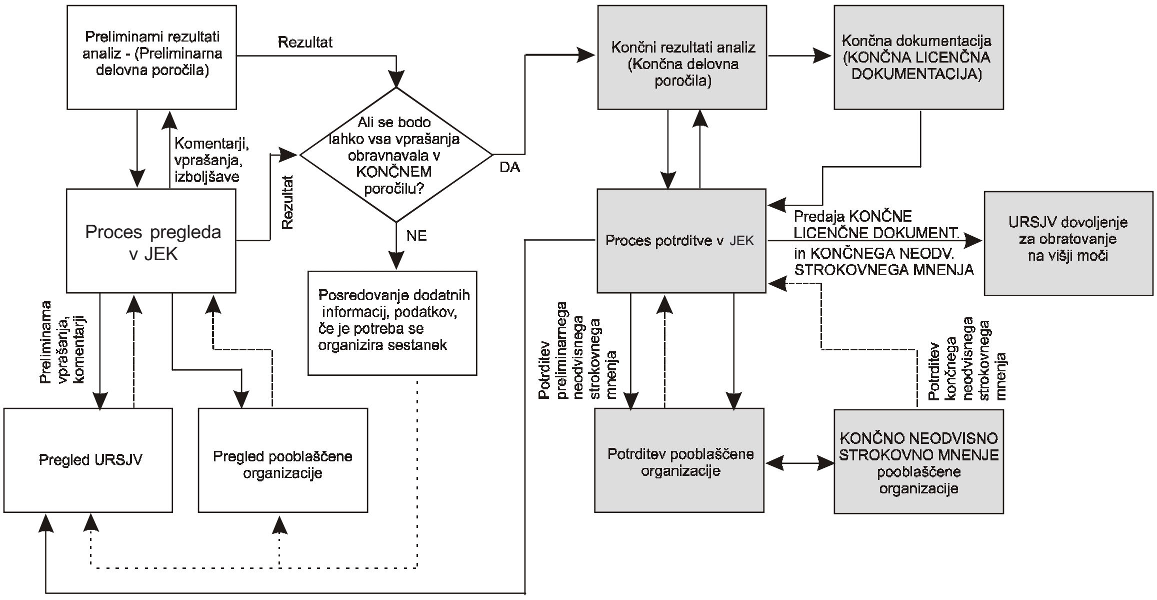
Each of the work reports was reviewed in parallel by the Krško NPP, the TSO and the SNSA, resulting in a list of comments, categorized as suggestions, recommendations and required changes (see Figure 1). After the clarification and/or resolution of all the comments, the TSO prepared Independent Evaluation Report(s), which were submitted together with other licensing documentation to the SNSA for their approval. With this approach (shown in Fig. 1) all licensing issues (questions, concerns) were addressed and resolved in due time with the SNSA and should enable the authorized restart of the plant after the power uprating and SG replacement is authorized.

The following Technical Support Organizations were selected by Krško NPP in agreement with the SNSA to act as independent reviewers for the analyses performed by Westinghouse:

- Institute Jožef Stefan, Ljubljana,
- Faculty of Electrical Engineering and Computing Department of Power Systems, Zagreb, Croatia,
- Faculty of Civil and Geodetic Engineering, Ljubljana,
- Enconet, Vienna, Austria.

Their findings support the appropriate safety level of the Krško NPP after power uprating and the SG replacement.

An additional review and evaluation will be conducted for the steam generator replacement activities and associated modifications as a regular part of the SNSA and TSO activities during each outage.



Sl. 1. *Postopek pridobivanja dovoljenj varnostnih analiz za zamenjavo uparjalnikov ter povečanje moči uparjalnikov*

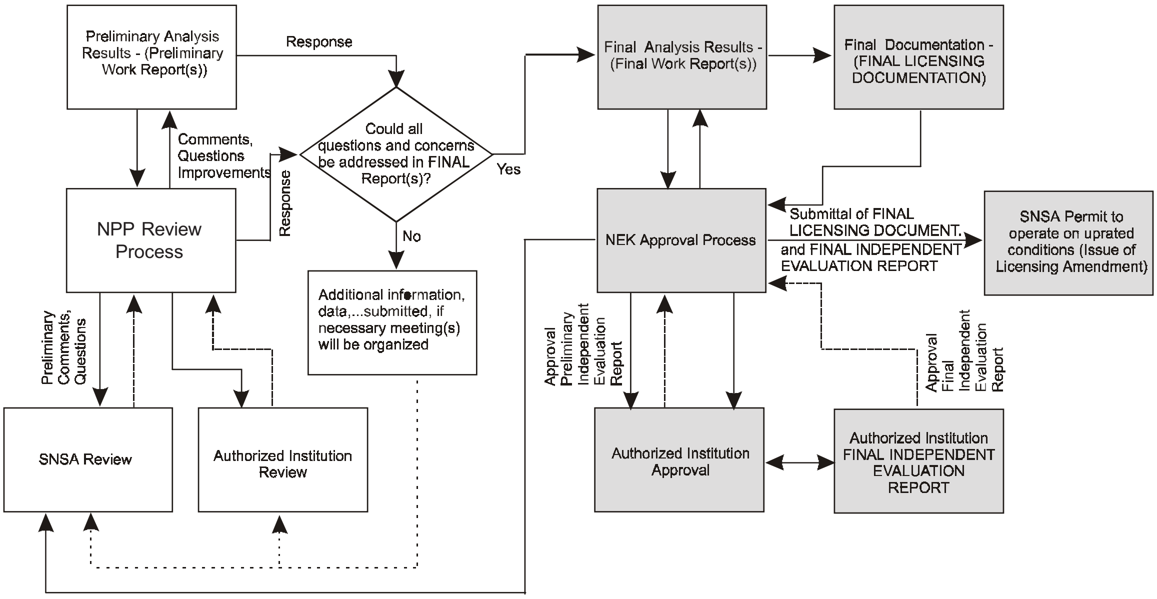


Fig. 1. *Licensing process for safety analysis for the SGs' replacement and power uprating*

3.2 Postopki pridobivanja dovoljenj za posege, ki ne vplivajo na jedrsko varnost

Pridobivanje dovoljenj za posege, ki ne vplivajo na jedrsko varnost, je prav tako del celotnega postopka pridobivanja dovoljenj. Sem spadajo naslednje dejavnosti v skladu z Zakonom o graditvi objektov:

- spremembe dokumentacije elektrarne, ki jih potrjuje občinska uprava,
- spremembe na transportni poti za omogočanje prevoza nadomestnih uparjalnikov,
- gradbeno dovoljenje za simulatorsko zgradbo,
- gradbeno dovoljenje poslopja za shranjevanje starih uparjalnikov,

3.2 Non-nuclear licensing process

The non-nuclear licensing process is also part of whole licensing process regarding the Krško steam-generator replacement and power uprating project. It covers the following licensing activities in accordance with Civil construction law:

- changes to site documents approved by local authorities,
- modifications of the roads to accommodate transportation of the replacement steam generators,
- building permit for the simulator building,
- building permit for old steam generators' building,

- remontna dela ter povečanje moči, ki so opredeljeni kot rekonstrukcija.

4 SKLEPI

Poglavitni namen projekta modernizacije JE Krško - analiz ter postopka pridobivanja dovoljenj je izvedba vseh potrebnih analiz, ki dokazujejo združljivost nadomestnih uparjalnikov s sistemi elektrarne ter zagotavljajo, da bodo parametri normalnega obratovanja in nezgodnih stanj znotraj področja sprejemljivosti tudi po zamenjavi uparjalnikov. Hkrati pa naj bi analize pokazale tudi vse spremembe, potrebne za zadovoljitev omenjenega kriterija.

Vse analize so hkrati pregledane od JE Krško, pooblaščenih organizacij ter URSJV. Njihove ugotovitve lahko kategoriziramo kot komentarje, priporočila ter zahtevane spremembe. Po pregledu in/ali razrešitvi vseh komentarjev, pripravi pooblaščena organizacija neodvisno strokovno oceno, ki se skupaj s preostalo potrebno dokumentacijo pošlje URSJ v odobritev. Z omenjenim postopkom se skupaj z URSJV pregledujejo in rešujejo vsa odprta vprašanja, sam postopek pa je voden tako, da omogoča uspešen konec procesa pridobivanja dovoljenj.

KRATICE

JEK - Jedrska elektrarna Krško, SG - uparjalnik, NSSS - jedrski sistem za proizvodnjo pare, URSJV - Uprava Republike Slovenije za jedrsko varnost, SAR - varnostno poročilo, TS - Tehnične specifikacije, QA - zagotovitev kakovosti, KOC - Strokovni svet pogona, KSC - Varnostni komite Krško, LBB - puščanje pred zlomom.

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- replacement activities and power increase defined as reconstruction.

4 CONCLUSIONS

The main purpose of the Krško Modernization – Analysis and Licensing project is to perform all the analyses needed to prove compatibility of the replacement SGs' with the plant systems and to prove that all normal operation and accident conditions after replacement the SGs' installation and power uprating remain within the acceptance criteria. Identification of all the changes needed to satisfy the above criteria was included.

All analysis and work was reviewed in parallel by the Krško NPP, the TSO and the SNSA, resulting in a list of comments, categorized as suggestions, recommendations and required changes. After the clarification and/or resolution of all comments, the TSO prepared Independent Evaluation Report(s), which were submitted together with other licensing documentation to the SNSA for their approval. With this approach, all licensing issues (questions, concerns) were addressed and the whole process was performed in such a way as to enable the successful completion of the licensing process.

ABBREVIATIONS

NPP - Nuclear Power Plant, SG - Steam Generator, NSSS - Nuclear Steam Supply System, SNSA - Slovene Nuclear Safety Administration, SAR - Safety Analysis Report, TS - Technical Specification, TSO - Technical Support Organization, QA - Quality Assurance, KOC - Krško Operating Committee, KSC - Krško Safety Committee, LBB - Leak before break.

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Specifični simulator JEK - orodje za učinkovito usposabljanje operaterjev

The Plant-Specific Simulator - A Tool for the Efficient Training of Operators

Franc Pribožič

Popolni simulator je pomembna pridobitev za Jedrsko elektrarno Krško. Namen tega sestavka je prikazati uporabo simulatorjev na področju usposabljanja in povedati nekaj o specifičnem popolnem simulatorju naše elektrarne. Podanih je nekaj osnovnih informacij o načinu usposabljanja osebja z dovoljenjem za operaterja in vlogi simulatorja za prihodnje usposabljanje v JE Krško.
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(Ključne besede: simulatorji, analize, modeliranje, usposabljanje)

The acquisition of a full-scope simulator represents a very important achievement for the Krško Nuclear Power Plant (NPP). The aim of this paper is to present the use of simulators in the training process and to give some basic information about our plant-specific simulator. Some basic information is provided about the training process for licensed personnel and the role of the simulator in the future training activities at Krško NPP.
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(Keywords: simulators, analysis, modelling, training)

0 UVOD

Upravljanje s sistemi jedrske elektrarne je nedvomno zahtevno in zapleteno opravilo. Za opravljanje takšnih opravil je seveda najpomembnejše ustrezno usposabljanje osebja, ki upravlja z napravami elektrarne iz glavne komandne sobe. Pri usposabljanju za taka opravila je uporaba simulatorjev bistvena in takega usposabljanja ni mogoče nadomestiti z drugimi tehnikami.

Bistvena vloga simulatorja je, da omogoča hitrejše, kakovostnejše, ponovljivo in nenevarno usposabljanje za upravljanje procesov, ki zahtevajo visoko stopnjo znanja ter veščin za ročne posege. Glavna področja človekove dejavnosti, pri katerih se uporabljajo simulatorji za usposabljanje, so: usposabljanje vojaških in civilnih pilotov, operaterjev jedrskih elektrarn, operaterjev v petrokemijski industriji, operaterjev elektrarn na standardna goriva ter upravljanje železniškega in cestnega prometa.

Glavni namen procesa modernizacije jedrske elektrarne Krško je zanesljivo, stabilno, varno in gospodarno obratovanje. Stabilnost obratovanja, znižanje števila nenačrtovanih zaustavitev, dvig razpoložljivosti, dvig moči (in s tem letne

0 FORWORD

The operation of nuclear power plant systems is a demanding and complex task. The most important element in ensuring that the capability exists for handling such tasks is the efficient training of the operations personnel who manipulate controls in the main control room. The use of simulators during the training process is essential and cannot be substituted by other techniques.

The simulator's advantage is that it enables faster, higher quality, repeatable, and safe training of the personnel responsible for handling the processes that require a high level of knowledge and skills. Simulators are widely used for training purposes in the following areas of human activity: training of military and civil pilots, nuclear power plant operators, operators in the petrochemical industry, operators in conventional power plants, the control of railroad traffic and the control of road traffic.

The main goal of the Krško nuclear power plant (NPP) modernization project is to provide reliable, stable, safe and economical operation. Stable operation, the reduction of unplanned shutdowns, an increase of in availability, power upgrade (and consequentially, annual production) and the organizaproizvodnje) ter nenazadnje, zagotovitev mednarodno primerljivega obsega usposabljanja so naloge, ki ob bok in tudi v oporo preostalim projektom modernizacije postavljajo tudi specifični simulator s pripadajočo opremo.

Specifični popolni simulator podrobno ponazarja delovanje jedrske elektrarne Krško. Na taki napravi bo mogoče opravljati usposabljanje operativnega osebja v enakem obsegu, kakor je to praksa v razvitih državah. Veljalo bi omeniti tudi to, da je popolne simulatorje kupilo že kar precej elektrarn iz držav v razvoju. Simulator za JE Krško je izdelalo kanadsko podjetje CAE Electronics, Ltd., ki je eno od vodilnih proizvajalcev različnih tipov simulatorjev in sistemov za upravljanje postrojenj.

1 SPLOŠNO O SIMULATORJIH

Kaj sploh je simulator? Pravzaprav je to vsaka naprava, ki v umetnem okolju ponazarja določen dejanski proces. Industrija si z različnimi simulacijskimi napravami pomaga pri preverjanju in izpopolnjevanju svojih izdelkov. Simulatorji so se pokazali tudi kot zelo učinkovito sredstvo za usposabljanje osebja za upravljanje zapletenih sistemov. Ena od bistvenih značilnic simulatorja za usposabljanje je ta, da mora delovati v realnem času, torej se mora odzivati z enako dinamiko kakor dejanski proces.

Prve enostavne simulatorje so izdelali za potrebe hitrejšega usposabljanja pilotov za vojaške namene. Lahko bi rekli, da so nastali, zaradi potrebe, da se v čim krajšem času osebje usposobi za obvladovanje določenih zahtevnih nalog. Te izkušnje so se s pridom začele uporabljati v preostali industriji in pričela se je doba, ko simulacijske naprave v čedalje večji meri pomagajo pri usposabljanju osebja, ki nadzoruje in upravlja postopke v zahtevnih tehnologijah.

Razvoj simulatorjev tesno sledi razvoju računalniške tehnologije in elektronike na splošno. Glavne omejitve simulatorjev so bile v preteklosti povezane s pomnilniško in procesno močjo računalnikov. Skokovit razvoj računalništva je pripomogel k temu, da današnji simulatorji uporabljajo zelo podrobne modele, kar omogoča prožnost pri uporabi.

Simulatorje, ki se uporabljajo za usposabljanje osebja jedrskih elektrarn, lahko v grobem razdelimo na tri skupine: simulatorji osnovnih principov, delni simulatorji in popolni simulatorji.

1.1 Simulatorji osnovnih principov se uporabljajo v postopku začetnega usposabljanja in tudi za periodično obnavljanje znanja predvsem za pridobivanje in vzdrževanje teoretičnega znanja. Interakcija med človekom in simulatorjem običajno

tion of training at a comparable level for the nuclear industry, are important plant missions. A full-scope simulator with associated infrastructure, supports the plant’s goals and other plant projects.

The plant specific full scope simulator replicates the Krško NPP systems’ operation in detail, thus representing a suitable vehicle for performing the training of operations personnel at the same level of quality and quantity as is standard practice in developed countries. It is worth mentioning that a lot of power plants in developing countries have already acquired simulators. The Krško NPP simulator was built by the Canadian company, CAE Electronics Ltd., which is one of the leading manufacturers of simulators and control systems.

1 GENERAL INFORMATION ABOUT SIMULATORS

What is a simulator? As a matter of fact, it is any device that replicates a real process in an artificial environment. Simulation devices are widely used by various industries for functionality verification or research and development of their products. Simulators have proved themselves to be very efficient tools for training the personnel that operate/handle complex systems. One of the essential training simulator characteristics is operation in real time, it must dynamically perform in the same time scale as the real process.

The construction of the first primitive simulators was driven by the need to train military pilots more quickly. We could say that the development of simulators was pushed by a necessity to train people for complex tasks in a short period of time. Good experience with simulation devices encouraged an industry-wide implementation of this method. The use of simulation devices is increasingly helpful in supporting the training of operators for complex technologies.

The development of simulators closely follows the development of computer technology, and electronics in general. In the past, the major limiting factors in simulation were linked to computer memory and processing power. Rapid development of computer technology has facilitated the development of sophisticated models, giving practical flexibility to modern simulators.

The simulators that are used for the training of nuclear power plant personnel can be divided roughly into three categories: basic principle simulators, part task simulators and full scope simulators.

1.1 Basic Principle Simulators are typically used during initial training phases and for periodic refresher training, mainly in the area of fundamental theoretical knowledge. The man-machine interface is usually achieved via a computer keyboard and omogočata računalniška tipkovnica in grafični monitor. V prvi vrsti so namenjeni za interaktivno ponazarjanje teoretičnih načel, sistemi elektrarne so simulirani v omejenem obsegu.

1.2 Delni simulatorji se prav tako uporabljajo v začetnem in stalnem usposabljanju za pridobivanje in vzdrževanje teoretičnega znanja, obenem pa v omejenem obsegu omogočajo tudi urjenje za upravljanje sistemov elektrarne. Taki simulatorji so običajno kombinacija računalniških prikazovalnikov in poenostavljene komandne plošče, ki navadno nima enakih upravljalnih mehanizmov kakor v pravi upravni sobi.

1.3 Popolni specifični simulator je po videzu stvarna kopija glavne upravne sobe elektrarne z vsemi upravljalnimi mehanizmi in prikazovalniki podatkov, z računalniškim modelom pa stvarno ponazarja obnašanje elektrarne. Simulirani so vsi sistemi, ki se iz glavne komandne sobe upravljajo, ter tudi sistemi, katerih odzivi so bistveni za urjenje v uporabi obratovalnih postopkov. Na takšnem simulatorju je mogoče v celoti izvajati ukrepe v skladu z obratovalnimi postopki. Simuliranja pokrivajo celoten spekter normalnih obratovalnih stanj elektrarne in tudi nezgodne primere. Tak simulator omogoča usposabljanje za pridobivanje in vzdrževanje znanja ter spretnosti, potrebnih za normalno delo v komandni sobi.

2 POPOLNI SIMULATOR JEK

Obseg simuliranja je opredeljen na osnovi namembnosti simulatorja, torej usposabljanja v našem primeru. Simulator JE Krško bo omogočal urjenje za vse dejavnosti operaterjev, ki se izvajajo iz glavne komandne sobe ter iz lokalnih komandnih pultov za zasilno zaustavitev. Zgrajen je v skladu z ameriškim standardom ANSI/ANS-3.5 [1]. Ta standard upoštevajo tudi druge države kot kriterij za ustreznost simulatorja.

Nivo natančnosti modeliranja posameznih sistemov je odvisen od zapletenosti in pomembnosti posameznega sistema. Zahteve za natančnost modeliranja sistemov so podrobno opredeljene v tehnični specifikaciji za simulator. Sistemi so lahko modelirani popolnoma dinamično, poenostavljeno dinamično ali funkcionalno. Od 80 simuliranih sistemov jih je 48 modeliranih popolno dinamično (primeri: sistem primarnega hladila, reaktorska sredica, hlajenje komponent, turbina, sistem napajalne vode itn.), 14 poenostavljeno dinamično (primeri: sistem pomožne pare, protipožarni sistem, sistem inštrumentacijskega zraka, sistem za vzorčenje itn.) in 18 funkcionalno (primeri: sistem zapornic na Savi, sistem zaznave potresa, dodajanje kemikalij, sistem za pripravo vode itn.). Popolno dinamično

graphical monitor. These simulators enable an interactive representation of theoretical principles, supported by the limited simulation of real plant systems.

1.2 Part Task Simulators are also used during initial training phases and refresher training. These simulators also serve well for training on fundamental theoretical principles. In addition, they enable limited training on plant systems manipulation. Such simulators are typically a combination of computer terminals and simplified control boards. The control boards typically do not replicate real control room mechanisms.

1.3 A Full Scope Replica Simulator is a realistic copy of a real control room. It has all the relevant control mechanisms and data monitoring devices; a computer model realistically reproduces the plant's operational characteristics. All the essential plant systems that have controls in the main control room or respond to operator requested actions, and are essential for training on operating procedures are simulated. Such a simulator enables the full implementation of all actions that are required by operating procedures. The simulation covers the full spectrum of normal operating states and abnormal/accidental conditions. Such a simulator serves for the acquisition and for the retention of the knowledge and skills necessary to work as a control room operator in a NPP.

2 THE KRŠKO NPP FULL-SCOPE SIMULATOR

The scope of the simulation is defined, based on the purpose of the simulator, which in our case is training. The Krško NPP full scope simulator will enable the training of all operator activities that are performed from the main control room and from local shutdown panels. It has been built in accordance with the American standard: ANSI/ANS-3.5 [1]. This standard is used by most counties as acceptance criteria for determining simulator conformance.

The system's simulation modeling fidelity is based on system complexity and importance. Requirements for the system's modeling fidelity are defined in the technical specification for the simulator. Plant systems are simulated either fully dynamically, simplified dynamically or functionally. The Krško NPP simulator has 80 systems simulated. 48 systems are fully dynamically modeled (examples: reactor coolant system, reactor core, component cooling, main turbine, feedwater system, etc.), 14 systems simplified dynamically (examples: auxiliary steam system, fire protection system, instrument air system, sampling system, etc.) and 18 systems are modeled functionally (examples: Sava river dam system, seismic instrumentation system, chemical addition, demineralized water system, etc.). Full dynamic simuliranje sistemov upošteva fizikalne zakonitosti, kot so: ohranitev mase, gibalne količine in energije, termo- in hidrodinamiko, električne veličine, tehnične korelacije in topologijo sistemov. Fizikalne zakonitosti upošteva tudi poenostavljeno dinamično simuliranje, sistem pa je poenostavljen v delu, ki ne vpliva na odzivanje indikacij parametrov ali stanja mehanizmov za upravljanje na glavni komandni plošči. Funkcionalno simuliranje se uporablja za sisteme, ki imajo na glavni komandni plošči samo omejene indikacije.

Modeli vseh sistemov so ponazorjeni z objektno usmerjenim programskim orodjem. S posebnim programom je modelirana samo sredica. Kakovost objektno usmerjenih orodij za modeliranje je bila ena večjih prednosti izbranega dobavitelja. Vsak sistem je modeliran v treh nivojih: termo- in hidrodinamika, električna napajanja ter upravljanje. Sredica reaktorja je modelirana tridimenzionalno. Zaradi potrebe po potrditvi obnašanja simulatorja je bil osnovni model zgrajen glede na stanje elektrarne v 15. gorivnem ciklu. V končni fazi bo model simulatorja upošteval novo projektno stanje elektrarne z vgrajenima novima uparjalnikoma in z drugimi posodobitvami (17. gorivni cikel). Obratovalno osebje se bo tako lahko usposabljalo na simulatorju pred zagonom elektrarne z vgrajenima novima uparjalnikoma.

Za upravljanje samega simulatorja je inštruktorjem namenjena posebna delovna postaja. Ta omogoča zagon, zaustavitev simulatorja, shranjevanje trenutnega stanja v pomnilnik in ponovno proženje s shranjene točke, preverjanje pravilne lege upravljalnih mehanizmov, vrnitev nazaj po poteku dogodkov in ponovni zagon simuliranja, pripravo, shranjevanje in proženje vaj za usposabljanje, vnašanje napak, upravljanje opreme prek interaktivnih shem sistemov itn.

Simulator omogoča tudi pospešeno in upočasnjeno simuliranje nekaterih specifičnih pojavov: gretje primarnega kroga, gretje turbine, vzpostavljanje podtlaka v kondenzatorju, nastajanje in razgradnja razcepkov, ki absorbirajo nevtrone v sredici ipd. Takšna funkcionalnost je zelo pomembna pri usposabljanju, ker omogoči relativno hiter prehod stanj, ki v praksi zahtevajo veliko časa.

Vhodni podatki za projektiranje simulatorja so: dokumentacija elektrarne, podatkovna baza varnostnih analiz, rezultati analiz ter različni standardi. Med elektrarniško dokumentacijo spadajo: pretočne, električne in inštrumentacijske sheme, izometrični načrti, opisi sistemov, inštrukcijske knjige, končno varnostno poročilo, obratovalni postopki, sistemski postopki, obratovalni podatki. Pri pripravi nekaterih podatkov so sodelovali tudi domači inštituti.

Trenutno je simulator v sklepni fazi preverjanja in bo pripravljen za uporabo pri usposabljanju v začetku aprila 2000.

simulation is achieved by the application of the conservation laws of mass, momentum and energy, other physical laws of thermal and hydrodynamics, electric power engineering correlations and system topology. Simplified dynamic simulation is also achieved by the application of physical laws but the systems are simplified in a way that does not impact on parameter indications or control mechanisms on the main control board. Functional simulation is applied for the systems that have only limited indications on the main control board.

All plant system models are built by using object oriented software modeling tools. Special programming has been used for the reactor core modeling only. The quality of the object oriented modeling tools was one of the recognized advantages of the selected vendor. Each system is modeled from three aspects: thermo and hydrodynamics, electrical power and instrumentation as well as control. The reactor core is modeled three dimensionally. The basic simulator model was developed based on a plant configuration in core cycle 15 to enable simulator validation using good, operational plant data for comparison. The final deliverable simulator configuration will take into account new steam generators and major plant modifications (configuration for core cycle 17). Training will be conducted on this configuration prior to plant startup with the new steam generators.

The simulator control is performed through a special instructor workstation. This workstation enables simulator startup, shutdown, current simulation status storage, activation of old stored points, control board switch checks, backtrack capability. It also enables preparation, storage and activation of training exercises and the insertion of malfunctions. The entire simulator can be fully controlled from the instructor station using interactive schematics and a graphical interface.

The simulator has the capability of fast and slow simulation of certain specific evolutions, for example: primary system heatup, turbine heatup, establishing condenser vacuum, neutron absorbers buildup/decay, etc. This functionality is very useful during training as it enables faster passage through certain sequences that would, in real time, take a lot of time.

Simulator design input data were obtained from plant documentation, the safety analysis input database, analysis results and applicable standards. The following plant documentation categories were used: flow diagrams, electrical wiring diagrams, instrumentation interconnecting diagrams, isometric drawings, system descriptions, equipment manuals, safety analysis reports, operating procedures, operational data. Domestic institutes were also involved in the preparation of specific data.

Currently, the simulator is in its final phase of acceptance testing and will be ready for training in the first half of April 2000.

Uporaba lastnega specifičnega popolnega simulatorja vsekakor pomeni dvig ravni jedrske varnosti in izboljšanje kakovosti usposabljanja z namenom, da bi izboljšali razpoložljivosti. Nabava popolnega specifičnega simulatorja pomeni hkrati tudi izpolnitev upravne odločbe [2] Uprave Republike Slovenije za jedrsko varnost ter upoštevanje priporočil mednarodnih misij.

3 SIMULATOR PRI USPOSABLJANJU

Popolni simulator je nenadomestljiv pripomoček pri usposabljanju osebja elektrarne z dovoljenjem za operaterja. Omogoča kakovostno doseganje ustrezne usposobljenosti za opravljanje zahtevnih del. Operativno osebje mora imeti potrebno znanje in spretnosti za hitro in kakovostno prepoznavanje informacij procesa, njihovo interpretacijo, načrtovanje potrebnih akcij, koordiniranje akcij s sodelavci in izvedbo ustreznih manipulativnih posegov prek komandnih mehanizmov glavnih komandnih pultov.

Simulator omogoča urjenje posegov za normalna obratovalna stanja elektrarne, kot so: gretje sistemov, zagon, obratovanje na moči, spremembe moči, zaustavitev in ohlajevanje.

Simulator prav tako omogoča urjenje posegov za reševanje nenormalnih stanj. To so predvsem odpovedi različnih komponent sistemov, odpovedi instrumentarija, pojav manjših puščanj in podobno. To so pričakovani dogodki, ki se v elektrarnah občasno pojavljajo.

Naslednja skupina stanj, ki jih simulator ponazarja, so projektne nezgode. Take nezgode so: zlom cevi v uparjalniku, zlom primarnega cevovoda, zlom parnega voda, zlom napajalnega cevovoda itn. Takšne nezgode so sicer zelo redke. Do sedaj je v nekaj jedrskih elektrarnah prišlo do zloma cevi v uparjalniku.

V nekaterih primerih je za omejitev posledic pomembno časovno pravilno in usklajeno ukrepanje, kar se dosega samo z zadostno pogostostjo urjenj. V procesu usposabljanja je pomembno, da se lahko posamezne vaje po potrebi ponovijo, kar omogoča edino simulator.

Treba je poudariti, da večina jedrskih elektrarn pretežni del gorivnega cikla obratuje pri stalni moči. Manevriranje z močjo se po navadi izvaja načrtovano za potrebe preskušanj ali v primeru motenj v procesu ali v elektroenergetskem sistemu. Število načrtovanih in tudi nenačrtovanih zaustavitev je sorazmerno majhno. Zaradi tega je ravnanje s komandami v normalnem obratovanju relativno omejeno. Nedvomno pa je sposobnost operaterjev, da kar najbolje ravnajo z opremo, ključnega pomena, tako za varnost kakor tudi za razpoložljivost elektrarne. Za vzdrževanje ustrezne usposobljenosti (znanje in spretnost) so torej potrebna ponavljajoča urjenja.

Use of our own, plant specific full scope simulator represents an improvement in the nuclear safety and training system, as well as improving availability. Acquisition of a plant specific full scope simulator also represents the fulfillment of a licensing amendment requirement [2] of the Slovenian Nuclear Safety Administration and the recommendations of international missions.

3 THE SIMULATOR IN THE TRAINING PROCESS

A full scope simulator is an irreplaceable tool in the training process for operations licensed personnel. The preparation for the performance of demanding tasks requires an efficient and high quality training process. Operations personnel have to attain the necessary knowledge and skills in order to be able to: efficiently recognize relevant information from the process; interpret such information; as well as plan, coordinate and perform adequate manipulation of system control mechanisms from the main control boards.

The simulator supports the training of activities during normal plant evolutions such as: plant systems heatup, startup, power operation, power changes, shutdown and plant cooldown.

The simulator also supports training for the mitigation of abnormal plant conditions. Such conditions are: different component malfunctions, instrumentation failures, system leaks and similar problems. Such failures are expected at the plants and do happen occasionally.

The next category of simulated states which the simulator is capable of representing, are design basis accidents. Such accidents are: steam generator tube rupture, a loss of coolant accident, main steam line break, feed line break, etc. Such accidents are infrequent, few nuclear power plants have experienced steam-generator tube ruptures.

When an event occurs, timely performance and coordination of the appropriate actions is important for the successful mitigation of consequences. The necessary skills can be obtained only by the appropriate training. Repetition of such occurrences can only be done safely with a simulator.

It has to be pointed out that majority of nuclear power plants operate at stable power for most of their fuel cycle. Power maneuvers are usually planned because of system testing requirements or are performed due to disturbances on the electrical grid. The number of planned and forced plant shutdowns is relatively low. For this reason, manipulation of real control mechanisms is relatively limited. However, the ability of operators to act and optimally operate the equipment is extremely important for safety reasons and achieve good availability. To maintain the necessary abilities, skills and knowledge to act properly, periodic training must be performed.

4 USPOSABLJANJE KLJUČNEGA OPERATIVNEGA OSEBJA – dosedanja praksa

4.1 Program začetnega usposabljanja

Usposabljanje operaterja reaktorja traja približno dve leti in pol. Usposabljanje je razdeljeno na več faz. V prvi fazi slušatelji pridobijo potrebno teoretično znanje. V drugi fazi pridobijo osnovno znanje o sistemih elektrarne in obratovalnih postopkih. Prvi dve fazi sta izvedeni pretežno v obliki predavanj. V prvo fazo je vključen delež praktičnih laboratorijskih vaj, v drugi fazi pa delež praktičnega usposabljanja na lokalnih delovnih mestih v elektrarni. V prvi in drugi fazi se uporablja tudi simulator osnovnih principov kot podpora predavanjem. V tretji fazi poteka usposabljanje na popolnem simulatorju, ki je kombinacija predavanj in praktičnih urjenj na simulatorju. V četrti fazi poteka usposabljanje v elektrarni. V tej fazi usposabljanje zajema poglobljen in voden individualni študij sistemov in elektrarniške dokumentacije ter praktično usposabljanje v glavni komandni sobi. Usposabljanje se konča z izpitom pred strokovno komisijo Uprave Republike Slovenije za jedrsko varnost (URSJV). Uspešnim kandidatom URSJV podeli dovoljenja za operaterja reaktorja.

V preteklosti je JEK za začetno usposabljanje na popolnem simulatorju sklepala pogodbe s tujimi ponudniki takih storitev, kakor sta podjetji Westinghouse in NUS (Nuclear Utility Services). To usposabljanje traja približno štiri mesece in je izredno intenzivno. Za to usposabljanje se morajo udeleženci popolnoma prilagoditi sistemom in odzivom nespecifičnega simulatorja ter anglosaškim merskim enotam. Po koncu te faze je seveda potreben precejšen napor, da se udeleženci miselno prilagodijo našim merskim enotam in sistemom ter komandni plošči JEK. Uporaba lastnega specifičnega simulatorja bo pomenila izjemen pozitiven premik in bo praktično odstranila težave zaradi prilagajanja.

4.2 Program stalnega usposabljanja

Slovenski predpisi trenutno zahtevajo 28 ur urjenj na simulatorju letno za osebje, ki mora imeti dovoljenje za operaterja reaktorja ali glavnega operaterja. Zadnja leta je osebje z dovoljenjem operaterja JEK opravljalo redno letno usposabljanje na simulatorju elektrarne Ginna v Rochestru, ZDA. Pogodbeni izvajalec je bilo podjetje General Physics. V preteklosti sta vsakoletno usposabljanje izvajali še podjetji Westinghouse in NUS.

Nobeden od simulatorjev, do sedaj uporabljanih za usposabljanje naših operaterjev, ne ustreza natančno komandni sobi JEK in odzivom naše elektrarne. V okviru zmožnosti so za potrebe rednega

4 KEY OPERATIONS PERSONNEL TRAINING – practice up to the present

4.1 Initial Training Program

The initial training for a reactor operator lasts approximately two and a half years. The training is divided into several phases. During the first phase, trainees receive the necessary theoretical knowledge. During the second phase, knowledge of basic systems and operating procedures is attained. The first two phases consist mostly of classroom training. The first phase also includes practical laboratory exercises also and the second phase includes practical training on local (field) operator positions at the plant. The basic principle simulator is used during the first two phases to support classroom presentations. The third phase is simulator initial training, consisting of a combination of classroom presentations and simulator practical exercises. The fourth phase is conducted on site. This training includes in-depth guided self study of the plant systems using plant documentation, and practical on-the-job training in the plant's main control room. Training is concluded with a final examination, administered by a special commission, appointed by the Slovenian Nuclear Safety Administration (SNSA). Successful candidates receive a reactor operator license.

In the past, the initial simulator training on a full scope simulator was contracted as a service. Contractors like Westinghouse or Nuclear Utility Services (NUS) were used. This high intensity training lasted approximately four months. For such initial training the trainees needed to adapt to the systems and response of the full scope simulator that was not Krško NPP specific and used British units of measurement. After completion of this phase, considerable effort was required to re-adapt to the metric units and to Krško NPP systems and the main control boards layout. Use of our own, plant specific full scope simulator represents a very positive move and will eliminate adaptation difficulties.

4.2 The Continuing Training Program

Slovenian legislation currently requires 28 hours of simulator exercises per year for personnel holding reactor operator or senior reactor operator licenses. For the last few years, the annual retraining for the Krško NPP licensed personnel was organized at the Ginna plant full scope simulator in Rochester, USA. The contracted company was General Physics. In the past, regular retraining has also been conducted by Westinghouse and NUS.

None of the simulators used in the past for training of our operators was identical to the configuration of the Krško NPP plant control room and to its dynamic behavior. For the annual retraining, letnega usposabljanja prilagodili računalniški program simulatorja, tako da je obnašanje sistemov podobno kakor v naši elektrarni. V to področje spada prilagoditev karakteristik črpalk, nastavitvene vrednosti za krmilne in zaščitne sisteme in podobno. Seveda je to le približek, ker se osnovni program ne spreminja.

Uporaba nespecifičnega simulatorja ima seveda določene pomembne pomanjkljivosti pri izvedbi stalnega usposabljanja. Udeleženci usposabljanja se morajo vsako leto znova privajati na drugačno komandno ploščo in je tako del časa za urjenje na simulatorju neoptimalno izkoriščen. Odziv sistemov je, ne glede na prilagoditve, različen od odzivov naše elektrarne in seveda obstaja določena stopnja nezaupanja v tisto, kar operater vidi in na kar se mora odzvati. Obstaja tudi nevarnost, da bi operater glede na izkušnje iz nespecifičnega simulatorja pričakoval enako obnašanje naše elektrarne. Problemi se seveda lahko pojavijo tudi v primerih, ko so potrebne akcije časovno odvisne. Kakovostno lahko te probleme rešuje le uporaba specifičnega popolnega simulatorja.

5 USPOSABLJANJE V PRIHODNJE

Uporaba lastnega specifičnega simulatorja in ustrezna oprema bosta imeli velik neposredni vpliv na usposabljanje operativnega osebja, omogočena pa bo tudi uporaba v druge namene s skupnim ciljem: izboljšati varno in zanesljivo obratovanje elektrarne.

Celotni krog usposabljanja bo lahko izveden bolj optimalno, kakor do sedaj. V začetnem usposabljanju bodo udeleženci uporabljali simulator že v fazi spoznavanja sistemov. Odpadla bo potreba po spoznavanju sistemov, nespecifičnih za JEK. Preverjanje usposobljenosti bo strokovna komisija URSJV opravljala na simulatorju. Redno letno usposabljanje z uporabo simulatorja bo potekalo štirikrat na leto. To pomeni okrog 80 ur urjenja (trikrat več kakor doslej) na simulatorju za vsakega udeleženca. Tolikšno število simulatorskih ur je v razvitih državah standardna praksa. Ob tem pa bo dejanska odsotnost udeležencev od dela v izmeni zaradi rednega usposabljanja enaka kakor doslej.

Uporaba simulatorja, ki specifično modelira sisteme JEK, zagotavlja visoko stopnjo zaupanja v skladnost obnašanja simulatorja v primerjavi z obnašanjem elektrarne. Omogočena je neposredna uporaba obratovalnih postopkov elektrarne.

Lastni simulator, nameščen na sami lokaciji JEK, bo omogočil tudi izvajanje usposabljanja tik pred izvedbo načrtovanih obratovalnih posegov (zagon, zaustavitev), kar bo vplivalo na optimizacijo dela ter zmanjšalo možnost za zakasnitve.

the contractor was responsible for changes to the simulator model to replicate the Krško NPP systems as closely as possible. This included the adjustment of pump characteristics, setpoints for control and protection systems and similar alterations. This was, at best, only an approximation, as the basic system models could not be changed.

Use of a non-specific simulator has various important drawbacks and limitations for the conduct of continuing training. Attendees have to adapt to the non-specific simulator control board each year. Because of this, some of the time dedicated for training was not efficiently used. The system's response, regardless of the implemented changes, was very different. This unavoidably leads to a certain lack of confidence in what the operator sees and what represents the basis for his actions. One other concern is that the operator might expect a response from our plant based on his experience from the non-specific simulator. The problems might also appear in situations when the timing of actions is important. Such problems can be avoided only by the use of a plant specific full scope simulator.

5 TRAINING IN THE FUTURE

Use of our own plant specific simulator and the associated infrastructure will have a significant effect on the operations personnel training process, and in addition, it will also support secondary uses, with the common aim of improving safety and plant reliability.

The simulator will enable the optimization of the entire training cycle. The simulator will be used for demonstrations much earlier in the training process, during familiarization with plant systems. The need to become familiar with the systems of other plants will be eliminated. The licensing examination by the SNSA-appointed commission will be performed with the use of the simulator. Regular, annual, simulator retraining will be conducted four times per year. This means approximately 80 simulator hours (three times more than in the past) for every participant. Such a number of simulator hours is standard practice in developed countries. This will be achieved by using the same amount of time allocated for training as before.

The use of the simulator with software models specifically replicating the plant systems will assure confidence in the consistency of simulator performance compared to the real plant's behavior. Plant operating procedures will also be directly implemented on the simulator.

Having the simulator installed on site will enable just-in-time training. It will be possible to run training sessions just before the execution of planned plant evolutions (startup, shutdown), which will support work optimization in operations, and decrease the possibilities of time delays.

V usposabljanje z uporabo simulatorja bodo v določenem obsegu vključeni tudi strojniki opreme, ki po navodilih osebja glavne upravne sobe lokalno ravnajo z opremo. To bo zelo pomembno za izboljševanje skupinskega dela celotne izmenske skupine.

Simulator se bo uporabljal tudi za podporo pri pripravi in izvedbi usposabljanja po programu, ki izhaja iz načrta ukrepov v primeru izrednih dogodkov v elektrarni. Na ta način bodo scenariji bolj stvarni in s tem izurjenost za ukrepanje še boljša.

Za zagotovitev ustrezne uporabe simulatorja v usposabljanju so pomembni tudi priprava inštruktorske ekipe, prilagoditev učnih programov in prilagoditev sedanjih in priprava novih učnih gradiv. Vzporedno s projektom simulatorja se je JEK pripravljala tudi na uporabo simulatorja.

Programi usposabljanja osebja z dovoljenjem za operaterja (začetni in redni letni) bodo izpopolnjeni v skladu s sistematskim pristopom k usposabljanju, na podlagi rezultatov analize del in nalog za operaterje. Analiza je bila izvedena po vzoru metodologije ameriškega inštituta za obratovanje jedrskih elektrarn (INPO), ki se uporablja v ZDA. Sistematski pristop k usposabljanju priporoča tudi Mednarodna agencija za atomsko energijo (MAAE) [3]. Analiza temelji na elektrarniških obratovalnih postopkih in drugi dokumentaciji. Rezultat analize je nabor znanj in spretnosti, ki jih mora obvladati oseba z dovoljenjem za operaterja. Program usposabljanja bo oblikovan tako, da bo zagotovljeno izvajanje vseh evidentiranih spretnosti in uporabo potrebnega znanja. Bistvena prednost sistematskega pristopa k usposabljanju je tudi celovito dokumentiranje procesa. JEK si je zastavila cilj, da mora prilagojeni program ustrezati normativom za ugotavljanje ustreznosti usposabljanja, ki jih v ZDA uporablja INPO. Pričakuje se, da bo v prihodnosti v mednarodnem merilu takšno vlogo imelo Svetovno združenje upravljalcev jedrskih elektrarn (WANO). Tako bomo lahko pokazali, da je usposabljanje pri nas popolnoma primerljivo s stanjem v razvitih državah.

JEK je prav tako dopolnila tudi inštruktorsko skupino za obratovalno osebje. Celotna skupina je sodelovala pri projektu simulatorja v fazi izdelave in v fazi sklepnih preskušanj in si s tem pridobila že veliko potrebnih izkušenj za delo s simulatorjem. Seveda inštruktorsko delo zajema mnogo več kakor samo upravljanje s simulatorjem. V podporo lastnih inštruktorjev pa bo JEK v začetni fazi usposabljanja z lastim simulatorjem uporabljala tudi tuje strokovnjake.

6 SKLEP

V načinu usposabljanja operativnega osebja je Jedrska elektrarna Krško vedno sledila oziroma v okviru dejanskih možnosti vsaj skušala slediti

Field operators, responsible for operating plant equipment as directed by the crew of the main control room, will also participate occasionally in simulator training, together with licensed personnel. This will enhance team-work practices of the entire shift crew.

The simulator will also be used to support the preparation and conduction of drills required by the plant emergency response plan training program. In this way, the scenarios for emergency drills will be more realistic, resulting in better preparedness.

Certain prerequisites are essential to assure adequate introduction of our own simulator training. Such prerequisites are: well trained instructors, prepared training programs, written training materials, etc. The preparation for the simulator utilization has been done in parallel with the simulator project.

Licensed operator training programs (initial and continuing training) will be revised in accordance with the systematic approach to training (SAT) methodology, based on the results of job and task analyses that were performed for licensed job duties. The analyses were done based on the methodology developed by the Institute for Nuclear Power Operations (INPO), which is used in the United States. SAT methodology is also used by the International Atomic Energy Agency (IAEA) [3]. The analysis is based on plant operating procedures and other relevant plant documentation. The final result of the analyses is a list of knowledge and skills the licensed person must obtain and retain. The program will be appropriately structured to ensure training on all the identified skills and knowledge items. One essential benefit of using the systematic approach to training is a well documented process. It is Krško NPP's goal to structure the revised training programs in compliance with the criteria for training program accreditation used by INPO in the USA. It is expected that in the future the World Association of Nuclear Operators (WANO) will assume this function on an international scale using practically the same criteria. This means that we will be in position to prove that training is comparable to that in developed countries.

Krško NPP has increased the number of instructors for operations personnel. All instructors were also involved in the simulator project, during the construction phase as well as during the testing. In this way they gained a lot of experience for future work with the simulator. Running the simulator is, of course, just one part of the instructor duties. In the early phases of simulator utilization Krško NPP will use experienced consultants to support the work of our own instructors.

6 CONCLUSION

It was always Krško NPP policy to follow western practices in training. The circumstances in the past did not allow us to fully comply. During the zahodnemu svetu. V času, ko so zahodne elektrarne nabavljale svoje simulatorje, je neizogibno zaostala.

Elektrarna je tudi v preteklosti že večkrat poskušala uresničiti svojo željo po lastnem simulatorju. Ob pomoči upravne odločbe in z velikim vloženim delom elektrarne je sedaj simulator tukaj. Elektrarna bo v prihodnje imela možnost kakovostno izvajati tako redno usposabljanje kakor tudi usposabljanje novih generacij operaterjev.

period when western countries were building their simulators, we lagged behind.

The acquisition of our own full scope simulator has fulfilled a long held desire of Krško NPP. With the additional effect of the licensing amendment and considerable effort from the plant's side, the simulator has become a reality. Now the plant has in place the requiements for improving the quality of annual retraining and the initial training of new generations of operators.

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DOKTORATI

Na Fakulteti za strojništvo Univerze v Ljubljani je *dne 2. marca 2000* mag. **Pavle Žerovnik**, dipl.inž. z uspehom zagovarjal svojo doktorsko disertacijo z naslovom: "Raziskava integritete površin z magnetnimi metodami".

S tem je navedeni kandidat dosegel akademsko stopnjo doktorja tehničnih znanosti.

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Na Fakulteti za strojništvo Univerze v Ljubljani so z uspehom zagovarjali svoja magistrska dela, in sicer:

dne 9. marca 2000: **Slavko Božič,** dip.inž., magistrsko delo z naslovom: "Vplivi vrste in mase jekel ter ohlajevalnih sredstev na njihove lastnosti po kaljenju";

dne 16. marca 2000: **Boštjan Pečnik**, dipl.inž., magistrsko delo z naslovom: "Analizna stanja materiala na osnovi magnetnega Barkhausnovega šuma" in **Petar Orbanić**, dipl.inž., magistrsko delo z naslovom: "Izpopolnjevanje razvojnih postopkov z vrednotenjem konstrukcij na zagotovljivost";

dne 23. marca 2000: **Matjaž Samarin**, dipl.inž., magistrsko delo z naslovom: "Naprava za merjenje enoosne relaksacije viskoelastičnih materialov";

dne 24. marca 2000: **Pavel Kaiba**, dipl.inž., magistrsko delo z naslovom: "Programsko okolje za analizo in konstruiranje vrtljivih zvez";

dne 27. marca 2000: **Boštjan Lukančič**, dipl.inž., magistrsko delo z naslovom: "Vpliv togostne strukture vrtljive zveze na nosilnost vijačnih zvez in zobniške dvojice".

S tem so navedeni kandidati dosegli akademsko stopnjo magistra tehničnih znanosti.

DIPLOMIRALI SO

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Instructions for Authors

Navodila avtorjem
Instructions for Authors

Članki morajo vsebovati:
- naslov, povzetek, besedilo članka in podnaslove slik v slovenskem in angleškem jeziku,
- dvojezične preglednice in slike (diagrami, risbe ali fotografije),
- seznam literature in
- podatke o avtorjih.

Strojniški vestnik izhaja od leta 1992 v dveh jezikih, tj. v slovenščini in angleščini, zato je obvezen prevod v angleščino. Obe besedili morata biti strokovno in jezikovno med seboj usklajeni. Članki naj bodo kratki in naj obsegajo približno 8 tipkanih strani. Izjemoma so strokovni članki, na željo avtorja, lahko tudi samo v slovenščini, vsebovati pa morajo angleški povzetek.

Vsebina članka

Članek naj bo napisan v naslednji obliki:
- Naslov, ki primerno opisuje vsebino članka.
- Povzetek, ki naj bo skrajšana oblika članka in naj ne presega 250 besed. Povzetek mora vsebovati osnove, jedro in cilje raziskave, uporabljeno metodologijo dela,povzetek rezulatov in osnovne sklepe.
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- Teorija.
- Eksperimentalni del, ki naj vsebuje podatke o postavitvi preskusa in metode, uporabljene pri pridobitvi rezultatov.
- Rezultati, ki naj bodo jasno prikazani, po potrebi v obliki slik in preglednic.
- Razprava, v kateri naj bodo prikazane povezave in posplošitve, uporabljene za pridobitev rezultatov. Prikazana naj bo tudi pomembnost rezultatov in primerjava s poprej objavljenimi deli. (Zaradi narave posameznih raziskav so lahko rezultati in razprava, za jasnost in preprostejše bralčevo razumevanje, združeni v eno poglavje.)
- Sklepi, v katerih naj bo prikazan en ali več sklepov, ki izhajajo iz rezultatov in razprave.
- Literatura, ki mora biti v besedilu oštevilčena zaporedno in označena z oglatimi oklepaji [1] ter na koncu članka zbrana v seznamu literature. Vse opombe naj bodo označene z uporabo dvignjene številke[1].

Oblika članka

Besedilo naj bo pisano na listih formata A4, z dvojnim presledkom med vrstami in s 3 cm širokim robom, da je dovolj prostora za popravke lektorjev. Najbolje je, da pripravite besedilo v urejevalnilku Microsoft Word. Če uporabljate kakšen drug urejevalnik besedil, prosimo, da besedilo konvertirate v navadno ASCII (tekstovno) obliko. Hkrati dostavite odtis članka na papirju, vključno z vsemi slikami in preglednicami ter identično kopijo v elektronski obliki.

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Enačbe naj bodo v besedilu postavljene v ločene vrstice in na desnem robu označene s tekočo številko v okroglih oklepajih

Enote in okrajšave

V besedilu, preglednicah in slikah uporabljajte le standardne označbe in okrajšave SI. Simbole fizikalnih veličin v besedilu pišite poševno (kurzivno), (npr. *v, T, n* itn.). Simbole enot, ki sestojijo iz črk, pa pokončno (npr. ms^{-1}, K, min, mm itn.).

Papers submitted for publication should comprise:
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- Information about the authors.

Since 1992, the Journal of Mechanical Engineering has been published bilingually, in Slovenian and English. The two texts must be compatible both in terms of technical content and language. Papers should be as short as possible and should on average comprise 8 typed pages. In exceptional cases, at the request of the authors, speciality papers may be written only in Slovene, but must include an English abstract.

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The paper should be written in the following format:
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- A Theory
- An Experimental section, which should provide details of the experimental set-up and the methods used for obtaining the results.
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- A Discussion section, which should describe the relationships and generalisations shown by the results and discuss the significance of the results making comparisons with previously published work. (Because of the nature of some studies it may be appropriate to combine the Results and Discussion sections into a single section to improve the clarity and make it easier for the reader.)
- Conclusions, which should present one or more conclusions that have been drawn from the results and subsequent discussion.
- References, which must be numbered consecutively in the text using square brackets [1] and collected together in a reference list at the end of the paper. Any footnotes should be indicated by the use of a superscript[1].

The layout of the text

Texts should be written in A4 format, with double spacing and margins of 3 cm to provide editors with space to write in their corrections. Microsoft Word for Windows is the preferred format for submission. If you use another word processor, please convert to normal ASCII (text) format. One hard copy, including all figures, tables and illustrations and an identical electronic version of the manuscript must be submitted simultaneously.

Please do not use a LaTeX text editor, since this is not compatible with the publishing procedure of the Journal of Mechanical Engineering. Graphs, tables and equations in LaTeX may be supplied in good quality hard-copy format, so that they can be copied for inclusion in the Journal.

Equations should be on a separate line in the main body of the text and marked on the right-hand side of the page with numbers in round brackets.

Units and abbreviations

Only standard SI symbols and abbreviations should be used in the text, tables and figures. Symbols for physical quantities in the text should be written in Italics (e.g. *v, T, n,* etc.). Symbols for units that consist of letters should be in plain text (e.g. ms^{-1}, K, min, mm, etc.).

Vse okrajšave naj bodo, ko se prvič pojavijo, napisane v celoti, npr. časovno spremenljiva geometrija (ČSG).

Slike

Slike morajo biti zaporedno oštevilčene in označene, v besedilu in podnaslovu, kot sl. 1, sl. 2 itn. Posnete naj bodo v kateremkoli od razširjenih formatov, npr. BMP, JPG, GIF. Za pripravo diagramov in risb priporočamo CDR format (CorelDraw), saj so slike v njem vektorske in jih lahko pri končni obdelavi preprosto povečujemo ali pomanjšujemo.

Pri označevanju osi v diagramih, kadar je le mogoče, uporabite označbe veličin (npr. *t, v, m* itn.), da ni potrebno dvojezično označevanje. V diagramih z več krivuljami, mora biti vsaka krivulja označena. Pomen oznake mora biti pojasnjen v podnapisu slike.

Vse označbe na slikah morajo biti dvojezične.

Za vse slike po fotografskih posnetkih je treba priložiti izvirne fotografije ali kakovostno narejen posnetek. V izjemnih primerih so lahko slike tudi barvne.

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Vsi podnaslovi preglednic morajo biti dvojezični.

Seznam literature

Vsa literatura mora biti navedena v seznamu na koncu članka v prikazani obliki po vrsti za revije, zbornike in knjige:

[1] Tarng, Y.S., Y.S. Wang (1994) A new adaptive controler for constant turning force. *Int J Adv Manuf Technol* 9(1994) London, pp. 211-216.
[2] Čuš, F., J. Balič (1996) Rationale Gestaltung der organisatorischen Abläufe im Werkzeugwesen. *Proceedings of International Conference on Computer Integration Manufacturing,* Zakopane, 14.-17. maj 1996.
[3] Oertli, P.C. (1977) Praktische Wirtschaftskybernetik. *Carl Hanser Verlag,* München.

Podatki o avtorjih

Članku priložite tudi podatke o avtorjih: imena, nazive, popolne poštne naslove, številke telefona in faksa ter naslove elektronske pošte.

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All abbreviations should be spelt out in full on first appearance, e.g., variable time geometry (VTG).

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Figures must be cited in consecutive numerical order in the text and referred to in both the text and the caption as Fig. 1, Fig. 2, etc. Figures may be saved in any common format, e.g. BMP, GIF, JPG. However, the use of CDR format (CorelDraw) is recommended for graphs and line drawings, since vector images can be easily reduced or enlarged during final processing of the paper.

When labelling axes, physical quantities, e.g. *t, v, m,* etc. should be used whenever possible to minimise the need to label the axes in two languages. Multi-curve graphs should have individual curves marked with a symbol, the meaning of the symbol should be explained in the figure caption.

All figure captions must be bilingual.

Good quality black-and-white photographs or scanned images should be supplied for illustrations. In certain circumstances, colour figures may be considered.

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All table captions must be bilingual.

The list of references

References should be collected at the end of the paper in the following styles for journals, proceedings and books, respectively:

[1] Tarng, Y.S., Y.S. Wang (1994) A new adaptive controler for constant turning force. *Int J Adv Manuf Technol* 9(1994) London, pp. 211-216.
[2] Čuš, F., J. Balič (1996) Rationale Gestaltung der organisatorischen Abläufe im Werkzeugwesen. *Proceedings of International Conference on Computer Integration Manufacturing,* Zakopane, 14.-17. maj 1996.
[3] Oertli, P.C. (1977) Praktische Wirtschaftskybernetik. *Carl Hanser Verlag,* München.

Author information

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