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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

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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.

(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.

(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.

6 LITERATURA
6 REFERENCES

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