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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

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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.

(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.

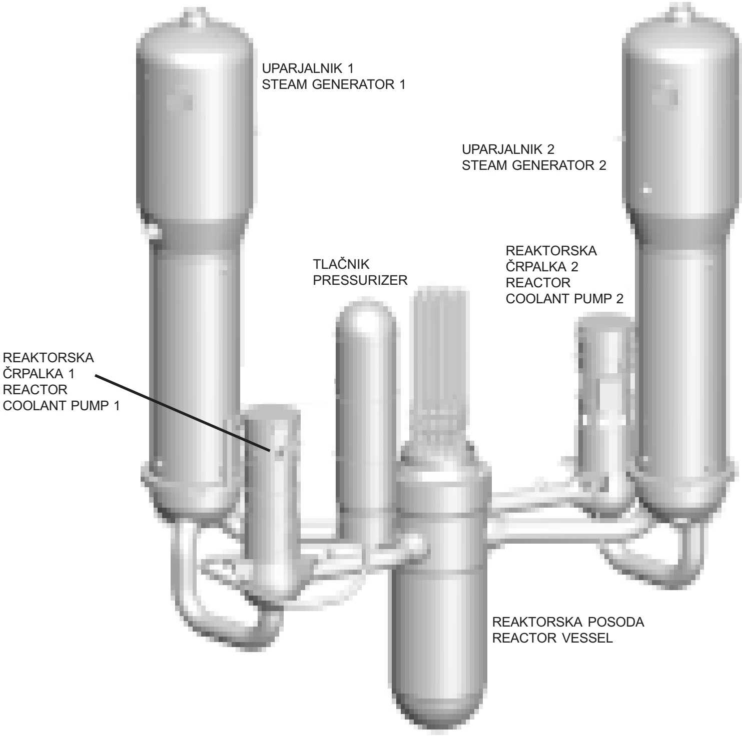
(Keywords: integrity, mechanical analysis, structural analysis, leak before break (LBB))

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.

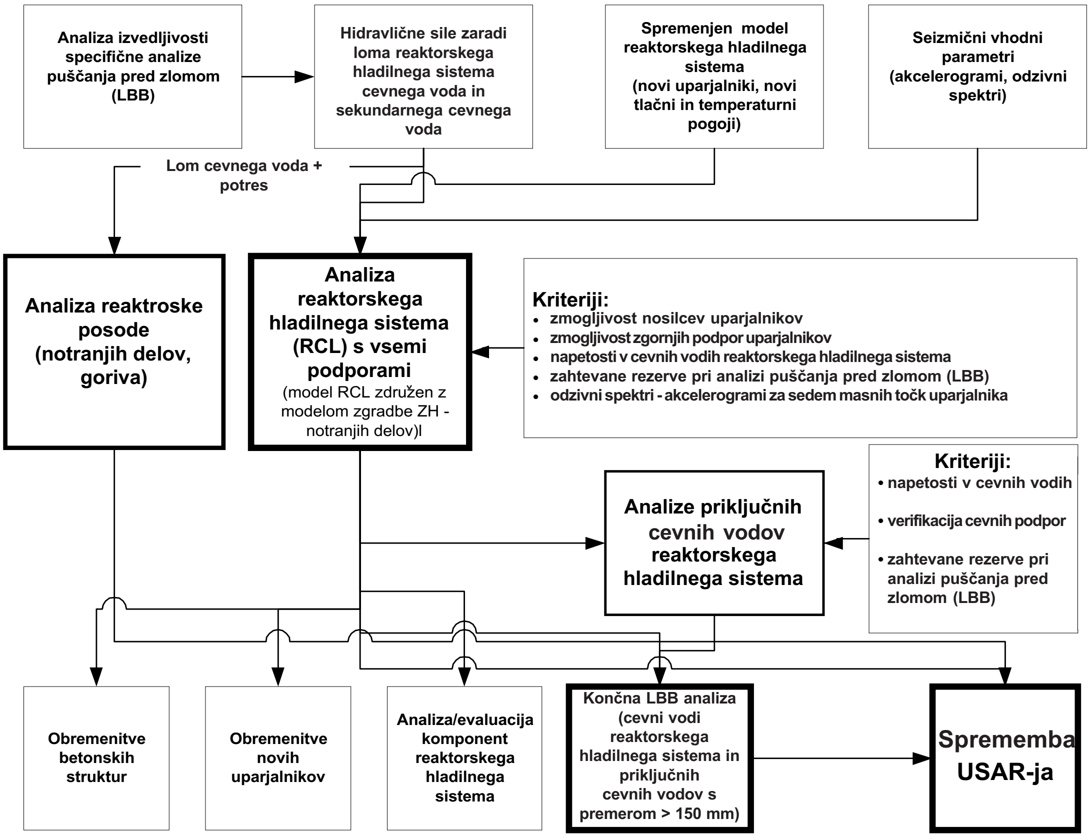
Zaradi naštetih razlogov je bilo treba pripraviti tudi nove vhodne podatke za mehanske in trdnostne analize:

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*

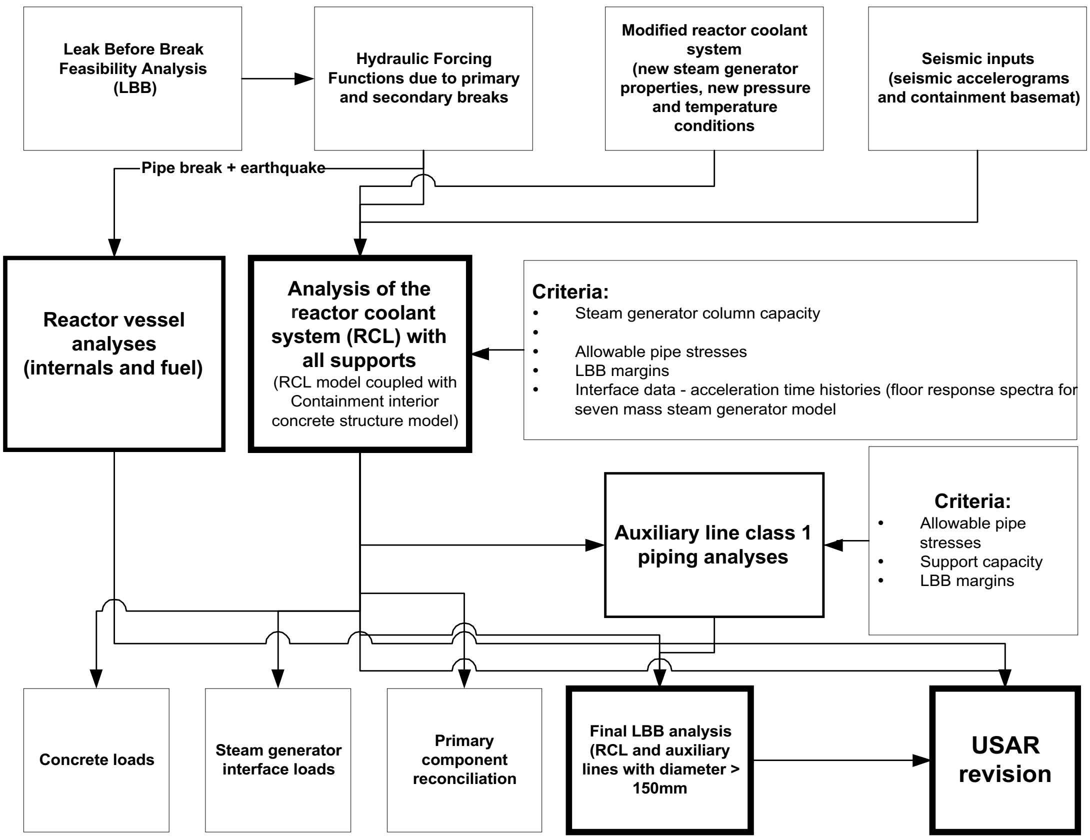


Fig. 2. *Structural and mechanical analyses flowchart for Krško NPP*

- 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.

- 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 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.

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.

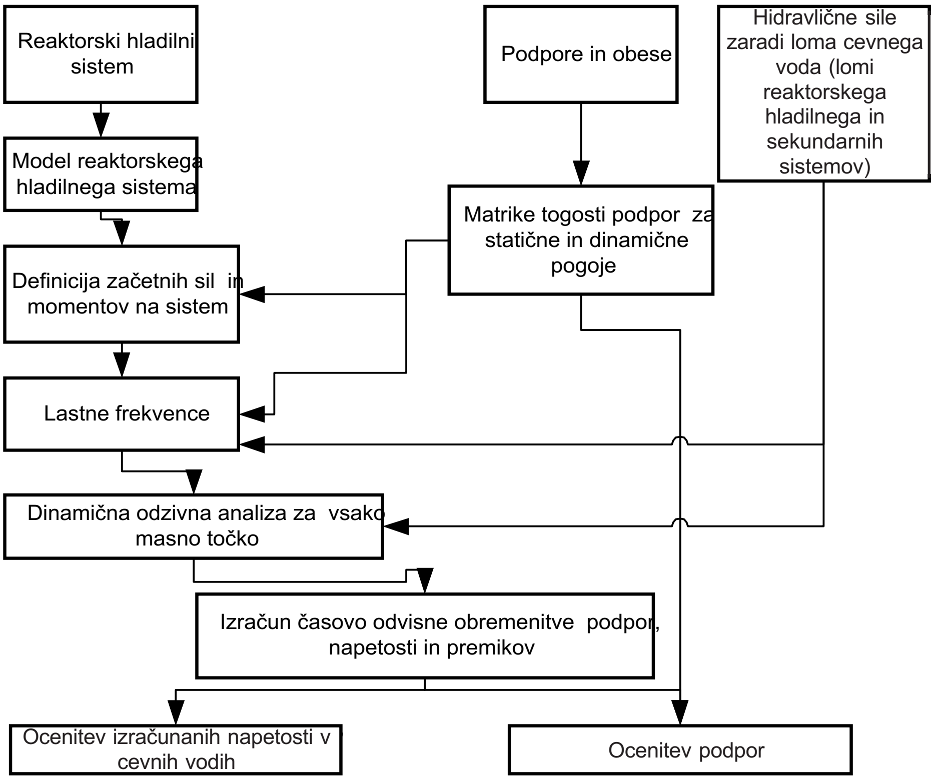
2.4 Fatigue Analysis

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*

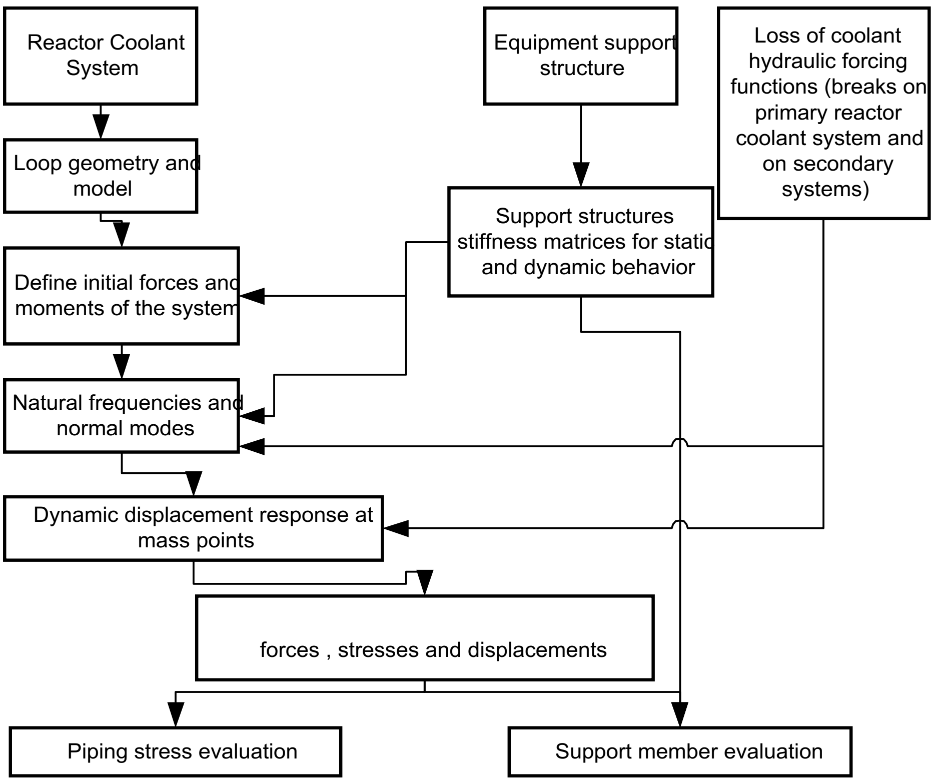


Fig. 3. *Flowchart of the required steps in a loss-of-coolant accident (including FW and MS break) RCL dynamic response analysis*

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.

4.1 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.

4.2 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 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 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 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 reanalyzed 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]).

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.

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.

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 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 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

8 LITERATURA
8 REFERENCES

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