International Agreement Report

Assessment of RELAP5/MOD3.3 against Single Main Steam Isolation Valve Closure Events at the Krško Nuclear Power Plant

Prepared by:
I. Parzer, B. Mavko

Jožef Stefan Institute
Jamova cesta 39
SI-1000 Ljubljana, Slovenia

A. Calvo, NRC Project Manager

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ABSTRACT

Recently, the last frozen version of RELAP5/MOD3.3 was released, before the code will be merged with another best estimate thermal-hydraulic system code (TRAC) into an integrated thermal-hydraulic code. Therefore, it is of utmost importance to assess models built using the RELAP5/MOD3.3 code against real plant transients before the code merger.

The paper presents the RELAP5/MOD3.3 analysis of two abnormal events that occurred at the Krško Nuclear Power Plant in September 1995 and January 1997, before replacement of the steam generators (SGs) in 2000. The events resulted from the sudden closure of a main steam isolation valve (MSIV). In the September 1995 event, a malfunction in the MSIV in SG-1 caused inadvertent valve closure. In the January 1997 event, the valve stem in SG-2 was broken, also causing sudden valve closure. Researchers obtained valuable plant data from these real plant transients and performed the RELAP5/MOD3.3 code assessment.

The analyses used a full two-loop plant model, developed at the Jožef Stefan Institute in Ljubljana, Slovenia. The model assumed old Westinghouse D4-type SGs with preheaters. It assumed that 18-percent of the U-tubes were plugged in both SGs.

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ACKNOWLEDGEMENTS

The plant data are courtesy of Krško Nuclear Power Plant.

1. INTRODUCTION

Recently, the last frozen version of RELAP5/MOD3.3 was released, before the code will be merged with another best estimate thermal-hydraulic system code (TRAC) into an integrated thermal-hydraulic code. Therefore, it is of utmost importance to assess models built using the RELAP5/MOD3.3 code against real plant transients before the code merger.

Available plant data from various abnormal events or incidents are of great importance for assessing large system thermal-hydraulic computer codes like RELAP5/MOD3.3. Two abnormal events with similar characteristics occurred at the Krško Nuclear Power Plant (NPP), which is a two-loop Westinghouse pressurized-water reactor (PWR) plant, on September 25, 1995, at 10:22:06 and on January 1, 1997, at 8:33:30. Both transients resulted from the sudden closure of a main steam isolation valve (MSIV).

The first event, in 1995, resulted from a malfunction in the regulation of the MSIV for steam generator (SG)-1, while the 1997 event occurred because of an SG-2 MSIV stem breach. The 1997 event resulted in slightly faster MSIV closure than the event in 1995. Both events occurred with the original Westinghouse D4 SGs installed. In 1995, these SGs were already considerably degraded, with high levels of SG tube plugging (18.87 percent for SG-1 and 17.27 percent for SG-2). The plugging level was reduced during the 1996 outage to 16.27 percent for SG-1 and 10.05 percent for SG-2; therefore, the plant was in a somewhat different state for the 1997 event. Nevertheless, for code assessment purposes, such data from real plant transients are of exceptional value, even if parts of the primary and secondary systems have been replaced.

2. PLANT DESCRIPTION

The Krško NPP is a Westinghouse two-loop PWR plant, in commercial operation since 1983.

2.1 Containment

The Krško NPP uses a cylindrical steel shell with a hemispherical dome and ellipsoidal bottom designed to accommodate normal operating loads, functional loads resulting from a loss-of-coolant accident, and the most severe loading predicted for seismic activity. A concrete shield building surrounds the steel shell to provide biological shielding for both normal and accident conditions and to collect and hold leakage from the containment vessel. Inside the containment structure, concrete shields the reactor and other nuclear steam supply system components. In addition to a containment spray system, a containment recirculation and cooling system removes postaccident heat.

2.2 Turbine Building

The turbine building contains the turbine generator and all the accessories related to power conversion. The building is of closed construction and is designed in accordance with local and national building codes. It does not contain any safety-related equipment.

2.3 Auxiliary Building

The auxiliary building structures are of reinforced concrete design with shear walls and beam and slab floor systems. The portion of the auxiliary building that is below grade is suitably protected with a waterproofing membrane to prevent the intrusion of ground water. In addition, redundant safety equipment below grade is located in separate compartments to preclude simultaneous flooding in the event of a fluid-system rupture.

2.4 Intake Structure

The intake structure consists of two separate substructures: (1) a nonsafety-category structure containing the main condenser circulating pumps and related equipment and (2) a safety-category structure containing the service water pumps and the related equipment. A dam across the Sava River, with the pumping station and water intake and discharge structures, provides cooling water intake. Two batteries of cooling cells provide combined cooling (Sava River and cooling towers) in the event of low river flow rates.

2.5 Fuel Handling Building

The fuel handling building is an integral part of the auxiliary building. It is a reinforced concrete structure that uses shear walls and beam and slab floor systems. The spent fuel pool within the fuel handling building is lined with stainless steel to prevent water leakage.

2.6 Nuclear Steam Supply System

The power rating of the Krško NPP nuclear steam supply system is 1,882 megawatts thermal (MWt), composed of 1,876 MWt core power output plus 6 MWt of reactor coolant pump (RCP) heat input. The nuclear steam supply system consists of a PWR, reactor coolant system (RCS), and associated auxiliary fluid systems. The RCS is arranged as two closed reactor coolant loops connected in parallel to the reactor vessel, each containing an RCP and an SG. An electrically heated pressurizer is connected to one of the loops.

2.7 Reactor Core

The multiregion-type reactor core is composed of 121 fuel assemblies, and 33 of them contain control rod clusters. Square spacer grid assemblies and the upper and lower end fitting assemblies support the fuel rods in the fuel assemblies. Each fuel assembly is composed of 16x16 rods; fuel rods use only 235 of these places. Of the remaining 21 places, 20 are distributed evenly and symmetrically across the cross-section of the assembly and contain thimble tubes that may be reserved for control rods, and one place contains a control instrumentation tube for the incore thimble.

All fuel assemblies are mechanically identical, although they differ in terms of fuel enrichment. Fuel assemblies with the highest enrichments are placed in the core periphery, and the two groups of lower enrichment fuel are arranged in a pattern in the central region. The core design approach depends on the strategy for plant operation (length of cycle) and improvements in fuel design.

2.8 Reactor Coolant System

The RCS consists of two reactor coolant loops connected in parallel to the reactor vessel, each loop containing an RCP and an SG. The coolant loops are filled with high-pressure water driven by the RCPs. The water circulates through the reactor core to remove the heat generated by the nuclear chain reaction from the fuel assemblies. The heated water exits from the reactor vessel and passes through the coolant loop piping to the SG. Once at the SG, the water gives up its heat to the feedwater to generate steam for the turbine generator.

The RCPs, one per coolant loop, are Westinghouse vertical, single-stage, centrifugal pumps of the shaft-seal type. The power supply system to the pumps is designed so that adequate coolant flow is maintained to cool the reactor core under all conceivable circumstances. The pump capacity is about 17,000 tons/hour. All pump parts in contact with the coolant are made of austenitic steel or are covered in stainless steel.

The SGs, one per loop, are Westinghouse D-4 vertical U-tube units with preheaters. Internal moisture separation equipment reduces the moisture content of the steam to 0.10 percent or less.

The reactor coolant piping and all of the pressure-containing and heat transfer surfaces in contact with the reactor water are made of stainless steel, except for the SG tubes and fuel tubes, which are Inconel and zircaloy, respectively. Reactor core internals, including control rod drive shafts, are primarily made of stainless steel.

An electrically heated pressurizer connected to one reactor coolant loop maintains RCS pressure during normal operation, limits pressure variations during plant load transients, and keeps system pressure within design limits during abnormal conditions.

2.9 Auxiliary Systems

Auxiliary system components are provided to charge the RCS and to add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactivity control (chemical and volume control system (CVCS)), cool system components (component cooling system), remove residual heat when the reactor is shut down (residual heat removal system), change fuel and cool the spent fuel storage pool, sample the reactor coolant water, provide for emergency safety injection (SI), and vent and drain the RCS.

2.10 Engineered Safety Features Systems

Engineered safety features are provided to prevent accident propagation or to limit the consequences of postulated accidents that might otherwise lead to system damage and the release of fission products. The principal criterion for these features is that under the conditions of a hypothetical loss-of-coolant accident, the system can, even when operating with partial effectiveness, maintain the integrity of the containment and limit the potential offsite radiation dose to less than the values of applicable Federal regulations (Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(1)). The Krško NPP includes the following engineered safety features:

- containment spray system
- hydrogen control system
- emergency core cooling system
- component cooling water system
- essential service water system
- auxiliary feedwater system

3. TRANSIENT DESCRIPTION

Both events at the Krško NPP occurred at 100-percent reactor power. The initial steam flow in SG-1 was slightly higher than in SG-2, and the corresponding SG-1 pressure was slightly lower than in SG-2 because of asymmetric SG tube plugging (18.87 percent in SG-1 and 17.27 percent in SG-2 in 1995, and 16.27 percent in SG-1 and 10.05 percent in SG-2 in 1997). In the 1995 event, a malfunction in the SG-1 MSIV control circuit caused inadvertent valve closure, while in the 1997 event, the SG-2 valve stem was broken, causing sudden valve closure. Plant parameters stayed within safety margins.

Table 1 shows the initial conditions for both events, according to the available plant data.

Table 1 Initial Conditions

Parameter	1995 Event	1997 Event
Core power	1876 MWt	1876 MWt
pressurizer pressure	15.47 MPa	15.47 MPa
pressurizer level	62.3%	62.8%
$T_{average}$	578.5 K	578.8 K
T_{hot} (hot leg 1/hot leg 2)	596.2/596.8 K	596.4/596.4 K
T_{cold} (cold leg 1/cold leg 2)	560.6/560.5 K	561.4/560.9 K
SG-1/SG-2 pressure	5.19/5.24 MPa	5.32/5.35 MPa
SG-1/SG-2 WR level	64.3/64.3%	63.1/63.1%
SG-1/SG-2 NR level	60.1/60.3%	60.6/60.8%

The flow through the affected SG MSIV stopped in a few seconds in the first event and almost instantaneously in the second. The pressure in the affected SG rose to the SG power-operated relief valve (PORV) setpoint shortly after. Since the heat was still generated in the core, the flow in the intact SG increased rapidly and the pressure dropped. The reactor protection system detected this drop in pressure. The SI and reactor trip signals were generated a few seconds later. Reactor scram followed, causing turbine trip and main feedwater isolation. In addition, the intact SG was isolated at the same time the SI signal was produced, and the pressure in this SG also started to rise. Thus, the pressure in both SGs was soon stabilized at the SG PORV setpoint. Closer examination of the secondary pressure indicated possible steam leakage through an MSIV or through some steamline drain valves.

4. INPUT MODEL DESCRIPTION

The analyses used a full two-loop plant model, developed at Jožef Stefan Institute (Refs. 1–6). The model assumed old Westinghouse D4-type SGs and that 18 percent of the U-tubes were plugged in both SGs, reflecting the condition for which the plant was licensed before SG replacement and power uprate in 2000. Although the levels of SG plugging differed slightly for each event, researchers used the same RELAP5/MOD3.3 master input model to simulate both transients.

The model consisted of 183 volumes, connected with 200 junctions. Plant structure was represented by 203 heat structures with 705 mesh points, while the reactor protection and regulation systems, safety systems operational logic, and plant instrumentation were represented by 109 logical conditions (trips) and 180 control variables. Figure 1 shows the plant nodalization used for transient simulation.

In the following the hydrodynamic components are described, representing the reactor vessel, the primary piping, emergency core cooling system (ECCS), steam generators, main feedwater system and auxiliary feedwater system.

The following components, numbered from 101 to 112 in Figure 1, represented the reactor vessel:

101 and 102	lower downcomer
103 and 104	lower head
105	lower plenum
106	reactor core
107	core bypass
108 and 113	core outlet
109	upper plenum
110	upper head
111	upper downcomer
112	guide tubes

The active core region was divided into an odd number of volumes to better capture the conservative “chopped cosine” axial power profile. Point reactor kinetics model is used, setting “Fission product yield factor” on card 300000001 to the best estimate value of 1.0. The coolant temperature in the reactor vessel upper head was initialized to a value of T_{cold} + 0.75 degrees C, according to the vendor’s information. This was achieved by balancing flows from the upper plenum and the upper downcomer and the flow through the guide tubes.

Components 202 and 220 represented the pressurizer surge line and the pressurizer vessel, respectively. Pressurizer spray lines were connected to the top of the pressurizer vessel with valves 230 and 231. Valves 211 and 212 modeled only the two pressurizer PORVs.

The following components represented the primary piping:

801 and 802	hot leg 1
803 and 804	intermediate leg 1

805 and 806	cold leg 1
701	hot leg 2
703 and 704	intermediate leg 2
705 and 706	cold leg 2

Loops were symmetrical except for the pressurizer surge line and the CVCS connections layout.

The main RCP was modeled by a special RELAP5/MOD3.3 pump model, using built-in homologous curves.

Figure 1 shows the ECCS nodalization and connections separately. The hydrodynamic components representing high-pressure injection system pumps were time-dependent junctions 704 and 804, while time-dependent junctions 750 and 850 represented low-pressure injection system pumps. Accumulators, numbered 701 and 801, provided cold leg injection only. The ECCS connected to both cold legs (junctions 819 and 719). Direct vessel ECCS injection through junctions 746 and 748 opened with a 3-minute delay after generation of the SI signal.

At the time of both events, the plant had the original Westinghouse type D-4 SGs with preheaters. In the original design, all of the main feedwater flow was directed into the cold part below the riser. Because of strong vibrations, the main feedwater piping was modified, redirecting 30 percent of the main feedwater flow through the auxiliary feedwater nozzle and feeding it from the top into the SG downcomer.

As shown in Figure 1, the inlet and outlet plenum represented the primary side of the SG, with a single pipe representing the U-tube bundle:

401-01 and 301-01	inlet plenum (hot side)
401 and 301	U-tubes
413 and 313	outlet plenum (cold side)

The hot side of each tube bundle was discretized in eight volumes, while the cold side was discretized in only five volumes.

The following hydrodynamic components represented parts of the SG secondary side:

422 and 322	mixing part of the preheater
403 and 303	preheater
402 and 302	riser
404 and 304	boiling region
411, 408 and 311, 308	SG pool
407 and 307	separator
406, 428 and 306, 328	separator bypass
409 and 309	downcomer
410 and 310	steam dome

Main steamlines were represented by two pipes (420, 430 and 320, 330 respectively), divided by MSIVs. Relief valves were only modeled and were situated upstream of the isolation valves. The turbine and steam dump flow are regulated by the logic.

Main feedwater piping was modeled according to the modified design. The 70-percent branch was modeled by two pipes (400, 300 and 442, 342), while the 30-percent part was modeled by volumes (432, 332 and 436, 336).

Auxiliary feedwater injected into the common pipe with the 30-percent part of the main feedwater (volumes 437 and 337). Both flows were then directed into the SG pool volumes (408 and 308).

The logic of the following regulation systems was modeled:

- control rod movement
- pressurizer level control
- pressurizer pressure control
- SG level control
- steam dump operation
- turbine regulation (digital electrohydraulic control system)

The reactor protection system included the following:

- manual trip by operator
- overpower ΔT (OPΔT) and over temperature ΔT (OTΔT) protection
- low primary flow in any of the loops
- high pressurizer pressure
- low pressurizer pressure
- high pressurizer level
- low-low SG level
- SI signal
- turbine trip

The following were inputs for the modeled safety systems' triggering logic:

- manual operator action
- reactor trip
- feedwater isolation
- auxiliary feedwater pumps start
- low steamline pressure
- low pressurizer pressure

The plant was operating at 1,876 MWt core power and approximately 578.5 K average primary temperature for both events. The primary pressure was about 15.5 megapascals (MPa) and the secondary pressure was approximately 5.7 MPa, differing slightly for the two events. Researchers introduced some adjustments into the original master input deck, mostly concerning valve closing times and characteristics (faster area reduction versus stem position). MSIV-1 and MSIV-2 closure times were shortened from 5 seconds to 1 second for the modeling the 1995 event. The MSIV-2 closure time was shortened to 0.1 second to simulate the 1997 event to capture fast closure because of stem breach.

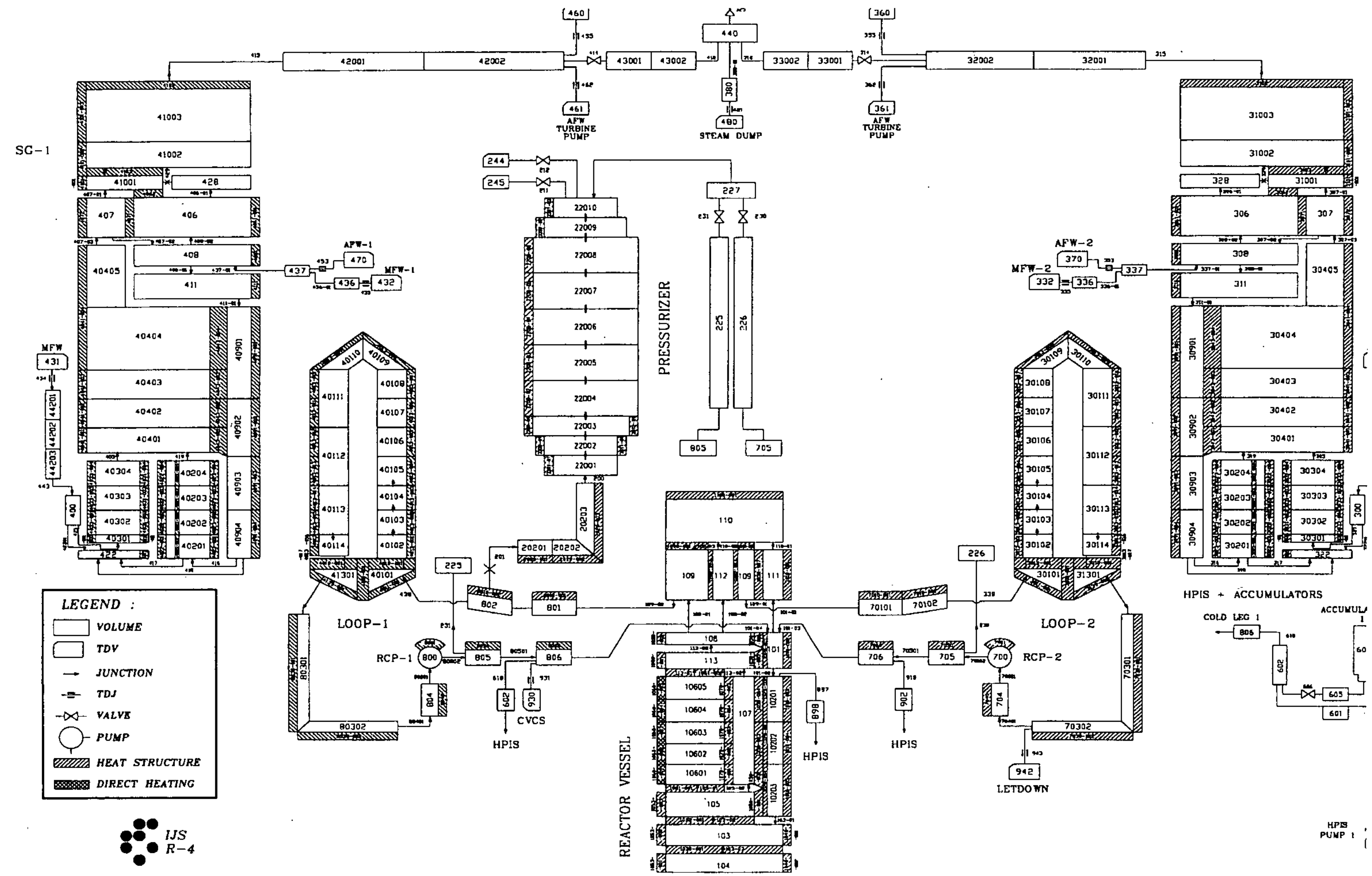

Figure 1 Krško NPP nodalization scheme

5. RESULTS

The objective of this study was to assess the RELAP5/MOD3.3 code against plant measured data in the initial transient stage, when the response of the plant and automatic regulation, protection, and safety systems to the initial event could be verified.

Figures 2 to 7 compare only the first 50 seconds of the calculation to the plant measured data, since the course of the transient after this period depends strongly on the operator actions (SI reset, CVCS charging and letdown flow, auxiliary feedwater flow). The CVCS and auxiliary feedwater flows were set to the values matching the plant data.

After the initial event, the secondary steam flow in the affected SG was reduced rapidly in the RELAP5/MOD3.3 simulation, which caused a sudden pressure rise. Figures 2 and 3 show the opposite for the intact SG; namely, the steam flow rapidly increased and the SG pressure dropped. After about 2–3 seconds, this simultaneously triggered the SI signal and closure of the intact SG MSIV. The turbine tripped immediately after the intact MSIV closure, while the reactor scram occurred 1 second later, followed by the main feedwater closure another second later. The timing of events is shown in Table 2.

Table 2 Sequence of Events in MOD3.3 Calculation

Event	1995 Event	1997 Event
Initial event	0 s	0 s
SI signal	2.9 s	1.7 s
Intact MSIV closure	2.9 s	1.7 s
Turbine trip	3.0 s	1.7 s
Scram	4.0 s	2.7 s
Main feedwater closure	5.2 s	3.8 s
RCP trip	did not occur	did not occur
CVCS behavior	plant data	plant data
Auxiliary feedwater flow	plant data	plant data

Plant data for the secondary pressure evolution show that the 1995 event (Figure 4) was slightly slower than the 1997 event (Figure 5) due to slower closing of the valve. The RELAP5/MOD3.3 simulation captured this for both events.

These differences in secondary pressure can be explained by analyzing the details of both initiating events. In 1995, the affected MSIV-1 closed smoothly in response to the malfunction in the regulation system, while in 1997 the stem breach caused uncontrolled oscillatory closing of MSIV-2. In 1997, a large flow oscillation at the very beginning of the transient caused strong pressure oscillations, and the MSIV closure signal was triggered earlier. Since the oscillatory closing of the MSIV could not be modeled in RELAP5/MOD3.3 using a motor valve component, the RELAP5/MOD3.3 calculation could not demonstrate those differences between the two events. Nevertheless, RELAP5/MOD3.3 predicted primary and secondary parameters that were mostly very close to the plant data.

A comparison of the RELAP5/MOD3.3 prediction indicates that the primary parameters did not match as closely with the plant data as the secondary parameters did, but they still reached good agreement (Figure 6). The primary pressure rise in the RELAP5/MOD3.3 simulation was slightly faster after the initial event than the plant recordings showed. The small disagreement in initial conditions and the expected error band in the RELAP5/MOD3.3 prediction could justify relatively small discrepancies. Plant data show that the initial pressure rise during 1995 event was slightly higher than during the 1997 event, which RELAP5/MOD3.3 correctly reproduced.

The last parameter compared to the plant data was the pressurizer level (Figure 7). As with the pressurizer pressure, the RELAP5/MOD3.3 simulation showed a slightly sharper rise in the initial level than did the plant data. Researchers observed a somewhat more moderate rise in the initial level for the 1997 event than for the 1995 event, which confirms the findings related to the pressurizer pressure. Some discrepancy originates in the difference in the initial pressurizer level between the plant data and the RELAP5/MOD3.3 calculation, but RELAP5/MOD3.3 correctly predicted the parameter evolution trends.

Another possible explanation for the differences in the prediction of primary parameters could lie in the modeling of spray flow and RCP seal leak flow, which contribute to the ability to correctly predict primary pressure and inventory. The plant data did not include spray flow and RCP seal leak flow measurements, introducing another potential source of inaccuracy.

Figures 8–11 show the long-term development of some primary and secondary parameters in comparison with the plant data. Secondary parameters such as SG levels matched the plant measured data closely over the long term (1,800 seconds), because the feedwater flow input into the model was based on the plant measured data. In addition, researchers based the secondary leak introduced into the model on the plant data; therefore, the SG pressures also matched the plant measured data closely over the long term (1,800 seconds).

The simulation resulted in almost perfect agreement for the long-term development of SG pressure for the 1995 event (Figure 8) and very good agreement for the 1997 event (Figure 9), because researchers used plant data as the basis for the steam leak from both SGs introduced into the model.

The primary parameters experienced more discrepancies. For example, the long-term development of the pressurizer level in the simulation did not match the plant data for the 1995 event very closely (Figure 10), while the agreement was much better for the 1997 event (Figure 11).

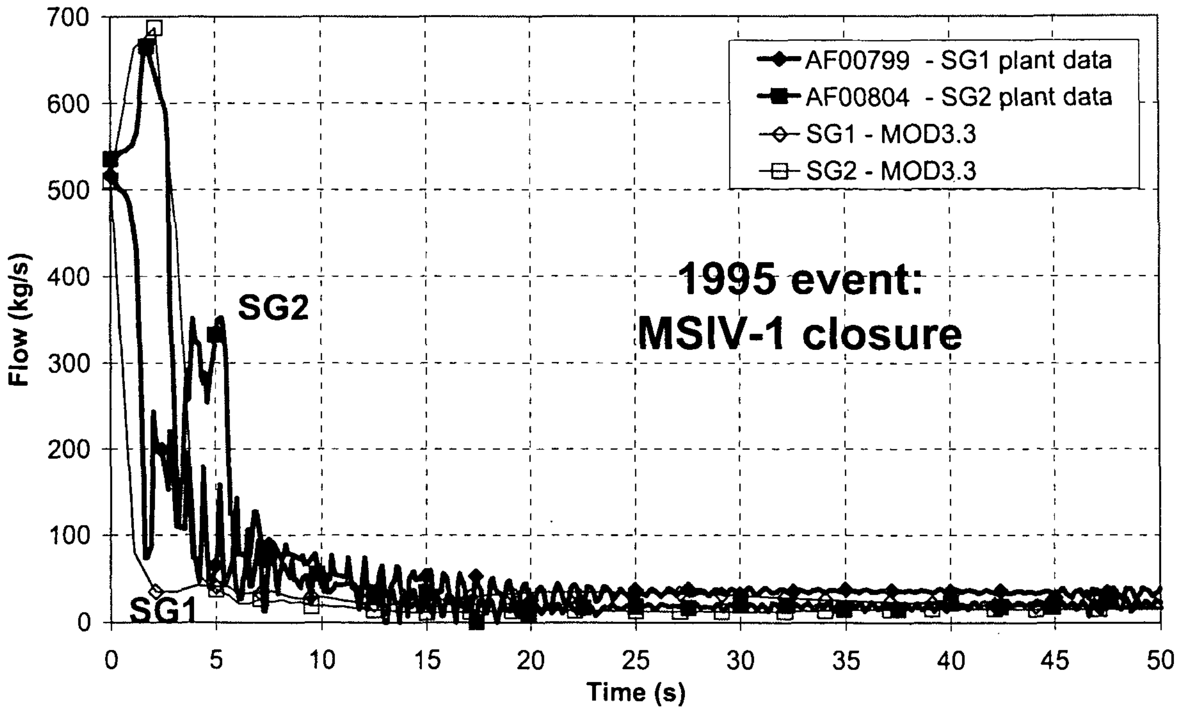

Figure 2 1995 event—steam flow

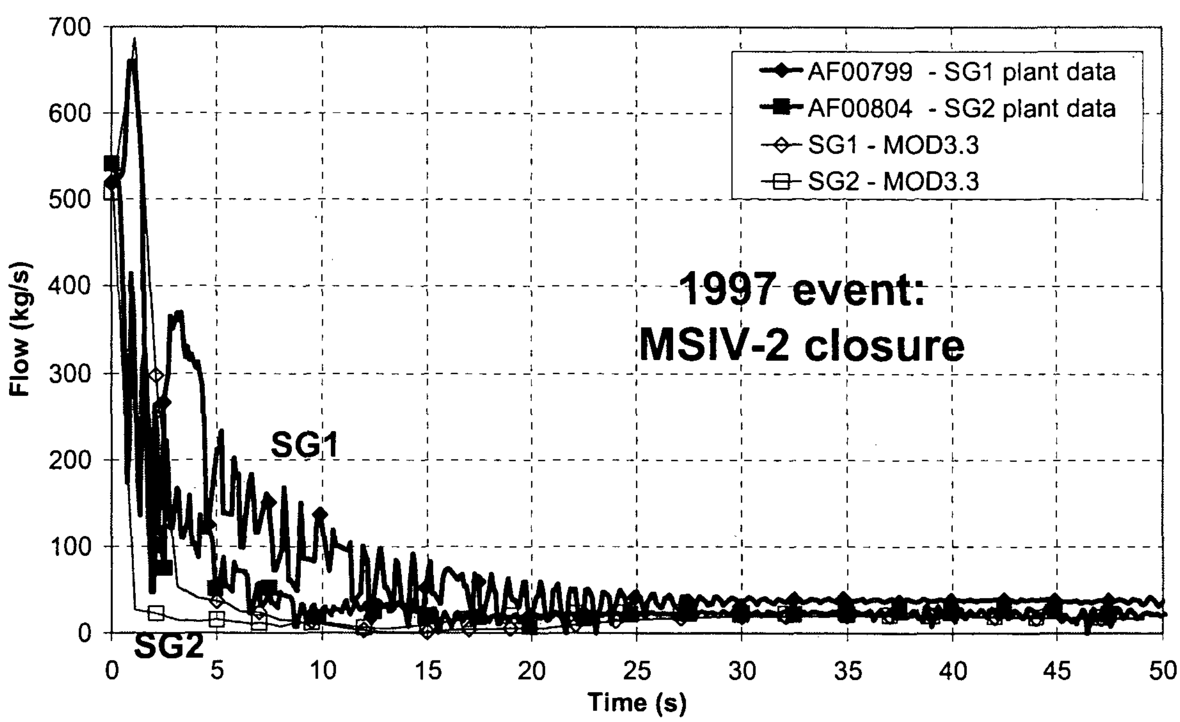

Figure 3 1997 event—steam flow

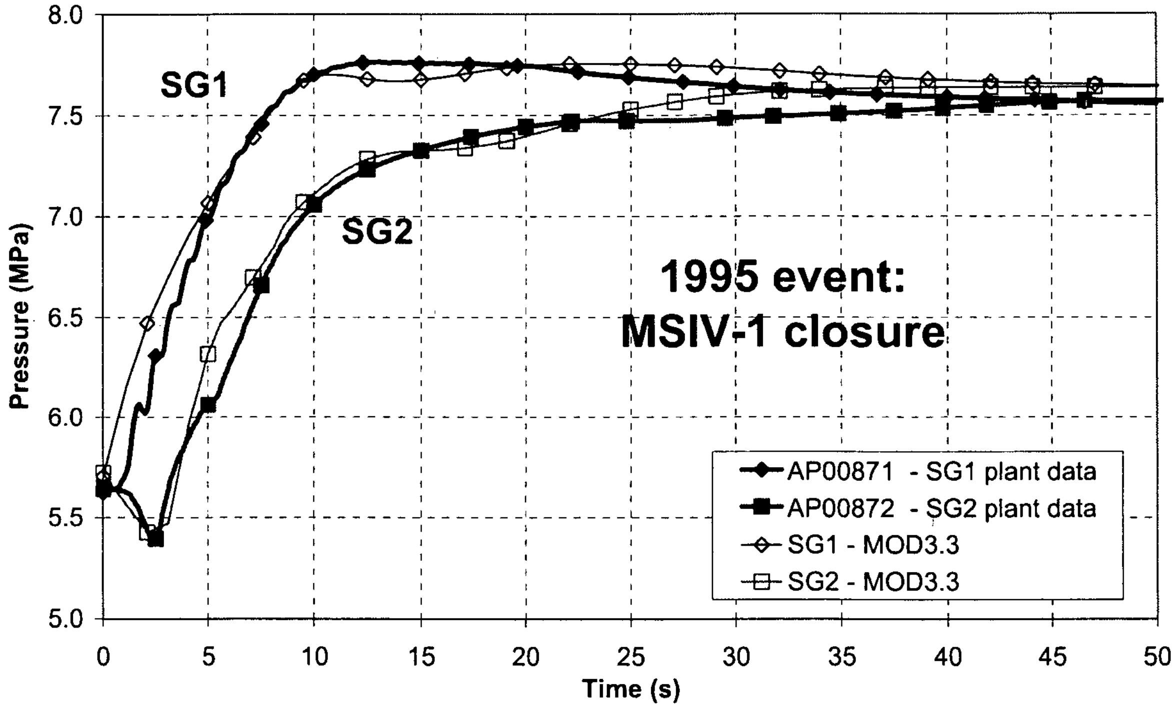

Figure 4 1995 event—SG pressure

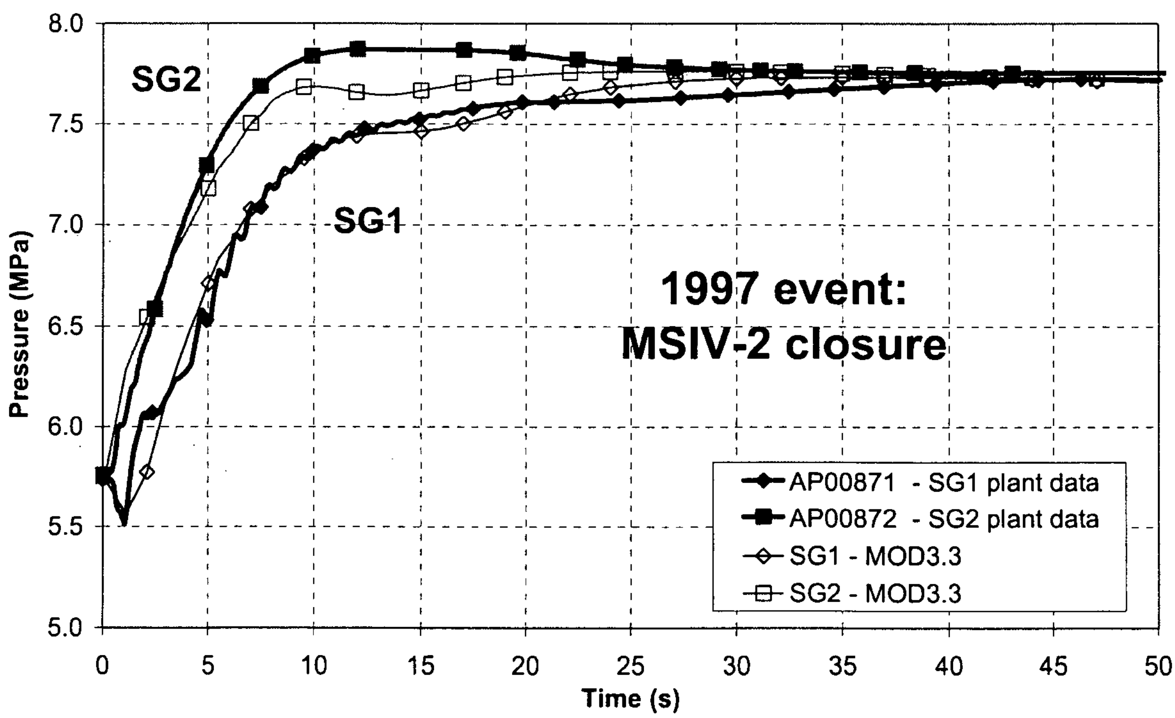

Figure 5 1997 event—SG pressure

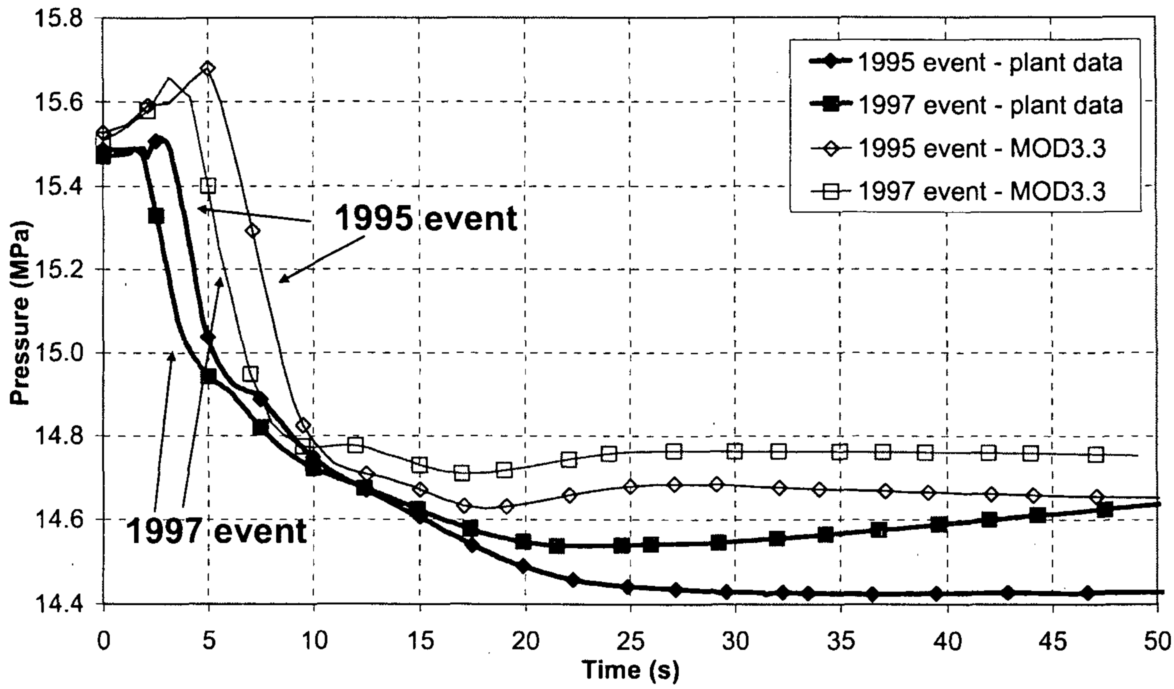

Figure 6 Pressurizer pressure

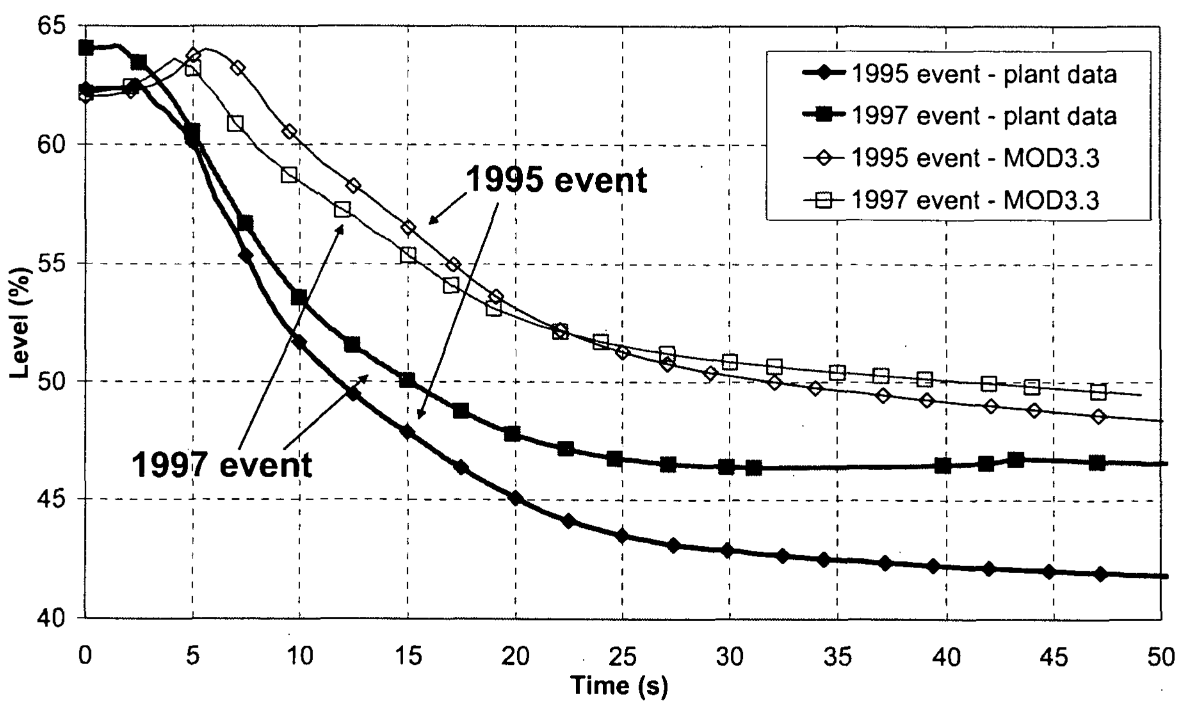

Figure 7 Pressurizer level

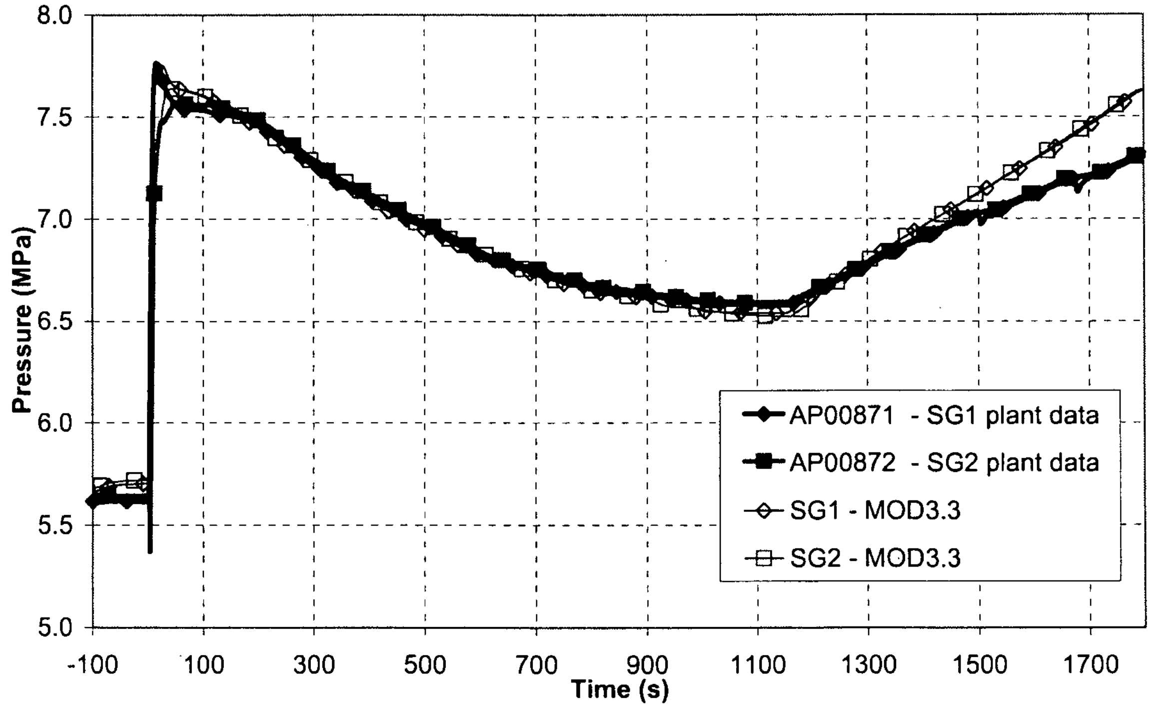

Figure 8 1995 event—SG pressure—long term

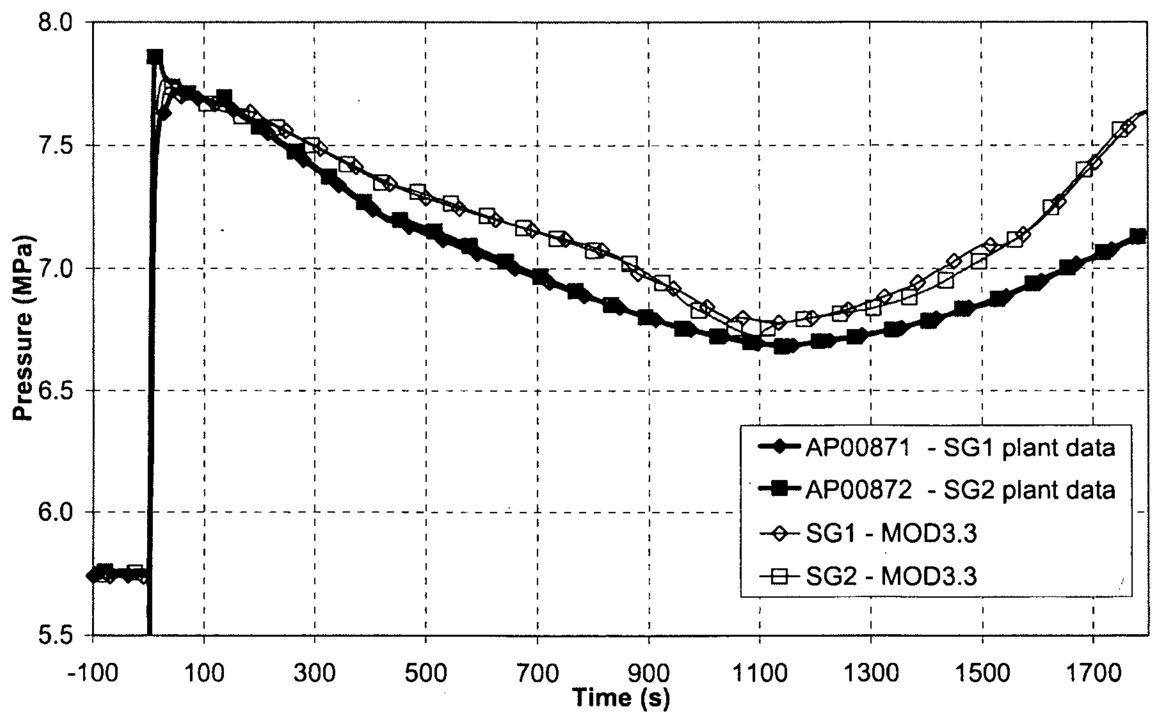

Figure 9 1997 event—SG pressure—long term

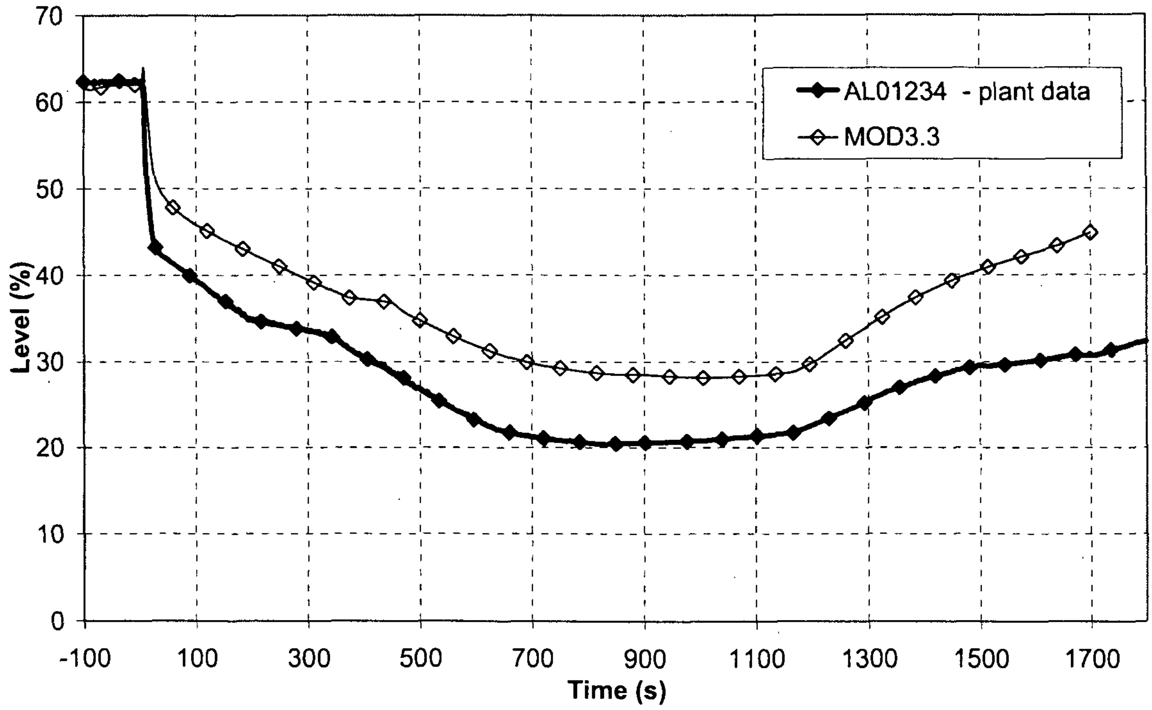

Figure 10 1995 event—pressurizer level—long term

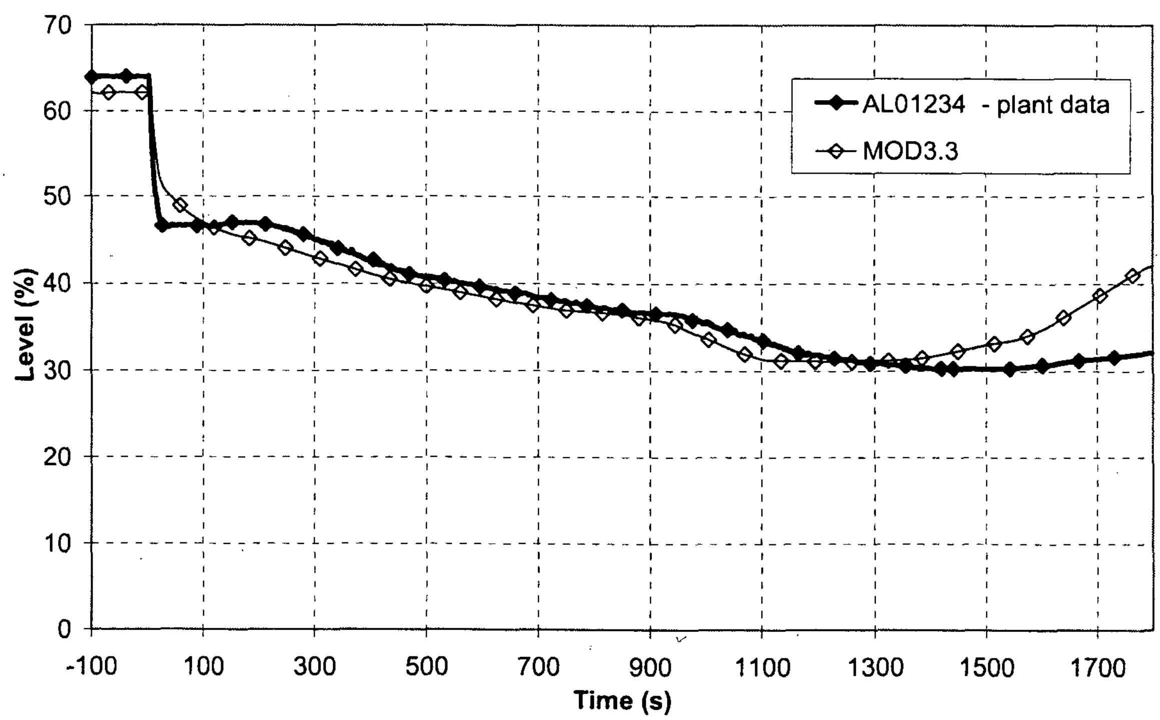

Figure 11 1997 event—pressurizer level—long term

6. RUN STATISTICS

Researchers performed MOD3.3 calculations on a SUN Enterprise 450 work station with a 4x400 megahertz central processing unit (CPU) and 4x2 gigabytes of random-access memory, running under a SOLARIS 8 operating system.

Table 3 presents the run times for different parts of the simulation.

Table 3 Simulation Run Times

Event	Computer Time (CPU) (s)	Total Number of Time Steps (NT)	Number of Volumes (N)	Grind Time CPU/(N*NT)
Steady State (100 seconds)				
1995	1.80	1,234	183	7.97E-06
1997	1.81	1,302	183	7.60E-06
Initial 50 Seconds				
1995	25.62	555	183	2.52E-04
1997	34.56	585	183	3.23E-04
Entire transient simulation (1.800 seconds after steady state)				
1995	849.29	19,456	183	2.39E-04
1997	1147.58	20,052	183	3.13E-04
Overall (steady state + transient) (1.900 seconds)				
1995	876.71	21,245	183	2.26E-04
1997	1,183.95	21,939	183	2.95E-04

7. CONCLUSIONS

The RELAP5/MOD3.3 simulation reproduced plant data very well. In particular, secondary parameters matched well because for the main feedwater flow the plant measured data were used and the MSIV leakage was considered, indicated from plant measured steam flow. Actual turbine flow at the plant was unknown, but turbine runback may have occurred, causing additional reduction in secondary steam flow.

The exact event sequence was not recorded at the plant, but MOD3.3 seems to have captured it closely.

Some discrepancies were evident between plant data and the RELAP5/MOD3.3 prediction because of differences in the initial conditions of the events and because of the expected accuracy of RELAP5/MOD3.3 modeling. Some additional uncertainty originated in the modeling of certain parameters that were not measured at the plant and therefore could not be verified and compared to the results calculated by RELAP5/MOD3.3.

The simulation of the 1997 event initially predicted a closure of the intact MSIV that was too slow, but it improved after researchers further shortened the closure time for the affected MSIV to 0.1 second.

8. REFERENCES

1. Petelin, S., O. Gortnar, B. Mavko, "Steam Generator Model for Design Accident Calculations," *ZAMM*, Berlin, Akademie Verlag GmbH, 72(6):T607–T611, 1992.
2. Mavko, B., S. Petelin, O. Gortnar, "RELAP5 Modelling of the Westinghouse Model D4 Steam Generator," *Nucl. Tech.*, American Nuclear Society, La Grange Park, Illinois, 101:181–192, 1993.
3. Parzer, I., S. Petelin, B. Mavko, "RELAP5 verification of a plant analyzer model for SGTR transients," *Trans. Am. Nucl. Soc.*, 71:536–538, 1994.
4. Parzer, I., S. Petelin, B. Mavko, "Feed-and-bleed procedure mitigating the consequences of a steam generator tube rupture accident," *Nucl. Eng. Des.*, 154:51–59, 1995.
5. Parzer, I., S. Petelin, B. Mavko, "Modelling operator rediagnosis during an SGTR event," *Nucl. Eng. Des.*, 159:143–151, 1995.
6. Hrvatin, S., S. Petelin, M. Tuma, "RELAP5 new steam generator model for Krško NPP," *Proc. Int. Conf. Nuclear Energy in Central Europe '98*, Terme Čatež, Slovenia, September 7–10, Nuclear Society of Slovenia, 1998, pp. 227–233.

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SI-1000 Ljubljana, Slovenia

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10. SUPPLEMENTARY NOTES
A. Calvo, NRC Project Manager

11. ABSTRACT *(200 words or less)*
Recently, the last frozen version of RELAP5/MOD3.3 was released, before the code will be merged with another best estimate thermal-hydraulic system code (TRAC) into an integrated thermal-hydraulic code. Therefore, it is of utmost importance to assess models built using the RELAP5/MOD3.3 code against real plant transients before the code merger.

The paper presents the RELAP5/MOD3.3 analysis of two abnormal events that occurred at the Krško Nuclear Power Plant in September 1995 and January 1997, before replacement of the steam generators (SGs) in 2000. The events resulted from the sudden closure of a main steam isolation valve (MSIV). In the September 1995 event, a malfunction in the MSIV in SG-1 caused inadvertent valve closure. In the January 1997 event, the valve stem in SG-2 was broken, also causing sudden valve closure. Researchers obtained valuable plant data from these real plant transients and performed the RELAP5/MOD3.3 code assessment.

The analyses used a full two-loop plant model, developed at the Jožef Stefan Institute in Ljubljana, Slovenia. The model assumed old Westinghouse D4-type SGs with preheaters. It assumed that 18-percent of the U-tubes were plugged in both SGs.

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