Abstract

The present Master’s Thesis is part of the Master’s degree in Nuclear Engineering of the Universitat Politècnica de Catalunya and the ENDESA Escuela de Energia, and it was developed during the internship in a Spanish Pressurized Water Reactor (PWR).

The objective of the project is to update and test the nuclear plant model used for the Safety Analysis department which belongs to the Licensing Department mainly for Severe Accidents phenomenology studies to prepare for and respond to emergencies.

The code used for this purpose is the Modular Accident Analysis Program (MAAP), actually developed by the Electric Power Research Institute (EPRI). MAAP simulates the response of Light Water Reactors (LWR) during severe accidents; given a set of initiating events and operator actions, MAAP predicts the plant’s response as the accident progresses.

The project includes a brief description of the severe accident phenomenology and an explanation of the MAAP code, including a summary of the reactor coolant primary system model, the containment systems and the Engineered Safeguards contemplated in the Reference Plant MAAP5 model.

The latest version of the code that has been used to model the PWR Reference Plant is MAAP4.0.6 and several accidents have been simulated for the Probabilistic Risk Assessment. Therefore, starting from the MAAP4.0.6 model of the PWR Reference Plant it has been developed the new MAAP5.0.1 model, the so called Parameter File. Later on, the new model has been tested with the aim of gain confidence in the values of the Parameter File and hence, in the results of the accident sequences obtained with the new developed model.

It has been tested the new Reference Plant MAAP5.0.1 model by comparison against reliable results obtained with the RELAP5 code and the MAAP5 code. For the first case, it has been used the RELAP5 model of the Reference Plant and for the second one, the well-validated reference PWR Parameter File of the ZION Nuclear Power Plant, which is delivered with the MAAP code.

Afterwards, it’s shown the results of the testing calculations, which accommodates a brief description of each simulated accident followed by the plottable results. Finally, the conclusions of the project are presented.
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Acronyms

AC: Alternate Current
AFW: Auxiliary Feed Water
BWR: Boiling Water Reactor
CANDU: CANada Deuterium Uranium (type of nuclear reactor)
CCI: Core-Concrete Interaction
CNA: Reference Plant
CSN: Consejo de Seguridad Nuclear
DC: Direct current
EPRI: Electric Power Research Institute
FW: Feed Water
HPI: High Pressure Injection
IDCOR: Industry Degraded Core Rulemaking
LBLOCA: Large Break Loss Of Coolant Accident
LOCA: Loss Of Coolant Accident
LPI: Low Pressure Injection
MAAP: Modular Accident Analysis Program
MCP: Main Coolant Pump
MFW: Main Feed Water
NPP: Nuclear Power Plant
PORV: Power Operated Relief Valve
PRA: Probabilistic Risk Assessment
PWR: Pressurized Water Reactor
RCS: Reactor Cooling System
RELAP: Reactor Excursion and Leak Analysis Program
RPV: Reactor Pressure Vessel
RV: Relief Valve
SAMG: Severe Accident Management Guidelines
SBO: Station Black Out
SG: Steam Generator
SGTR: Steam Generator Tube Rupture
SV: Safety Valve
TDP: Turbine Driven Pump
VVER: Vodo-Vodyanoi Energetichesky Reactor (type of nuclear reactor)
1. Preface

1.1. Project Origin

The present Master’s Thesis is part of the Master’s degree in Nuclear Engineering of the Universitat Politècnica de Catalunya and the ENDESA Escuela de Energía, and it was developed during the internship in a Spanish Nuclear Power Plant.

The objective of the project is to update and test the nuclear plant model used for the Safety Analysis department which belongs to the Licensing Department mainly for Severe Accidents phenomenology studies to prepare for and respond to emergencies.

The code used for this purposes is the Modular Accident Analysis Program (MAAP), actually developed by the Electric Power Research Institute (EPRI). MAAP simulates the response of LWR power plants during severe accidents; given a set of initiating events and operator actions, MAAP predicts the plant’s response as the accident progresses.

The latest version of the code that has been used to model the Reference Plant is MAAP4.0.6 and several accidents have been simulated for the Probabilistic Risk Assessment.

However, after the Fukushima Nuclear Power Plant accident in Japan, the Regulatory Body required the utilities to perform new safety related studies. The new capabilities of MAAP5.0.1 will allow the utility to perform these calculations and therefore, it showed the necessity to update the model to the new version.

1.2. Motivation

Nuclear Safety is the principal aspect that concerns the operation of a nuclear power plant and it’s in this context where a reliable thermal-hydraulic model used for safety-related studies is crucial. Therefore, the new Regulatory Body requirements aren’t the unique motivation of the project; the new capabilities added to the new MAAP version code include an enhanced Best Estimate approximation that improves the behavior of the models.

Starting from the MAAP4.0.6 model of the PWR Reference Plant and using the old Calculation Note, it has been developed the new MAAP5.0.1 model of the Reference Plant, the so called Parameter File. Later on, following the several recommended steps of the MAAP4 Application Guidance [1], the new model has been tested with the aim of gain confidence in the values of the Parameter File and hence, in the results of the accident sequences obtained with the new developed model.
2. Introduction

2.1. Objectives

The objectives of the project are updating a MAAP plant model from MAAP4.0.6 to MAAP5.0.1 for a PWR Reference Nuclear Power Plant and testing it by comparison against retrievable results obtained with the RELAP5 Mod3.2 code and the MAAP5.0.1 code. For the first case, it’s used the RELAP5 model of the Reference Plant and for the second case, the well-validated reference PWR Parameter File of the ZION Nuclear Power Plant, which is delivered with the MAAP code.

2.2. Scope

The project includes a brief description of the severe accident phenomenology and an explanation of the MAAP code, including a summary of the reactor coolant primary system model, the containment systems and the Engineered Safeguards contemplated in the Reference Plant MAAP5 model.

With the aim of gain confidence with the new developed Parameter File, a comparison between the results obtained with the Reference Plant models of the MAAP5.0.1 code and RELAP5 code was made, in order to evaluate and adjust the model during the thermal-hydraulic phase of the accident, that is, before core damage. It has been analyzed seven accidents that were chosen to cover the major primary and secondary systems and different initiating events.

In absence of the MAAP5 Application Guidance, it has been detailed and met the recommendations of the MAAP4 Application Guidance [1], which includes:

- Testing the Parameter File with a steady-state sequence.
- Testing the Reference Plant model with three standard sequences to validate the file's overall performance: a station blackout, a small LOCA, and a large LOCA.

Afterwards, it’s shown the results of the testing calculations, which accommodates a brief description of each simulated accident followed by the plottable results. Finally, the conclusions of the project are presented.
3. Severe accident

3.1. Definition

Consideration of beyond design basis accidents at nuclear power plants is an essential component of the defence in depth approach used in ensuring nuclear safety. The probability of occurrence of a beyond design basis accident is very low, but such an accident may lead to significant consequences resulting from the degradation of nuclear fuel [2].

A design basis accident is defined as accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

A beyond design basis accident comprises accident conditions more severe than a design basis accident, and may or may not involve core degradation. Accident conditions more severe than a design basis accident and involving significant core degradation are termed severe accidents.

3.2. Accident phenomenology

The types of phenomenological challenges that arise during a PWR severe accident can be identified for each of the fission product barriers. The descriptions of phenomena are intended to provide further insight related to the type of function that would be capable of stabilizing the accident and the distinct damage conditions at which recovery of a quasi-stable state could occur. The phenomena that challenge the fission product barriers include the following [3]:

3.2.1. Fuel cladding

Ballooning and rupture

In low-pressure accident scenarios, when the core temperature reaches between 726 °C and 927 °C, this failure mode becomes possible. The time and temperature at which ballooning and rupture of the fuel cladding occurs depend on the internal pressure in the fuel rod relative to the external RCS pressure. In the case of high pressure sequences, the cladding will tend to collapse onto the fuel at much lower temperatures than those at which balloon rupture could occur. Although failure of the cladding in this manner will lead to release of fission products, there still exists a high probability of arresting the accident progression if cooling water is reintroduced.

Over-temperature and oxidation

During a severe accident, the exposure of fuel cladding to reduced cooling will lead to an escalation of cladding temperature. The rate at which the fuel and cladding will heat up will depend on the nature of the accident. For accidents with high heat-up rates, the temperature of the Zircaloy cladding can
escalate to its melting temperature, which would cause cladding failure and relocation. For accidents characterized by more gradual heat-up rates, which are typical in loss-of-heat-sink events, the temperature of the Zircaloy cladding will rise to about 1227 °C, at which point the rate of the exothermic interaction between Zircaloy and steam, generating hydrogen as a byproduct of the reaction, will begin to increase rapidly.

This Zircaloy oxidation reaction will lead to a rapid escalation of fuel temperature. If not arrested, interactions between fuel, fuel cladding, and structural materials will lead to the formation of molten material at temperatures possibly below the individual melting points of the respective materials. Even after the start of oxidation of the fuel cladding, damage can be arrested by the restoration of cooling (sufficient to flood the uncovered fuel).

3.2.2. Reactor coolant system

Hot leg creep rupture

If the core is uncovered for a significant period, natural circulation flows will be established between the core and the upper plenum, with additional natural circulation flows extending into the hot legs and the steam generator tubes. If creep rupture of a steam generator tube occurred, this would induce a bypass of containment. However, creep rupture of the hot leg or even the surge line could occur before an induced steam generator tube rupture.

This challenge is of increasing relevance from the onset of significant fuel cladding oxidation through more extensive fuel damage. Actions that stabilize RCS temperatures, quenching damaged fuel and fuel debris, will be capable of mitigating this challenge. Actions that depressurize the primary system will also be capable of greatly reducing the rate of creep of these RCS components, depending on the extent of depressurization. Actions that lead to a rise in RCS pressure, such as restarting the main coolant pumps, could exacerbate this challenge.

Overpressure

This challenge is primarily related to situations in which water is injected into the RPV downcomer at a substantial rate. An example of an action that could cause such injection rates is the restart of an RCP in a PWR. For these situations, substantial pressurization could occur as the water interacts with the hot damaged fuel. Depending on the conditions in the RCS at the time of a RCP restart, there is the potential for the pressurization to challenge RCS integrity.

3.2.3. Reactor pressure vessel

Overpressure

Under most operating and accident conditions, RPV overpressure does not pose a direct challenge to the structural integrity of the vessel. If high-pressure injection is inadvertently initiated at a time when the RCS has been cooled down, however, the overpressurization of the vessel could promote a brittle fracture of the vessel.
In-vessel steam explosion

The consequences of relocation of held-up molten fuel debris into the lower plenum has been extensively investigated. Of particular concern has been the potential for an in-vessel steam explosion to occur, which has been postulated in the past to lead to the failure of the upper head in such a way as to generate a missile that could impact and fail the containment. Consensus has developed that the likelihood of such an event is very small. The primary means of preventing the interaction between molten core debris and water in the lower plenum altogether is to restore cooling to the fuel before the fuel is significantly damaged.

Molten jet attack

In the absence of core cooling, core degradation will result in a large mass of debris forming on or above the lower core support plate. If cooling is not restored, this held-up molten debris will eventually relocate into the lower plenum. When molten material relocates into the lower plenum, there is the potential for debris jet contact with the lower head and some ablation of the lower head over the region of contact. However, water that is present in the lower plenum at the time of molten debris relocation out of the core region will tend to promote instability and fragmentation of the molten jet. This will greatly limit the extent of thermal attack of the lower head vessel wall.

Creep failure of the lower head

When molten corium has relocated into the lower head of the vessel, the thermal load on the lower head wall will escalate substantially, as internal convection within the molten pool draws approximately a quarter of the decay heat load out through the lower head wall in contact with debris. In addition to the thermal loading of the vessel wall, the vessel is subjected to mechanical loading from the dead weight of the core debris and the internal pressure of the vessel if the RPV has not been depressurized. The extent to which the lower head wall temperature will increase will depend on whether the cavity is flooded and the lower head is cooled externally by water.

In addition, the heat flux from the molten pool will depend on the morphology of the debris bed in the lower head (that is, the potential for formation of metal layers that tend to promote high heat fluxes through limited regions of the vessel wall). If the lower head is not adequately cooled, creep failure of the lower head will eventually occur. The internal flooding of the vessel has the potential to minimize the thermal loading of the lower head wall. However, for higherpowered cores with a large mass of molten debris relocating into the lower head, the integrity cannot be guaranteed by this action alone. Additional strategies have been considered in which the cavity is flooded and the lower head is entirely submerged.

Penetration failure

The relocation of debris into the lower head of the RPV could also be accompanied by relocation of molten material into the numerous penetrations of the bottom head typically found in operating LWRs (the AP1000 has eliminated penetrations in the RPV lower head). Should molten material not freeze in the penetration, a melt-through of the penetration would be expected. If initial freezing does occur, the
penetration can still fail by melting if the debris remelts due to internal decay heat. Furthermore, if the RPV remains at high pressure with molten material in the lower head, the weakening of the penetration weld to the lower head could result in an ejection of the penetration from the lower head. The relocation of molten material through a failed penetration will tend to increase due to the ablation-assisted growth in the size of the failure area. This failure will result in vessel depressurization and relocation of debris into the cavity. As with other lower head failures, flooding of the vessel and the cavity could aid in limiting the extent of challenge to the integrity of penetrations.

### 3.2.4. Containment building

#### Core–concrete interaction

The relocation of debris into direct contact with concrete has the potential to induce melting of the concrete. If the debris does not remain adequately quenched, the temperature at the interface with the concrete will escalate to the melting temperature of the concrete. Attack of the concrete structure will then begin, and the physicochemical processes associated with core–concrete interaction (CCI) will ensue. CCI tends to lead to the release of large quantities of hydrogen and carbon monoxide and promotes the release of less-volatile fission products into the containment. The occurrence of CCI will be influenced by the degree to which an overlying water pool can remove the heat generated within the debris bed. However, ensuring that the debris remains covered by water will serve to aid in scrubbing fission products released from the debris bed and, it will prevent direct radiative heat transfer from the surface of the debris bed into the containment atmosphere.

#### Static overpressure

The discharge of steam and noncondensable gases such as hydrogen, carbon monoxide, and carbon dioxide into containment will tend to increase the internal pressure of containment. Efforts to cool the overheated fuel can also lead to further steam generation at lower rates of cooling water addition to the core. If no action is taken to reduce the pressure, the static overpressure will eventually exceed the upper limit of the containment, and failure will occur. Such a failure could occur through a containment penetration or a gross failure of the structure. The rate at which the gases in containment will leak out depends on the size of the failure. Subsequent to this failure, containment will depressurize. The overpressure could be reduced by restoration of containment cooling (such as condensation of steam) or controlled venting.

#### Over-temperature

The release of steam and hot noncondensable gases such as hydrogen into the containment atmosphere will result in an escalation of the containment temperature. Furthermore, following core relocation into the reactor cavity, if the core is not covered by water, radiation from the hot surface of the debris will serve to radiatively heat the containment atmosphere. At sufficiently high temperatures, the containment will become susceptible to localized failures due to the degradation of, for example, penetration seals. As with reduction in containment static pressure, the restoration of containment cooling or controlled venting will reduce the temperature of containment.
Containment bypass/isolation

Depending on the initiating event, the containment could be bypassed before the onset of core damage or as a consequence of events occurring during the progression of the accident. Initiating events such as interfacing system loss of-coolant accidents (LOCAs) or a steam generator tube rupture will provide a direct path for fission products to escape the primary containment. Events during the course of an accident, such as a consequential steam generator tube rupture, can induce containment bypasses. Alternatively, a path outside containment can develop if there is a failure in the containment isolation system. Potential means of minimizing the transport of fission products through breaches in the containment involve limiting the extent of overpressure in containment or flooding the location of the breach in either the primary or secondary containment. In addition, sprays provide a means of removing fission products from the atmosphere.

Flammable gas combustion

The oxidation of fuel cladding and other structural materials in the reactor will produce hydrogen. In addition, when CCI has initiated, hydrogen and carbon monoxide will be generated and released into the containment atmosphere. At sufficient concentrations of hydrogen and carbon monoxide—which depend on the relative proportion of hydrogen, carbon monoxide, steam, and oxygen in the mixture—the containment atmosphere will become flammable. If a weak ignition source exists, such as a spark from electrical equipment or igniters, combustion of these flammable gases will begin. Depending on the concentration of flammable gases, high-speed flames could occur. In certain geometries, these high-speed flames could accelerate beyond the speed of sound to initiate an explosion. Such dynamic loads from fast or slow flames could induce structural failure of the containment. The challenge to structures from combustion events can be mitigated by measures to control the concentration of flammable gases in the containment atmosphere, such as the use of igniters or passive auto-catalytic recombiners. In addition, the introduction of nitrogen into containment atmospheres in sufficient quantities serves to inert the containment atmosphere.

Ex-vessel steam explosion

In the event that the reactor cavity contains water at the time of RPV lower head failure and corium relocation, an energetic interaction between molten debris and cavity water (that is, a steam explosion) may be possible. The occurrence of a steam explosion relies on rapid and sufficient fragmentation of molten debris to enhance the water-melt contact surface area and promote substantial heat transfer from the corium to the water. A shock wave could potentially result. The formation and propagation of a shock wave will mechanically load external structures. However, the occurrence of an ex-vessel steam explosion is by no means a certain occurrence. The fragmentation of molten debris upon entering the cavity water could also lead to strong quenching of debris and enhance the coolability of the debris in the reactor cavity. The occurrence of and energy released by an ex-vessel steam explosion will depend on the amount of water present in the reactor cavity. There is no direct means to mitigate an ex-vessel steam explosion, aside from preventing relocation of core debris outside the RPV. The evaluation of the benefit from flooding the reactor cavity in terms of long-term, in-vessel debris retention or even enhancing debris coolability through greater fragmentation must be weighed against the likelihood and consequences of a steam explosion. This type of evaluation is typically plant specific.
Direct containment heating

If the RPV fail at high pressure, the discharge of molten corium and steam (referred to as a high-pressure melt ejection) will result in complex flows of steam and corium within the reactor cavity. The consequence of such flows is strongly dependent on the geometry of the reactor cavity. Corium that is entrained by steam could be carried into regions of the containment connected to the reactor cavity. Furthermore, oxidation of corium by the steam will generate hydrogen. The transport of steam and corium into containment will induce pressurization and heating of the containment atmosphere. The extent of containment pressurization depends on the degree of core debris dispersal into containment. This issue has been resolved and is not considered, across the range of LWR designs, to pose a potential challenge to containment.
4. Modular Accident Analysis Program (MAAP)

4.1. Introduction

The Modular Accident Analysis Program (MAAP) is an integral systems analysis code for assessing off-normal transients that can progress to and include severe accidents. It was initially developed during the industry-sponsored IDCOR Program. Ownership of MAAP was transferred to EPRI at the completion of IDCOR. Subsequently, the code evolved into a major analytical tool for supporting the plant-specific Individual Plant Examinations (IPEs) requested by NRC Generic Letter 88-20.[4]

The first version of the code, MAAP3B, was developed early in 1980’s and it was updated to MAAP4 in the mid 1990’s. Periodically EPRI released subversions with improvements and new lessons learned. The last version of MAAP4 is MAAP4.0.8. In 2011 EPRI released MAAP5.0.1 which includes enhancements that will be explained in the following chapter.

There are parallel versions of MAAP that support BWRs and PWRs and unique versions for Russian Federation pressurized light water reactor (VVER), Canadian-designed pressurized heavy water reactor (CANDU), and advanced thermal reactor (ATR) designs.

These versions contain the same core model, containment model, fission product model, and input and output schemes. In addition, they have distinct primary system models and engineered safeguards (ESF) models. The code is applicable to both current and advanced LWR designs, with models that represent the passive features of the latter.

MAAP simulates the response of LWR power plants during severe accidents. Given a set of initiating events and operator actions, MAAP predicts the plant’s response as the accident progresses. The code is used for the following:

- To predict the timing of key events (for example, core uncovering, core damage, core relocation to the lower plenum, and vessel failure)
- To evaluate the influence of mitigative systems and the impact of the timing of their operation
- To evaluate the impact of operator actions
- To predict the magnitude and timing of fission product releases
- To investigate uncertainties in severe accident phenomena

MAAP results are primarily used to determine level 2 PRA success criteria and accident timing to support human reliability analyses. MAAP considers the full spectrum of important phenomena that could occur during an accident, simultaneously modeling those that relate to the thermal-hydraulics and to the fission products. It also simultaneously models the primary system and the containment and reactor/auxiliary building.
MAAP treats steam formation, core heatup, cladding oxidation and hydrogen evolution, vessel failure, core debris-concrete interactions, ignition of combustible gases, fluid (water and core debris) entrainment by high velocity gases, and fission product release, transport, and deposition. MAAP treats all of the important engineered safety systems such as emergency core cooling, containment sprays, fan coolers, and power operated relief valves.

In addition, MAAP allows operator interventions and incorporates these in a flexible manner, permitting the user to model operator behavior in a general way. Specifically, the user models the operator by specifying a set of variable values and/or events which are the operator intervention conditions combined with associated operator actions. Lastly, the auxiliary or reactor building can be modeled for sequences in which it is important.

4.2. MAAP input and a output files

MAAP requires two input files. The first is the parameter file, which contains plant-specific information, output specifications, and user-controlled phenomenological parameters. The second is the sequence input file, which specifies the accident initiators, operator actions, and sequence control times (end time and print interval). After processing the information in the two files, the code predicts the sequence of events and corresponding plant conditions. It generates a number of output files, including a synopsis of the sequence, a summary of events, tables of time-dependent results in a form suitable for plotting, and tabulated results that provide the details of the plant's status at selected times.

4.3. Reactor coolant primary system model

The MAAP5 PWR primary system model calculates the thermal-hydraulic conditions in the RPV, the hot legs, the cold legs and the primary side of the steam generators (SGs). The pressurizer is treated in a separate model. The primary system is structured to evaluate the individual response of each coolant loop and the steam generator in that loop. The user specifies how many actual loops are and which loop contains the surge line to the pressurizer.

Specifically, the common elements for those Westinghouse PWR designs, ranging from two-loop to four-loop designs, are:

- One reactor coolant pump (RCP), one cold leg, one hot leg and one crossover leg from a steam generator to the RCP for each loop,
- A reactor pressure vessel (RPV) downcomer, lower plenum, reactor core, reactor vessel upper plenum and the volume above the reactor vessel dome plate, and
- One steam generator for loop.

The PWR MAAP5 RCS models calculate the generation of steam and hydrogen gas in the core and the lower plenum, the overheating and possible melting of the fuel, clad and control components in the core, and mobile fission products in the core, the liberation, transport and deposition of fission products,
the possible relocation of core debris into the lower plenum, the thermal response of the molten debris in the lower plenum.

It also calculates the release of hydrogen, steam, and water, to the containment, the release of core debris to the containment if the accident sequence leads to failure of the Reactor Pressure Vessel (RPV) lower head, local gas and structure temperatures in the RCS, heat losses from the RCS to containment and forced and natural circulation flows within the RCS.

The pressurizer is modeled as a single control volume, with one water pool and one gas node. The water and gas can be at different temperatures (which are also distinct from the primary system fluid temperatures). Calculations of the thermal-hydraulic conditions in the pressurizer account for evaporation, condensation, steam stripping due to steam and non-condensable gases sparging through the water pool, and water and gas exchange with the primary system via the surge line and with the containment through relief and safety valves. Mass and energy contributions from pressurizer sprays and heaters and from heat transfer to structures are also included.

The core model predicts the thermal-hydraulic behavior of the core and the water and gas contained within the core boundary along with the response of core components during all phases of a sequence. The calculations are performed on a nodal basis, typically 13 axial nodes (10 for the active core; 2 below the active core for the core support plate, the lower tie plate, and lower gas plenum; and 1 above the active core for the upper tie plate and upper gas plenum) and 5 to 7 radial rings provide adequate resolution.

The code tracks the mass, energy, and temperature of the following constituents in each PWR node: Fuel (UO2), cladding (Zr, ZrO2, stainless steel, and steel oxide), control rod or water rod (Ag-In-Cd or B4C, stainless steel, steel oxide, Zr, and ZrO2) and structural materials (Zr, ZrO2, stainless steel, and steel oxide).

Input quantities include the initial masses of the different materials, the geometry of the constituents, and axial and radial peaking factors. The initial core power is specified by the user. Decay power can be determined by using the American National Standards Institute/American Nuclear Society (ANSI/ANS) decay heat power correlation or the user can specify the decay power as a function of time as an input table.

4.4. Containment systems

The MAAP containment model is an interconnection of compartments nodes and flow paths. Several compartments and flow junctions can be modeled in this way. The reference NPP containment is modeled as five compartments nodes and the auxiliary building is modeled as two compartments. Eleven junctions connect the different compartments among themselves and the environment.

The containment regions have been conveniently selected and defined to coincide with physical partitions such as walls, gratings, etc. Flow junctions are based on openings such as stairwells, doorways, and gratings.
The following figure corresponds to the nodalization of the Containment and the Auxiliary Building of reference NPP model.

![Figure 4.1: Reference NPP Containment Nodalization.](image)

### 4.5. Engineered Safeguards Systems

Consistent with the purposes of the MAAP code, it is necessary to model the important safety and other systems influencing the sequence of events occurring during applications ranging from short-term, non-severe transients to long-term severe accidents. The safety systems include engineered safeguards systems (such as emergency core cooling systems and containment cooling systems) as well as control and standby systems which are actuated during accident transients.

The reference NPP model includes the quench tank, the containment sprays, the auxiliary feedwater system, the low and high injection systems, fan coolers and relief valves of pressurizer and steam generators.
5. Updating plant model

5.1. Introduction

The latest version of the code that has been used to model the Reference Plant is MAAP4.0.6 and several accidents have been simulated for the Probabilistic Risk Assessment. However, after the Fukushima accident, the Regulatory Body required the utilities to perform new safety related calculations. The new capabilities of MAAP5.0.1 allow the utility to perform these calculations and therefore, it showed the necessity to update the model to the new version.

5.2. Main differences between version codes

The following differences between version codes [5] are important to know in order to compare and evaluate the discrepancies that will be found when simulating the same scenarios with MAAP4.0.6 and MAAP5.0.1.

**Advanced Containment Model**

The first major development included in the MAAP5 version was the advanced containment modeling. The model has the following features:

1. It models circulation of the containment gas space that is induced by the substantial momentum of the jet discharging from a break in the reactor coolant system. The initial circulation in the break compartment is mechanistically transferred to adjacent compartments. The circulation flow is in addition to pressure-driven and density-driven flows already in the previous MAAP4 model.

2. Induced circulation significantly enhances forced convective heat transfer to structural heat sinks, which lowers peak pressures during large LOCA and main steam line break analysis.

3. Induced circulation also results in entrainment of condensate from wall surfaces. The condensate is recirculated as water droplets into the freeboard gas space. The droplets act as a spray that virtually eliminates superheated temperatures caused by gas space compression.

4. The model enhancements have been extensively benchmarked against a variety of small-scale separate effects benchmarks and large-scale integral benchmarks.

**Advanced PWR RCS Model**

While the MAAP4 PWR RCS model was adequate for the original MAAP4 mission of supporting PRA and SAMG assessments, this model was not sufficient to cover the entirety of the broad scope of
applications necessary to support the addition of day-to-day engineering evaluations. Therefore, the MAAP5 model was given the following features.

5. It models the individual performance of each coolant loop. Previously, MAAP4 was limited to two loops, and one- and two-phase water circulation was assumed identical in the two loops. The MAAP5 model demonstrates that, during asymmetric transients such as blowdown of only one steam generator loop, this loop has considerable circulation. However, in the other loops, circulation is stagnated by the absence of a thermal gradient, so these loops do not cooldown.

6. The coolant loop models are fully mechanistic. This includes a fully-pose momentum equation that accommodates flow reversal in the event of a large transient, such as a large break or an asymmetric trip of reactor coolant pumps. It also includes a mechanistic phase separation (phase disengagement) model in each node.

7. The coolant loop modeling also extends to the vessel, where the downcomer, lower plenum, core and upper plenum are each divided into quadrant nodes. Each quadrant node is connected azimuthally to the adjacent quadrant nodes and axially to the associated coolant loop. This nodalization accommodates pressure-driven flow and turbulent mixing that occurs between adjacent quadrant nodes during a large transient, such as a large break or an asymmetric trip of reactor coolant pumps.

8. The MAAP4 model currently represents the response of the RPV lower head on relocation of molten core material into the lower plenum using five nodes through the RPV wall and five azimuthal nodes. These characterize the response of the lower head, specifically the temperature profile within the vessel carbon steel that is of key importance in representing the structural response (material creep) of the RPV lower head.

To better represent the thermal response of the steel, in particular the potential for axial conduction in the azimuthal direction, this nodalization scheme is increased substantially. With this additional detail, the thermal conduction in this direction can be represented and thereby provide insights into the rate at which the lower head could be deformed under accident conditions. Such evaluations are important to the assessment of when the lower head could be expected to fail should such accident conditions be developed.

9. The mechanistic water flow includes backflow through a fraction of steam generator tubes, which can occur during post-RCP trip natural circulation, thereby reducing steam generator heat transfer. The model also includes the potential for leakage flow from the hot leg to the downcomer across the hot leg boss on the core barrel.

10. The two-region steam generator model from MAAP4 has been upgraded to accommodate the individual loop model. This includes benchmarking against the steam generator transients performed at the Westinghouse MB-2 large-scale test facility.
11. The MAAP5 model represents the boric acid precipitation within the core as well as the reverse process when hot leg injection is initiated for those conditions where global circulation through the loops cannot be established.

12. The entire RCS model has been benchmarked against separate effects benchmarks, scaled integral benchmarks, and actual plant transients.

5.3. **MAAP5 basic plant model**

For helping users to prepare and update the Parameter File, MAAP code includes a Westinghouse four-loop reference plant model, the ZION5.par Parameter File, which contains the description of all necessary variables. Therefore, a comparison between the reference plant MAAP4.0.6 Parameter File and Zion MAAP5.0.1 Parameter File was done in order to evaluate which parameters were new, obsolete, modified or equal to the previous version.

The Parameter File can be tested only when the file is essentially complete; that is, it is not possible to test individual components of the file such as the sections that relate to the primary system or the containment. For this reason, only the minimum variables and models required for running the updated MAAP5 Parameter File have been incorporated, in order to simplify the comparison of the results between code versions.

5.4. **Testing of Parameter File**

After the parameter file was created, it has to be tested before it is used for plant sequences. There are several recommended steps in this process to gain confidence in the values in the parameter file and hence in the results of their accident sequences [6]:

5.4.1. **Steady State sequence**

Testing the parameter file with a steady-state sequence is used to accomplish the following tasks.

1. Check that the lines in the file can be read by the code (for example, screens for typographical errors or hidden formatting characters).

2. Do a preliminary assessment of the initialization phase of a calculation and make adjustments to parameter values.

3. Check that the parameter values are self-consistent.

5.4.2. **Visual review of the input lines**

The parameter file contains substantially more comment lines than lines actually processed by the code. A useful step in validating the entries in a file is to strip out the comment lines and then visually
inspect the remaining lines. Without the comment lines, it is easier to read the data lines and hence detect errors (for example, typographical errors or units mismatches).

5.4.3. **Test with several standard sequences**

It is recommended that the parameter file be tested with three standard sequences to validate the file’s overall performance: a station blackout, a small LOCA, and a large LOCA.

First, each sequence should be run in order to solve any problems with the input and/or parameter file that may come out until the sequences run are completed. Then the results should be compared with the corresponding results from the sample sequences. The comparison consists of looking at the trends of the sequences as indicated by the key events and debris and fission product distributions in the figures-of-merit tables, system actuations as indicated by events in the summary files, and primary system and containment conditions shown in the plottable results.

The mass and energy balances and the diagnostic message counts should also be checked, and any warning messages in the log files should be evaluated. Major differences between the test and sample sequences should be investigated. The differences may be the result of plant features. Other types of differences may indicate to the user areas in which their parameter file values could be refined.

5.4.4. **Test user-defined events**

All plant-specific user-defined events that have been included in the Userevt section of the parameter file should be tested. PWR users should also check that the events for creep rupture in the Userevt section have been modified to match the plant geometry, for example, the break elevations. Sequences should be devised that force the events to happen. The input files should contain logic or plot files that can be used to check the user-defined logic and the resultant actions.

This is an important step and should not be skipped because it is easy to make mistakes when creating user-defined events. An iterative process is recommended to ensure that all potential errors or omissions have been addressed (for example, that units are correct, that the logic works as intended, and that deadbands and repeat statements are included as needed).

5.4.5. **MAAP5.0.1 versus MAAP4.0.6 code comparison**

It’s recommended to compare the results with MAAP5 and MAAP4 in order to validate the new model by explaining the differences between code’s versions. A Transmittal Document is delivered with each MAAP code version and it provides an executive summary of the notable modifications that were added in each code revision.

5.4.6. **MAAP5.0.1 versus RELAP5 comparison**

A comparison between the results obtained with MAAP5.0.1 code and RELAP5 code can be done in order to evaluate and adjust the model during the thermal-hydraulic phase of the accident, that is, before core damage.
RELAP5 MOD 3.2 has been widely used by the utility of the reference plant for simulating several accidents of the level 1 probabilistic risk assessment and therefore, the RELAP5 model has been tested and validated so that the comparison will add an extra confidence for the MAAP5.0.1 model.
6. Testing the Parameter File

6.1. MAAP5.0.1 vs RELAP5 MOD3.2 comparison

The purpose of MAAP5 allows simplify the safeguards systems of their models. However, the control systems included in the RELAP5 model are quite complex and detailed and therefore they need to be explored in order to know the adjustments that have to be implemented in the MAAP5 model to obtain similar results between both codes.

In order to gain confidence with the MAAP5.0.1 model of the Reference Plant, several accidents have been simulated with MAAP5.0.1 and they have been compared with the results of the Reference Plant RELAP5 model. The present chapter includes the results of the comparative analysis of five accidents that were chosen to cover the major primary and secondary systems and different initiating events. These are:

1. Loss of heat sink
2. Double-ended large break LOCA with unavailability of AFW
3. Small LOCA with unavailability of AFW
4. Steam generator tube rupture
5. Station blackout

Two more accidents were simulated for testing the new Parameter File and are presented in the Annex A due to their similarity of the previous scenarios. The simplicity of some MAAP aspects has concluded in certain limitations in the accident selection to adequately reproduce (according to the RELAP5 plant model) the control of some plant actuations. From the comparison of the results, significant conclusions of the MAAP5.0.1 thermal-hydraulic modeling capacity can be extracted.

The following assumptions have been considered for all the cases:

- MAAP assumes that MSIV closure, MFW termination and AFW actuation are initiated after certain delay time introduced by the user after the reactor scram. For the reference plant, the MSIV’s are closed 0.5 seconds after the scram hence the same actuation has been assumed in RELAP5.

- The make-up and the letdown flows are not included in the MAAP5 model however they are considered in RELAP5 and therefore, they are stopped after the initiating event.

- In contrast to RELAP5, the containment model in MAAP is modeled in detail and therefore, for all the simulations made with MAAP5, the four Fan Coolers are always available in order to cool the containment building and reduce its pressure, and the containment sprays are not available to have the same time to empty the water of the refueling water storage tank.
6.1.1. Plant model and steady state

The new model was updated from the last MAAP4.0.6 Reference Plant model and the main sources of information are: the Final Safety Analysis Report (FSAR), the Precautions, Limitations and Setpoints (PLS), primary and secondary system nodalization and the control implemented in the RELAP5 plant model, and the building, installations and systems schemes of the Reference Plant supplied by the utility.

The following table shows the data obtained after simulating a steady state of 1000 seconds with MAAP5.0.1 and RELAP5 mod3.2 (null transient).

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>RELAP5 mod3.2</th>
<th>MAAP5.0.1</th>
</tr>
</thead>
<tbody>
<tr>
<td>Primary system pressure</td>
<td>1.5517E7 Pa</td>
<td>1.5459E7 Pa</td>
</tr>
<tr>
<td>Secondary system pressure</td>
<td>6.4744E6 Pa</td>
<td>6.4776E6 Pa</td>
</tr>
<tr>
<td>Hot leg temperature</td>
<td>598.15 K</td>
<td>598.91 K</td>
</tr>
<tr>
<td>Cold leg temperature</td>
<td>561.49 K</td>
<td>561.54 K</td>
</tr>
<tr>
<td>Pressurizer water level</td>
<td>54.56 %</td>
<td>57.27 %</td>
</tr>
<tr>
<td>Steam generator wide range level</td>
<td>81.60 %</td>
<td>81.73 %</td>
</tr>
<tr>
<td>Nuclear power</td>
<td>2939.40 MW</td>
<td>2940.60 MW</td>
</tr>
<tr>
<td>Primary to secondary heat transfer</td>
<td>2945.68 MW</td>
<td>2954.19 MW</td>
</tr>
<tr>
<td>RCP’s power (3 pumps)</td>
<td>13.90 MW</td>
<td>13.89 MW</td>
</tr>
<tr>
<td>Pressurizer heaters</td>
<td>0.68 MW</td>
<td>0 MW</td>
</tr>
<tr>
<td>Letdown - Makeup + RCP’s seal flow (should be)</td>
<td>8.30 MW</td>
<td>Not modeled</td>
</tr>
<tr>
<td>Primary water inventory (without pzr)</td>
<td>177610.27 kg</td>
<td>175829.11 kg</td>
</tr>
<tr>
<td>Primary system mass flow (one loop)</td>
<td>4650.36 kg/s</td>
<td>4657.21 kg/s</td>
</tr>
<tr>
<td>Steam mass flow (one SG)</td>
<td>543.26 kg/s</td>
<td>546.45 kg/s</td>
</tr>
<tr>
<td>MFW water temperature</td>
<td>500.41 K</td>
<td>500.37 K</td>
</tr>
</tbody>
</table>

Table 7.1 – Steady State at nominal conditions for RELAP5 mod 3.2 and MAAP5.0.1
6.1.2. Loss of heat sink

Accident conditions

The initiating event, the loss of Main Feed Water (MFW), is produced after 50 seconds at nominal conditions. It's postulated the unavailability of the Low Pressure Injection (LPI) system, the accumulators and one train of High Pressure Injection (HPI) system. It's also postulated the loss of Auxiliary Feed Water (AFW) and the impossibility of its recovery.

Taking into account the precautions of the Emergency Operation Procedure to response in front of a loss of heat sink, the plant recovery is manually started when the primary pressure is higher than 164 kg/cm² or 2/3 steam generator wide range levels (R.A) are less than 12%. At this point, the Main Coolant Pumps are manually stopped and the feed and bleed maneuver is initiated using one train of HPI and one PORV of the pressurizer.

The simulation is stopped after 10000 seconds, with the accident controlled and the temperatures decreasing.

Accident description

The loss of MFW produces in the Reference Plant a control signal to shut down the reactor in order to avoid the steam generator (SG) tubes to be discovered waiting for the low SG level scram signal. After the automatic shutdown, the nuclear power and the primary to secondary heat transfer decrease rapidly increasing the subcooled margin 30 ºC (see Figure 6.1, Figure 6.2 and Figure 6.3).

Because of the unavailability of the auxiliary feed water system, the steam generator relief valves play an important role on the primary to secondary heat transfer until the feed&bleed maneuver is initiated (see Figure 6.2 and Figure 6.10).

The plant recovery started when the feed and bleed conditions are reached by stopping the MCP’s, manually opening the pressurizer relief valve and initiating the HPI (see Figure 6.6, Figure 6.8 and Figure 6.14). The objective of these actuations is to cool the core injecting water into the cold legs and bleeding through the pressurizer relief valve maintaining the primary water mass inventory (see Figure 6.4). These actuations make the primary pressure decrease and the pressurizer water level increase (see Figure 6.5 and Figure 6.7).

Since the Auxiliary Feed water system is not available, the steam generator level decreases until the feed&bleed is started (see Figure 6.9). The secondary pressure diminishes 4000 seconds after the scram, when the secondary started to heat the primary (see Figure 6.11).

The accident is controlled 10000 seconds after the initiating event with the cold and hot leg temperatures decreasing (see Figure 6.12 and Figure 6.13).
**Adjustments**

The following figures show the evolution of the most significant variables for three different calculations: One with RELAP5, another with the MAAP5 base case and another with the MAAP5 modified case. The following adjustments have been introduced in the MAAP5 modified model in order to approximate its behavior to the RELAP5 calculation.

The decay heat in MAAP5 is introduced by the user using a table depending on time after reactor shutdown. In RELAP5 the decay heat is calculated using the reactor kinetics of the cycle. Therefore, the residual power table has been adjusted in the MAAP5 modified case in order to coincide with the RELAP5 decay heat calculation.

In MAAP5 there are two different methods to calculate the safety valve flow areas. The method used in the MAAP base model calculates the PORV’s flow area using the reference mass nominal flow rate at a given opening pressure introduced by the user. In the second method, which has been used in the MAAP5 modified model, the flow area and the discharge coefficient associated to the relief valve are introduced by the user. For the MAAP5 modified model it has been assumed the same flow area than in RELAP5 and it has been tested the discharge coefficient that best approximates the results, which is 0.8. The following figures show that the pressurizer relief model plays an important role in the primary water mass inventory and therefore, the primary pressure and temperatures.

Since the initiating event is the loss of main feed water and the auxiliary feed water is not available, no further adjustments need to be made for the steam generator level.

The feed and bleed conditions are reached approximately at 1340 s, 1440 s and 1640 s for MAAP5 base model, MAAP5 modified model and RELAP5 respectively due to low level in two out of three steam generators. MAAP5 slightly anticipates the actuation since the primary to secondary heat transfer is higher and therefore, the water evaporation rate of the steam generator is faster.

The simulation of this scenario allowed adjusting the behavior of the pressurizer relief valve model and improving the performance of the MAAP5 model, which is now capable to simulate significantly better the evolution of this accident.
Figure 6.1: Loss of FW – Nuclear Power

Figure 6.2: Loss of FW – Primary-secondary heat transfer
Figure 6.3: Loss of FW – Subcooled margin

Figure 6.4: Loss of FW – Primary water mass inventory
LOSS OF FEED WATER - LOSS OF HEAT SINK

PRIMARY PRESSURE

Figure 6.5: Loss of FW – Primary pressure

LOSS OF FEED WATER - LOSS OF HEAT SINK

HPI FLOW

Figure 6.6: Loss of FW – High pressure injection system
Figure 6.7: Loss of FW – Pressurizer level

Figure 6.8: Loss of FW – Pressurizer relief valves flow
Figure 6.9: Loss of FW – Steam generator wide range level

Figure 6.10: Loss of FW – Steam generator relief mass flow
Figure 6.11: Loss of FW – Steam generator pressure

Figure 6.12: Loss of FW – Hot leg A temperature
Figure 6.13: Loss of FW – Cold leg A temperature

Figure 6.14: Loss of FW – Primary loop A flow
6.1.3. Double-ended large break LOCA with unavailability of AFW

Accident conditions

The initiating event, a double-ended large break LOCA in the cold leg A, is produced after 50 seconds at nominal conditions. It’s postulated the unavailability of one train of the Low Pressure Injection system (LPI), two out of three accumulators, one train of High Pressure Injection system (HPI) and the Auxiliary Feed Water system (AFW). The main coolant pumps are manually stopped when the subcooled margin is less than 5 ºC.

The simulation is finished after 5000 seconds, just after the refueling water storage tank is emptied.

Accident description

The double-ended large break LOCA in the cold leg A depressurizes the primary system in few seconds. The reactor scramed due to low pressure in the pressurizer and rapidly two out of three accumulators discharged borated water into the cold legs of the primary system. The LPI system started and it was available to cool the core (see Figure 6.15, Figure 6.20 and Figure 6.21).

The mass flow leaked just after the break is higher in MAAP5 and therefore, the safety system is capable to rapidly inject borated water during the first few seconds, what it’s traduced in a faster recovery of the water mass inventory of the primary circuit, higher RPV water level and higher subcooled margin. Due to the break mass flow oscillations, other thermal-hydraulics properties also fluctuate (see Figure 6.16, Figure 6.18, Figure 6.19 and Figure 6.22).

The peak containment pressure reached after the large break LOCA in MAAP5 can’t be predicted by RELAP5, where the leakage is sent to a time dependent volume. In order to minimize the effect of the containment contra pressure on the LOCA mass flow and the thermal-hydraulics properties of the primary system, the fan coolers are connected in all MAAP5 simulations and consequently, the containment pressure falls exponentially with time. This behavior of the containment pressure is introduced in RELAP5 using a pressure versus time table constructed using the MAAP5 results. In this way, the evolution of the primary pressure and hot leg temperatures between both codes converged (see Figure 6.20 and Figure 6.25).

Due to the loss of coolant, the primary to secondary heat transfer in RELAP5 becomes negative so that the secondary system heats the primary system reducing its pressure (see Figure 6.17).

However, the heat transfer became positive for the first few minutes in MAAP5 after the primary depressurization and consequently the secondary pressure was increased. Once the primary to secondary heat transfer became negative, the SG pressure started to decrease. In MAAP5 it’s contemplated the SG heat losses through the containment environment and in consequence, the secondary depressurization is faster in MAAP5 than in RELAP5 (see Figure 6.17, Figure 6.23, and Figure 6.24).
The cold leg temperature of the cold leg A oscillates, in RELAP5 is due to the oscillations of the primary to secondary heat transfer while in MAAP5 is result of the fluctuations of the break mass flow (see Figure 6.17, Figure 6.16 and Figure 6.26).

**Adjustments**

The following figures show the evolution of the most significant variables for three different calculations: One with RELAP5, another with the MAAP5 base case and another with the MAAP5 modified case. The following adjustments have been introduced in the MAAP5 modified model in order to approximate its behavior to the RELAP5 calculation.

The decay heat in MAAP5 is introduced by the user using a table depending on time after reactor shutdown. In RELAP5 the decay heat is calculated using the reactor kinetics of the cycle. Therefore, the residual power has been adjusted in the MAAP5 modified case in order to coincide with the RELAP5 decay heat.

Different simulations changing the break discharge coefficient and the orientation of the break have been made in order to evaluate their effect on the thermal-hydraulics properties but no significant differences have been noted. The MAAP5 base model is appropriate to simulate a large break LOCA and to obtain good results.
200 % LOCA IN COLD LEG WITHOUT AFW

**NUCLEAR POWER**

Figure 6.15: Large break LOCA - Nuclear Power

200 % LOCA IN COLD LEG WITHOUT AFW

**LOCA MASS FLOW**

Figure 6.16: Large break LOCA - Combined liquid and vapor break flow rate
Figure 6.17: Large break LOCA – Primary to Secondary heat transfer

Figure 6.18: Large break LOCA - Subcooled margin
200 % LOCA IN COLD LEG WITHOUT AFW

Figure 6.19: Large break LOCA - Primary water mass inventory

Figure 6.20: Large break LOCA - Primary Pressure
200 % LOCA IN COLD LEG WITHOUT AFW

**TOTAL SAFETY INJECTION MASS FLOW**

Figure 6.21: Large break LOCA - Total safety injection mass flow

200 % LOCA IN COLD LEG WITHOUT AFW

**VESSEL WATER LEVEL**

Figure 6.22: Large break LOCA - Vessel water level
200 % LOCA IN COLD LEG WITHOUT AFW

STEAM GENERATOR A LEVEL (R.A.)

Time (s)

Figure 6.23: Large break LOCA - Steam Generator Level A (R.A)

200 % LOCA IN COLD LEG WITHOUT AFW

STEAM GENERATOR A PRESSURE

Time (s)

Figure 6.24: Large break LOCA - Steam Generator A Pressure
Figure 6.25: Large break LOCA - Hot leg A temperature

Figure 6.26: Large break LOCA - Cold leg A temperature
6.1.4. Small LOCA with unavailability of AFW

**Accident conditions**

The initiating event, a two-inch break LOCA in the cold leg A, is produced after 50 seconds at nominal conditions. It’s postulated the unavailability of the Low Pressure Injection (LPI) system, the three accumulators, one train of High Pressure Injection (HPI) system and the Auxiliary Feed Water (AFW) system. The main coolant pumps are manually stopped when the subcooled margin is less than 5 ºC.

After the initialization of the HPI system, one pressurizer relief valve is manually opened to establish the coolant circulation through the core by injecting water into the cold legs and bleeding through the pressurizer. The simulation is finished after 22500 seconds, with the accident controlled and the temperatures decreasing.

**Accident description**

The two-inch break LOCA in the cold leg A depressurizes, in few seconds, the primary system until approximately 8.0E6 Pa where the pressure remains constant for some minutes and close to the secondary system pressure (see Figure 6.31).

The reactor SCRAMed due to low pressure in the pressurizer at 89.2 s, 78.8 s and 71.7 s for the MAAP5 base model, MAAP5 modified model and RELAP5 model respectively (see Figure 6.27). The safety injection signal is activated at 93.7 s, 82.9 s and 79.2 s for the MAAP5 base model, MAAP5 modified model and RELAP5 model respectively. At this point, one train of the HPI system started to inject borated water into the cold leg and the pressurizer relief valve was opened (see Figure 6.34 and Figure 6.37).

The water mass inventory is lightly increasing from 5000 s to the end of the accident so that the flow coming in (HPI) is slightly greater than the flow coming out (the break flow and the pressurizer relief valve flow) (see Figure 6.30).

The break mass flow is mainly composed by liquid until the U-tubes of the steam generator and the cold legs are emptied. At this time, vapor started to leak through the break and the primary to secondary heat transfer is inverted. In RELAP5 the break vapor flow lasts until 5000 seconds while MAAP5 estimates vapor leaking during all the simulation time (Figure 6.28, Figure 6.32 and Figure 6.33).

The primary system pressure was reduced from 8.0E6 Pa when vapor started to leak through the break. The HPI, the pressurizer relief flow and the two-inch break LOCA maintain the water level just below the hot leg elevation and the loop seals oscillated from block to unblock during the period between 1000 seconds to 1400 seconds in MAAP5. However, in RELAP5, there are two clear loop seal unblocking at 2500 seconds and 16000 seconds. The first one reduced the primary system pressure however; the second one incremented the core heat transfer by injecting water into the core and consequently, the primary pressure and hot leg temperature were increased (see Figure 6.31 and Figure 6.41).
The pump bowls blocking and the liquid and gas break mass flow rates impact the evolution of the primary system pressure between both codes. In RELAP5 the RCS pressure at the end of the simulation is equal to 8.1E5 Pa while in MAAP5 base model is equal to 33.7E5 Pa and in MAAP5 modified model is equal to 23.7E5 (see Figure 6.31).

The peak containment pressure reached after the 2" break LOCA in MAAP5 can't be predicted by RELAP5, where the leakage is sent to a time dependent volume. In order to minimize the effect of the containment contra pressure on the LOCA mass flow and the thermal-hydraulics properties of the primary system, the fan coolers are connected in all MAAP5 simulations and consequently, the containment pressure falls exponentially with time. The quench tank rupture occurred in MAAP5 in less than ten minutes and the consequent discharge of mass and energy to the containment also increases its pressure. This behavior of the containment pressure is introduced in RELAP5 using a pressure versus time table constructed using the MAAP5 results. However, the primary system pressure and the hot leg temperature are greater in MAAP5 (see Figure 6.31 and Figure 6.41).

The primary temperatures are decreasing after 22500 seconds. The MAAP5 hot leg temperature is higher than the RELAP5 temperature and its evolution is similar to the primary system pressure (see Figure 6.41). The evolution of the cold leg temperature is determined according to the fluid temperature and consequently, when the liquid is vaporized, the temperature is illusively changed. However, the cold leg temperature general trend in MAAP5 is slightly colder than in RELAP5 (see Figure 6.42).

The vessel boiled-up water level is lower than the active core level. Once the pump bowl is unblocked, the water accumulated in the loop seal goes to the core increasing promptly the vessel water level. This effect is more notable in MAAP5 than in RELAP5. The higher temperature and pressure in MAAP5 allow a small positive subcooled margin. In RELAP5, it's null and consequently the vessel water level in RELAP5 is generally lower than in MAAP5 (see Figure 6.29 and Figure 6.35).

The pressurizer water level behavior differs notably between both codes. In MAAP5 the pressurizer is decreasing with the primary pressure until it is stabilized around 18 % however, in RELAP the level is maintained until the second loop seal unblocking (see Figure 6.36). The dome vessel bubble pushes the liquid water into the pressurizer and it can suddenly collapse into the hot leg if the RCS pressure is decreased. This water is injected into the core and therefore, the water vessel level and the subcooled margin are increased. Because of the MAAP5 higher primary pressure, the pressurizer relief valve flow is higher (see Figure 6.37).

The steam generator relief valves are opened while the primary to secondary heat transfer is not null. Once the primary water level does not reach the U-tubes, the SG wide range level remains constant and the secondary pressure decreases; faster in MAAP5 because it also considers the SG heat losses to the environment; by transferring the heat to the primary circuit (see Figure 6.38, Figure 6.39 and Figure 6.40)
Adjustments

The following figures show the evolution of the most significant variables for three different calculations: One with RELAP5, another with MAAP5 base case and another with MAAP5 modified case. The following adjustments have been introduced in the MAAP5 modified model in order to approximate its behavior to the RELAP5 calculation.

The decay heat in MAAP5 is introduced by the user using a table depending on time after reactor shutdown. In RELAP5 the decay heat is calculated using the reactor kinetics of the cycle. Therefore, the residual power has been adjusted in the MAAP5 modified case in order to coincide with the RELAP5 decay heat.

It has been tested again the same modified pressurizer PORV's model used in the previous simulated accidents (same flow area than in RELAP5 and discharge coefficient equal to 0.8) in order to evaluate the convenience of using this method for the pressurizer relief valve flow areas calculation. Although the pressurizer relief model does not play the important role on the primary water mass inventory, primary pressure and temperatures that showed before, it's still improving the evolution of the accident. Again it's confirmed the better behavior of the new PORV's model.

As we have seen, MAAP assumes MFW termination certain delay time after the reactor scram that it's introduced by the user. The steam generator main feed water coast down time is assumed to be zero. With the aim of adjust the steam generator level, it has been modified the signal delay time for main feedwater isolation, which is defined in MAAP5 as the delay from the receipt of signal (reactor scram) until actual isolation. For the MAAP5 base model this time is 3.0 seconds. However, in RELAP5 the isolation of the MFW valves is controlled by the average primary temperatures. In order to match the steam generator level of the MAAP5 modified case to the RELAP5 calculation, the delay time has been extended to 7.5 seconds, the same delay calculated by RELAP5.

By default, the discharge coefficient of the break is assumed to be 0.75. Different simulations changing the break discharge coefficient have been made in order to evaluate its effect on the thermal-hydraulics properties. The best approximation is reached with the discharge coefficient set to 1.

Break area sensibilities showed no significant reduction of the primary pressure.
Figure 6.27: Small break LOCA – Nuclear power

Figure 6.28: Small break LOCA – Primary to secondary heat transfer
2" LOCA IN COLD LEG A WITHOUT AFW

SUBCOOLED MARGIN

Figure 6.29: Small break LOCA – Subcooled margin

2" LOCA IN COLD LEG A WITHOUT AFW

PRIMARY WATER INVENTORY

Figure 6.30: Small break LOCA – Primary water mass inventory
2" LOCA IN COLD LEG A WITHOUT AFW

PRIMARY PRESSURE

Figure 6.31: Small break LOCA – Primary pressure

2" LOCA IN COLD LEG A WITHOUT AFW

BREAK LIQUID MASS FLOW

Figure 6.32: Small break LOCA – Break liquid mass flow
Figure 6.33: Small break LOCA – Break vapor mass flow

Figure 6.34: Small break LOCA – High pressure injection mass flow
Figure 6.35: Small break LOCA – Vessel water level

Figure 6.36: Small break LOCA – Pressurizer level
Figure 6.37: Small break LOCA – Pressurizer relief valve flow

Figure 6.38: Small break LOCA – Steam generator wide range level
**Figure 6.39:** Small break LOCA – Steam generator A relief valve mass flow

**Figure 6.40:** Small break LOCA – Steam generator A pressure
Figure 6.41: Small break LOCA – Hot leg A temperature

Figure 6.42: Small break LOCA – Cold leg A temperature
6.1.5. Steam generator tube rupture

Accident conditions

The initiating event, a double guillotine steam generator B tube rupture, is produced after 50 seconds at nominal conditions followed by the main steam isolation valves closure. It’s postulated the unavailability of one train of High Pressure Injection system, the AFW motor driven pump B and the AFW turbo driven pump.

The AFW motor driven pump A will feed the steam generator A. At time 1850 seconds, the steam generators B and C are isolated and one pressurizer relief valve is manually opened in order to reduce the pressure of the steam generator B and the break mass flow. The steam generator A relief valve is manually controlled from 1850 seconds to the end of the simulation to avoid a maximum cooling greater than 55 ºC in any period of one hour. The main coolant pumps are working during all the accident.

The simulation is finished after 10000 seconds, with the accident controlled and the temperatures decreasing.

Accident description

The main steam isolation valves closure vaporizes the steam generator water and consequently, in MAAP5.0.1, the water level falls below the steam generator water level setpoint for low level reactor trip in just 2.6 seconds. In RELAP5, the SG level didn’t fall below this setpoint and the scram is produced after 9.5 seconds due to OTDT (see Figure 6.43 and Figure 6.52).

The MFW is isolated few seconds after reactor scram and the AFW motor driven pump A is started to inject water only into the SG A. MAAP models the start of the AFW system as a step function; that is, after the reactor scram, MAAP waits for expiration of the delay time specified by the variable TDAFW, after which the AFW flow is enabled. In accordance with the AFW system specifications it should be initiated if there is Safety Injection signal, low level in any SG, MFW turbo driven pumps shut off, AMSAC actuation or low voltage in the MCP lines. The AFW control of the RELAP5 PWR model is accordingly simulated to these requirements and this discrepancy is the source of the different SG water level behavior.

The AFW mass flow is regulated to maintain the steam generator water wide range level around 80%. The RELAP5 water level control is quite simple; the AFW valve is opened if the water level is between 78% and 83% and therefore, the AFW mass flow is not constant (see Figure 6.52 and Figure 6.58). This control makes oscillate the SG water level and therefore, the relief and safety valves mass flow and the primary to secondary heat transfer fluctuate (see Figure 6.44 and Figure 6.55).

The steam generator tube rupture depressurizes the primary system until the safety injection signal is activated, at 290.2 s, 478.4 s and 480.0 s for the MAAP5 base model, MAAP5 modified model and RELAP5 model respectively. At this point, one train of the HPI system started to inject borated water into the cold leg (see Figure 6.47 and Figure 6.49).
The relief and safety valves of the steam generators are automatically opened after MSIV's closure with the aim of avoiding the over pressure of the secondary system. After 1850 seconds, the SG B relief and safety valves are manually closed in order to isolate the broken SG. The relief valve of the steam generator A is manually controlled to cool the primary system preventing a maximum cooling greater than 55 °C in any period of one hour and the relief valve of the SG C is manually closed due to MAAP input options (see Figure 6.55, Figure 6.56 and Figure 6.57).

The PORV opening reduces the primary pressure and consequently the secondary pressure decreases (see Figure 6.59, Figure 6.60 and Figure 6.61). The break mass flow direction is inverted for few seconds due to the pressure mismatch between primary and secondary systems and therefore, some SG B liquid water returned into the RCS (see Figure 6.48).

Notice that in MAAP5, the SG B pressure is not reduced below the RCS pressure while it is reduced in RELAP5. This is consequence of the faster depressurization of the steam generators B and C calculated by RELAP5 (see Figure 6.60).

In order to avoid the steam generator B relief valves opening and attempting to minimize the break mass flow, the primary system pressure is reduced until 7.5E6 Pa by manually opening the pressurizer relief valve at 1850 seconds. Accordingly to the RCS rapid depressurization, the subcooled margin is reduced for few minutes and the boiled-up vessel water level is decreased below the active core elevation (see Figure 6.45 and Figure 6.64).

After the PORV opening, the pressurizer became solid and the water mass flow relieved through the PORV calculated by RELAP5 made nonphysical oscillations so even reducing the time step (see Figure 6.50 and Figure 6.51).

The PORV opening held back the RCS mass flow of all loops for few minutes except for the loop A in RELAP5 (see Figure 6.65 and Figure 6.66). The primary water mass inventory is also reduced after the PORV opening until the HPI mass flow is able to compensate the water leaked (see Figure 6.46).

The hot and cold leg temperatures decreased after the SG A relief valve and the pressurizer PORV opening (see Figure 6.65 and Figure 6.66).

Adjustments

The following figures show the evolution of the most significant variables for three different calculations: One with RELAP5, another with MAAP5 base case and another with MAAP5 modified case. The following adjustments have been introduced in the MAAP5 modified model in order to approximate its behavior to the RELAP5 calculation.

The decay heat in MAAP5 is introduced by the user using a table depending on time after reactor shutdown. In RELAP5 the decay heat is calculated using the reactor kinetics of the cycle. Therefore, the residual power has been adjusted in the MAAP5 modified case in order to coincide with the RELAP5 decay heat.
It has been tested again the same modified pressurizer PORV's model used in the previous simulated accidents (same flow area than in RELAP5 and discharge coefficient equal to 0.8) in order to evaluate the convenience of using this method for the pressurizer relief valve flow areas calculation. Although the pressurizer relief model does not play the important role on the primary water mass inventory, primary pressure and temperatures that showed before, it's still improving the evolution of the accident. Again it's confirmed the better behavior of the new PORV's model.

As we have seen, MAAP assumes MFW termination certain delay time after the reactor scram that it's introduced by the user. The steam generator main feed water coast down time is assumed to be zero. With the aim of adjust the steam generator level, it has been modified the delay time signal for the main feedwater isolation, which is defined in MAAP5 as the delay from the receipt of signal (reactor scram) until actual isolation. For the MAAP5 base model this time is 3.0 seconds. However, in RELAP5 the isolation of the MFW valves is controlled by the average primary temperatures. Tempting to match the steam generator level of the MAAP5 modified case to the RELAP5 calculation, the MFW system was kept after scram until $t=97.5$ seconds, the same delay time calculated by RELAP5.

By default, the discharge coefficient of the break is assumed to be 0.75. Different simulations changing the break discharge coefficient have been made in order to evaluate its effect on the thermal-hydraulics properties. The best approximation is reached with the discharge coefficient set to 0.45.
STEAM GENERATOR TUBE RUPTURE

NUCLEAR POWER

Figure 6.43: Steam generator tube rupture – Nuclear power

STEAM GENERATOR TUBE RUPTURE

PRIMARY-SECONDARY HEAT TRANSFER

Figure 6.44: Steam generator tube rupture – Primary to secondary heat transfer
STEAM GENERATOR TUBE RUPTURE

SUBCOOLED MARGIN

Figure 6.45: Steam generator tube rupture – Subcooled margin

STEAM GENERATOR TUBE RUPTURE

PRIMARY WATER INVENTORY

Figure 6.46: Steam generator tube rupture – Primary water inventory
STEAM GENERATOR TUBE RUPTURE

PRIMARY PRESSURE

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<th>MAAP5 Modified</th>
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Figure 6.47: Steam generator tube rupture – Primary system pressure

STEAM GENERATOR TUBE RUPTURE

BREAK MASS FLOW

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Figure 6.48: Steam generator tube rupture – Liquid break mass flow
Figure 6.49: Steam generator tube rupture – HPI mass flow

Figure 6.50: Steam generator tube rupture – Pressurizer water level
Figure 6.51: Steam generator tube rupture – PORV combined water and vapor flow

Figure 6.52: Steam generator tube rupture – Steam generator A water wide range level
Figure 6.53: Steam generator tube rupture – Steam generator B water wide range level

Figure 6.54: Steam generator tube rupture – Steam generator C water wide range level
Figure 6.55: Steam generator tube rupture – SG A relief and safety valves mass flow

Figure 6.56: Steam generator tube rupture – SG B relief and safety valves mass flow
STEAM GENERATOR TUBE RUPTURE

STEAM GENERATOR C RELIEF & SAFETY VALVES FLOW

Figure 6.57: Steam generator tube rupture – SG C relief and safety valves mass flow

AFW MASS FLOW TO STEAM GENERATOR A

Figure 6.58: Steam generator tube rupture – AFW mass flow entering SG A
Figure 6.59: Steam generator tube rupture – Steam generator A pressure

Figure 6.60: Steam generator tube rupture – Steam generator B pressure
STEAM GENERATOR TUBE RUPTURE

STEAM GENERATOR C PRESSURE

Figure 6.61: Steam generator tube rupture – Steam generator C pressure

HOT LEG A TEMPERATURE

Figure 6.62: Steam generator tube rupture – Hot leg A temperature
Figure 6.63: Steam generator tube rupture – Cold leg A temperature

Figure 6.64: Steam generator tube rupture – Boiled-up vessel water level
STEAM GENERATOR TUBE RUPTURE

Figure 6.65: Steam generator tube rupture – Primary loop A mass flow

Figure 6.66: Steam generator tube rupture – Primary loop B mass flow
6.1.6. Station Blackout

**Accident conditions**

The initiating event, a station blackout, is produced after 50 seconds at nominal conditions. It's postulated the unavailability of the AFW turbine driven pump and the MFW isolation just after the reactor trip. Although the actual and RELAP5 MFW isolation is produced after the SCRAM when the average primary temperatures are lower than 568.15 K, is postulated the MFW isolation just after the scram with the aim of matching the RELAP5 and MAAP5 behavior of the steam generator level.

Due to the loss of AC power, the control rods are de-energized and the reactor is SCRAMed, the main coolant pumps and MFW motor driven pumps are tripped and the Safety Injection system (except the accumulators), the pressurizer’s heaters and pressurizer’s sprays are not available.

It's assumed the loss of coolant through the MCP seals just after the loss of AC power; first the leakage rate is 1.3 liters per second and after 10 minutes, 30 liters per second.

It’s postulated the loss of DC power two hours after the station blackout thus, from this point, the steam generator and pressurizer’s PORVs won’t be available.

The simulation is finished at 6380 seconds, when core damage is produced in RELAP5 and before the loss of DC power.

**Accident description**

After the loss of AC power, the reactor is tripped due to the de-energization of the control rods thus, the nuclear power and the primary to secondary heat transfer decrease rapidly (see Figure 6.67 and Figure 6.68).

The mass flow rate of the primary system is rapidly decreased after the MCPs shutdown to a greater extent in MAAP5 than in RELAP5 and in consequence, in MAAP5, the primary to secondary heat transfer is decreased faster after the loss of AC power and oscillated once. The heat transfer is not similar between codes until the primary mass flow is almost matched (see Figure 6.82 and Figure 6.68). The lower heat transfer in MAAP5 reduces more the subcooled margin since the hot leg temperature is higher and the primary pressure is lower than in RELAP5 (see Figure 6.69, Figure 6.71 and Figure 6.79).

The secondary pressure increases after the reactor trip faster in MAAP5 than in RELAP5 and it’s almost constant during all the simulation. Because of the unavailability of the auxiliary feed water system, the steam generator relief valves play an important role on the primary to secondary heat transfer until the steam generator and the primary system are emptied (see Figure 6.70, Figure 6.76, Figure 6.77 and Figure 6.78).
Since the AFW system is not available and the MFW system is tripped after reactor shutdown, the steam generator level decreases during all the accident and they are emptied approximately 45 minutes after the loss of AC power (see Figure 6.76).

The pump seal failure depressurizes the primary system at 1.3 liters per second during the first 10 minutes and then, at 30 liters per second (see Figure 6.72). The primary pressure is always over the accumulator’s pressure and therefore, there’s no safety injection entering the primary system. The water natural circulation is stopped due to the reduction of the water mass inventory and in consequence, the primary pressure, the cold temperature and the pressuriser water level started to increase (see Figure 6.71, Figure 6.80 and Figure 6.74). In MAAP5, the pressurizer’s relief valves were automatically opened to reduce the RCS pressure but not in RELAP5 (see Figure 6.75). The loss of coolant reduced the vessel water level and the core was uncovered (see Figure 6.73).

The simulation is finished after core damage in RELAP5, that is, when the core is uncovered and the cladding temperature is higher than 1350 K (see Figure 6.81).

**Adjustments**

The following figures show the evolution of the most significant variables for three different calculations: One with RELAP5, another with MAAP5 base case and another with MAAP5 modified case. The following adjustments have been introduced in the MAAP5 modified model in order to approximate its behavior to the RELAP5 calculation.

The decay heat in MAAP5 is introduced by the user using a table depending on time after reactor shutdown. In RELAP5 the decay heat is calculated using the reactor kinetics of the cycle. Therefore, the residual power table has been adjusted in the MAAP5 modified case in order to coincide with the RELAP5 decay heat calculation.

In MAAP5 there are two different methods to calculate the safety valve flow areas. The method used in the MAAP base model calculates the PORV’s flow area using the reference mass nominal flow rate at a given opening pressure introduced by the user. In the second method, which has been used in the MAAP5 modified model, the flow area and the discharge coefficient associated to the relief valve are introduced by the user.

It has been tested again the same modified pressurizer PORV’s model used in the previous simulated accidents (same flow area than in RELAP5 and discharge coefficient equal to 0.8) in order to evaluate the convenience of using this method for the pressurizer relief valve flow areas calculation. Although the pressurizer relief model does not play the important role on the primary water mass inventory, primary pressure and temperatures that showed before, it’s still improving the evolution of the accident. Again it’s confirmed the better behavior of the new PORV’s model.

The actual and RELAP5 MFW isolation is produced after the reactor trip, when the average primary temperatures are lower than 568.15 K. For this accident in RELAP5, the MFW is not isolated until 103 seconds after reactor shutdown and during this period, the MFW mass flow is regulated according to the vapor mass flow leaving out the steam generator. This control is simplified in MAAP5, where the
MFW isolation is produced after a delay time introduced by the user or there is the option of keeping the MFW after reactor scram. Due to the impossibility to simulate the same MFW mass flow for both codes and in order to march the RELAP5 and MAAP5 behavior of the steam generator level, it's postulated the MFW isolation just after the reactor scram.

By default, the discharge coefficients of the breaks are assumed to be 0.75. Different simulations changing the break discharge coefficients have been made in order to evaluate the effect on the thermal-hydraulics properties. The best approximation is reached with the discharge coefficients of each break (each loop) set to 1.0.

The steam generator A level is emptied at time equal to 2940 s, 2360 s and 2550 s for RELAP5, MAAP5 base model and MAAP5 modified model respectively. The core is uncovered earlier in RELAP5 and therefore the core damage (hot cladding temperature equal to 1350 K) is also reached earlier in RELAP5 than in MAAP5.

The simplicity of this accident allows observing the different behavior of both codes in front of the natural circulation phenomena and how it modifies the timing of the steam generator and vessel dry-outs. Although there are some differences in the evolution between both codes, it's possible to say that MAAP5 foresees quite well the behavior of the thermal-hydraulic properties after a station blackout without AFW.
Figure 6.67 – Station Blackout – Nuclear power

Figure 6.68 – Station Blackout – Primary to secondary heat transfer
Figure 6.69 – Station Blackout – Subcooled margin

Figure 6.70 – Station Blackout – Primary water inventory
Figure 6.71 – Station Blackout – Primary system pressure

Figure 6.72 – Station Blackout – Combined liquid and vapor break mass flow seal A
Figure 6.73 – Station Blackout – Vessel water level

Figure 6.74 – Station Blackout – Pressurizer water level
Figure 6.75 – Station Blackout – Pressurizer relief valves mass flow

Figure 6.76 – Station Blackout – SG wade range level
Figure 6.77 – Station Blackout – SG and SV relief valves mass flow

Figure 6.78 – Station Blackout – SG A pressure
Figure 6.79 – Station Blackout – Hot leg A temperature

Figure 6.80 – Station Blackout – Cold leg A temperature
Figure 6.81 – Station Blackout – Cladding temperature and maximum core temperature

Figure 6.82 – Station Blackout – Primary loop A mass flow
6.1.7. Conclusions of RELAP5 vs MAAP5 comparison

The simplicity of the MAAP5.0.1 control for simulating the feedwater and steam dump behavior has limited the accident selection however; the MAAP containment is modeled in detail while it isn’t in the RELAP5 one and in consequence, the MAAP pressure evolution is introduced in the RELAP5 model to simulate the containment peak pressures after mass and energies releases.

The main detected differences between both codes are:

- The MFW isolation is produced after a certain delay time in MAAP5 however, in the RELAP5 model, the isolation time depends on the primary average temperatures and therefore, it is function of the accident simulated.

- The steam needed to feed the AFW turbine driven pump is not considered in the MAAP5.0.1 model while it is taken into account, and it is extracted from the SG B and C, in the RELAP5 model.

- The AFW mass flow control, which regulates the steam generator wide range levels around 80%, is simple and discrete in the RELAP5 model. Thereupon it introduces some oscillations in the primary to secondary heat transfer however; the SG level control in MAAP5 is able to stabilize the level around 80% without introducing so many fluctuations. This different control modifies the behavior of the plant between both codes.

- The steam dump is not considered in the MAAP5 model and the MSIVs are closed 0.5 seconds after reactor trip. The same actuation is imposed in RELAP5 in all the simulated accidents.

- In contrast to RELAP5, the containment model in MAAP is modeled in detail and therefore, for all the simulations made with MAAP5, the four Fan Coolers are available in order to cool the containment building and reduce its pressure. In the RELAP5 model, the containment is considered as a time dependent volume and it can’t predict peak pressures after large amount of mass and energy releases to the containment building. In consequence, the containment pressure is introduced according to the MAAP5 containment pressure evolution in order to converging temperatures and pressures of the primary system between both codes and models.
• In RELAP5 is not possible to simulate the recirculation period after the refueling water storage tank emptying and in consequence, the containment sprays are not available in the MAAP5 simulation in order to match the time to empty the water of the RWST.

Taking into account the previous limitations of both codes, seven accidents were selected and simulated from which good results were obtained and they showed the need to adjust some MAAP5 model aspects.

The adjustments performed that can be introduced permanently in the MAAP5 model are:

• As we have seen, in MAAP5 there are two different methods to calculate the safety valve flow areas. The method used in the MAAP base model calculates the pressurizer PORV’s flow area using the reference mass nominal flow rate at a given opening pressure introduced by the user. In the second method, which has been used in the MAAP5 modified model, the flow area and the discharge coefficient associated to the relief valve are introduced by the user.

It has been tested the same modified pressurizer PORV’s model in all the simulated accidents (same flow area than in RELAP5 and discharge coefficient equal to 0.8) in order to evaluate the convenience of using this method for the pressurizer relief valve flow areas calculation.

The new PORV’s flow area calculation allows simulating significantly better the evolution of the transients where the pressurizer’s PORVs need to be opened and therefore, it’s proposed to change its model.

• By default in MAAP5, the discharge coefficients of the primary breaks are assumed to be 0.75. Different simulations changing the break discharge coefficient have been made in all LOCA transients in order to evaluate its effect on the thermal-hydraulics properties. The best approximation to RELAP5 is reached for all the cold leg LOCA accidents with the discharge coefficient set to 1. For the steam generator tube rupture scenario, the best approximation is reached with the discharge coefficient equal to 0.45. It's proposed to change the default discharge coefficients.

The comparative analysis between MAAP5.0.1 and RELAP5 Mod3.2 calculations has showed the limitations of the Reference Plant MAAP5 model to simulate some accidents of the level 1 Probabilistic Risk Assessment. However, it allowed testing the Reference NPP MAAP5.0.1 model in front of the thermal-hydraulic phase of the severe accidents and justifying its behavior in front of different initiating events.
6.2. Reference plant vs Zion comparison with MAAP5.0.1

According to the MAAP4 Applications Guidance [1], it is recommended that the parameter file be tested with three standard sequences to validate the file’s overall performance: a Station Blackout, a small LOCA, and a large LOCA.

Meeting the recommendations, a comparison between the results of the sample sequences obtained with the Reference Plant MAAP5.0.1 model and the Zion MAAP5.0.1 model has been done in order to test and validate the new plant model for Reference Plant I-II NPP.

The present chapter includes the results of the comparative analysis using the three PWR Large Dry Plant standard sequences that are delivered with the MAAP5.0.1 code. These are:

1. LBLOCA: A double-ended cold LOCA with failure of recirculation
2. SBLOCA: A 2” break at bottom of horizontal cold leg with failure of recirculation
3. Station blackout with pump seal LOCA

While not completely exhaustive in terms of exercising every single line of the code, these sequences do an adequate job of exercising all major models and confronting those models with a variety of anticipated boundary conditions. These sequences also enable/disable various operator actions and system actuations. In so doing, these sequences provide the broadest general evaluation of the code in the most productive manner possible.

6.2.1. Plant model and steady state

The new Reference Plant model is updated from the last MAAP4.0.6 Reference Plant model and the main sources of information are: the Final Safety Analysis Report (FSAR), the Precautions, Limitations and Setpoints (PLS) and the building, installations and systems schemes of the Reference Plant supplied by the Utility.

The Reference Plant and Zion NPP are Westinghouse PWR Large Dry Plants. The main difference is the thermal power and hence, the number of loops in each plant. The Reference Plant has 3 loops while Zion is a four-loop plant. Another significant difference is the reactor cavity compartment size and the orientation and area of the junctions that connect this compartment with the other ones. Zion’s cavity is smaller than Reference Plant’s cavity and the disposition of its junctions may flood the reactor cavity and cool externally the reactor vessel. This phenomenon is not possible in the Reference Plant.
In the following Table 6.1 are summarized the cavity’s compartment features of the models.

<table>
<thead>
<tr>
<th>Cavity Compartment</th>
<th>ZION model</th>
<th>Reference Plant model</th>
</tr>
</thead>
<tbody>
<tr>
<td>Volume</td>
<td>217.0 m³</td>
<td>407.5 m³</td>
</tr>
<tr>
<td>Elevation relative to ground floor</td>
<td>-10.0 m</td>
<td>-22.55 m</td>
</tr>
<tr>
<td>Height</td>
<td>11.0 m</td>
<td>14.835 m</td>
</tr>
</tbody>
</table>

Table 6.1 – Cavity’s compartment features of the Reference Plant and Zion MAAP5.0.1 models

In the following Table 6.2 are summarized the cavity compartment’s junctions features of the models.

<table>
<thead>
<tr>
<th>Model</th>
<th>Junction Number</th>
<th>Junction Orientation</th>
<th>Upstream compartment</th>
<th>Downstream compartment</th>
<th>Junction Area (m²)</th>
</tr>
</thead>
<tbody>
<tr>
<td>ZION</td>
<td>1</td>
<td>Horizontal</td>
<td>Cavity Comp.</td>
<td>Lower Comp.</td>
<td>0.500</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Vertical</td>
<td>Cavity Comp.</td>
<td>Lower Comp.</td>
<td>5.920</td>
</tr>
<tr>
<td>Reference Plant</td>
<td>1</td>
<td>Horizontal</td>
<td>Coolers Comp.</td>
<td>Cavity Comp.</td>
<td>4.534</td>
</tr>
<tr>
<td></td>
<td>5</td>
<td>Vertical</td>
<td>Cavity Comp.</td>
<td>Upper Comp.</td>
<td>0.762</td>
</tr>
<tr>
<td></td>
<td>8</td>
<td>Horizontal</td>
<td>Cavity Comp.</td>
<td>Lower Comp.</td>
<td>1.479</td>
</tr>
</tbody>
</table>

Table 6.2 – Cavity compartment’s junctions features of the Reference Plant and Zion MAAP5 models

Other differences between models are summarized in the following tables and they will be explained in detail in the following chapters if they are relevant for clarifying the different evolution between both models. As can be seen in the following tables, some system actuation setpoints and some steady state conditions are different between the Reference Plant and Zion NPP models.

The following Table 6.3 shows the main setpoints and features of the principal systems used at the Reference Plant and Zion models.

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>Reference Plant MAAP5 model</th>
<th>Zion MAAP5 model</th>
</tr>
</thead>
<tbody>
<tr>
<td>Low pressurizer trip point</td>
<td>1.3536E7 Pa</td>
<td>1.2507E7 Pa</td>
</tr>
<tr>
<td>Pressure setpoint for charging pumps</td>
<td>Not modeled</td>
<td>1.262E7 Pa</td>
</tr>
<tr>
<td>Pressure setpoint for HPI</td>
<td>1.285E7 Pa</td>
<td>1.172E7 Pa</td>
</tr>
<tr>
<td>Pressure setpoint for LPI</td>
<td>1.285E7 Pa</td>
<td>1.276E6 Pa</td>
</tr>
<tr>
<td>Pressure setpoint for containment sprays</td>
<td>0.16995E6 Pa</td>
<td>0.245E6 Pa</td>
</tr>
<tr>
<td>Accumulator initial pressure</td>
<td>4.516E6</td>
<td>4.24E6 Pa</td>
</tr>
<tr>
<td>Accumulator initial water mass</td>
<td>2.7564E4 kg</td>
<td>2.382E4 kg</td>
</tr>
</tbody>
</table>
### Table 6.3 – Setpoints for The Reference Plant I-II and Zion MAAP5.0.1 models

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>Reference Plant MAAP5 model</th>
<th>Zion MAAP5 model</th>
</tr>
</thead>
<tbody>
<tr>
<td>Opening pressure SG PORV</td>
<td>7.8093E6 Pa</td>
<td>7.24E6 Pa</td>
</tr>
<tr>
<td>RWST initial water mass</td>
<td>1.44768E6 kg</td>
<td>1.316E6 kg</td>
</tr>
<tr>
<td>Delay time for MFW isolation</td>
<td>3.0 s</td>
<td>0.0 s</td>
</tr>
<tr>
<td>Delay time for AFW actuation</td>
<td>49.84 s</td>
<td>10.0 s</td>
</tr>
<tr>
<td>Pressurizer PORV opening pressure</td>
<td>1.6209E7 Pa</td>
<td>1.583E7 and 1.6148E7 Pa</td>
</tr>
<tr>
<td>CST initial water mass</td>
<td>0.6414E6 kg</td>
<td>1.0E6 kg</td>
</tr>
</tbody>
</table>

The following Table 6.4 shows the data obtained after simulating a steady state of 1000 seconds with the Reference Plant and Zion models.

### Table 6.4 – Nominal conditions for The Reference Plant I-II and Zion MAAP5.0.1 models

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>Reference Plant MAAP5.0.1 model</th>
<th>Zion MAAP5.0.1 model</th>
</tr>
</thead>
<tbody>
<tr>
<td>Primary system pressure</td>
<td>1.5459E7 Pa</td>
<td>1.5732E7 Pa</td>
</tr>
<tr>
<td>Secondary system pressure</td>
<td>6.4776E6 Pa</td>
<td>5.0669E6 Pa</td>
</tr>
<tr>
<td>Hot leg temperature</td>
<td>598.91 K</td>
<td>599.92 K</td>
</tr>
<tr>
<td>Cold leg temperature</td>
<td>561.54 K</td>
<td>562.88 K</td>
</tr>
<tr>
<td>Pressurizer water level</td>
<td>6.628 m</td>
<td>8.527 m</td>
</tr>
<tr>
<td>Steam generator wide range level</td>
<td>12.299 m</td>
<td>12.479 m</td>
</tr>
<tr>
<td>Nuclear power</td>
<td>2940.60 MW</td>
<td>3565.00 MW</td>
</tr>
<tr>
<td>Primary to secondary heat transfer</td>
<td>2954.19 MW</td>
<td>3577.01 MW</td>
</tr>
<tr>
<td>RCP’s power (1 pump)</td>
<td>4.630 MW</td>
<td>3.175 MW</td>
</tr>
<tr>
<td>Convective heat losses</td>
<td>1.795E6 W</td>
<td>2.00E6 W</td>
</tr>
<tr>
<td>Primary water inventory (without pzr)</td>
<td>175829.11 kg</td>
<td>209453.14 kg</td>
</tr>
<tr>
<td>Primary system mass flow (one loop)</td>
<td>4657.21 kg/s</td>
<td>4250.75 kg/s</td>
</tr>
<tr>
<td>Steam mass flow (one SG)</td>
<td>546.45 kg/s</td>
<td>483.53 kg/s</td>
</tr>
<tr>
<td>MFW water temperature</td>
<td>500.37 K</td>
<td>493.30 K</td>
</tr>
</tbody>
</table>

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*ETSEIB*
6.2.2. Large Break LOCA

**Accident conditions**

The initiating event, a double-ended LOCA in the cold leg 1, is produced at time zero at nominal conditions. It's postulated the unavailability of the four Fan Coolers and the recirculation mode failure after the Refueling Water Storage Tank (RWST) depletion. The simulation is stopped after 20000 s.

The following table summarizes the key timings for the Double-ended cold leg LOCA sequence, which were simulated with MAAP5.0.1 using the Reference Plant and Zion NPP models.

<table>
<thead>
<tr>
<th>MAAP5.0.1 PWR LBLOCA Figures-of-Merit</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
</tr>
<tr>
<td>Fraction of Clad Reacted in Vessel</td>
</tr>
<tr>
<td>Time of Core Uncovery (s)</td>
</tr>
<tr>
<td>Time of 1st Relocation to Lower Plenum (s)</td>
</tr>
<tr>
<td>Time of Vessel Failure (s)</td>
</tr>
</tbody>
</table>

Table 6.5 – Key timings for the sequence: Double-ended cold leg LOCA with failure of recirculation

**Accident Description**

The double-ended LOCA depressurizes the primary system in few seconds due to the large amount of water and gas discharged through the break (see Figure 6.83, Figure 6.84 and Figure 6.85). The reactor is automatically shutdown due to MCP trip, which is produced owing to low RCS mass flow. Then, accordingly to the MAAP control, the MFW system is stopped, the AFW system and the SI system are initiated increasing the vessel water level, and the MSIVs are closed. The pressurizer heaters are stopped due to low pressurizer water level and the core is uncovered. All these events took place in just one second. The loss of coolant increases the containment pressure so the Containment Sprays are initiated and ultimately, the accumulators are discharged just one minute after the break.

The RWST has run dry in around 50 minutes due to the water consumption of the SI and the Containment Sprays. Since it's postulated the failure of the recirculation mode, the LPI and the Containment Sprays are tripped after the RWST depletion (see Figure 6.86, Figure 6.87 and Figure 6.88). In consequence, the natural circulation is stopped in loops 2 and 3; there wasn't natural circulation in loop 1 where the break occurred; and hence, the core water temperature and the RCS void fraction are increased, abruptly after the core relocation into the lower plenum (see Figure 6.89, Figure 6.90, Figure 6.91 and Figure 6.92). The vessel water level is decreased due to the reduction of core water mass and consequently, the upper plenum gas temperature and the maximum core temperature increased (see Figure 6.93, Figure 6.94, Figure 6.95, Figure 6.96 and Figure 6.97).
The vessel’s downcomer can be blocked or unblocked for gas export depending on the vessel water level and this phenomena introduces higher oscillations in the RCS temperature, the vessel water level and the gas break mass flow just before the RCS is completely empty (see Figure 6.84, Figure 6.94 and Figure 6.95).

Due to the rapid depressurization of the primary system and the increase of RCS water temperature, the primary to secondary heat transfer is inverted. It means that the secondary system heated the primary system until the RCS temperature rose up due to the SI termination (see Figure 6.98).

The AFW mass flow is higher in Zion than in The Reference Plant and hence, the SG water level is increased until its desired collapsed water level faster in Zion (see Figure 6.99 and Figure 6.100). However, the divergent secondary system pressure evolution between models is explained by the different features of the SG (see Figure 6.101). Although the primary to secondary heat transfer is negative and very similar between both models, the secondary pressure is increased in Zion due to the addition of water into the SG while in The Reference Plant, the AFW water is not enough to compensate the energy lost and increase the secondary pressure.

The containment pressure is increased due to the large amount of mass and energy released into the containment building almost instantly after the break. The Containment Sprays are capable to reduce the containment pressure until the RWST is depleted. Then, the containment pressure is increased again, changing abruptly just after the relocation of core material to the lower head (see Figure 6.102).

The different reactor cavity compartment size and the different orientation and area of the junctions that connect the reactor cavity with other compartments explain why the vessel failure took place in the Reference Plant while it didn’t occur in Zion. Zion’s Cavity size and the disposition of its junctions have allowed flooding the reactor Cavity and hence, the reactor vessel external cooling (see Table 6.1 and Table 6.2).

The water mass accumulated in the Cavity compartment is higher in Zion than in The Reference Plant and the vessel water level in Zion is above the vessel height so the vessel didn’t fail (see Figure 6.103 and Figure 6.104). The vessel failure and the consequent corium ejection into the reactor Cavity made increase the containment pressure in the Reference Plant due to the faster vaporization of the water accumulated in the reactor Cavity (see Figure 6.102, Figure 6.103, Figure 6.105 and Figure 6.106).

By looking at the trends of the sequence, comparing the timing of the key events summarized in the figures-of-merit in the Table 6.5 and the primary system and containment conditions shown in the plottable results, and analyzing the system actuations as indicated by events in the summary files it’s possible to affirm that the Reference Plant model is responding as expected when simulating a Double-ended LOCA.
Figure 6.83: Double-ended LOCA – Primary pressure

Figure 6.84: Double-ended LOCA – Gas break flow through loop 1
Figure 6.85: Double-ended LOCA – Water break flow through loop 1

Figure 6.86: Double-ended LOCA – RWST water level
Figure 6.87: Double-ended LOCA – Safety injection mass flow rate

Figure 6.88: Double-ended LOCA – Containment spray mass flow rate
Figure 6.89: Double-ended LOCA – Average flow in cold leg loop 1

Figure 6.90: Double-ended LOCA – Average flow in cold leg loop 2
CNA5 vs ZION5: LBLOCA

**Figure 6.91: Double-ended LOCA – RCS void fraction**

CNA5 vs ZION5: LBLOCA

**Figure 6.92: Double-ended LOCA – Core water temperature**
Figure 6.93: Double-ended LOCA – Core water mass

Figure 6.94: Double-ended LOCA – Vessel water level
Figure 6.95: Double-ended LOCA – Upper plenum hot leg temperature

Figure 6.96: Double-ended LOCA – Maximum core temperature
CNA5 vs ZION5: LBLOCA

**Figure 6.97:** Double-ended LOCA – Total mass of core remaining in core

CNA5 vs ZION5: LBLOCA

**Figure 6.98:** Double-ended LOCA – Primary to secondary heat transfer
Figure 6.99: Double-ended LOCA – AFW mass flow to steam generator 1

Figure 6.100: Double-ended LOCA – Steam generator 1 water level
Figure 6.101: Double-ended LOCA – Steam generator 1 pressure

Figure 6.102: Double-ended LOCA – Cavity compartment pressure
Figure 6.103: Double-ended LOCA – Water mass in the Cavity compartment

Figure 6.104: Double-ended LOCA – Water level in the Cavity compartment
Figure 6.105: Double-ended LOCA – Corium mass in the Cavity compartment

Figure 6.106: Double-ended LOCA – Steam mass in Cavity compartment
6.2.3. Small Break LOCA

**Accident conditions**

The initiating event, a 2” break at the cold leg, is produced at time zero at nominal conditions. It’s postulated the recirculation mode failure after the Refueling Water Storage Tank (RWST) depletion. The simulation is stopped after 60000 seconds.

The following table summarizes the key timings for the 2” break at the cold leg LOCA sequence, which were simulated with MAAP5.0.1 for the Reference Plant and Zion NPP models.

<table>
<thead>
<tr>
<th>MAAP5.0.1 PWR SBLOCA Figures-of-Merit</th>
<th>Reference Plant Model</th>
<th>Zion Model</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fraction of Clad Reacted in Vessel</td>
<td>0.39</td>
<td>0.45</td>
</tr>
<tr>
<td>Time of Core Uncovery (s)</td>
<td>8043</td>
<td>9169</td>
</tr>
<tr>
<td>Time of Hot Leg Creep Rupture (s)</td>
<td>–</td>
<td>21534</td>
</tr>
<tr>
<td>Time of 1st Relocation to Lower Plenum (s)</td>
<td>27223</td>
<td>15384</td>
</tr>
<tr>
<td>Time of Vessel Failure (s)</td>
<td>33096 Due to creep rupture</td>
<td>N/A Due to ex-vessel cooling</td>
</tr>
</tbody>
</table>

Table 6.6 – Key timings for the sequence: 2” break at cold leg with failure of recirculation

**Accident Description**

The 2” break LOCA depressurizes the primary system in few seconds and the reactor is automatically shutdown due to low primary system pressure (see Figure 6.107, Figure 6.109 and Figure 6.110). Then, accordingly to the MAAP control, the MFW system is stopped, the AFW system and the Safety Injection System are initiated and the MSIVs are closed (see Figure 6.117). The main coolant pumps are tripped due to high vibration and the pressurizer heaters are stopped due to low pressurizer water level. The loss of coolant increases the containment pressure and the Fan Coolers are activated. All these actions are produced in just over one minute.

The upper containment sprays are initiated 980 seconds after the LOCA and it accelerates the RWST depletion (see Figure 6.118). Since it’s postulated the recirculation mode failure, the core water mass is reduced, the upper plenum gas temperature started to increase and the core is uncovered around 1000 seconds after the RWST emptying, at time 8043 seconds for the Reference Plant and at time 9169 seconds for Zion NPP (see Figure 6.108 and Figure 6.112).

The accumulators are injected 1550 seconds after the RWST depletion, once the RCS pressure is below the accumulator’s pressure. Then the RCS void fraction is increased and the primary system
pressure diminishes due to the LOCA mass flow and the absence of water injection into the primary circuit (see Figure 6.111).

In contradiction to The Reference Plant, in Zion the higher pressure and temperatures of the RCS produced the hot leg creep rupture and in consequence, the RCS pressure and temperature were suddenly reduced allowing the water of the accumulators to be injected quickly and increase the vessel water level for few seconds. The second reduction of the upper plenum gas temperature is due to the unblocking of the loop seals 1 and 2 which permitted the gas to flow and cool the core (see Figure 6.108 and Figure 6.114).

However, in The Reference Plant, the in-core instrument tube failure reduced the RCS pressure and the hot leg rupture didn’t take place. The failure was closed one hour later when the relocation of core materials to the lower head was initiated and consequently, the RCS pressure and water temperature augmented, increasing the gas break mass flow and the primary to secondary heat transfer, until the vessel failed by creep (see Figure 6.115 and Figure 6.116).

As in the previous scenario, the different reactor cavity compartment size and the different orientation and area of the junctions that connect the reactor cavity with other compartments explain why the vessel failure took place in The Reference Plant while it didn’t occur in Zion. Zion’s Cavity size and the disposition of its junctions have allowed flooding the reactor Cavity and hence, the reactor vessel external cooling (see Table 6.1 and Table 6.2).

The water mass accumulated in the Cavity compartment is higher in Zion than in The Reference Plant and the vessel water level in Zion is above the vessel height, so the vessel didn’t fail (see Figure 6.125 and Figure 6.126). The vessel failure and the consequent corium ejection into the reactor Cavity made increase the containment pressure in the The Reference Plant due to the faster vaporization of the water accumulated in the reactor Cavity (see Figure 6.124, Figure 6.127 and Figure 6.128).

The AFW mass flow is higher in Zion than in The Reference Plant and hence, the SG water level is increased until its desired collapsed water level faster in Zion (see Figure 6.122 and Figure 6.123). The steam generator PORV opening setpoints are different between both plants and the secondary pressure remains higher in The Reference Plant until it started to decrease due to the AFW water injection.

Once the water level got the desired water level, the AFW was stopped and the pressure started to increase. In The Reference Plant the evolution of the SG 1 differs from the evolutions of the SG 2 and SG 3, which are similar to each other. The asymmetry of the natural circulation flow and in consequence, the different energy balance in each steam generator, explain the divergent secondary pressures between plants. The pressure increment produced before the vessel failure is provoked by
the pressure and temperature increase of the RCS after the in-core instrument tube plugging. In Zion, the SG with different behavior is the number 3 (see Figure 6.119, Figure 6.120 and Figure 6.121).

The containment pressure is increased due to the large amount of mass and energy released into the containment building almost instantly after the break. The Containment Sprays are capable to reduce the containment pressure until the RWST is depleted, to a greater extent in The Reference Plant than in Zion due to the higher the Reference Plant’s sprays mass flow. At this point, the water accumulated in the Refueling Cavity in Zion (Compartment number 5) is completely evaporated, increasing the containment pressure faster than in the Reference Plant.

The abrupt changes in the Reference Plant’s containment pressure are produced by the energy and mass realized after the in-core instrument tube failure, the vessel rupture, and the evaporation of the water accumulated in the Cavity compartment. In Zion, the pressure peaks are due to the mass and energy released to the containment after the hot leg creep rupture and the unblocking of the loop seals (see Figure 6.124).

By looking at the trends of the sequence, comparing the timing of the key events summarized in the figures-of-merit in the Table 6.6 and the primary system and containment conditions shown in the plottable results, and analyzing the system actuations as indicated by events in the summary files is possible to affirm that the Reference Plant I-II NPP model is responding as expected when simulating a 2” small LOCA.
Figure 6.107: 2” Break LOCA – Primary Pressure

Figure 6.108: 2” Break LOCA – Upper plenum gas temperature
Figure 6.109: 2" Break LOCA – Water break mass flow

Figure 6.110: 2" Break LOCA – Gas break mass flow
Figure 6.111: 2" Break LOCA – RCS void fraction

Figure 6.112: 2" Break LOCA – Core water mass
Figure 6.113: 2" Break LOCA – Core water temperature

Figure 6.114: 2" Break LOCA – Vessel water level
Figure 6.115: 2" Break LOCA – Mass of core material remaining in core

Figure 6.116: 2" Break LOCA – Maximum core temperature
Figure 6.117: 2" Break LOCA – Safety injection mass flow

Figure 6.118: 2" Break LOCA – Containment sprays mass flow
Figure 6.119: 2” Break LOCA – Steam generator 1 pressure

Figure 6.120: 2” Break LOCA – Steam generator 2 pressure
CNA5 vs ZION5: SBLOCA

Figure 6.121: 2" Break LOCA – Steam generator 3 pressure

CNA5 vs ZION5: SBLOCA

Figure 6.122: 2" Break LOCA – Steam generator 1 water level
Figure 6.123: 2” Break LOCA – AFW mass flow of SG 1

Figure 6.124: 2” Break LOCA – Cavity compartment pressure
Figure 6.125: 2” Break LOCA – Water mass in Cavity compartment

Figure 6.126: 2” Break LOCA – Water level in Cavity compartment
Figure 6.127: 2" Break LOCA – Steam mass in Cavity compartment

Figure 6.128: 2" Break LOCA – Steam mass in Upper compartment
6.2.4. Station Blackout

**Accident conditions**

The initiating event, a station blackout, is produced at time zero at nominal conditions. It's assumed the unavailability of the Auxiliary Feedwater System and the steam generator relief valves. It's postulated the loss of coolant through the MCP seals 2700 seconds after the loss of AC power. The simulation is terminated after 40 hours.

The following table summarizes the key timings for the Station blackout sequence, which was simulated with both MAAP5.0.1 for the Reference Plant and Zion NPP models.

<table>
<thead>
<tr>
<th>MAAP5.0.1 PWR Station Blackout Figures-of-Merit</th>
<th>Reference Plant Model</th>
<th>Zion Model</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fraction of Clad Reacted in Vessel</td>
<td>0.32</td>
<td>0.50</td>
</tr>
<tr>
<td>Time of Core Uncovery</td>
<td>5884</td>
<td>7464</td>
</tr>
<tr>
<td>Time of Hot Leg Creep Rupture</td>
<td>10076</td>
<td>10835</td>
</tr>
<tr>
<td>Time of 1st Relocation to Lower Plenum</td>
<td>12763</td>
<td>16116</td>
</tr>
<tr>
<td>Time of Vessel Failure</td>
<td>19205</td>
<td>23314</td>
</tr>
<tr>
<td>Time of Containment Failure</td>
<td>---</td>
<td>123229</td>
</tr>
</tbody>
</table>

Table 6.7 – Key timings for the sequence: 2” break at cold leg with failure of recirculation

**Accident Description**

Due to the loss of AC power, the control rods are de-energized and the reactor is automatically shut down. The loss of both, the essential and nonessential electrical buses, produces the main coolant pumps trip and the unavailability of the AFW motor-driven pumps, the HPI and LPI systems. Since it's postulated the unavailability of the turbine-driven pump, the steam generators are emptied in less than one hour. The secondary pressure is increased until the SG’s safety valves are opened.

It's lost the RCS pressure control capability since the pressurizer's sprays and the heaters are off. In consequence, during the first hour of the accident the RCS pressure is increased until the pressurizer's PORV valves are opened (see Figure 6.129). It's postulated the loss of coolant through the MCP seals 2700 seconds after the loss of AC power, with an area of $4 \cdot 10^{-5}$ m$^2$ per loop, and because there is no safety injection, the RCS void fraction is increased and the natural circulation of the primary system is stopped (see Figure 6.131, Figure 6.132, Figure 6.133 and Figure 6.134). The water and break mass flows are similar for both plants but the primary circuit volume is greater in Zion therefore, the core uncover is produced earlier in the Reference Plant.
The seal LOCA depressurizes immediately the primary circuit in the Zion NPP model while it doesn’t happen in the Reference Plant model. Actually, the RCS pressure reduction in the Reference Plant started at 8200 seconds due to an in-core instrument tube failure. This kind of failure can only transport gas and fission products and it was plugged 1600 seconds later.

Few seconds after the plugging, the hot leg failed by creep rupture and the primary circuit is immediately depressurized until the Containment pressure. The accumulators are discharged and the water is introduced into the vessel, increasing the vessel water level and reducing drastically the core water temperature. After the accumulators’ injection and the relocation of core materials into the lower head, the loop seals remained unblocked for gas export in Zion NPP while they remained blocked in the Reference Plant (see Figure 6.135, Figure 6.136, Figure 6.137 and Figure 6.138).

The upper plenum gas temperature is increased after the core uncovering and it is drastically reduced after the hot leg creep rupture. Until the vessel failure, this temperature is oscillating due to the blocking and unblocking for the gas export of the loop seals and the downcomer of the vessel. After the lower head rupture, the temperature is decreasing until the Containment is failed (see Figure 6.130).

The maximum core temperature is increased after the core uncover and the whole core is melted and discharged into the Cavity compartment once the vessel has failed due to creep rupture in the Reference Plant and due to the ejection of one penetration tube in Zion NPP (see Figure 6.139 and Figure 6.140).

The steam generator water inventory is evaporated in less than one hour and the secondary pressure augmented after the scram until the safety valve were opened. After the hot leg rupture, the SG pressure suddenly decreased and from this point, the behavior of the steam generator differs between both models due to different evolution of the Containment pressure and hence, due to the different heat transfer rate from the SG to the Containment between models (see Figure 6.141 and Figure 6.142).

By looking at the trends of the sequence, comparing the timing of the key events summarized in the figures-of-merit in the Table 6.7 and the primary system and containment conditions shown in the plottable results, and analyzing the system actuations as indicated by events in the summary files is possible to affirm that the Reference Plant model is responding as expected when simulating an Station Blackout with pump seal LOCA.
Figure 6.129: Station Blackout – Primary Pressure

Figure 6.130: Station Blackout – Upper plenum gas temperature
CNA5 vs ZION5: TMLB

Figure 6.131: Station Blackout – Water break mass flow

CNA5 vs ZION5: TMLB

Figure 6.132: Station Blackout – Gas break mass flow
Figure 6.133: Station Blackout – RCS void fraction

Figure 6.134: Station Blackout – Average flow in hot leg loop 1
Figure 6.135: Station Blackout – Accumulators water mass

Figure 6.136: Station Blackout – Core water mass
Figure 6.137: Station Blackout – Core water temperature

Figure 6.138: Station Blackout – Vessel water level
Figure 6.139: Station Blackout – Total mass of core remaining in core

Figure 6.140: Station Blackout – Maximum core temperature
Figure 6.141: Station Blackout – Pressure of steam generator 1

Figure 6.142: Station Blackout – Water level of steam generator 1
Figure 6.143: Station Blackout – Cavity compartment pressure

Figure 6.144: Station Blackout – Water mass in Cavity compartment
Figure 6.145: Station Blackout – Corium mass in Cavity compartment

Figure 6.146: Station Blackout – Water mass in Annular compartment
6.2.5. Conclusions of Reference Plant vs Zion comparison

It has been compared several scenarios simulated with MAAP5.0.1 using the Reference Plant model and the Zion5 model with the aim of testing the migrated Parameter File. It has been studied the evolution of a Double-ended Break LOCA, a 2” Small LOCA and a Station Blackout.

The Reference Plant has 3 loops while Zion is a four-loop plant. It means that the thermal power, the water mass and the volume of the primary system are quite different between both plants. Furthermore, most of the setpoints that actuates the main systems are slightly different as detailed in Table 6.3. Therefore, the objective of the comparison has been limited to analyze and delimit the major discrepancies found in each scenario. It has been verified that the general trends of the sequences are as expected and the discrepancies are explained by the different features of these plants.

In that way, one significant difference is the reactor cavity compartment size and the orientation and area of the junctions that connect this compartment with the other ones. Zion’s cavity is smaller than the Reference Plant’s cavity and the disposition of its junctions may flood the reactor cavity and cool externally the reactor vessel. This phenomenon is not possible in the Reference Plant and in consequence, the vessel failed in the LOCA cases while it didn’t fail using the Zion’s model.

The system actuations as indicated by events in the summary files, the timing of the key events summarized in the figures-of-merit and the primary system and the containment conditions shown in the plottable results are responding as expected when simulating these standard sequences.

Summarizing, it has been met the recommendations of the MAAP4 Applications Guidance testing all major models and confronted those models with a variety of anticipated boundary conditions. In this way, the updated Reference Plant model has been validated.
7. Conclusions

It has been explained the severe accident phenomenology and the MAAP code used for the study of beyond design bases accidents.

It has been updated and tested the new parameter file for the PWR Reference Plant. First of all, it has been checked the steady state consistency and then, it has been compared the results obtained with the MAAP5.0.1 Reference Plant model with the results obtained with the RELAP5 model.

The comparative analysis between MAAP5.0.1 and RELAP5 Mod3.2 calculations has showed the limitations of the Reference Plant MAAP5 model to simulate some accidents of the level 1 Probabilistic Risk Assessment. However, it allowed testing the Reference NPP MAAP5.0.1 model in front of the thermal-hydraulic phase of severe accidents and justifying its behavior in front of different initiating events.

It has been compared several scenarios simulated with MAAP5.0.1 using the Reference Plant model and the Zion5 model with the aim of validating the updated Parameter File. It has been studied the evolution of a Double-ended Break LOCA, a 2” Small LOCA and a Station Blackout.

By looking at the trends of these sequences, comparing the timing of the key events summarized in the figures-of-merit and the primary system and containment conditions shown in the plottable results, and analyzing the system actuations as indicated by events in the summary files is possible to affirm that the Reference Plant model is responding as expected when simulating these standard sequences.

Summarizing, it has been met the recommendations of the MAAP4 Applications Guidance testing all major models and confronted those models with a variety of anticipated boundary conditions. In this way, the updated Reference Plant model has been validated.
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Bibliography

References


