11. ACCIDENT ANALYSIS

Operating nuclear power plants require a safety analysis report which confirms the original design basis and describes the behavior of the plant for all potential accidental conditions. In accordance with regulatory requirements, the safety analysis should be based on the current status of the systems, structures and components of the NPP, and should consider all the modifications carried out during upgrading outages including those changes which are committed for implementation. For the Ignalina NPP this information is presented in several reports. This includes the TOB [74], the Safety Analysis Report [62] and additional anticipated transients without scram analyzed in the SAR [62].

The initial safety studies were performed by the Russian design institute, RDIPE [74]. For the evaluation of break flow of a steam-water mixture from the rupture an equilibrium two-phase flow model taking into account the hydraulic losses along the pipeline length was used. Critical flow of subcooled water through the break was calculated using a non-equilibrium flow model, which approaches the equilibrium model as the degree of subcooling is reduced. Calculations of the transient pressure response were performed using quasi-static correlations for energy and mass transfer processes. The RDIPE calculations ware performed before 1989 and therefore used the design thermal power level of 4800 MW. However, after the Chernobyl accident the maximum permissible thermal power level of Ignalina reactors was reduced up to 4200 MW.

The SAR computations reflect the present operational power level of about 4200 MW. The accident analysis performed in the SAR were undertaken using Western state-of-the-art computer codes. System codes such as RELAP5 and ATHLET were used for thermal-hydraulic analyses and modern Russian codes such as the 3-dimensional codes SADCO and MOUNT which incorporate coupled neutronic-thermal-hydraulic calculations were used for evaluating reactivity initiated accidents. A review of the verification and validation studies which had been performed for each of these codes was undertaken as part of the quality assurance program. The Western codes had been validated extensively for PWR and BWR reactors but had only limited validation for conditions relevant to the RBMK. The Russian codes had undergone varying degrees of verification. In order to compensate for this lack of extensive verification, the codes were used cautiously when any of the critical and unverified regimes were encountered.

A number of accidents sequences which have to be analyzed in accordance with current Lithuanian regulations were not explicitly addressed either in the Ignalina TOB [74] or in the SAR [62]. As noted in Section 10, the SAR was initially conceived as a Western-style safety analysis report, but the completion of such a SAR would have consumed several times the resources budgeted for the in-depth safety assessment of Ignalina NPP. The scope, especially the scope of the accident analysis, was therefore defined as including assessment specific essential items [75]. A list of 23 accidents was developed which was intended to cover the "worst case" for each accident category in the sense that these sequences bounded those accidental events which were not included. In order to ensure that no important sequence was omitted an assessment was made by Task Group which undertook the development of a Fault Schedule. The goal of this task was to prepare a summary of all the accidental conditions which can be identified as having the potential to lead to fuel damage or a release of radioactivity from the plant. However, a thorough comparison of the accidents considered in the Ignalina SAR with initiating events of an extended Fault Schedule showed that they are bounding most of the credible events and no sequences were found which would have required a modification of the essential items list of accidents specified in the Guidelines for production and review of Ignalina SAR [75].

This Section incorporates material from the SAR Report [62] and Barselina Phase 4 Report [63].

11.1 REQUIREMENTS FOR ACCIDENT ANALYSIS

Design basis accidents are events which bound accident categories (e.g. the guillotine break of the largest pipe in a system). The response of the plant to DB accidents is evaluated using conservative assumptions. The nuclear power plant, its systems, structures and components is then designed to withstand the evaluated loads for such events without releasing harmful amounts of radioactive materials to the outside environment. A set of DBAs is postulated for each type of reactor, covering the consequences of all failure combinations. The following groups of design basis accidents are considered for RBMK-type nuclear power plants [76] :

- Accidents initiated by equipment failures, including loss of flow transients.
- Loss of coolant accidents.
- Reactivity initiated transients.
- Fuel handling accidents.
- Other accidents.

Design basis accidents are classified according to the type of initiating events. A list of initiating events which should be analyzed for each group of DBAs is given in the Subsections which follow.

Depending on the accident sequence, the process used to assess consequences of a particular design basis accident in the Ignalina SAR involves different assessment tasks [62]. If the fuel cladding loses its integrity, a key barrier to a release of fission products is breached, and the coolant in the heat transport system becomes further contaminated by radioactive released from fuel. In turn, the contaminated coolant can be released into the environment by means of normal leakage or by means of accidental discharge either inside or outside of the Accident Confinement System. If the accident does not challenge the fuel cladding and pressure tube integrity, no detailed analysis of other accident issues need to be performed. If there are fuel failures, mass, energy and fission product transport paths must be defined for explicit analysis of radiological consequences. The maintenance of pressure tube integrity is one of the design targets for the design basis accidents. Should a pressure tube fail, it must be shown that the integrity of the reactor cavity is not jeopardized. In addition, for all accidents with mass and energy discharge into the Accident Confinement System, the integrity of this system needs to be verified in order to confirm that the transport path used in analysis of radiological consequences are correctly defined. These steps ensure that, for all accidents addressed in analysis, the compliance with the regulatory dose limits will be demonstrated with adequate confidence.

One of the tasks undertaken in the SAR project was the development of a set of acceptance criteria for each type of accidents [62]. The following acceptance criteria are used in accident analysis:

- fuel cladding integrity criteria,
- pressure tube integrity criteria,
- heat transport circuit integrity criteria,
- reactor cavity integrity criteria,
- ACS integrity criteria,
- permissible doses.

Regulatory document [77] prescribes the acceptable conditions in terms of how many fuel rods can have perforated cladding, and what type of fuel cladding failure is permissible:

- number of fuel rods with perforated cladding is not to exceed 1 % for all the rods in the reactor,
- number of rods that are perforated such that the coolant can come into contact with ceramic fuel is not to exceed 0.1 % of all the fuel in the reactor.

Regulatory document [77] defines also that the peak cladding temperature must not exceed 1200 ^{o} C and that the local fuel cladding oxidation must not exceed 18 % of the initial wall thickness. These criteria are pertinent to the maintenance of coolable fuel geometry during an accident and beyond.

Fuel cladding integrity criteria conservatively define the cladding failure thresholds for all fuel cladding failure mechanisms. The following conditions are sufficient to confirm that the fuel cladding integrity is maintained in an accident [62]:

- maximum fuel enthalpy remains below 712 kJ/kg,
- fuel temperature does no reach the UO_2 melting point of about 2800 ^{o} C,
- fuel cladding temperature does not exceed 700 ^{o} C.

These simplistic criteria are useful for a fast screening of accident analysis results. If these conditions are not exceeded, no further analysis is required to confirm that the accident does not threaten the fuel cladding integrity. If any of these criteria is exceeded, it does not necessarily mean that fuel failures have occurred. It means that supplementary analysis is required. During an accident, fuel cladding can fail due to thermal-mechanical interaction between the fuel and the cladding, or due to thermal deformations of the cladding under positive or negative pressure differentials. The first type of failure is prototypic of rapid and large fuel power excursions where a hot, and possibly molten, UO_2 material may come into contact with the cladding material. The other failure mechanisms are associated with cladding temperature excursion, either when the external pressure is higher than the internal one, or when internal pressure is higher than external one. In first case fuel cladding could fail due to collapses onto the fuel pellet stack and deformation into any gaps between fuel pellets, while in the last case fuel cladding could fail due to ballooning of hot cladding away from the fuel pellet stack. Cladding temperatures at which the failure occurs due to cladding collapse are listed in the Table 11.1 [62]. These failure conditions were quantified for the operating pressure of 7 MPa and the lowest internal pressure within the fuel element as a function of axial gap between the fuel pellets. Cladding temperatures at which the failure occurs due to cladding ballooning are listed in the Table 11.2 [62].

Table 11.1 Temperatures of failure by cladding collapse at P=7 MPa [62]

δ, mm	**2**	**4**	**6**	**8**	**10**	**14**	**20**
T,^{o}C envelope	1300	1300	1280	1260	1240	1120	900
T, ^{o} C onset	1200	1200	1180	1150	700	700	700

Table 11.2 Temperature of failure by cladding ballooning [62]

ΔP, MPa	**1.0**	**2.0**	**4.0**	**6.0**	**8.0**
T, ^{o} C envelope	1000	830	800	790	780
T, ^{o} C onset	850	730	700	700	700

Pressure tube integrity criteria conservatively define the pressure tube failure thresholds. The following conditions are sufficient to confirm that the pressure tube integrity is maintained in an accident [62]:

- pressure in the pressure tube does not exceed 13.4 MPa,
- pressure tube temperature in any cross-section of its wall does not exceed 650 ^{o} C.

If any of these criteria were to be exceeded, the affected pressure tube can potentially fail and supplementary analysis must be performed to establish whether or not the pressure tube integrity is maintained.

The requirement for the integrity of the heat transport circuit is not prescribed by regulations, but has been employed in the Ignalina SAR [62] to avoid complex and costly analyses of accident consequences following pressure boundary failures. The heat transport circuit can withstand three pressure levels. The pressure tubes can withstand at least 13.4 MPa. All fuel channels are hydrostatically tested at this pressure. The piping between the MCP check valve and the pressure header is designed and hydrostatically tested to withstand at least 12.3 MPa. The rest of the of the heat transport circuit piping is designed and tested to withstand at least 10.4 MPa. These test values are applicable to operating temperatures because the ratio of yield stress at the two temperatures is less than 1.4. The lowest of the test pressures is taken to be the acceptance criterion for the accidental pressurization of the heat transport system.

The maintenance of reactor cavity integrity is a derived requirement of acceptable plant response to any accident that involves a discharge of coolant into reactor cavity. Permissible pressure loads on the reactor cavity structures were quantified by the designers of these structures. The permissible loads were evaluated for casing, upper and lower plates. The lower plate can withstand 294 kPa cavity pressure, the casing can withstand 255 kPa cavity pressure. The lowest pressure value corresponds to conservative estimates of pressure needed to lift the upper plate. Cavity pressure exceeding 214 kPa has been described as having possibility to lift the upper plate breaking the reactor seal, the pressure tubes, and affecting the operating of other safety functions. The smallest of these loads is taken as a conservative criterion for maintaining the integrity of the reactor cavity.

The Ignalina NPP is protected against accidental discharges of contaminated coolant by an Accident Confinement System. This system is described in Section 6.3. In accident analysis the maintenance of ACS integrity is a derived acceptance criterion. Permissible pressure loads in ACS compartments are summarized in Table 6.5. These values are adopted as the acceptance criteria in accident analysis. An acceptance hydrogen concentration in any ACS compartment is taken to be 4 % by volume.

Table 11.3 Meteorological parameters

Discharge height, m	Pasquill weather category	Wind speed, m/s	Deposition factor, s/m^3
0	F	2	5.8 10^{-5}
50	F	2	9.15 10^{-6}
150	B	2	1.40 10^{-6}

The regulatory dose limits are taken to be the key criteria of acceptance. Permissible radiological doses to the population after an accident are defined by [78] as follows:

- whole body dose to a member of the population not to exceed 50 mSv,
- critical organ, i.e. thyroid, dose to member of the population is not to exceed 300 mSv.

For design basis accidents, the doses are to be accumulated for a period of one year after the accident at and beyond the Ignalina NPP exclusive zone, i.e., beyond a 3 km radius from the plant. In analyses of design basis accidents doses are evaluated by conservative analyses that assume:

- fission products escaping from the plant are released as a single “puff” at the elevation appropriate to the postulated accident,
- least favorable meteorological conditions are assumed, Table 11.3,
- a critical individual, i.e. child, is assumed to remain at the boundary of the plan exclusion zone indefinitely.

One of the requirements for the accident analysis is to account for the effect of single failure in the accident analysis. The single failure criterion is defined in the IAEA Code of Practice on Design [79]. In order to comply with IAEA practice, analyses would ideally be performed as follows:

- assume that each mitigation system is operating at the start of the accident, with as much equipment out of service as is allowed by the operating procedures,
- perform the accident simulation assuming that all systems operate as described above plus assume that one component of the system has failed,
- repeat the simulations as many times as necessary, each time assuming a different single failure of one component of one system,
- evaluate consequences of all cases and select that which produced the worst consequence.

This ideal approach is difficult to apply in practice and in order to meet the intent of [79] a conservative approach was adopted in the Ignalina SAR project:

- assume that all mitigating systems with equipment credited in the analysis simultaneously experience a single failure of one of its components when the component is called upon to act,
- if the analysis with the plant configuration assumed above produces results which meet the acceptance criteria, no further analysis of this accident sequence is required. However, if the results indicate a non-compliance with acceptance criteria, it is permissible to analyze a less conservative progression of the accident sequence. This less conservative analysis can be achieved by assuming that one or more systems operate in a manner consistent with operating limits, as opposed to having all mitigating system failed simultaneously,
- additional failures of mitigating systems caused as a consequence of initiating event are taken into account.

The analyses evaluate two initial plant states: Design Reference state where all processes and protective systems function as designed, and a plant state where a failure is assumed in each system that is active during the Design Reference accident. This last state is refered to as the Multiple Failures (or Limiting) Plant state.

In addition to the rules for evaluating the effect of single failure for each initiating event, the following deterministic rules are also applied in accident simulations:

- the most effective absorber rod is assumed to be unavailable,
- the second trip parameter is credited in accidents where two parameters are available. Where the second parameter is not available, it must be shown that a sufficient time is available from manual intervention following an unambiguous annunciation of the accident in control room,
- in order to cover any undetected failures in the signals or trip activation logic, where the system has 2 out of n logic, the trip is credited when last signal is reached.

11.2 ACCIDENTS INITIATED BY EQUIPMENT FAILURES

All accidents initiated by equipment failure occur in the intact heat transport system. Therefore, the following issues are relevant to this family of accidents:

- an imbalance between the heat generation in the reactor core and the convective heat removal from the core if and when the forced circulation is lost or impaired,
- a pressurization of the heat transport system if and when the turbines are disconnected,
- a long term coolant makeup to the heat transport system if and when the main heat sink is lost.

Some accidents in this group are subject to only one of the above issues, e.g. a pump power seizure concerns only the issue of the power-cooling mismatch in the channels. Other accidents encounter several above issues simultaneously, e.g. a loss of AC power supply encounter a loss of circulation as well as pressurization. The equipment failure accidents addressed in SAR are as follows:

- single MCP trip,
- multiple MCP trips,
- MCP seizure,
- loss of normal AC electrical power supply,
- turbine generator trip,
- loss of main heat sink,
- loss of feedwater,
- grid frequency reduction,
- spurious opening and failure to re-close of the main safety relief valve.

However, in accordance with regulatory requirements [76] the following accidents initiated by equipment failure should be also analyzed:

- break of GDH check valve disk,
- break of disk of MCP check valve or main gate valve,
- reduction or loss of flow in one fuel channel,
- complete station blackout,
- failure of feedwater system.

Consequences of all the accidents initiated by equipment failure are explored by three cases that are simulated explicitly: MCP seizure, loss of AC power and loss of feedwater supply. The remaining accidents are assessed qualitatively. It is explained how these latter cases relate to the simulated cases, or it is shown that adequate provisions are available in the current plant to make the accident benign.

For the pump failure cases the automatic power reduction is the only required mitigation action. Analysis of the most severe conceivable power-cooling mismatch shows that cladding dry-out is avoided. A combination of a timely power trip, a pump costdown, and relatively early ECCS water injection maintains the cladding and pressure tube wall temperatures below their initial values for accidents that involve a global impairment of forced circulation, i.e. a loss of AC power and a loss of feedwater supply. There is no potential for power-cooling mismatch in accidents that maintain forced circulation, e.g. turbine trip and loss of main heat sink. The accidents that lead to an impairment of steam removal from the heat transport system, i.e. loss of AC power, loss of turbines and loss of heat sink, activate the MCC over-pressure protection system. The SAR analysis shows that this system is adequate, if the timely power reduction is given.

The SAR analysis shows that the reactor power is reduced in a timely manner in all accidents initiated by equipment failures. Either power setbacks AZ-3 or AZ-4, or a trip AZ-1 are performed by the CPS on signals by the EPPS. There are at least two EPPS signals issued in close succession, based on diverse process parameters. Hence, reliable signals are available to activate the reactor power reduction.

The short-term ECCS is not activated in any accidents initiated by equipment failures because there is no break in the MCC to produce the necessary conditioning signal of high pressure in one of reinforced leak-tight compartments. However, the long-term emergency core cooling function is activated quite early in accidents that involve an impairment of steam removal or feedwater supply. The long term emergency feed water supply is preferentially provided by the AFWPs drawing hot water from the deaerators. If AFWPs cannot provide this emergency supply, the ECCS pumps, already running in a re-circulation mode, supply "cold" water from the condensate chambers in the ACS. No automatic system is available to regulate the emergency water supply in the long term, and to establish a long-term heat sink for the removal of decay and stored heat. These functions are performed by operators. Analysis shows, that adequate time is available to initiate the manual operator actions.

Thus, results of analysis show, that the class of events included under accidents initiated by equipment failures are unlikely to cause power plant conditions that would result in violation of the design criteria to avoid fuel damage, maintain integrity of pressure boundaries, and not exceeded regulatory dose limits. The existing protective system at the Ignalina NPP are adequate to bring the plant into a safe state following all accidents initiated by equipment failures.

11.3 LOSS OF COOLANT ACCIDENTS

Pipe breaks in one of the two main circulation loops, the service water system and purification and coolant system as well as steam and feed water line breaks are classified as loss of coolant accidents. The full range of loss of coolant accidents have been assessed. Piping breaks resulting in a loss of coolant from the circuit may occur within the reinforced leak-tight compartments of the ACS or in compartments that are connected to the outside environment. In accordance with regulatory requirements [76] the following loss of coolant accidents should be analyzed for nuclear power plants with RBMK-type reactors:

- break of MCP header or pipeline (e.g., full break of MCP pressure header),
- break of a group distribution header downstream and upstream of check valve,
- break of steam separator downcomer pipe,
- main feedwater line break,
- main steam line break,
- break of water communication line,
- break of steam-water communication line,
- break of pressure tube,
- break of service water pipeline,
- break of purification and cooling system pipeline.

The LOCAs addressed in the SAR include the following accidents:

- full break of the MCP pressure header,
- full break of the GDH downstream of the check valve,
- full break of steam separator downcomer pipe in drum separator compartments,
- partial breaks in a GDH downstream of the check valve which can lead to flow stagnation conditions,
- partial breaks in the pressure header which could potentially lead to stagnation conditions,
- full feedwater line break,
- full steam line break in different compartments.

The SAR concluded that the Ignalina NPP is quite well protected against the breaks that occur in the reinforced leak-tight compartments if they do not result in local flow degradation. A prompt activation of the ECCS occurs for breaks with large discharge rates and for breaks with coincident failures that impair global circulation. However, the emergency core cooling system activation is not fast enough to ensure that dangerous, early temperature excursion do not occur following partial breaks in one GDH. However, note that if local deterioration of channel cooling occurs during this LOCA scenario, the contaminated coolant discharges to the ACS. Analysis also shows that four emergency core coolant pumps, i.e. either the ECCS pumps, or the AFWPs, are sufficient for adequate long term cooling.

In the LOCA scenarios analyzed, the peak fuel temperature did not exceed 1200 ^{o}C, and the fuel cladding oxidation did not reach the maximum allowable levels. The fuel cladding failure criterion of 700 ^{o}C is exceeded in the following LOCA scenarios: full break of the pressure header accompanied with multiple failures, full break of the GDH, and partial break of the GDH. Analysis shows that, except for the last case, the fuel cladding failure criteria are violated for only a very short period of time during the initial phase of accident. Thus, fuel cladding failure is not expected in the first two cases. In the LOCA scenario with flow stagnation conditions in one GDH, fuel elements could fail in several channels. Design modification to improve the activation of the short-term ECCS was recommended and accepted by the Ignalina NPP. This improvement would be implemented during implementation of the SIP-2.

The SAR analysis shows that for all LOCAs which occur inside the reinforced leak-tight compartments, pressure tube temperatures do not exceeded the failure criterion of 650 ^{o}C. Results of analysis also states that for all breaks inside the reinforced leak-tight compartments,

the existing prescribed public dose limits would not be exceeded.
However, for breaks outside the ACS, especially for main steam line breaks, peak cladding and pressure tube temperatures as well as doses could exceed acceptance criteria. The main reason of this is that breaks outside of the reinforced leak-tight compartments do not trip the reactor nor do they activate the ECCS. Violation of acceptance criteria could also result due to a large number of pre-existing cladding failures permitted during normal operation, and due to a limited drainage capacity in the vented compartments. The SAR analysts propose a number of hardware modifications and changes in regulations and procedures to overcome the design weaknesses and to better protect the surrounding population against radiological exposure after steam rupture events. First of all an additional early reactor trip and emergency coolant injection for all break locations, based on the dP/dt measurements in steam separators should be installed. This modification will be implemented in the immediate future at the Ignalina NPP. The SAR also recommended as a safety enhancement measure to keep the number of pre-existing fuel rod failures as low as achievable. Means to rapidly remove the contaminated water from compartments that are in direct communication with the environment will be developed and implemented.

Downcomer breaks outside the ACS do not result in violation of safety criteria. However, reactor hall over-pressure protection may not be sufficient to prevent the release of contaminated coolant to the environment and provisions to improve the reactor hall over-protection will be installed during implementation of the SIP-2.

11.4 REACTIVITY INITIATED ACCIDENTS

Reactivity initiated accidents are accidents which are induced by postulated faults in the CPS. In accordance with regulatory requirements [76] the following reactivity initiated accidents should be analyzed for nuclear power plants with RBMK-type reactors:

- continuous single rod withdrawal,
- continuous rod bank withdrawal,
- rod ejection,
- rod drop,
- faulty actuation of reactor emergency protection,
- refueling error including improper fuel placement,
- inadvertent emergency core cooling system actuation,
- voiding of or gas ingress into control rod channel cooling system.

In the SAR the following cases were analyzed:

- withdrawal of a single rod in the center of the core and at the periphery at full power (4200 MW) and during start-up (240 MW),
- withdrawal of a group of three rods in the center of the core at full power and during start-up,
- voiding of CPS including:
  - loss or interruption of water supply to the upper storage tanks,
  - various CPS coolant flow blockages,
  - air entrapment in the CPS circuit following outage.

Initial conditions have been defined to account for most unfavorable operational conditions. Perturbed axial and radial power distributions have been defined which maximize the effect of the reactivity insertion. For dynamic simulations, power setback signal, the first neutronic trip signal and any trip signal based on process parameters were neglected.

For the single rod or group of three rods withdrawal accidents both at full power and during start-up no safety problems arise because the absolute power remains low and the maximum values of key safety parameters are maintained well below their limiting values. The analysis covers reactivity insertion for high and low rod worth's. The consequences of high reactivity insertion are limited due to the generation of early trip signals, which terminate the transient earlier than in cases with low reactivity insertion. Also, neglecting the first shutdown signal does not create problems concerning the safety limits.

Total voiding of the CPS channels in the reactor at operational conditions can cause a reactivity insertion of up to 4-5 β. The highest reactivity insertion is obtained for low values of the operational reactivity margin, i.e. when most of the rods are withdrawn from the reactor. The worst case of CPS voiding is a loss of coolant above the reactor core, producing a draining of all CPS channels. The water level in the channels decrease by gravitational forces, thus the process is not very fast. Due to different types of control rods and different control rod insertion depths, the flow velocities differ significantly in the different channels. Thus, the reactivity insertion is non-uniform in the CPS channels. In addition, the reactivity insertion is not very fast. The fastest possible complete voiding of CPS channels in the core occurs in about 10 seconds, while the slowest voiding occurs in about 50 seconds. Multiple scram signal are generated and if the reactor shutdown function is available on demand, no safety limits are exceeded.

Assessments of reactivity initiated accidents show that the Ignalina NPP is adequately protected against this type of accidents. The fuel channels remain adequately cooled both in cases where all systems operate as designed, and when additional equipment or component failures are postulated to coincide with the initiating event. Multiple signals are available either to reduce the reactor power or to shut down the reactor. The main issue is detector coverage, which is shown to be adequate for central and peripheral, single and multiple control rod withdrawals as well as CPS voiding accidents. The single failure criterion is applied through the loss of signals due to the loss of one detector group of six. The loss of a group of detectors does not significantly impact detector coverage because there are many redundant signals based on the remaining detectors, i.e. acceptable consequences are obtained whether or not these signals are available.

11.5 ANTICIPATED TRANSIENTS WITHOUT SCRAM

Anticipated Transients Without Scram (ATWS) are accident sequences involving a non-LOCA transient of moderate frequency (about 1.0/ year, e.g. turbine trip) or infrequent incidents (about $3{\cdot}10^{-2}$/year, e.g. reactivity events) and failure of automatic reactor scram. Major objectives of ATWS analyses are to demonstrate that the pressure boundary of the reactor coolant will not fail, the pressure suppression system will not fail, safe long term shutdown is reached and heat removal capacity is sufficient. The ATWS are commonly considered as design basis accidents or as accidents to be dealt with in the licensing process for Western reactors. For RBMK reactors ATWS are not design basis accidents and no previous analyses of such accidents were performed. The ATWS studies in the Ignalina SAR are the first of the kind for RBMK reactors. These analyses have a different purpose from DBA studies. The purpose of the ATWS studies in this project is to identify the need for possible future design modifications to the shutdown system, to determine the minimum time available for accident mitigation and to make a step towards developing accident management measures and procedures. The ATWS scenario can lead to unacceptable consequences. The failure probability of the overall scram system is the major concern at Ignalina NPP. According to the assessment the failure probability may be $4{\cdot}10^{-4}$ per demand or higher. The magnitude of this failure probability highlights the importance of the ATWS issue for Ignalina NPP. Four different Anticipated Transients Without Scram were addressed in SAR:

- maximum reactivity insertion by continuous single rod withdrawal at full power and during start-up,
- partial loss of flow due to MCP failure,
- loss of main heat sink (loss of both turbines with loss of condenser vacuum),
- loss of preferred AC power.

The analyses were carried out using the following initial and boundary assumptions: All systems that affect the reactor power and are not active during normal reactor operation are assumed unavailable. This applies to the 24 FASS rods, 24 LSR rods and the CPS operation modes BAZ and AZ-1. All systems active during normal operation remain functional during the accident as long as they are not affected by the consequences of the accident, e.g. LAC system, pressure and level controllers. Systems that do not affect the reactor power and are poised to be activated by the accident, e.g. relief valves, ECCS, are assumed available. The base-case simulations is performed until one of the following conditions is reached:

- the plant achieves a new steady state,
- acceptance criteria are violated,
- conditions are encountered in the simulation that cannot be reliably described by the available mathematical models.

Results provided by the base-case simulation include list of all available scram actuation and power set-back signals as well as the minimum time available for accident mitigation. The issues addressed in the analysis include:

- the ability of fuel channels to withstand a local power rise due to rod withdrawal and sensitivity of protective systems for this postulated events,
- thermal-hydraulic stability under reduced flow rates and high power levels,
- rate of pressure rise when steam generation rates exceed steam relief rates,
- time necessary for reaching critical values of safety parameters,
- plausibility of effective operator intervention.

The following conclusions were drawn regarding the ATWS sequences for the Ignalina NPP. Continuous withdrawal of one control rod with ATWS from full power are controlled by local automatic control/protection system. Total reactor power is kept nearly constant, while the maximum local power excursion at full power was 175 %. Detector coverage is such that the reactor setback or trip signals are generated within about 10 and 16 seconds of the start of rod motion for star-up and full power levels, respectively. Redundant trip signals are generated within a short time span, so single failure of trip signal are inconsequential. At powers below the normal operating range acceptance criteria in fuel channels are not violated.

Failure of one MCP is inconsequential because the flow from the operating pumps compensates for the trip of 1 out of 3 MCPs in one circulation loop. The local automatic control/protection system maintains the plant within a safe range of operation. Flow instability is not encountered even when the power is not reduced. The acceptance criteria for fuel and pressure boundary are met. This conclusion applies to the whole normal operation range from 1000 MW to 4200 MW. There is adequate time for operator action.

During reactor operation at full power a turbine trip with loss of main heat sink leads to failure of the pressure boundary within about 3.5 minutes (likely between core outlet and MCP suction header) because steam production exceeds the steam removal capacity of 2 SDV-A and 12 MSRVs. Total reactor power is maintained nearly constant by local automatic control/protection system. However, eight different power reduction signals were identified before pressure boundary failure. Effective operator intervention, i.e. manual scram is possible. If this ATWS were to occur at some steady state operation power

level higher than 2650 MW, the sequence of event will remain the same, only there will be more time available for operator intervention. The relief capacity is sufficient at reactor power level below 2650 MW, so the manual scram is a highly probable terminator of transient, since long delay can be tolerated.
Loss of preferred AC power results in constant reactor power due to functioning of the local automatic control/protection system. Due to costdown of the MCPs and loss of main feedwater steam production rises considerably and will be in excess of the steam removal capacity of the 14 discharge valves (2 SDV-A and 12 MSRVs). Flow instability could occur after 10 seconds and dangerous cladding and pressure tube wall temperatures after 40 seconds. The acceptance criterion for main coolant circuit pressure of 10.4 MPa is violated after about 1 minute. Multiple pressure tube ruptures are likely to occur. Although the operator may be able to manually insert control rods, this may not prevent a pressure boundary failure.

The results of ATWS studies demonstrate the lack of inherent safety features in the RBMK design. The power is not reduced by means of inherent physical processes such as steam generation. The reactivity loss due to fuel temperature rise (Doppler effect) is not effective enough to prevent major damage of the core. The local automatic control/protection system assumed available under analysis rules turns out to be detrimental in some cases since it tries to maintain the power level.

The apparent lack of the effective inherent safety features in RBMK reactors leads to one high priority recommendation, that a second fast acting, independent and fully diverse reactor shutdown system needs to be installed. The second shutdown system has to be designed to ensure its functionality at conditions prevailing during and after the accident, and to provide safe long term reactor shutdown. Development of second reactor shutdown system is under progress, but its implementation requires 3-4 years. Compensatory measures which have the potential to reduce the overall risk are implemented at Ignalina NPP until a second shutdown system is in place.

11.6 POTENTIAL INITIATORS OF MULTIPLE PRESSURE TUBE RUPTURE

The SAR project evaluated the capability of the Ignalina NPP reactor cavity structures to withstand coolant discharges that might be encountered during accidents which involve fuel channels ruptures. The range of coolant discharge conditions from the ruptured pressure tubes into graphite moderator stack has been quantified. The consequences of various coolant discharges into the hot and rather confined reactor stack in terms of peak pressures within the reactor cavity have been evaluated. The venting capacity of the reactor cavity over-pressure protection system is expressed in terms of the number of fuel channels that can rupture simultaneously or sequentially without damaging the reactor by exceeding the 214 kPa peak pressure load on the upper lid of the reactor cavity. The capacity of the existing reactor cavity over-pressure protection system introduced at the end of 1996 is 9 ± 4 simultaneous or closely-spaced-in-time channel ruptures at full system pressure. If the above channel ruptures occur at reduced system pressures, the discharge of the coolant will be smaller, and hence capacity of the reactor cavity over-pressure protection system to relieve this discharged coolant will be higher. This capacity rises to 25 ± 12 at 4 MPa.

The range of uncertainty associated with the analysis is quite large, i.e. about 50 %. The primary reasons for this are the scenario-specific variability of the break flow, and the uncertain characteristics of the flow path through the reactor cavity. In addition, there are uncertainties concerning the deformation of the graphite stack after channel rupture, uncertainties regarding the graphite surface area that will be in contact with the discharged water and the stored heat that will evaporate this discharged water. If the stack deformation remains small, the discharge water-steam mixture flow rate will be rather “isotropic” and evaporation could be almost complete, i.e. 100 %. If the stack deformation creates vertical free “channels”, the discharged water will be quickly forced out of the graphite stack and additional evaporation will be small. Therefore the range of the amount of steam generated from the discharge into reactor cavity is quite wide - from about 30 % to almost 100 %. Improved analytical methods which might decrease the noted uncertainties are not available in the short term. Since the consequences of the multiple pressure tube rupture can be catastrophic, it is necessary to continue investigations related to this issue in order to better understand physical phenomena which can lead to multiple pressure tube ruptures and to develop more accurate prediction methods for the reactor cavity over-pressure issue.

11.7 PROBABILISTIC SAFETY ASSESSMENT

A probabilistic safety assessment of the Ignalina NPP was performed in conjunction with the Barselina project [63]. The project is a multilateral co-operative study conducted by Lithuanian, Russian and Swedish experts. The Barselina project, four phases of which have been completed, was initiated in the summer of 1991. Its long term objective is to establish common perspectives and unified bases for assessing severe accident risk and establishing requirements for remedial measures for RBMK reactors. In this project the Swedish BWR Barseback is being used as a reference plant and the RBMK-1500 at the Ignalina NPP is being used as the applicant plant.

The Barselina project has been split into four phases. Phase 1 included familiarization with and analysis of a limiting number of safety systems and one single initiating event. It ran from October 1991 to the end of March 1992. Phase 2 included analysis of the principal components for all important safety systems and extension to several initiating events, but excluding external events and with limited treatment of human factors. This phase ran from April 1992 to February 1993. During phase 3, from March, 1993 to June, 1994, a full scope Probabilistic Safety Assessment (PSA) model of the Ignalina unit 2 was developed in order to identify the reduction of risk that can be achieved with possible safety improvements. The probabilistic methodology was applied on a plant specific basis for a channel type reactor of RBMK design. To increase the realism of the risk model a set of deterministic analyses were performed and plant-specific data base were developed and used. A general concept for analyzing this type of reactors was developed. During phase 4, July 1994 to September 1996, the Ignalina PSA model was further developed, taking into account plant changes, improved modeling methods and extended plant information concerning dependencies (area events, dynamic effects, electrical and signal dependencies). The PSA model is also updated to reflect the “as built” plant. The phase 4 PSA work used insights from the peer review performed by Battelle Pacific Northwest Laboratories on the phase 3 work. Another review is planned for phase 4.

The scope of the PSA study in the Barselina project is as follows. The source of radioactivity is the reactor core. The PSA also is based only on full power operation. Internal initiating events such as transients, LOCAs and Common Cause Initiators as well as internal hazards, such as fire, flooding and missiles are taken into consideration. Final consequence of the accident is core damage, equal to level 1 PSA. During the work, however the core damage states have been defined in such a way, that the results can be used partly as level 2 results - the damage stages represent 4 classes of environmental impact.

The hazard states in the core are evaluated on the basis of the development of accident event sequences resulting in conditions of either “safe conditions”, “violation”, “reactor core damage” and “severe accident”. The plant is considered to have met the “safe condition” requirements when temperature limits are not exceeded or exceeded in no more then 3 fuel channels, but cladding temperature of 800 ^oC are not exceeded in any channels. Safe operation limits are listed in Table 6.2. If the fuel cladding integrity is breached in more than three channels due to cladding defects and damages or because the cladding temperature limit of 800 ^oC is exceeded, the state is classified as “violation”. The “violation” category can be regarded as belonging to relatively mild consequences. The reactor core damage category is characterized by severe accidental conditions caused by significant deviation from the design scenario which lead to cladding temperatures above 800 ^oC in no less than 3 and no more than 90 fuel channels of the reactor. Such accidents do not lead to loss of core structural integrity and this category can been looked upon as resulting in medium severity consequences. The “severe” accident category is characterized by severe accidental conditions caused by significant deviation from the design scenario and accompanied by the rupture at high pressure of more than 3 and less then 9 pressure tubes before the

reconstruction of reactor cavity over-pressure protection system and 9 pressure tubes after reconstruction. Such an event can be accompanied by fuel melting or fuel damage in more than 90 fuel channels. This is the most severe consequence.

The accident sequence model for reactor cooling is a phased mission model divided into three time period:

- Short term cooling 0-2 minutes.
- Intermediate term cooling 2 minutes - 1 hour.
- Long term cooling 1 hour - 24 hours.

The phase 4 results indicate that the overall core damage frequency is lower than the phase 3 results. The reason for this is the implementation of plant safety improvement features, and improved analytical procedures which eliminated unnecessary conservatism’s. The new results are also balanced by the improvements in the modeling of the CPS and ACS systems. The quantitative results obtained are based partly on plant specific data and partly on generic data. The results are not intended to show absolute risk levels, but to give a risk topography and to serve as a basis for identifying risk dominant features and systems design aspects and hence serve as a basis for safety improvement.

The general results show a probability of the “violation” end state to be in the order of 10^{-2} per reactor year. This probability is dominated by single channel blockage events. The assessment of probability value is based on operational data. To date 3 such cases have occurred in the RBMK reactors. However, the design of control isolation valves has been changed, which should have a positive impact on the initiating event probability. The “damage” and “accident” end states show probabilities together on the order of 10^{-5} per reactor year, the same range as is expected for “core damage” as defined for Western reactors.

The risk typography is shown schematically in Fig. 11.1. The characteristic of the risk topography is that for “damage” and “accident” end states transients dominate the risk rather than loss of coolant accidents. Transients contribute more than half of the total frequency. Furthermore it is the long term failure to cool the core that produces the dominating contributions, Fig. 11.2. The distribution of risk between short term, intermediate term and long term contribution shows that most of the sequences lead to damage or accident only in the long term. Only the core blockage sequences lead to damage in the short term. This demonstrates both the high redundancy of the front line engineered safety systems and the “forgiving” features of the reactor. Low power density and a high heat capacity enables the reactor to survive at least a one hour total loss of electrical power without core damage. In the long term, support functions

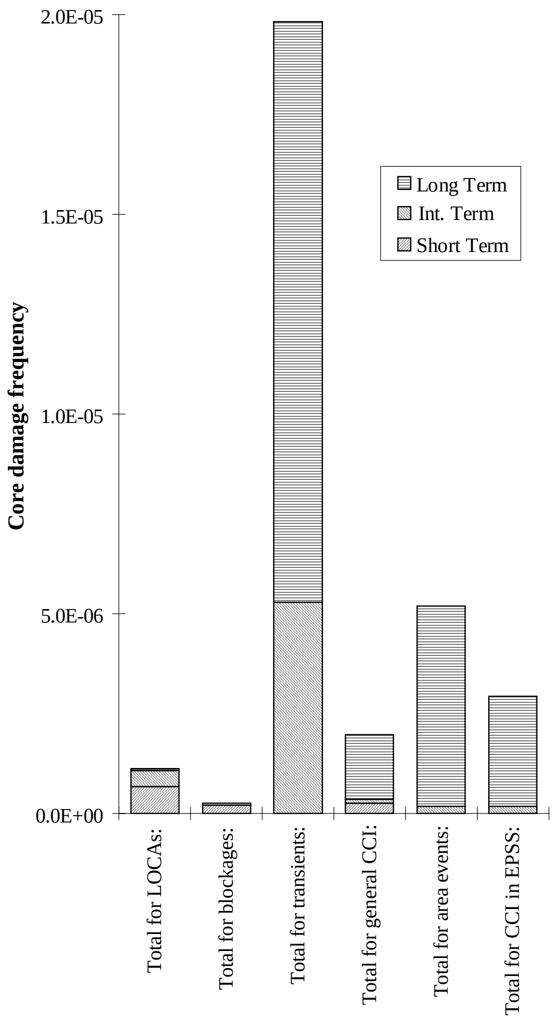

Fig. 11.1 Damage and accident contributors in different initiating event classes [63]

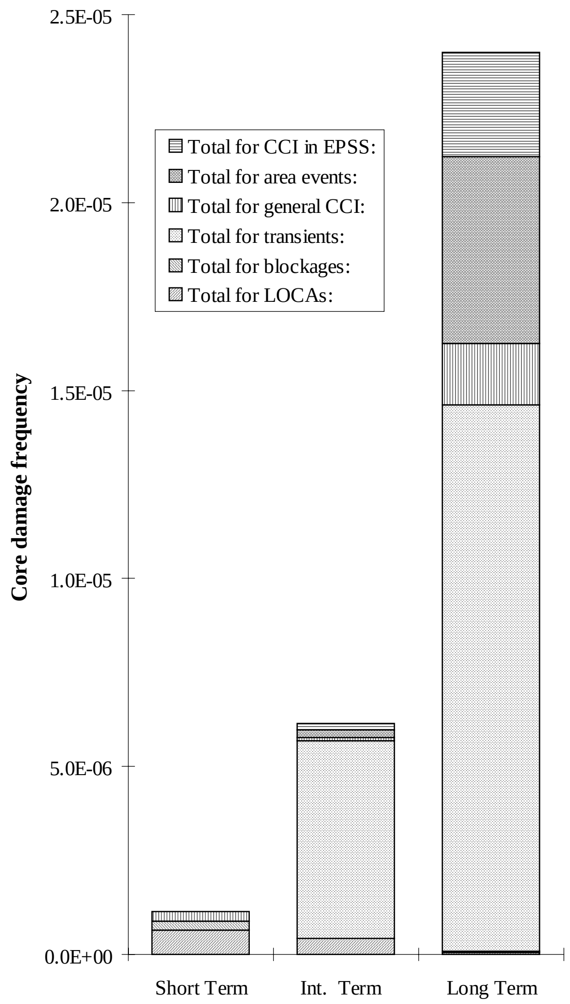

Fig. 11.2 Damage and accident contributors in short, intermediate and long term cooling [63]

become more important and their failures become thedominating contributions. The results indicate that a long term lack of coolant leads to severe environmental consequences because the core damage is assumed to occur at high reactor pressure. Human factors also contribute significantly to the core damage frequency. However, the development and introduction of event-based Emergency Operating Procedures is still not accounted for in the phase 4 results.

Since January 1996 a newly formed internal PSA group at Ignalina NPP is responsible for the probabilistic safety assessment. The experience and information from the Barselina PSA phases provides valuable information to other projects, e.g., the in In-Depth Safety Assessment of the Ignalina NPP project, for development of the event-based Emergency Operating Procedures and Reliability and Maintenance Management System [63].