

# Comparison of the N Reactor and Ignalina Unit No. 2 Level 1 Probabilistic Safety Assessments

Prepared for the U.S. Department of Energy  
Office of Environmental Restoration and  
Waste Management



**Westinghouse**  
**Hanford Company** Richland, Washington

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## COMPARISON OF THE N REACTOR AND IGNALINA UNIT NO. 2 LEVEL 1 PROBABILISTIC SAFETY ASSESSMENTS

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### ABSTRACT

A multilateral team recently completed a full-scope Level 1 Probabilistic Safety Assessment (PSA) on the Ignalina Unit No. 2 reactor plant in Lithuania. This allows comparison of results to those of the PSA for the U.S. Department of Energy's (DOE) N Reactor. The N Reactor, although unique as a Western design, has similarities to Eastern European and Soviet graphite block reactors.

### I. INTRODUCTION

Differences exist in the scope and boundaries of the N Reactor and Ignalina studies, so certain comparisons could be misleading. Additionally, PSA methods were developed first for Western reactor systems. So, it is difficult to know how much cultural differences affecting facility operation, management, and environment, as well as infrastructure, are reflected in the PSA results. Therefore, it is not the intent of this paper to draw conclusions, but rather to identify parallels and contrasts. Comparisons include differences in the plant design, the PSA methods and assumptions, and the PSA results and insights.

Both the N Reactor and Ignalina reactors are light-water, multichannel, graphite-block core designs. Ignalina Unit No. 2 is a commercial Reactor Bolshoi Moschnosti Kanalnyi (RBMK) reactor rated at 1,500 MWe and 4,200 MWt. It operates at about 1,300 MWe and supplies Lithuania with about 90% of its electrical power. The N Reactor is a shutdown DOE reactor that was used for the production of weapons grade plutonium, but it also co-generated 860 MW of electrical power and was rated at 4,000 MWt.

Although somewhat similar, these reactors have important design differences. For example, the Ignalina reactor operates in the boiling mode while the N Reactor was a pressurized water reactor. Also, the process tubes run vertically in the Ignalina reactor and hold commercial ceramic fuel, but run horizontally in the N Reactor and hold metallic fuel.

The Ignalina PSA<sup>1</sup> is the first complete Level 1 PSA performed on an RBMK reactor. The N Reactor is the only modern Western-style graphite-moderated reactor with a power-generating history. It was the first DOE production reactor for which a full-scope PSA was performed.

The Ignalina reactor PSA was performed by the Barselina Project, which is a cooperative effort among Lithuania, Russia, and Sweden. The PSA work was finished and the results published in the summer of 1994. The results are being used to identify plant-specific improvements in the system design and operating procedures.

The Barselina PSA work was formally reviewed by Battelle Pacific Northwest Laboratory (at the DOE Hanford Site where the N Reactor is located). Westinghouse Hanford Company (WHC), the last operating contractor of the N Reactor, provided technical assistance and insights. The review was financed by the U.S. Agency for International Development and the DOE.

Westinghouse Hanford Company first published the results of the N Reactor Level 1 PSA<sup>2</sup> in August 1988. A full-scope Level 3 and external events analysis were subsequently performed. The Level 3 assessment,<sup>3</sup>

published in April 1990, was a joint effort by DOE contractors. Methods (NUREG-1150\*) developed by Sandia National Laboratories and WHC were employed.

## II. CONCLUSIONS AND INSIGHTS

The total core damage frequencies for Ignalina and N Reactor are  $6.4E-05$  and  $1.5E-05$  per year, respectively. The scope of the Ignalina study includes sequences (i.e., area events such as internal fire and flood) not included until a later study for the N Reactor. The Ignalina total core damage frequency is a little higher but comparable to N Reactor as well as commercial reactors.

Uniquely, both studies include partial core damage states. This was more important at N Reactor than Ignalina, even though both reactors are multichannel. At N Reactor, 42% of the core damage frequency applies to less than one-sixteenth of the core. This is due partly to the horizontal versus vertical orientation of the process tubes. It is also due to the fact that when more than three process tubes are breached at high pressure in the Ignalina reactor, all tubes are eventually sheared off. This is caused by the displacement of the reactor cover.

For both reactors, no single fix would dramatically decrease risk. The Ignalina study reports that modest risk improvement measures in 14 areas would yield a decrease of a factor of 3; at N Reactor, the same was true. At its shutdown, improvements were occurring in a number of areas.

According to the Ignalina PSA, although no single failure is dominating, a significant component failure sensitivity exists: the group distribution header (GDH) blockage event. Certain catastrophic valve failures in the GDH can lead to core damage before manual emergency core cooling system (ECCS) actions are initiated. These valves are considered to be more reliable than comparable generic Western commercial valves. If less optimistic failure frequencies are used, core damage frequencies in the  $1E-3$ /yr range are possible.

The relative uniqueness of these plants compared to Western commercial reactors challenged the PSAs. For example: (1) both studies used multiple approaches to obtain comprehensive initiating event lists; (2) the Ignalina study incorporated an area events analysis as a common-cause failure mechanism; and (3) the N Reactor, because of its age, included a component aging analysis.

Many accident sequence failures reported in the Ignalina study include failure of adequate pressure relief, not critical to N Reactor (not a boiling water reactor). The ECCS failures were important at N Reactor as the ECCS is not as diverse as at Ignalina.

For Ignalina, only a Level 1 PSA is complete, although a safety analysis is ongoing. So, the important issue of health affects from the consequence of core damage is not yet addressed. Because Ignalina lacks containment or a confinement system to mitigate fission products, severe consequences are likely. Also, no probabilistic external events analysis (seismic, fire, wind, etc.) has yet been done. For N Reactor, external events contributed more to risk than did internal events. Like N Reactor, Ignalina displays some location-dependent weaknesses (e.g., certain control buses) and an unknown seismic fragility potential.

As a final note, the scopes of both PSAs were limited to only full-power events. Traditionally, it has been assumed in PSAs that the dominant risk contributors are associated with full power. This assumption is being challenged as PSAs are now including nonfull-power events. The Chernobyl accident (an RBMK) occurred at very low power. The Ignalina PSA staff, however, has determined that a Chernobyl-like accident, at least, is not credible given the changes in design and operating culture.

## III. COMPARISON

The following comparisons are made in three different areas: (1) plant design, (2) PSA methods, and (3) PSA results.

### A. Plant Design Comparison

1. N Reactor. The N Reactor is a pressurized water multichannel reactor. Zircaloy-clad, metallic uranium fuel is arrayed in 1,003 horizontal pressure tubes through which water coolant flows. The zircaloy tubes are supported by graphite moderator blocks. Twelve steam generators provide heat exchange between the primary and secondary systems.

Two diverse protection systems provide for rapid reactor shutdown: (1) a system of 84 horizontal boron carbide ( $B_4C$ ) control rods, and (2) a system of  $B_4C$  balls held in hoppers above the core. At reactor shutdown, up to 16 dump condensers are available to accept the heat load shed by the power-exporting turbine generators. The primary system experiences a severe transient but ECCS is never actuated.

The ECCS is once-through flow actuated only in an emergency.

The ECCS has independent, diesel-driven pumps that inject coolant into the fuel tubes through 16 separately valved headers. Four ECCS inlet valves and four exit valves open to allow flow. It is possible to lose cooling to a single tube or to a header without losing cooling to the remainder of the core, thereby creating partial core damage.

The graphite and shield cooling system (GSCS) removes heat from the graphite moderator during normal operation. The GSCS heat removal capacity is sufficient to prevent core damage without additional assistance from the ECCS after the reactor has been shut down and cooled for about 2 hours. A high-pressure injection system provides about 2,730 L/min of makeup during normal operation.

The N Reactor does not have a containment building, but is designed with a confinement system. Because of the wide exclusion zone (8.9 km), the N Reactor is able to conform to commercial reactor siting criteria using a confinement system. The confinement system works by (1) releasing the initial steam released from a pipe failure into the environs; and (2) controlled, low-pressure release of steam, air, and gases through charcoal and high-efficiency filters, which minimize radioactive release. The system works well because of the virtual absence of fission products in the initial steam release. This is due, in part, to the relatively low-specific heat and slow heatup rate of the fuel.

2. Ignalina. Ignalina is a boiling water reactor. The graphite stack consists of blocks arranged in the form of columns allowing some 2,000 vertical channels. These channels provide locations for the pressure tubes containing the uranium fuel, control rods, and instrumentation. The primary system consists of two parallel loops, arranged symmetrically with respect to the vertical plane of the reactor. There are two steam drum separators in each loop. They extract saturated steam from the incoming steam-water mixture given off from the reactor vessel.

There are two types of control rods for reactor shutdown: 211 normal rods and 24 fast scram (FASS) rods. The normal rods operate in water and fall into the core when they are released. The FASS rods operate in nitrogen-filled channels and can be dropped into the core or lowered in by electric motors.

At shutdown and other transient conditions, the steam dump system maintains allowable pressure in the primary circuit. This system includes 10 fast-acting valves (BRU-K and BRU-B) and 12 main steam relief valves. Also at shutdown, auxiliary and emergency feedwater pumps are put into service because the main feedwater pump capacity is too high. The ECCS pumps are started during this time.

The ECCS consists of two subsystems: the pressurized tanks for short-term cooling and the pump system for intermediate and long-term cooling. The short-term cooling mission is 2 minutes, enough to compensate for makeup lost in a large pipe break. The ECCS pumps take makeup from the accident localization pools. Pool makeup is from the emergency deaerator feeding system and from condensed steam.

The Ignalina reactor does not have a containment building, but does have an accident localization (ACL) system. The solid compact cubicles are reinforced, leak-tight rooms. Most of the primary pipes, with the exception of the steam drum separators and some connected piping, are inside these solid compact cubicles. Two ACL towers, which are concrete with an internal leak-tight liner, rise five stories above the suppression pools. If a pipe breaks in a primary pipe, the discharged steam is directed to the ACL towers where it is condensed.

## B. PSA Method Comparison

Both the N Reactor and Ignalina studies are comprehensive and incorporate state-of-the-art PSA methods: complex event and fault trees. The Ignalina study is still ongoing. For example, the reliability of the reactivity control system is being reexamined to incorporate common cause. In contrast, a full-scope Level 3 and external events analysis have been completed on the N Reactor. Important comparisons include: initiating events development, treatment area events, component failure data, and establishment of core damage states.

As is typically done, the scope of both PSAs addresses only the full-power operating mode. Recent PSAs have given more attention to other operating modes. Significant for the Ignalina PSA was that the Chernobyl reactor (also an RBMK) accident happened at very low power. The Ignalina PSA staff concluded that a Chernobyl-like accident is no longer credible given the changes in reactor operating culture and design. Therefore, the important risk contributors should be in the full-power operating mode.

Identification of initiating events was important in both studies because both reactors are significantly different in design from Western commercial reactors. Accordingly, both lack information generated by PSAs of similar designs. Both employed a comprehensive, multifunctional approach to initiating event identification.

In the Ignalina study, a detailed area event analysis was performed for internal fire, steam, and missiles, as well as other internal energetic events. The N Reactor Level 3 PSA showed certain area events to be risk dominant. Both reactors suffer from lack of separation between certain critical loads and/or cabling.

Both reactor studies initiated plant-specific component failure database development. The N Reactor is older and personnel kept good historical records, but generic data were still augmented. For N Reactor, a component aging analysis was included. The Ignalina study had to rely even more heavily on generic data.

For both plants, when plant-specific data were compared to generic data, there was good agreement. However, some estimated component failure frequencies were somewhat higher (e.g., heat exchanger plugging at Ignalina, for example) than generic values. This demonstrates the value of developing plant-specific databases.

An important similarity between the studies is the treatment of accident end states. Typically, Level 1 PSAs treat core damage as a single end state and do not discriminate between classes of damages. This is not true for these reactors as both have multichannel designs. The process tubes, which contain the fuel, are separated by a solid graphite moderator, and so several distinct core damage end states are possible. For the Ignalina PSA, three different end states were defined:

- Violation—Design limits exceeded in less than three channels, or limited exceedance of design limits
- Damage—Significant deviation from design scenarios in 3 to 90 channels, but with limited number of ruptured process tubes
- Accident—Severe accident conditions, with rupture of several pressure tubes and loss of reactor cavity integrity.

The N Reactor PSA also used several core damage end states which were based on the fraction of the core

affected and decay heat load. For example, if ECCS cooling flow from a single inlet riser (1 of 16) was interrupted, it would affect cooling to about one-sixteenth of the reactor core. In another case, GSCS can cool about two-thirds of the core if ECCS fails.

The Ignalina results (after discounting the 'violation' end state) are dominated by the 'accident' end state. According to the N Reactor PSA, a large fraction of the core damage frequency is related to the one-sixteenth core damage end state.

A final important difference is that the results of the Ignalina PSA are already being implemented, whereas the N Reactor is now shut down. Important changes to the N Reactor were initiated because of insights gained in the N Reactor PSA, but the final benefit will never be known because the plant was shut down in 1988 and terminated in 1991. Important hardware and procedure changes are ongoing at the Ignalina reactor.

### C. PSA Results Comparison

The PSA results can be compared by examining the corresponding accident frequencies and important analysis results.

The total core damage frequency is reported to be  $6.4E-05/\text{yr}$  for N Reactor and  $1.5E-04/\text{yr}$  for Ignalina. These values are mean point values and do not reflect any uncertainty distribution. The major contribution to the N Reactor core damage frequency is from loss-of-coolant accident (LOCA) events as opposed to transient events for Ignalina.

Both studies report the frequencies of individual dominant accident sequences. However, the N Reactor PSA grouped accident sequences according to core damage bins, whereas Ignalina accident sequence frequencies are reported according to initiator categories. Frequencies from four overall initiator categories are shown in Table 1. These categories are roughly comparable.

The accident frequency related to large LOCAs is much higher for N Reactor than for Ignalina. No medium LOCA is reported for N Reactor. So, if the medium and large Ignalina LOCA frequencies are combined, the frequencies are more nearly the same. The frequency for N Reactor large LOCAs would still be somewhat higher. The Ignalina reactor is apparently more forgiving to large LOCAs because there is more flexibility in getting water to the core.

Table 1. Accident Sequence Frequencies.

Accident sequences	N Reactor		Ignalina	
	IE	Total	IE	Total
Large LOCA	6.2E-04	4.68E-05	1.4E-04	3.9E-07
Medium LOCA	—	—	5.3E-03	1.1E-05
Small LOCA	2.3E-01	1.60E-05	1.3E-02	7.4E-07
Transients	—	1.71E-06	—	1.3E-04
Area events	3.1E-02	1.60E-07	4.5E-03	6.1E-06
Total	—	6.4E-05	—	1.5E-04

The accident frequency related to small LOCAs is much higher for N Reactor than for Ignalina. Both N Reactor and Ignalina have about the same amount although much more small piping than commercial vessel reactors. However, the small LOCA frequency reported for N Reactor is two orders of magnitude higher than that reported for Ignalina. Much of this difference is due to the contribution of spuriously opening relief valves. The remaining difference is from the difference in generic pipe failure frequency used for N Reactor versus the RBMK-specific data used for Ignalina.

Accident frequencies from transient events dominate the Ignalina results and are much higher than for N Reactor. Total transient initiator frequencies are shown in Table 1. Loss of offsite power, loss of certain control buses, and GDH blockage were key Ignalina transient initiators. Loss of offsite power was not a dominant accident sequence for N Reactor. At N Reactor, equipment was supplied from two separate buses, one from the onsite turbine generator run by reactor or boiler steam and one from offsite power.

Accident frequencies from area events were important contributors to the total core damage frequency for Ignalina. The N Reactor Level 1 PSA reports only a minor contribution; this is misleading. The area event analysis for the N Reactor was limited. Subsequent external events showed fires at certain locations to be damaging to key primary cooling and ECCS signal wiring.

For N Reactor, about 42% of the accident sequences leading to core damage affect only about one-sixteenth of the core. These accidents contain failures related to 1 of the 16 inlet risers. This is roughly

equivalent in the Ignalina reactor to a GDH blockage. However, in the Ignalina reactor this event can propagate to total core damage. Pressure from failed process tubes related to a blocked GDH can lift the head of the reactor vessel, failing all process tubes. The N Reactor has no such vessel or related lid.

Figures 1 and 2 show the accident sequence contribution according to time regime (time after shutdown). The time regimes are different and specific to each reactor. However, Figure 2 does show that the majority of accidents for Ignalina happen in the long term, whereas it is about evenly divided for N Reactor. For Ignalina, this demonstrates the redundancy of the front line systems and the low-power density and high-heat capacity that enable the reactor to resist a total loss of electrical power for at least 1 hour. This could also be true for N Reactor, except loss of power is less of an issue because ECCS pumps are individually powered diesel pumps.

The importance of individual basic events using the risk contribution measures (Fussler Vesely) are shown in Table 2.

For N Reactor, three events related to the delivery of ECCS water to the core contribute considerably to the core damage frequency. Two events are related to ECCS inlet valve (V-3s): failure to open on demand. The other event is failure of the ECCS outlet valves to open. These valves must work for most LOCAs and all transients. The water pathways to the Ignalina core are more diverse so that one particular set of ECCS valves is not so important.

Ignalina has 12 main steam relief valves (MSRV). If two or more MSRVs fail to reclose after a scram, it

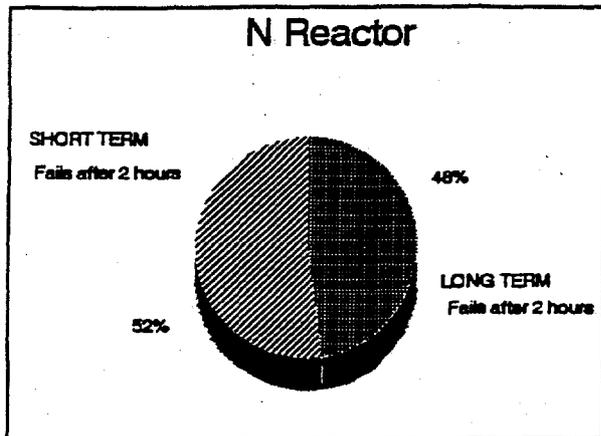


Figure 1. Accident Sequence Contribution for N Reactor.

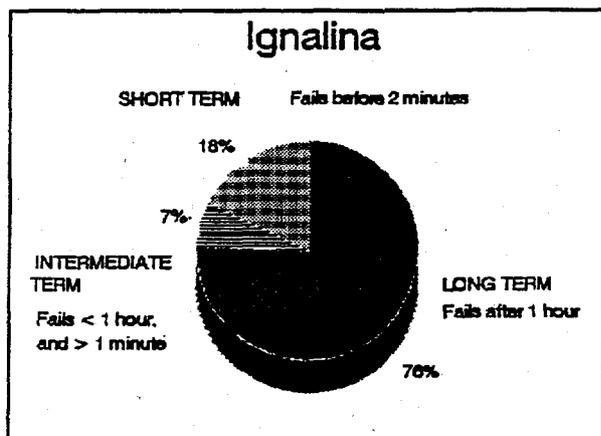


Figure 2. Accident Sequence Contribution for Ignalina.

is necessary to transfer from drum separator makeup to GDH injection. Several failure combinations exist. These failures, combined with operator error during transition to long-term cooling in the GDH injection mode, lead to several failure paths (cutsets).

Among the top events in both lists are a number of human errors. Both reactors require considerable operator interaction in upset situations compared to Western commercial reactors. In the case of N Reactor, the designs of certain reset switches and interlocks were reexamined. At Ignalina, the contribution of human errors is also being examined.

At the Ignalina reactor, certain control buses were found to be very important because they fed critical

loads. The PSA recommends that certain control power loads be reallocated. This would help reduce the contribution of corresponding component failure to core damage.

After the first 10 most important events, individual event contributions begin to approach 1% to the total for both reactors. No one failure significantly dominates. In each case, if the first 10 failures were eliminated, the contributions of the next 10 failure events would look about the same.

#### IV. ACKNOWLEDGEMENTS

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Table 2. Important Events.

N Reactor		Ignalina	
Event description and frequency contribution		Event description and frequency contribution	
AOOMOD	37.4	MSRV RAXOSJX_VS2BA (12 individual events)	41.0
Failure of sufficient number of ECCS inlet valves to open on demand.		A set of main steam relief valves fail to reclose mechanically (alone each event contributes 3.4%).	
E99MOD	18.3	MSRV RAXOSOX_VE1YA (12 individual events)	23.8
Loss of thermal flywheel steam to onsite turbine generator supporting B bus.		A set of main steam relief valves fails to open electrically (alone each event contributes about 2.0%)	
AAGV4CMA	12.8	FE V3.1_GDH-TRANS	18.5
Common-cause failure of all ECCS exit valves to open on demand.		Human error during transition from intermediate to long-term cooling.	
ADPHL1-G	5.1	AFWSRL10S05_VC2AA	6.7
Failure of high-lift ECCS diesel-driven pump to run.		Check valve between auxiliary feed water system and drum separators 1 and 2 fails to open.	
AAGV3CMA	5.0	AFWSRL20S05_VC2AA	6.7
Common-cause failure of all ECCS inlet valves to open on demand.		Check valve between auxiliary feed water system and drum separators 3 and 4 fails to open.	
XOP3H33V	4.8	FE U3 DS-MKP_LOW	4.4
Manual failure to restore ECCS interlock to normal during reactor startup sequence.		Low level in drum separators due to human flow regulation errors.	
MOPLP13Q	1.4	&CONDENSATE_SYS	3.4
Failure to manually start low-pressure injection water as a remedial action.		Failure of condensate system to condensate steam after scram.	
GSW2H33Q	1.4	FE U3 GDH-TRANSIT	3.0
Inadvertent trip to ECCS from failure to bypass switch 2H33 after a scram.		No manual action to start GDH injection after drum separator makeup failure.	
FOPHP1PU	1.4	EPSS2HZ19_SB1YA	2.0
Miscalibration of the high-pressure injection pump flow instrumentation.		Busbar HZ19 (<20 kV) fails.	
FCBHPASK	1.3	EPSS2HZ18_SB1YA	1.9
High-pressure injection pump power breaker fails to remain open.		Busbar HZ18 (<20 kV) fails.	

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