ASSESSMENT OF RADIOLOGICAL CONSEQUENCE HAZARDS FROM EXISTING NUCLEAR INSTALLATIONS

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Hungarian nuclear safety regulations require that external events must be considered in the site investigation and evaluation, and in the definition of design basis for nuclear installations. Radiological consequence hazards attributable to existing nuclear installations located an important component of man-made external hazards. The purpose of this study was the assessment of such hazards in the close vicinity of the Paks Nuclear Power Plant in Hungary by making use of the existing probabilistic safety assessment (PSA) for the plant. Since level 3 PSA that could support best the objectives of the study was not available, the level 2 PSA results and the underlying risk contributors were examined and analyzed to select a range of severe accidents of the four reactors and the associated spent fuel pools of the plant that should be considered in characterizing off-site radiological consequences. Detailed consequence analysis was performed for these accidents. The radiological effects of the accidents were determined by analyzing the environmental spread of radioactive isotopes released from the plant and determining health consequences in terms of effective dose. Calculations were performed to assess (1) the radiological impacts on personnel working open air at the area investigated, and (2) the impacts on habitability of service areas of a fictitious nuclear installation within this area. Subsequently, a bounding accident involving large releases of radioactivity from multiple sources was selected as an ultimate reference event to describe radiological effects. The ultimate reference event is a severe accident of the reactor operating at full power in unit 4 (the closest one to the area investigated), combined with a severe accident of the spent fuel pool in the same unit. This complex accident is induced by a strong seismic motion shaking the site, and leading to large releases of radioactivity outside the plant. The radiological consequences of the reference event were found significant, which confirms the importance of considering the potential effects of existing nearby nuclear installations during the construction of a nuclear installation; therefore fulfilment of nuclear safety requirements can be a demanding task based on the results of the study.

I. INTRODUCTION

In agreement with the recommendations of the International Atomic Energy Agency and with good practices world-wide, Hungarian nuclear safety regulations require that external events must be considered in the site investigation and evaluation, and in the definition of design basis for nuclear installations. This evaluation should cover man-made external hazards too. Events with radiological impacts on a nuclear installation from outside sources are an important but often less examined instance of external events. In this study we looked at the implications of safety regulations concerning the treatment of radiological consequence hazards. We selected the four VVER-440/213 type units of the Paks Nuclear Power Plant as the source of potential large releases of radioactivity and analyzed the radiological impacts in the vicinity of the plant from those accidental releases that would need to be considered as external events if a nuclear installation was to be located in the area investigated. The target area investigated was a territory within a distance of 1500 meters north of the northern edge of the Paks site.

II. APPROACH

According to nuclear safety requirements, man-made external events of more than 10^{-7}/year expected frequency have to be considered in the design basis of a nuclear power plant in Hungary. Had level 3 PSA been available for the Paks plant, the radiological consequence – frequency relationship in the vicinity of the pant (as it could be derived from PSA) could have been directly applied to identify all those radiological consequence events that should be within the design basis envelop of a nuclear power plant near the existing one in order to take the effects of potential large releases from nearby nuclear installations into account. Since level 3 PSA was not available for the Paks plant, and the development of a full scope level 3 analysis was beyond the scope of the study, a simplified but structured approach was followed. The key analysis steps included:

- Selection of reference events based on the existing level 1 and level 2 PSA for the plant
- Analysis of radiological consequences for the reference events
- Interpretation and evaluation of results.

II.A. Selection of Reference Events

In lack of a level 3 analysis we made use of the existing level 1 and level 2 PSA for the Paks NPP (Refs. 1, 2, 3 and 4) to select accident sequences to be analyzed from the point of view of radiological impact. The PSA was found much helpful for the purpose of the analysis, because it covers both reactor and spent fuel pool (SFP) accidents, it addresses full power as well as low power and shutdown states, and it includes a wide range of initiating events (internal events, internal and external hazards). However, a site risk model is not available for the plant. This was considered a limiting factor. Multi-unit or multi-source accidents can play an important role in characterizing radiological consequences as the releases from such accidents would typically exceed the release from a single-source accident.

The results of the single-source level 2 PSA were studied in detail to select the limiting severe accident sequences from the point of view of environmental release. It means that not just coarse categories of release magnitude (i.e. large vs. small or early vs. late) were considered, but the actual location, size and timing of the release were also accounted for. Considerations were of course given to accident frequencies too. This step led to the identification of four single-source accident sequences for consequence assessment, three of them as reference scenarios and a fourth one as an “extreme” scenario for sensitivity analysis. Of course, the location of the target area relative to the site of the Paks NPP also played a role in the definition of the reference events.

In the next step the minimal cut sets of the level 1 and level 2 accident sequences were analyzed in detail for both the reactor and the spent fuel pool to identify potential multi-unit and multi-source accidents. Various combinations of full power and low power and shutdown states were studied for the four power plant units. Emphasis was placed on assessing inter-unit and inter-source correlation of seismic induced failures. In addition to analyzing PSA results, the findings of the post-Fukushima targeted safety assessment (stress-test) (Ref. 5) and subsequent seismic safety analyses (Refs. 6, 7 and 8) for the plant were utilized in the assessment, although a great deal of expert judgement was also necessary. As a result, two multi-source severe accidents were selected as reference cases for consequence assessment, and three additional multi-source or multi-unit accidents were defined for sensitivity analysis. The frequency of the two reference multi-source scenarios was estimated to exceed the screening threshold. A summary of the selected single and multiple severe accident sequences is given in Table 1 with a much concise description of the events. In each case the initiator is a strong seismic motion not exceeding 0,30g in peak ground acceleration. All these events were subject to radiological consequence analysis.

TABLE I. Severe Accident Scenarios Selected for Consequence assessment

<table>
<tr><th colspan="2">Event</th><th rowspan="2">Release Source</th><th rowspan="2">Operational State</th><th rowspan="2">Description</th></tr>
<tr><th>Type</th><th>Category</th></tr>
<tr><td rowspan="4">Single</td><td>1. Reference</td><td>Reactor, Unit 4</td><td>Full Power</td><td>Early Containment Failure</td></tr>
<tr><td>2. Reference</td><td>Reactor, Unit 4</td><td>Open Reactor</td><td>Core Melt, Reactor Building Intact</td></tr>
<tr><td>3. Reference</td><td>SFP, Unit 3</td><td>Refueling</td><td>Fuel Melt, Reactor Building Intact</td></tr>
<tr><td>4. Extreme</td><td>SFP, Unit 3</td><td>Medium Power, Normal Level</td><td>Fuel Melt, Reactor Building Damage</td></tr>
<tr><td rowspan="10">Multiple</td><td rowspan="2">5. Reference</td><td>Reactor, Unit 4</td><td>Full Power</td><td>Early Containment Failure</td></tr>
<tr><td>SFP, Unit 4</td><td>Medium Power, Normal Level</td><td>Fuel Melt, Reactor Building Intact</td></tr>
<tr><td rowspan="2">6. Reference</td><td>Reactor, Unit 4</td><td>Open Reactor</td><td>Core Melt, Reactor Building Intact</td></tr>
<tr><td>SFP, Unit 4</td><td>Medium Power, Normal Level</td><td>Fuel Melt, Reactor Building Intact</td></tr>
<tr><td rowspan="2">7. Extreme</td><td>Reactor, Unit 3</td><td>Full Power</td><td>Early Containment Failure</td></tr>
<tr><td>SFP, Unit 3</td><td>Medium Power, Normal Level</td><td>Fuel Melt, Reactor Building Damage</td></tr>
<tr><td rowspan="2">8. Extreme</td><td>Reactors, Units 1-4</td><td>Full Power</td><td>Early Containment Failure</td></tr>
<tr><td>SFPs, Unit 3-4</td><td>Medium Power, Normal Level</td><td>Fuel Melt, Reactor Building Intact</td></tr>
<tr><td rowspan="2">9. Extreme</td><td>Reactor, Unit 4</td><td>Open Reactor</td><td>Core Melt, Reactor Building Intact</td></tr>
<tr><td>SFPs, Unit 3-4</td><td>Medium Power, Normal Level</td><td>Fuel Melt, Reactor Building Intact</td></tr>
</table>

II.B. ANALYSIS OF RADIOLOGICAL CONSEQUENCES

In preparation for the analysis of radiological effects we first characterized off-site releases of radioactivity for the single-source reference events. We did calculations by the MAAP4/VVER code (Ref. 9) for this purpose, and the results were used as input to determining radiological consequences near the plant. The results obtained for single-source events were aggregated to describe releases from multi-unit and multi-source scenarios to enable consequence analysis.

The time history of release was determined by MAAP calculations for each single-source reference event. The magnitude of off-site release from an accident was expressed in percentage of the initial inventory of radioactive isotopes for different groups of radionuclides. As dose calculations appeared time-consuming, a conservative stepwise approximation of the release functions was applied to generate input to radiological consequence analysis. We divided the release curves into eight discrete sections (sub-intervals) and assigned the highest release rate within each sub-interval to the lower bound of the interval. An example of release curves and their approximation is shown in Fig. 1 for the first reference reactor accident, i.e. for early containment failure at full power.

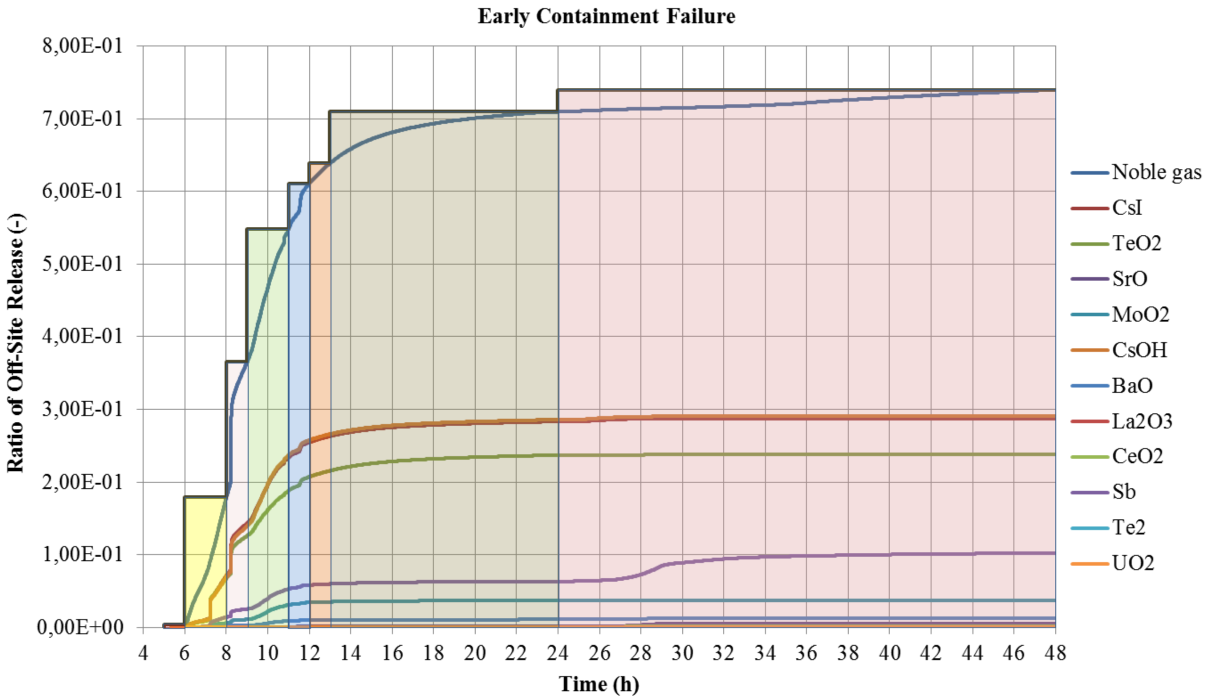

Fig. 1. Time History for Environmental Release of Fission Products during a Reference Reactor Accident

Based on the initial isotope inventory, the release ratios for radionuclide groups were converted to released activity of 31 important fission product isotopes for the purpose of dose calculations. These isotopes were selected by giving considerations to their half-life, activity and consequence. In addition to the time history of release, the expected location and size of the release source were also specified to prepare input data for analyzing environmental spread of radioisotopes.

Using the results of MAAP analysis as input we analyzed the environmental spread of radioactive materials in the vicinity of the plant and determined the corresponding dose consequences. We quantified consequence measures suitable for describing the effects of radioactivity on humans working open air and on humans working within building enclosures of a fictitious nuclear installation in the investigated area. For this purpose we first calculated the volumetric concentration of radioactivity as a function of time elapsed from the accident and horizontal distance from the release source for 16 discrete distance values, which provided the basis for subsequently determining the effective dose from inhalation. Similarly, we assessed the concentration of radioactivity on the ground and determined the expected effective dose from ground contamination. Finally, we estimated the dose consequences from cloud shine too. All the three considered components of radioactive dose were aggregated to characterize dose rate for personnel working open air, while the dose from inhalation was found relevant for personnel residing within building enclosures. The volumetric concentration of radioactivity and the associated inhalation dose rate were also determined as a function of altitude (vertical distance from the ground surface) for 6 values of vertical distance such data were considered important from the point of view of specifying design input to protection measures and equipment (e.g. air filtration systems) needed to ensure habitability of plant service areas. Dose consequences were assessed by assuming three different kinds of weather pattern: a most probable one characterized by Pasquill stability class F with straight-line wind of 1 m/s from source to target area, and two additional ones for sensitivity analysis: Pasquill stability class D and straight-line wind of 2 m/s with and without rain of 20 mm/h intensity. The PC COSYMA (Ref. 10) code was the basic tool of consequence analysis, although software from in-house development was also applied to overcome some limitations of COSYMA.

III. RESULTS

The overall numerical results of consequence assessment include quantified values describing the effects of radioactivity for all the selected reference and extreme accident sequences and for each of the three pre-defined weather patterns:

- Volumetric concentration of radioactivity concentration
- Concentration of radioactivity on ground surface
- Effective dose rate from inhalation, ground contamination and cloud shine
- Aggregated dose rate.

The results of consequence analysis for the various analysis cases were subject to comparative assessment to select an accident from the list of reference events that could be considered bounding concerning radiological impacts. This bounding reference accident – as an ultimate reference event – is event No. 5 in Fig. 1 involving large releases of radioactivity from multiple sources. It is a severe accident of the reactor operating at full power in unit 4 (the closest one to the area investigated), combined with a severe accident of the spent fuel pool in the same unit. This complex accident is induced by a strong seismic motion shaking the site and leading to large releases of radioactivity outside the plant. The frequency of this event is higher than the screening threshold defined in nuclear safety regulations and the radiological impacts of the event were found the most substantial from among the reference events examined. The results of consequence analysis are demonstrated below on the example of the ultimate reference event as it was the basis for specifying radiological effects that would need to be considered in siting and in defining the design basis of a nuclear installation.

III.A. Open Air Radiological Consequences

The accessibility of the investigated area adjacent to the existing nuclear power plant as well as the conditions and the allowable time for working open air in that area are primarily determined by the inhalation dose, the dose from cloud shine and the dose from ground surface contamination. Based on the results of radiological consequence analysis for the ultimate reference event, the dose rates for these three dose contributors are shown in Fig. 2, Fig. 3, and Fig. 4, respectively as a function of distance from the southmost edge of the investigated area that is closest to the operating units of NPP Paks. Fig. 5 presents the cumulative dose rates for selected times as a sum of the three contributors. The curves present the distance dependence of the dose rates for two reference values of time elapsed from the occurrence of the accident initiator (a seismic motion). These are the time values when the releases from the different release sources are at their maximum:

1. Unit 4 operating at full power (11 hours)
2. Spent fuel pool of unit 4 (92 hours).

As shown in Fig. 4, 93% of the cumulative dose rate is attributable to inhalation at each reference time point. About 35% of the maximum inhalation dose rate from the full power accident of the reactor comes from the ^{131}I isotope, 25% is due to the ^{106}Ru isotope, 11% is due to the ^{132}Te, and 10% is due to the ^{133}I isotope. 40% of the maximum inhalation dose rate from the spent fuel pool accident comes from the ^{134}Cs isotope, and the contributions of the ^{131}I and ^{137}Cs isotopes are 30% and 22%, respectively.

The 4 Sv/h effective inhalation dose rate obtained from the calculations is quite high. However, it can be effectively reduced by protective measures. A reduction of about two orders of magnitude can be achieved by using individual protection devices (e.g. masks). The dose rate can be reduced further by about two orders of magnitude if iodine prophylaxis is applied. The dose rate from ground surface contamination is also significant, although lower than dose rate from inhalation. Appropriate decontamination technologies and special transport vehicles can be used to reduce the effect of ground contamination below an acceptable level. The individual protection devices and the special transport vehicles are also a good means of defense against the effects of cloud shine.

The figures witness that the dose rate increases until a distance of 200 meters, and then it reduces with distance. Thus, over and above the protective measures discussed above, the application of appropriate safety distances is also a good means to reduce dose consequences. Furthermore, the exposure time of the operating personnel can be limited to limit the effective dose by strictly controlling the allowable time for working open air with considerations to the changes in the dose rate over time.

It is noted that the results obtained for the weather pattern characterized by stable atmosphere showed higher volumetric radioactivity concentrations and correspondingly higher inhalation dose rates as compared to the results for the reference weather pattern.

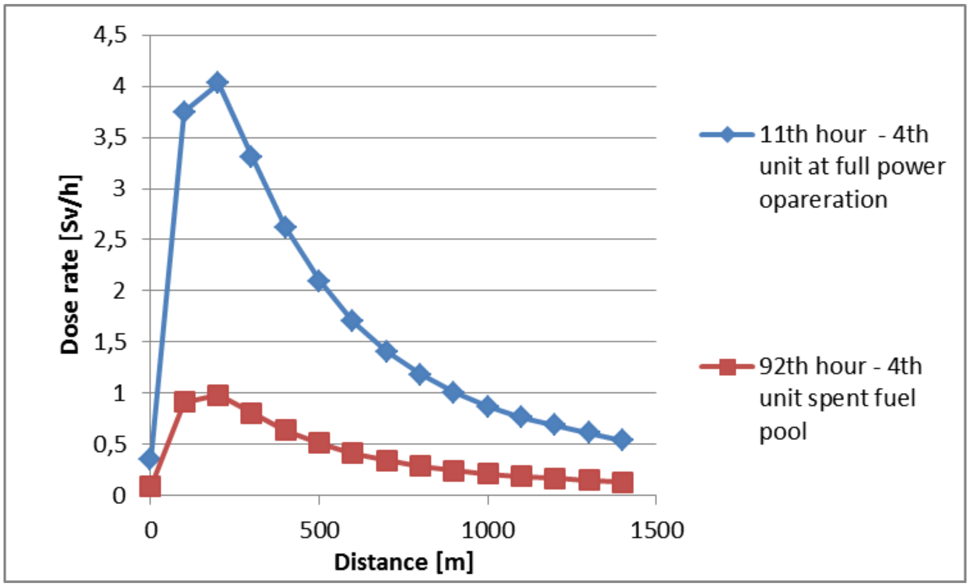

Fig. 2. Inhalation Dose Rate at Reference Time Points

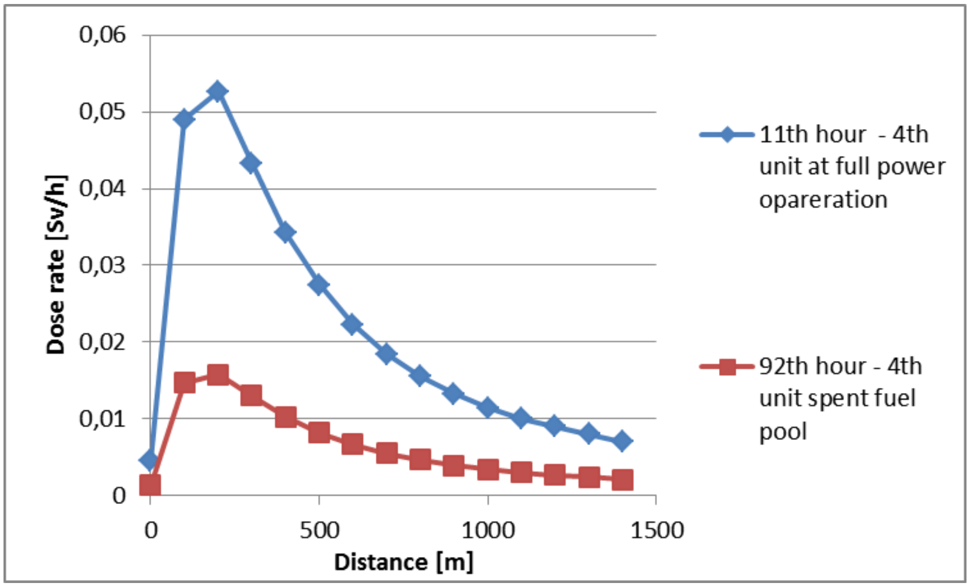

Fig. 3. Ground Dose Rate at Reference Time Points

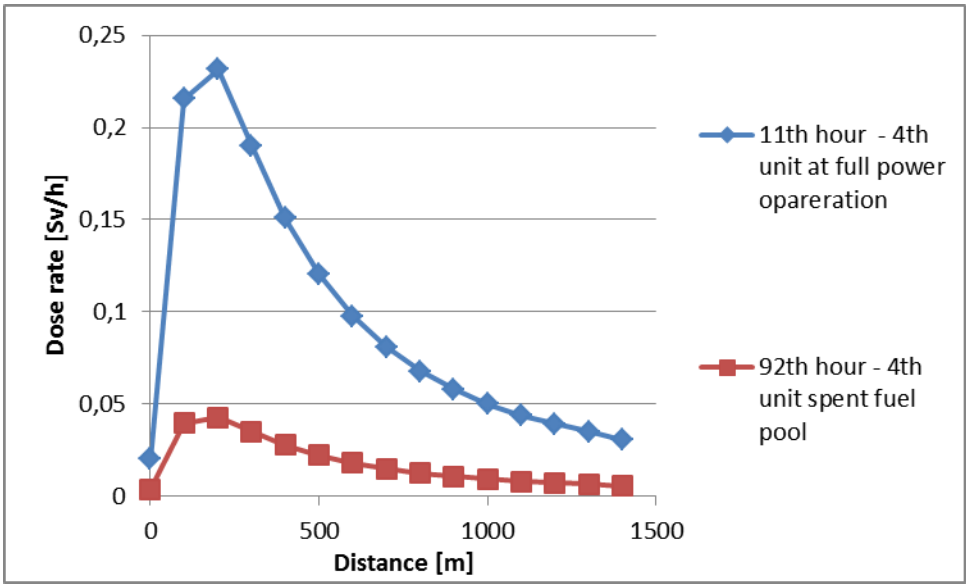

Fig. 4. Cloud Shine Dose Rate at Reference Time Points

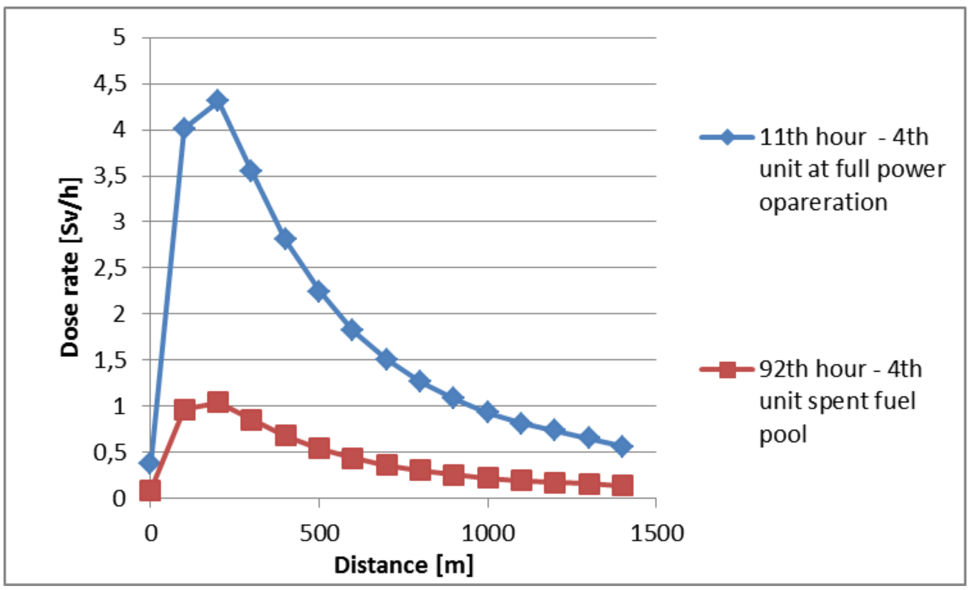

Fig. 5. Cumulative Dose Rate at Reference Time Points

III.B. Radiological Consequences within Building Enclosures

As for open air consequences, some results important to characterizing potential radiological consequences within building enclosures are presented in figures below for the ultimate reference event. Fig. 6 shows volumetric radioactivity concentration, while Fig. 7 gives the inhalation dose at the altitude of the release source (20 meters) as a function of distance from the most southmost edge of the site that is closest to the operating units. The reference time points of the curves are identical to those described in Section III.A. At each reference time point the contribution of the different isotopes is the same as that discussed in Section III.A.

The 25 Sv effective inhalation dose rate obtained from the calculations (see Fig. 7) is excessively high. However, it can be effectively reduced by protective measures. A reduction of about three orders of magnitude can be achieved by using similar air filtration systems that are in use at the operating units of the Paks nuclear power plant. The dose rate can be reduced further by about two orders of magnitude by using individual protection devices (e.g. masks) and similar additional reduction can be expected if iodine prophylaxis is applied. The dose rate significantly reduces with distance and it varies as a function of the altitude. Therefore, appropriate positioning and orientation of the air filtration equipment also helps to limit the dose rate within the plant buildings. Furthermore, the exposure time of the operating personnel can be limited to limit the effective dose by strictly controlling the allowable time at work. The habitability of vital service areas within the building enclosures of the plant can be ensured and thus the safe conditions of a nuclear power plant can be maintained by an appropriate use of the protective measures and administrative procedures referred to above.

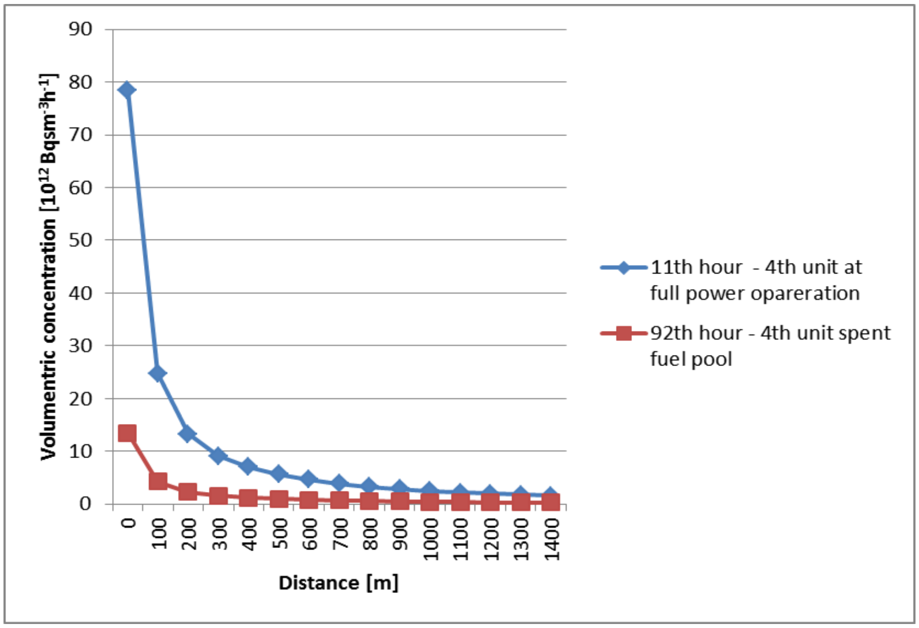

Fig. 6. Volumetric Radioactivity Concentration at Reference Time Points at 20 Meters Altitude

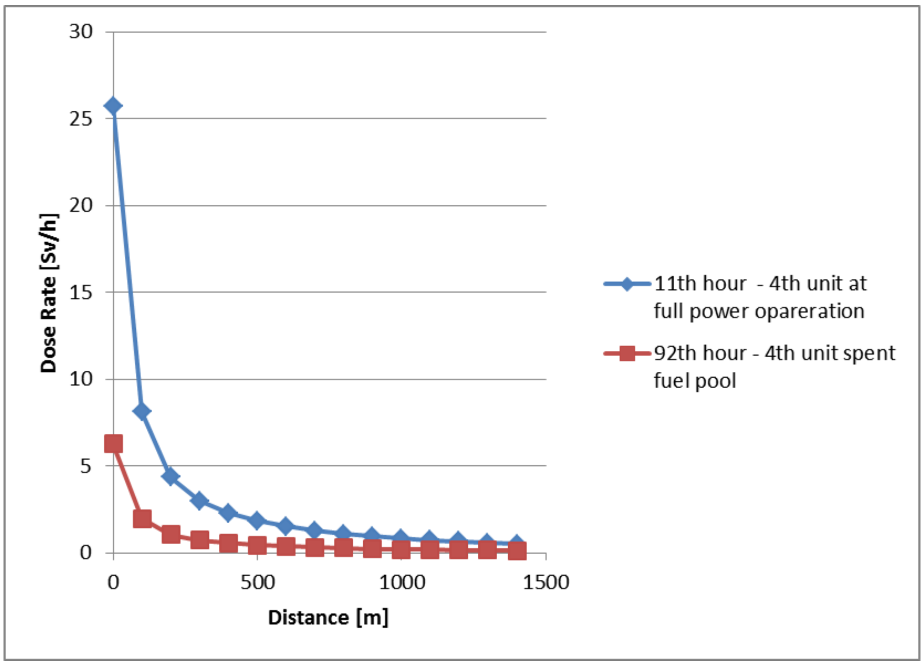

Fig. 7. Inhalation Dose Rate at Reference Time Points at 20 Meters Altitude

IV. CONCLUSIONS

The Hungarian nuclear safety regulations require that man-made hazards should be taken into account in a new nuclear facility design up to 10^{-7}/year frequency. Radiological consequence hazards have been found a most important area of site investigation. The approach followed and the analysis results obtained can support site evaluation and the definition of design data for ensuring the safe conditions of future nuclear installations if a severe accident occurs at the operating units of the Paks plant. Depending on the location of the nuclear installation relative to the existing power plant, fulfilment of nuclear safety requirements can be a demanding task based on the results of the study.

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