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ORIGINAL PAPER

Assessment of environmental public exposure from a hypothetical nuclear accident for Unit-1 Bushehr nuclear power plant

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Abstract Unit-1 of the Bushehr nuclear power plant (BNPP-1) is a VVER-type reactor with 1,000-MWe power constructed near Bushehr city at the coast of the Persian Gulf, Iran. The reactor has been recently operational to near its full power. The radiological impact of nuclear power plant (NPP) accidents is of public concern, and the assessment of radiological consequences of any hypothetical nuclear accident on public exposure is vital. The hypothetical accident scenario considered in this paper is a design-basis accident, that is, a primary coolant leakage to the secondary circuit. This scenario was selected in order to compare and verify the results obtained in the present paper with those reported in the Final Safety Analysis Report (FSAR 2007) of the BNPP-1 and to develop a well-proven methodology that can be used to study other and more severe hypothetical accident scenarios for this reactor. In the present study, the version 2.01 of the PC COSYMA code was applied. In the early phase of the accidental releases, effective doses (from external and internal exposures) as well as individual and collective doses (due to the late phase of accidental releases) were evaluated. The surrounding area of the BNPP-1 within a radius of 80 km was subdivided into seven concentric rings and 16 sectors, and distribution of population and agricultural products was calculated for this grid. The results show that during the first year following the modeled hypothetical accident, the effective doses do not exceed the limit of 5 mSv, for the considered distances from the BNPP-1. The results obtained in this study are in good agreement with those in the FSAR-2007 report. The agreement obtained is in light of many inherent uncertainties and variables existing in the two modeling procedures applied and proves that the methodology applied here can also be used to model other severe hypothetical accident scenarios of the BNPP-1 such as a small and large break in the reactor coolant system as well as beyond design-basis accidents. Such scenarios are planned to be studied in the near future, for this reactor.

Keywords Bushehr · Nuclear power plant · Public exposure · Effective dose · Hypothetical accident scenario · Dose assessment · Dispersion modeling · PC COSMYA · FSAR-2007

Introduction

Nuclear power plants (NPPs), such as the unit 1 of the Bushehr NPP (BNPP-1) which is the reactor of concern in this paper, are designed for safe operation so that the risk of accidents and thus public exposure is usually minimal. However, a severe accident may occur due to unexpected natural or human failures. The Fukushima accident is one recent example of a severe nuclear accident. During such accidents, copious amounts of radionuclides might be released into the environment which can cause severe public exposure. The impact of NPP accidents is thus of serious public concern, and studies on public exposure after a severe hypothetical nuclear reactor accident are indispensable, to develop tools for an adequate design of the countermeasures.

The BNPP-1 is a VVER reactor (Russian: Voda-Vodyanoi Energetichesky Reaktor) and is a pressurized water reactor with 1,000-MWe power; light-water-moderated and cooled. It was constructed at the Hallileh site near

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Bushehr at the coast of the Persian Gulf, Iran. The BNPP-1 has been recently made operational to near its full power. So, the assessment of the radiological impact of normal and accidental conditions of this reactor on public exposure is highly important and vital. The radiological impact of the BNPP-1 under its normal operation conditions has been studied and recently published (Sohrabi et al. 2012). In contrast, the present study focuses on the assessment of public exposure from environmental releases of radionuclides from a hypothetical accident scenario, as discussed below.

Many accidents have been hypothesized by the International Atomic Energy Agency (IAEA) as initiating events in its guidelines for accident analysis of water-cooled water-moderated power reactor (WWER) (IAEA 1996). These hypothetical accidents generally include (1) anticipated transients (anticipated operational occurrences, AOO) which can be caused by equipment failure, improper operation of any component or operator's error, (2) postulated design-basis accidents (DBA) where damages at an NPP can take place and immediate restoration of operation may prove impossible, and (3) beyond postulated accidents (beyond design-basis accidents, BDBA) which are certain incidents of very low probability, which are beyond DBA conditions and which may arise due to multiple failures of safety systems and may jeopardize the integrity of many or all the barriers preventing release of radioactive substances (IAEA 1996). Since the AOO scenarios do not reach the limits of nuclear safe operation and the BDBA accidents have very low probability, DBA conditions are usually considered, and among those, the loss of coolant mass in the reactor coolant system (LOCA) is chosen as the most important incident. This kind of accident contains primary-to-secondary leakage within a steam generator, a large break LOCA, and a small break LOCA (IAEA 1996). According to DBA analysis, the worst accident case—from the standpoint of radioactivity release into the volume of the first (steel) containment shell—is considered the large break LOCA. In case of the occurrence of such an accident, the activity of radionuclides in the atmosphere inside the first (steel) shell will be highest and, therefore, represent the most hazardous source of dose for reactor personnel. On the other hand, the worst accident case from the view point of maximum activity of radionuclides released into the open air is the leakage of radioactivity from the primary to secondary coolant loop (FSAR 2007).

The reactor accident scenario involving primary-to-secondary leakage within a steam generator has been recently studied and reported in the Final Safety Analysis Report (FSAR) of the BNPP-1 (FSAR 2007). This hypothetical scenario was also selected in the present study in order to (a) perform a comparative analysis of the results obtained in the present study using 2.1 version of the PC COSMYA code with those reported in the FSAR-2007 using another computational code (DOZA) and different sets of data, (b) develop the required computational and data analysis methodology and know-how, (c) use the experiences gained, and the know-how and lessons learned (such as adequacy of the computation code used, collection of adequate statistics, proper data selection and analysis) for modeling of other severe hypothetical accidents such as a small or large break in the reactor coolant system as well as beyond design-basis accidents.

Materials and methods

To study the above, version 2.01 of the PC COSYMA code was applied. This version is a simplified version of the mainframe COSYMA (Code System from MARIA) code first developed by the Radiation Protection Division of the Health Protection Agency in the United Kingdom and the Forschungszentrum Karlsruhe (FZK) in Germany as part of the European Commission's MARIA Program (Methods for Assessing Radiological Impact of Accidents) (NRPB and FZK 1996). The PC COSYMA code has a number of endpoints which include air concentration or deposition of particular nuclides, doses received by members of the public, the individual or collective risks of health effects in the exposed population, the extent and duration of countermeasures which might be imposed to reduce health effects, and the economic costs of the countermeasures and health effects, each of which requires separate investigations. In the present study, individual effective doses in the early phase and individual and collective doses in the late phase of the accident are estimated. The "effective dose" is called "dose" from now on in this paper.

For the early phase, the individual dose includes the external dose from the plume (cloud shine) and the ground (ground shine), and the internal dose to organs due to inhalation for each relevant radionuclide; radionuclide deposition and re-suspension are also considered. Also, during the late phase of the accidental releases, individual and collective doses were assessed with consideration of corresponding ingestion factors.

The first assumptions made here were similar to those given in the FSAR-2007, and the individual doses for the first year after the accident were calculated. A probabilistic run was also applied. The individual doses for the first year and 50 years (with consideration of external and internal exposure from ingestion and inhalation) and the collective doses were assessed. For the collective dose calculation, the surrounding area of the BNPP-1 within a radius of 80 km was subdivided into seven concentric rings (0–5, 5–10, 10–20, 20–30, 30–40, 40–60, and 60–80) with 16 sectors per ring. For this grid, the population data for 2006

(Population and Housing Census 2006) and the agricultural data for 2003 (Agricultural General Census 2003) as prepared by the Statistical Center of Iran were used.

Models used in PC COSYMA

A large number of parameters influence the process of radionuclide transfer from the point of release through the environment to the public, and calculation of the subsequent doses. By means of the PC COSYMA code, analyses of the following aspects have been carried out: (a) dispersion and deposition, (b) food chain transfer, (c) assessment of external and internal doses (due to inhalation and ingestion of radionuclides), and (d) early and late health effects.

Atmospheric dispersion and deposition

The atmospheric dispersion model in PC COSYMA is a Gaussian segmented plume model (MUSEMET) (NRPB and FZK 1996). This model takes the atmospheric conditions affecting the plume from a data file providing hourly averages of atmospheric parameters (wind speed and direction, atmospheric stability classes, precipitation intensity, and mixing layer depth). It allows estimating the effects of these parameters on the subsequent dispersion of the plume and takes, for example, the effects of wet and dry deposition of the dispersed radionuclides into account.

The basic formula of the Gaussian plume model describes the downwind time-integrated concentration $C(x, y, z)$ of any radionuclide in air resulting from the release of material from a point source at $(x = 0, y = 0, z = H)$:

$$C(x,y,z;H) = \frac{Q}{2\pi u \sigma_y \sigma_z}\left[\exp\left(-\frac{y^2}{2\sigma_y^2}\right)\left\{\exp\left(-\frac{(z-H)^2}{2\sigma_z^2}\right) + \exp\left(-\frac{(z+H)^2}{2\sigma_z^2}\right)\right\}\right],$$

where x is the distance (m) downwind from the emission source point; y is the crosswind from the emission plume centerline (m); z is the distance (m) above ground level; Q is the activity (Bq) released; u is the mean wind speed (transport speed) (m s^{-1}) in the downwind x direction; H is the height (m) of the plume centerline; and σ_z, σ_y (m) are the dispersion parameters describing the plume spread in the horizontal and vertical crosswind directions y (m) and z (m) which are dependent on a_y, b_y, a_z, and b_z through the relation:

$$\sigma_y = a_y x^{b_y}, \ \sigma_z = a_z x^{b_z},$$

where a_y, b_y, a_z, and b_z are stability-dependent parameters. The values of a_y, b_y, a_z, and b_z should be introduced into the code by the user.

Dry deposition of radioactive material from the plume to surfaces is modeled using three constant deposition velocities for elemental iodine, organically bound iodine, and aerosols. The plume depletion due to dry deposition of material is modeled by using the 'source depletion model.' This model assumes that depletion occurs over the whole depth of the plume rather than in the layer near to the surface.

Wet deposition of radioactive material is modeled using the so-called washout model. The washout constant Λ (s^{-1}) is defined as a function of the rain intensity, I, through the following power law:

$$\Lambda = a(I)^b$$

The coefficients a and b are washout proportionality coefficients and depend on the radioactive material to be washed out which should be introduced into the code by the user.

The plume depletion due to wet deposition is modeled by assuming that the radioactive material is washed out and removed from the whole vertical extension of the plume. The fraction of radioactive material remaining in the plume after time t is (Päsler-Sauer and Jones 2000):

$$F(t) = \exp(-\Lambda t)$$

The PC COSYMA code can be used for deterministic and probabilistic calculations. Deterministic assessments provide detailed results for a single set of atmospheric conditions, while probabilistic assessments provide results taking into account the full range of atmospheric conditions and their respective frequency of occurrence. In all cases, the atmospheric conditions for the analysis must be specified (NRPB and FZK 1996).

For deterministic runs, a constant set of conditions throughout the time taken for the plume to travel over the region is considered. In fact, the assumed values stability category, wind speed, direction, and rainfall rate apply for the whole plume travel period (NRPB and FZK 1996).

The PC COSYMA code has been also designed for probability safety assessment based on the statistical processing of results obtained in repeated (144 times) single runs. For probabilistic runs, many sets of atmospheric conditions are considered. Therefore, a data file including the atmospheric conditions every hour for a period of at least 1 year is required. The simplest method of selecting starting times for the sequences is to do so either at random or by selecting every nth sequence (cyclic sampling). The latter method tends to sample those conditions which occur frequently in the data file, and assigns equal probability to each of the sequences of conditions considered. The results are presented in probability distribution forms. A probability distribution is presented using complementary cumulative frequency distributions (CCFDs) which represent the probability that the radiological consequences on the public can be greater than or equal to a particular value (NRPB and FZK 1996).

Food chain models

PC COSYMA requires information on the concentration of radionuclides in foods as a function of time after an accident. The code does not include a food chain model but uses the results of such models through data libraries which provide the activity concentration for a range of radionuclides in a number of foods as a function of time following unit deposition. The concentration of radionuclides in foods depends on the time of the year at which the deposition occured.

PC COSYMA has access to two data libraries used in calculating the dose from ingestion of contaminated food. These libraries were derived from the model FARMLAND introduced by the Health Protection Agency in the United Kingdom and the model ECOSYS by the Gesellschaft für Strahlen und Umweltforschung (GSF) (Margeanu and Angelescu 2003).

Differences between the predictions of these models are generally small, and an agreed set of model parameter values for application in Europe was derived. The foods considered in FARMLAND are milk, meat, and liver from cattle, pork, meat, and liver from sheep, green vegetables, grain products, potatoes, and other root vegetables. In contrast, the foods considered in ECOSYS are milk, beef, pork, grain products, potatoes and other root vegetables, and leafy and non-leafy green vegetables. The values predicted by both models are generally applicable to Northern and Central Europe. When calculating the individual doses due to ingestion, the user must provide consumption rates for the foods included in the data library being used. PC COSYMA assumes that all food produced is consumed somewhere, and so calculates the collective intake of activity from the food produced at each grid location. The collective dose in the whole population is then calculated from these collective intakes (Margeanu and Angelescu 2003).

Exposure pathways

Following an accidental release of radioactive material into the environment, the public can be exposed through a number of pathways of radioactive material both from the atmosphere and after its deposition to the ground.

PC COSYMA considers the following pathways: (a) external γ exposure from material in the cloud, (b) internal exposure following inhalation of material in the cloud, (c) external β exposure from material deposited on skin or clothing, (d) external γ exposure from material deposited on skin or ground, (e) internal exposure following inhalation of material re-suspended after its initial deposition on the ground, and (f) internal exposure following the ingestion of food stuffs contaminated by deposited material.

The first pathway of radiation exposure only affects those people who were exposed to the plume of radionuclides as it passes overhead. External exposure from the cloud only occurs while the plume is present. Also, material is only inhaled while the plume is present, but some material can be retained in the body for long periods, and so the exposure can persist for long periods of time. The exposure from topic (c) stops once material is removed from the skin or clothing. The other three topics can result in long-time exposures. The dose from ingestion of contaminated food is not considered in calculating doses to be used for early deterministic health effects (NRPB and FZK 1996).

Dose calculations

External γ dose from radionuclides in the plume is calculated by multiplying the time-integrated air concentration by a pre-calculated factor and the cloud γ doses per unit air concentration which is obtained from a data library. The dose factor in the library was calculated assuming that the plume is semi-infinite and of uniform concentration.

In contrast, the dose from radionuclides inhaled from the plume is calculated as the product of the time-integrated air concentration, breathing rate and dose per unit activity inhaled, which is obtained from a data library included in the system. External β dose to skin is obtained from radionuclides deposited on skin and clothes (β particles reaching from cloths to skin), considering the time the corresponding radionuclides remain on skin or on clothes.

External γ dose from deposited radionuclides is calculated by multiplying the amount of material deposited by a quantity, the dose per unit deposition, which is obtained from a data library, while the dose due to inhalation of re-suspended radionuclides is also calculated as the product of the time-integrated air concentration, breathing rate, and dose per unit activity inhaled. Finally, the collective dose from ingestion is calculated from the radioactivity deposited in each grid element, the concentration of radioactive material in foods from unit deposition, the amount of food which is produced for human consumption in each grid element, and the dose per unit activity ingested. This calculation is based on the assumption that all food produced in the area of interest is consumed somewhere; the collective dose is independent of where the food is consumed. Individual doses are calculated on the assumption that all food is produced at the point of consumption. The deposition density is multiplied by the concentration in food per unit deposition, the individual consumption rate, and the dose per unit intake (NRPB and FZK 1996).

Input parameters

A considerable amount of information is needed before the PC COSYMA code can be run. The information can be considered in some sections. The calculations assumed no countermeasures. The other input parameters were based on default values in the PC COSYMA code.

Source term

The amount of radionuclides that might be released to the environment—the source term—is very dependent on plant design and can be estimated from the calculations made by computer programs. A list of the reactor data are presented in Table 1.

The IAEA guidelines provide methodological recommendations for accident analysis of WWERs (IAEA 1996). This IAEA document has been used for categorization and classification of the initiating events as used in the FSAR-2007 and includes (a) anticipated transients, anticipated operational occurrences (AOO), which can be caused by equipment failure, improper operation of any component, or operator's error. The given transients shall not have consequences from the viewpoint of safety and do not require cessation of the plant operation. The results of thermohydraulic calculations showed that in AOO scenarios, the limits of nuclear safe operation are not reached (FSAR 2007); (b) postulated accidents, design-basis accidents (DBA), at which damages at an NPP can take place and the immediate restoration of operation can prove impossible (FSAR 2007); (c) beyond postulated accidents and beyond design-basis accidents (BDBA); certain very low-probability events due to beyond design-basis accident conditions and may arise due to multiple failures of safety systems and may jeopardize the integrity of many or all the barriers preventing release of radioactive substances (FSAR 2007).

Table 1 List of the reactor data taken from FSAR (2007)

Parameters	Value
Reactor thermal power (MW)	3,120
Coolant flow rate via the reactor (m^3/h)	80,000
Coolant pressure at the core outlet (Mpa)	16.0
Coolant temperature at the reactor inlet (°C)	293
Steam pressure in steam header of the steam generator (Mpa)	6.37
Boiler water level in steam generator (m) (version 1)	1.87
Boiler water level in steam generator (m) (version 2)	1.97

Since the AOO scenarios do not reach the limits of safe operation and the BDBA accidents have very low probability, DBA conditions have been considered here, and among those, the loss of coolant mass in the reactor coolant system (LOCA) was chosen as the most important event in a nuclear power plant. This kind of accident includes primary-to-secondary leakage within a steam generator, large break LOCA, and small break LOCA.

Radionuclides released into the environment in case of an accident involving the primary coolant leakage to the secondary circuit which is the subject of this paper are presented in Table 2.

The amount and form of radionuclides released into the environment in case of a small break LOCA and a large break LOCA for the BNPP-1 are shown in Table 3 (FSAR 2007).

According to the design-basis accident analysis, in terms of radioactivity release into the volume of the first (steel)

Table 2 Release into the environment in case of an accident involving the primary coolant leakage to the secondary circuit (FSAR 2007)

Radionuclides	Release (Bq)	Radionuclides	Release (Bq)
Br-84	5.56×10^{12}	I-135	1.89×10^{13}
Kr-85m	4.44×10^{12}	Xe-135	4.08×10^{12}
Kr-85	1.22×10^{9}	Cs-137	1.04×10^{12}
Br-87	1.37×10^{13}	Xe-138	1.71×10^{13}
Kr-87	1.44×10^{13}	Cs-138	1.82×10^{13}
Kr-88	1.81×10^{13}	Ba-139	3.67×10^{11}
Rb-88	1.81×10^{13}	Ba-140	4.08×10^{9}
Kr-89	2.48×10^{13}	La-140	5.19×10^{8}
Rb-89	2.63×10^{13}	Ce-141	6.67×10^{8}
Sr-89	3.15×10^{9}	Ce-144	7.41×10^{7}
Kr-90	2.48×10^{13}	Pr-144	6.67×10^{7}
Rb-90	2.37×10^{13}	Zr-95	4.45×10^{8}
Sr-90	8.14×10^{6}	Nb-95	4.45×10^{6}
Sr-91	9.63×10^{10}	Zr-97	2.74×10^{10}
Sr-92	7.78×10^{10}	Nb-97	2.48×10^{10}
Mo-99	4.07×10^{8}	Na-24	1.00×10^{11}
Ru-103	3.44×10^{8}	K-42	4.45×10^{11}
Ru-106	4.82×10^{6}	Fe-59	7.04×10^{6}
Rh-106	4.82×10^{6}	Co-58	2.74×10^{7}
Te-131	3.45×10^{10}	Cr-51	5.19×10^{7}
I-131	1.15×10^{13}	Mn-54	7.04×10^{6}
Te-132	4.45×10^{9}	Co-60	7.41×10^{7}
I-132	3.08×10^{13}	*Organic iodine*	
Te-133	5.93×10^{10}	I-131	1.15×10^{11}
I-133	2.52×10^{13}	I-132	3.11×10^{11}
Xe-133	6.30×10^{12}	I-133	2.56×10^{11}
I-134	2.34×10^{13}	I-134	2.34×10^{11}
Cs-134	6.67×10^{11}	I-135	1.89×10^{11}

Table 3 Radionuclide release into the environment in case of small and large break LOCA (FSAR 2007)

Radionuclides	Small break LOCA Release (Bq)	Large break LOCA Release (Bq)
Molecular iodine		
I-131	4.00×10^{9}	9.26×10^{11}
I-132	3.47×10^{9}	2.07×10^{11}
I-133	4.95×10^{9}	2.07×10^{11}
I-134	1.47×10^{9}	5.58×10^{9}
I-135	3.04×10^{9}	4.42×10^{10}
Organic iodine		
I-131	2.17×10^{9}	5.00×10^{11}
I-132	8.28×10^{7}	4.78×10^{9}
I-133	5.66×10^{8}	2.31×10^{10}
I-134	2.27×10^{7}	8.26×10^{7}
I-135	1.39×10^{8}	1.96×10^{9}
Inert radioactive gases		
Kr-85m	1.03×10^{10}	6.17×10^{10}
Kr-87	8.99×10^{9}	1.77×10^{10}
Kr-88	2.52×10^{10}	9.52×10^{10}
Xe-133	4.06×10^{11}	3.28×10^{14}
Xe-135	1.88×10^{10}	2.13×10^{11}
Xe-138	2.14×10^{9}	2.27×10^{9}
Aerosols		
Cs-134	1.75×10^{7}	4.94×10^{10}
Cs-137	2.86×10^{7}	9.34×10^{9}

containment shell, the worst accidental case is the large break LOCA. In case of such an accident, the activity of radionuclides in the atmosphere inside the first (steel) shell will be highest, and this is the most hazardous source of dose for reactor personnel. On the other hand, in terms of maximum activity of radionuclides released into the open air, the worst accidental case is the leakage of radioactivity from the primary to the secondary coolant loop. Therefore, an accident involving primary-to-secondary leakage within a steam generator is considered the most adverse one from the viewpoint of radionuclide release into the environment, as considered in this study. This accident can be identified by some symptoms such as (a) increase in the activity in the affected steam line of the steam generator, (b) decrease in the primary coolant pressure, (c) increase in the level of the affected steam generator, and (d) lowering the level of pressurizer.

Calculation of the amount of radionuclides released into the environment stemming from operational occurrences and design-basis accidents involving leakage of radioactive fluids within the limits of containment was made by the LEAK3 code used in the FSAR-2007 report. The computation model realized by LEAK3 describes the behavior of radionuclides inside the reactor containment. It uses a linear system of differential equations with constant or piecewise constant factors and accounts for the following processes: (a) radionuclide precipitation on the surface of civil structures and equipment, (b) radionuclide entrainment by the spraying system, (c) desorption of volatile radionuclides from the surface of civil structures and equipment, (d) release of iodine radionuclides from the solution sprayed to steam-gas medium, (e) radioactive decay of radionuclides, (f) propagation of radionuclides beyond the rooms, and (g) radionuclide cleaning on iodine and aerosol filters prior to their release into the environment.

Distribution of population and/or agricultural production

For the collective dose calculations, information on the distribution of the population and/or the agricultural production around the site is required. For this reason, the surrounding area of the BNPP-1 plant was subdivided into seven concentric rings and 16 sectors with an outer radius of 80 km. The plant is located at the center of the concentric rings. The distance of the outer boundaries of the concentric rings from the NPP are 5, 10, 20, 30, 40, 60, and 80 km, respectively.

The population and agricultural distribution around the site was calculated for this grid based on data prepared by the Statistical Center of Iran (Population and Housing Census 2006; Agricultural General Census 2003).

Atmospheric conditions

In this study, deterministic and probabilistic calculations for atmospheric conditions were used separately. For the deterministic runs, a set of conditions which were constant throughout the time taken for the plume to travel over the region was considered. In contrast, for the probabilistic runs, hourly meteorological data between May 2006 and April 2007 measured at the BNPP-1 site were employed and edited as input in 8,760 weather sequences (BNPP, On-site meteorological station 2006–2007), and cyclic sampling was used (see above).

This method tends to sample those conditions which occur frequently in the data file, and assigns equal probability to each of the sequences of conditions considered. For the probabilistic runs, the hourly meteorological data file also contained hours of rain. Because precipitation has a low probability in the study area, for the deterministic run, no precipitation was assumed.

Food

The FARMLAND model was selected because, compared to the ECOSYS model, it contains a greater number of food groups and it allows to model consumption habits that are more similar to the dietary habits in the Bushehr region. The food produced in and the food consumption rate typical for the study region, fraction of fresh food consumed, and average delay time between food harvest and consumption were considered.

Methods used in the FSAR

In FSAR-2007 report, dose calculations with special assumptions have been carried out for some hypothetical accidents by using the DOZA code. This code allows calculation of individual doses of the population living in the region around the BNPP-1 site. The dose due to external and internal exposures including cloud, surface contamination, and inhalation within 30 km from the BNPP-1 for a primary-to-secondary circuit leak within steam generator accident is presented in the FSAR-2007 report.

Results

By using assumptions similar to those presented in FSAR-2007 report, a deterministic dose calculation was performed. These assumptions were (a) the D category for Pasquill stability, (b) the average wind speed for various sectors according to Table 4, (c) a height of radionuclide release of 50 m, (d) no precipitation, and (e) the amount of radionuclides presented in FSAR-2007 report as source term. The inhalation, cloud shine, and ground shine exposure pathways were considered in the PC COSYMA

Table 4 Typical values of wind direction and velocities at the location of the BNPP-1 (FSAR 2007)

Sector of the location	Wind direction	Wind velocity (m/s)
S	N	3.1
SSW	NNE	2.7
SW	NE	2.3
WSW	ENE	2.1
W	E	2.3
WNW	ESE	4.4
NW	SE	4.0
NNW	SSE	4.3
N	S	4.2
NNE	SSW	3.8
NE	SW	3.5
ENE	WSW	3.6
E	W	4.4
ESE	WNW	5.5
SE	NW	5.6
SSE	NNW	4.0

individual dose calculations for 1 year following the accident.

The dose for various sectors around BNPP-1 at 15 km was calculated. This is the distance where Bushehr city is located, which has a higher population in comparison with other locations in the study area. The results are compared with those values in FSAR-2007 report (Fig. 1).

Individual doses at various distances from 5 to 30 km in NNW direction were also calculated. This sector includes the city of Bushehr. The results are also compared with those given in FSAR-2007 report (Fig. 2).

A probabilistic run was performed for the same accident, that is, primary-to-secondary circuit leak within the steam generator. Calculations were done for distances of 5, 10, 20, 30, 40, 60, and 80 km, respectively. As an input, the population and agricultural distribution around the site for the mentioned grids and the hourly meteorological data as described in the "input parameters" section were used, while the amounts of radionuclides released were taken from Table 2. The results are presented in terms of complementary cumulative frequency distributions (CCFDs) which represent the probability that the radiological consequences on the public can be greater than or equal to a particular value.

In Fig. 3, the CCFDs of mean individual doses cumulated over 365 days and 50 years are shown, for a distance of 15 km. Percentile of the distribution for individual doses is presented in Tables 5 and 6 for 365 days and 50 years, respectively.

The collective dose with consideration of ingestion factors is presented in Table 7 including some mean values and percentiles of the distribution. The probability distribution of the endpoints selected in this study, as discussed above, at a chosen distance includes values in all sectors around the site at that distance, and in all sequences of atmospheric conditions considered.

The contribution from different pathways to the individual dose for 365 days for whole body and thyroid alone at 15 km is presented in Fig. 4.

Discussion

The doses due to inhalation and cloud shine as well as ground shine for the first year post-accident were calculated by using the PC COSYMA code. The highest dose was obtained in the WSW sector, in agreement with the results shown in FSAR-2007 report. This sector lies in the Persian Gulf. The highest dose in the populated sectors was calculated to be 0.184 mSv for the first year after the accident in the NE direction, at 15-km distance from the site, which does not exceed the 5 mSv dose limit defined in FSAR-2007 report for the first year post-accident. This is

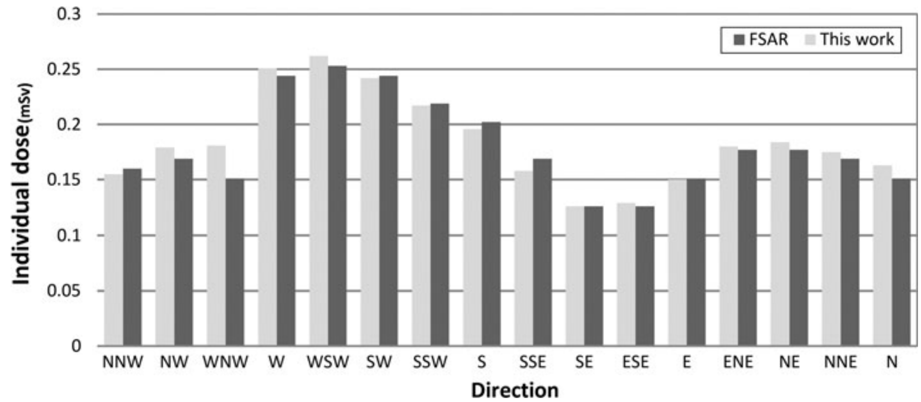

Fig. 1 Comparison of the doses estimated in the present study with those reported in FSAR (2007) for various sectors around BNPP-1 at 15-km distance from the reactor

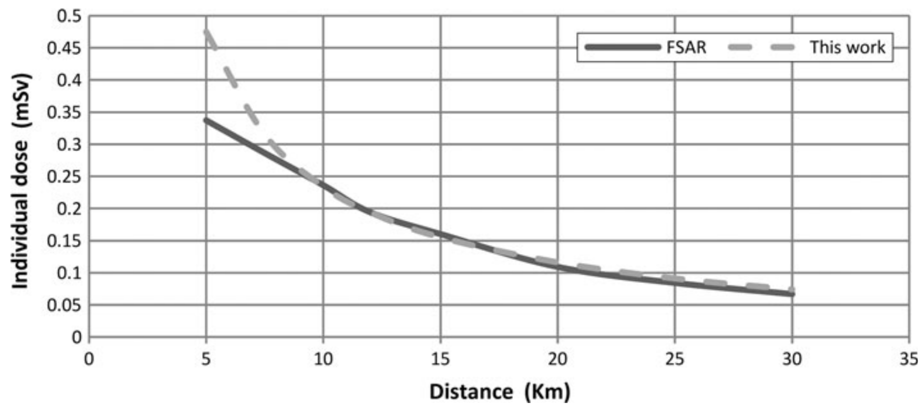

Fig. 2 Comparison of the doses estimated in the present study with those reported in FSAR (2007) for the NNW sector

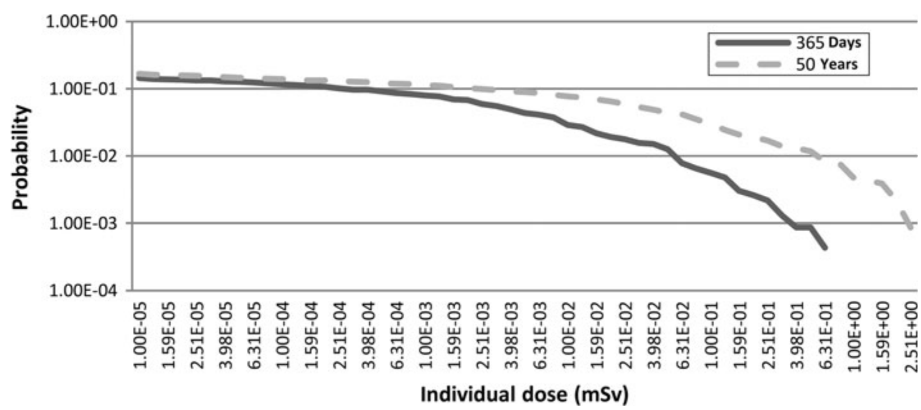

Fig. 3 Complementary cumulative frequency distributions (CCFDs) of mean individual doses for the first year and 50 years post-accident time at 15-km distance from the reactor

Table 5 Percentile of the distribution for individual doses (mSv) for the first year after the accident

x-km	Mean	Max	99th	95th	90th
2.5	4.81×10^{-2}	9.98×10^{0}	1.10×10^{0}	1.59×10^{-1}	1.51×10^{-2}
7.5	6.02×10^{-3}	1.40×10^{0}	1.32×10^{-1}	1.38×10^{-2}	1.02×10^{-3}
15	2.50×10^{-3}	6.47×10^{-1}	5.37×10^{-2}	3.98×10^{-3}	2.75×10^{-4}
25	8.15×10^{-4}	2.25×10^{-1}	1.78×10^{-2}	1.38×10^{-3}	7.22×10^{-5}
35	5.93×10^{-4}	1.42×10^{-1}	1.74×10^{-2}	9.55×10^{-4}	1.74×10^{-5}
50	2.04×10^{-4}	3.60×10^{-2}	5.62×10^{-3}	3.47×10^{-4}	–
70	1.32×10^{-4}	2.52×10^{-2}	3.89×10^{-3}	2.14×10^{-4}	–

Table 6 Percentile of the distribution for individual doses (mSv) for 50 years after the accident

x-km	Mean	Max	99th	95th	90th
2.5	4.06×10^{-1}	$6.90 \times 10^{+1}$	9.55×10^{0}	1.55×10^{0}	1.29×10^{-1}
7.5	5.64×10^{-2}	$1.84 \times 10^{+1}$	1.32×10^{0}	1.26×10^{-1}	1.02×10^{-2}
15	2.13×10^{-2}	4.40×10^{0}	5.62×10^{-1}	3.63×10^{-2}	2.29×10^{-3}
25	7.66×10^{-3}	2.38×10^{0}	1.59×10^{-1}	1.55×10^{-2}	6.61×10^{-4}
35	5.49×10^{-3}	9.34×10^{-1}	1.62×10^{-1}	8.71×10^{-3}	1.48×10^{-4}
50	2.19×10^{-3}	4.41×10^{-1}	7.08×10^{-2}	3.72×10^{-3}	2.63×10^{-5}
70	1.41×10^{-3}	3.22×10^{-1}	4.37×10^{-2}	2.24×10^{-3}	–

Table 7 Percentile of the distribution for collective doses (man Sv) for 50 years after the accident

Mean	Max	99th	95th	90th	50th
7.35×10^{0}	5.62×10^{1}	5.61×10^{1}	4.79×10^{1}	2.00×10^{1}	2.88×10^{0}

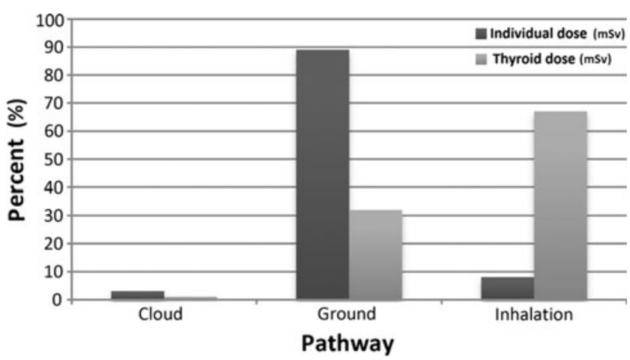

Fig. 4 Exposure pathways for total body and thyroid doses at 15-km distance from the reactor

similar to the NCRP 160 (2009) statement that "for human-made sources, annual dose limits for members of the public are 1 mSv for continuous exposures and 5 mSv for infrequent exposures. It is an ALARA standard, and has a constraint of 25 % of the limit from one single source." As for the individual doses, the results obtained in the present study are slightly different to those given in the FSAR-2007 report, for various sectors around the BNPP-1 at 15 km. The largest difference is 0.03 mSv for the WNW direction.

Doses at various distances from 5 to 30 km in NNW direction were also calculated, and good agreement was found between the values obtained in the present work and those given in the FSAR-2007 report. The largest difference was observed at 5-km distance (0.14 mSv in the first year).

The differences observed can be explained by the fact that two different codes were used including different parameters and assumptions. In addition, for the distances <5 km, the stability class C was assumed in the FSAR-2007 report, while in the present study, the stability class E was used, which is more conservative. Thus, for all distances >5 km for which the same stability classes were used, the results were very similar, while at distances <5 km, there were some pronounced differences. It is noted that the dose obtained within the 5 km circle is 0.475 mSv which does not exceed the 5 mSv dose limit for the first year post-accident, as discussed above.

The results of the probabilistic calculation show that at 2.5-km distance, for the worst atmospheric conditions (i.e., very stable class, low wind speed, and precipitation) and for an individual who stays under the plume for 1 year, the dose can be greater or equal to 1.1 mSv, while for 50 years with consideration of ingestion factors, it can be greater or equal to 9.6 mSv. The collective dose under these conditions is greater or equal to 56.1 man Sv.

The probabilistic calculations showed that for most atmospheric conditions, the wind direction is toward the E, ESE, and SE directions. This is in agreement with the predominant wind directions which are 270° and 315° from the north, as reported by Iran Meteorological Organization (Bushehr Meteorological Station Report 2008).

For an exposure of 365 days, individual thyroid doses were mainly due to inhalation, while whole body doses were mainly due to exposure from the ground shine.

Conclusions

In the present study, radiation doses received by the public living around the site of the BNPP-1 were estimated, due to an accidental release of radionuclides into the environment involving a hypothetical accident caused by the primary coolant leakage to the secondary circuit. Version 2.01 of the PC COSYMA code was applied for this purpose.

At first, assumptions similar to those made in the FSAR-2007 report were used, and individual doses due to inhalation of airborne radionuclides and due to external exposure from cloud and ground shine for the first year after the accident were calculated. The results obtained here are in good agreement with those in the FSAR-2007 report.

A probabilistic run was also performed for distances of 5, 10, 20, 30, 40, 60, and 80 km from the BNPP-1 site. As an input for this run, population and agricultural distributions around the site, hourly meteorological data, and releases of radionuclides as presented in the FSAR-2007 report were used. Cumulated individual doses for the first year and for 50 years (with consideration of the ingestion factor) after the accident and collective doses were estimated.

The results obtained in the present study showed that:

1. For the assumed accidental scenario, the doses to the public do not exceed 5 mSv for the first year after the accident at the considered distances from BNPP-1, and thus, no over-exposure of the population is expected if such a hypothetical accident would occur.
2. The results obtained here are in good agreement with the values reported in the FSAR report (FSAR 2007) in spite of the fact that two different computational codes and different sets of input data and assumptions have been used in the two studies.
3. The agreement of the results in this paper and also the agreement of the results of a parallel study on the radiological impact of normal BNPP-1 operation conditions on the public (Sohrabi et al. 2012) with those in the FSAR-2007 report suggest that the methodologies applied here are reasonable for such studies for the BNPP-1 in the future. It is therefore planned to apply the computational methodologies developed here to other severe hypothetical accident scenarios of the BNPP-1 such as a small and large break in the reactor coolant system as well as beyond design-basis accidents.

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References

Agricultural General Census (2003) Statistical Center of Iran

BNPP on-site Meteorological Station Report (2006–2007) Atomic Energy Organization of Iran

Bushehr Meteorological Station Report (2008) Data Processing Center, Iranian Meteorological Organization, Iran

Final Safety Analysis Report for BNPP (FSAR) (2007) Moscow

IAEA-EBP-WWER-01(1996) Guidelines for accident analysis of WWER nuclear power plants. IAEA, Vienna. Date of Issue: 1996-01-23

Margeanu S, Angelescu T (2003) Ingestion modeling in COSYMA. Romanian Rep Phys 55(3):224–229

NCRP (2009) Ionizing radiation exposure of the population of the United States, Report 160. National Council on Radiological Protection and Measurements

NRPB and FZK, Final Report (1996) PC COSYMA (Version 2) An accident consequence assessment package for use on a PC. EUR 16239 EN

Päsler-Sauer J, Jones J (2000) A probabilistic accident consequence uncertainty analysis of the atmospheric dispersion and deposition module in the COSYMA package. Radiat Prot Dosim 90:331–337

Population and Housing Census (2006) Statistical Center of Iran, Tehran, Iran

Sohrabi M, Parsouzi Z, Amrollahi R, Khamooshy C, Ghasemi M (2012) Public exposure from environmental release of radioactive material under normal operation of unit-1 Bushehr nuclear power plant. Ann Nucl Energy. doi:10.1016/j.anucene.2012.12.002