Development of A Framework for Establishment of Risk-informed Safety Goals for Nuclear Power Plants Operation in the UAE

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Abstract: A framework for establishment of risk-informed safety goals for nuclear power plants (NPPs) operations in the UAE was developed in this study. The current regulatory circumstance to the safety goals in the UAE was addressed as well. Representative parameters related to the core integrity (Level 1 PSA) and containment integrity (Level 2 PSA) are used as surrogate measures, Core Damage Frequency (CDF) for cancer (latent) fatality and Large Early Release Frequency (LERF) for early (prompt) fatality, for risk-informed safety goals. Under this framework a conservative evaluation of risk-informed safety goals was performed on the basis of conservative assumptions and data which were obtained and/or derived from the PSA results of APR-1400, the same type of the Barakah NPPs which are under construction in the UAE, and public health risk assessments. The current safety targets specified in the regulatory guideline (FANR-RG-004) in the UAE were examined to be appropriately determined with sufficient conservatism from the evaluation results. Limitations of the study and recommendations for appropriate applications of the risk-informed safety goals were provided as well.

Keywords: safety goal, quantitative health objective (QHO), probabilistic safety assessment (PSA)

1. INTRODUCTION

Basically requirements for safe design and operation of Nuclear Power Plants (NPPs) have been based on deterministic principles. Under legal requirements, they are incorporated in some forms of dose limits and performance criteria on the radiation barriers including fuel rods, reactor coolant system, and reactor containment building. In addition to these deterministic requirements based on defense-in-depth (DID) principle, quantitative health objectives (QHOs) for NPPs are or can be set up by either stipulating them in regal requirements or guidelines or policy statements. In the UAE there are regulations for NPP design, radiation dose limits, and the application of Probabilistic Risk Assessment (PRA, the same term as Probabilistic Safety Assessment (PSA)) requirements which should be reviewed for the establishment of risk-informed safety goal. Also regulatory guides regarding the evaluation criteria for probabilistic safety targets and design requirements are provided in a regulatory guidance (FANR-RG-004).

In this study, considering the regulatory circumstance, a framework for establishment of risk-informed safety goals for NPPs operation in the UAE was developed as shown in Figure 1. Risk-informed safety goals are defined in this study as quantitative safety targets evaluated on the basis of available risk insights. Generally QHO values limiting public health risks in the vicinity of the plant site cannot be directly used to check whether an NPP or NPPs satisfies them or not because the health risks cannot be obtained unless Level 3 PSA is performed. Therefore, it is convenient to define surrogate measures for risk-informed safety goals which correspond to QHOs and can be directly compared to the safety level of an NPP or NPPs such as CDF (Core Damage Frequency) and LERF (Large Early Release Frequency). Hence the CDF and the LERF are selected and used as surrogate measures for early (prompt) and cancer (latent) fatalities, respectively. The next step after the selection of surrogate measures for risk-informed safety goals is to determine quantitative values so that they can be used as the safety targets. General risks other than those due to NPPs operation are evaluated with statistical data such as accident and cancer fatalities to estimate the GHOs for early and cancer fatalities, respectively. Public health risks are assessed for the evaluation of conditional probabilities of early and cancer fatalities. Finally, CDF and LERF criteria as risk-informed safety goals are determined by comparing the QHO values and the conditional probabilities of early and cancer fatalities, respectively. Under this framework a conservative evaluation of risk-informed safety goals was performed on the basis of conservative assumptions and data which were obtained and/or derived from the PSA results of APR-1400, the same type of the Barakah NPPs which are under construction in the UAE, and public health risk assessments. Appropriateness of existing safety targets provided in regulatory guides in the UAE was examined by comparing those values with the risk-informed safety goals evaluated in this study.

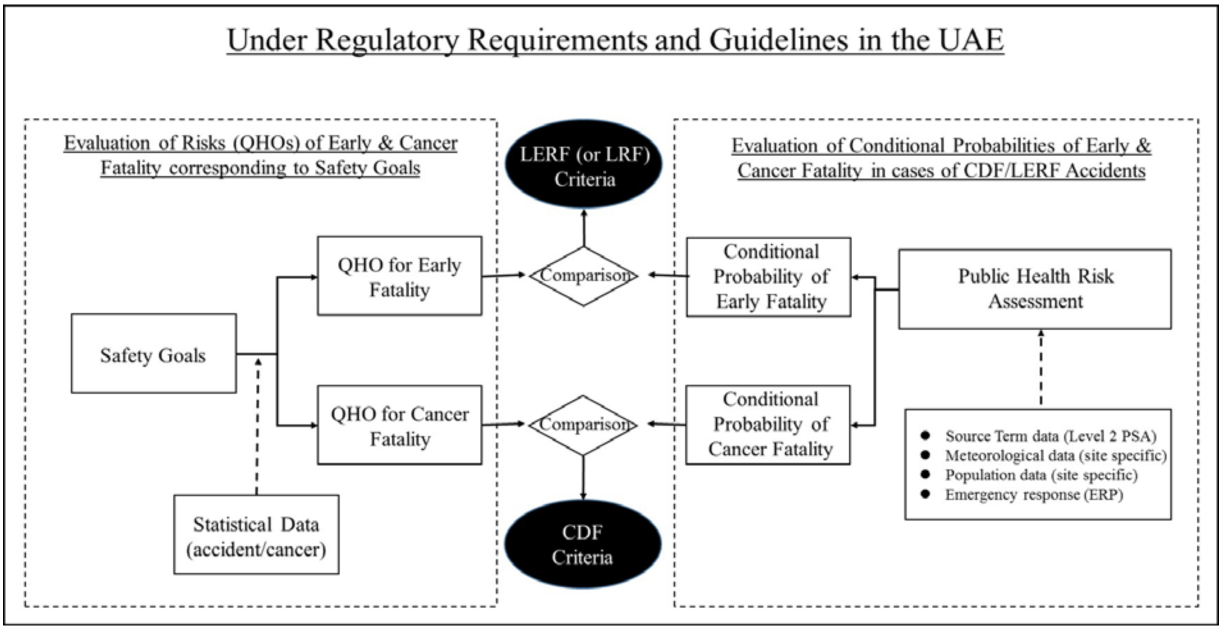

Figure 1: Framework for establishment of risk-informed safety goals

2. LEGAL REQUIREMENTS AND POLICIES TO SAFETY GOALS

2.1. UAE Current Situation

An analysis that carried out by official UAE entities and published in the *Policy of the United Arab Emirates on the Evaluation and Potential Development of Peaceful Nuclear Energy* in 2008 [1] has concluded that, the projections of the energy demand is expected to double in 2020. Based on this feasibility study, an evaluation has been done to find viable options, that capable to fit to the predicted energy demand with environmental and economic considerations. Nuclear power-generation was one of the competitive options that has been adopted. Nuclear policy has set an outlines of highest commitment to nuclear safety, security, and non-proliferation. The Federal Authority for Nuclear Regulation (FANR) as an independent regulator has been formally established with *Federal Law by Decree No 6 of 2009, Concerning the Peaceful Uses of Nuclear Energy* [2]. The federal law has set the responsibilities and requirements for regulating and licensing of nuclear sector in the UAE toward the peaceful purposes. The Emirates Nuclear Energy Corporation (ENEC) has been established with Abu Dhabi Law No. (21) of 2009. ENEC submitted a construction license application to FANR in December 2010, to construct the first two units (Barakah Units 1 and 2) of a nuclear facility at the western region of Abu Dhabi. The applicant submitted a Preliminary Safety Analysis Report (PSAR) which is based on the reference plant of the Shin-Kori Units 3 and 4 facility in Korea, and of the type of Korean APR1400 reactors. The construction license of units 1 and 2 of the Barakah nuclear facility and related regulated activities, is granted in July 2012 which specifies activities authorized and license conditions [3].

2.2. Regulation for Design Requirements

Part of the FANR activities and responsibilities is issuing regulations and regulations guidance, one of these essential regulations was the *Regulation for the Design of Nuclear Power Plants* (FANR-REG-03) [4], which is consistent with International Atomic Energy Agency (IAEA) safety requirements NS-R-1, "*Safety of Nuclear Power Plants: Design*" which superseded by new equivalent publication of IAEA SSR-2/1 [5]. The (FANR-REG-03) aims to establish the design requirements for Structures, Systems and Components (SSCs) important to safety and requirements for safety assessment for different plant states (operational states and incident/accident conditions). The requirements for safety assessment process state that, the complementary techniques of deterministic safety analysis and Probabilistic Risk Assessment (PRA) have to be included. In the safety analysis requirements (Article 44), stipulate "*It shall also be demonstrated that the nuclear facility as designed is capable of meeting any approved limits or criteria for radioactive releases and potential radiation doses for each category of plant operation and that defense-in-depth will be maintained*". In the principal technical requirements, (Article 7), it requires that "*in the design process, the defense-in-depth shall be incorporated to provide multiple physical barriers to the uncontrolled release of radioactive materials to the environment*"; and to provide safety margin to ensure maintaining of safe operation and preventing accidents. The defense-in-depth (Article 7.f), also aims to "*Provide multiple means for ensuring that each of the fundamental safety functions, i.e. control of the reactivity, heat removal, and the confinement of radioactive materials is performed, thereby ensuring the effectiveness of the barriers and mitigating the consequences of any Postulated Initiated Events (PIEs)*". The regulation of the design also include the requirements for safety classification, general design basis, PIEs, internal events and external events, site related characteristics, Design Basis Accidents (DBAs), severe accidents, and other design related requirements. In the part of severe accidents (Article 24), "*considerations shall be given to severe accidents by providing in the design reasonably practicable preventive and/or mitigative measures*". These events which have very low probability but could lead to the core degradation and release of radioactive materials have to be analyzed to provide correspondence preventive and/or mitigative measures.

2.3. Regulation of Radiation Dose Limits

The *Regulation for Radiation Dose Limits and Optimization of Radiation Protection for Nuclear Facilities* (FANR-REG-04) [6] aims to establish the radiation dose limits and the requirements for optimization of radiation protection that are relevant to a nuclear facility during its design, construction, normal operation and decommissioning. The limit for the effective dose during the operation of the nuclear facility does not exceed the dose limits as shown in Table 1:

Table 1: Dose Limits

Category	Worker who is Occupationally Exposed during the normal operation of a nuclear facility	Member of the public
Effective Dose	An average of 20 milli sieverts (mSv) per year averaged over a period of five years (100 mSv in 5 years), and 50 mSv in any one year.	1 mSv (this includes persons working in the nuclear facility other than those categorized under the Worker definition)
The annual Equivalent Dose in the lens of the eye	150 mSv, nor shall the annual Equivalent Dose exceed 500 mSv at any point on the hands, feet or skin.	15 mSv, nor shall the annual Equivalent Dose at any point on the skin exceed 50 mSv.

2.4. Regulation for the Application of Probabilistic Risk Assessment

The objective of the *Regulation for the Application of Probabilistic Risk Assessment (PRA) at Nuclear Facilities* (FANR-REG-05) [7] is to require the applicant or licensee constructing or operating a nuclear facility to conduct a high quality PRA to support the construction and operating licensing. The scope of the PRA has to be defined including internal and external events and all modes of plant operation. The regulatory guide (FANR-RG-004) [8] regarding the *Evaluation Criteria for*

Probabilistic Safety Targets and Design Requirements is defining the evaluation criteria the staff will use in assessing plant Safety assessments associated with probabilistic safety targets and the design requirements in FANR-REG-03.
The probabilistic safety targets – as indicated in the evaluation criteria of regulatory guide (Article 6) are as follows:

1. Core Damage Frequency (CDF) to < 10^{-5}/yr (mean value from the PRA considering internal and external events and all modes of Operation).
2. Overall Large Release Frequency (LRF) to < 10^{-6}/yr (mean value from the PRA considering internal and external events and all modes of Operation).

3. EVALUATION OF SAFETY GOALS UNDER THE FRAMEWORK

3.1. Risk-Informed Safety Goals

Performance measures for the safety goal must be chosen based on prevention and mitigation of core damage and radioactive material release. Representative parameters of facilities related to the core integrity (Level 1 PSA) and containment integrity (Level 2 PSA) can be used as performance measures. Two selected measures for the risk-informed safety goals are the CDF and the LERF. The CDF is defined as the frequency of an accident which can cause the fuel in the reactor to be damaged. The LERF is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects (NUREG/CR-6595) [9]. The CDF and LERF have been adopted as performance measures in similar studies performed for other countries [10].

3.2. Quantitative Health Objectives

Qualitative safety goals for securing safety due to NPPs operation were announced in the USA as follows [11]:

- *"Individual members of the public should be provided a level of protection from the consequences of NPP operation such that individuals bear no significant additional risk to life and health."*
- *"Societal risks to life and health from NPP operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks."*

The qualitative safety goals were supported by quantitative objectives (QHO: Quantitative Health Objectives) used in determining achievement of the qualitative safety goals as follows:

- *"The risk to an average individual in the vicinity of a NPP of prompt fatalities should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed."*

- *"The risk to the population in the area near a NPP of cancer fatalities that might result from NPP operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes."*

South-Korea also published *The Policy on Severe Accident of Nuclear Power Plants* in August, 2001 [12], which declared the similar safety goals to the USA's and recommended implementations of Probabilistic Safety Assessments (PSAs) for NPPs.
To develop the quantitative safety goals, risks from other causes than NPPs should be evaluated, which is called "general risk" hereafter. Annual death numbers of accident and cancer per 100,000 people were evaluated 28.7 and 59.2 on average, respectively from literature survey on death statistics

[13-15]. The corresponding general risks due to accident and cancer were calculated 2.87 x 10^{-4}/year and 5.92 x 10^{-4}/year, respectively. If the 0.1% rule from the QHOs is applied to the general risks, the values of 2.87 x 10^{-7}/year and 5.92 x 10^{-7}/year could be calculated as crude quantitative safety goals (QHOs) for early (prompt) and latent (cancer) fatalities due to NPPs operations, respectively.

3.3. Public Health Risks Assessments

Code: The WinMACCS code has been used to calculate plume dispersion and dose risk assessment. The code includes three modules of ATMOS, EARLY and CHRONC which evaluate atmospheric dispersion, emergency phase impact, and intermediate/long-term impact, respectively [16].

Source Term: The most important information in consequence analysis is source term data which can be released to the environment. The source term obtained from the Level 2 PSA of APR-1400 consists of seventeen source term categories (STCs). The accident set of the early containment failure includes 9 STCs and that of the late containment failure contains the other 6 STCs (see Table 4). In addition to the release quantity of radioactive material, the source term information includes the timing of the radioactive material release, the amount of energy associated, the release height and the predicted frequency of the release. 72 hours release duration after accidents was applied and the frequencies of internal accidents and external accidents including fire and earthquake were considered. For more conservative calculation, radioactive sources were assumed to be released from 0 m height and thermal plume rise was not considered. The radionuclides included in the analysis have been categorized according to their chemical properties. Table 2 shows nine categories of chemical group. The code library of radionuclides and decay chain data is "Indexr.dat" which can handle 825 kinds of radionuclides and six generation decay series. Database has been provided by the Radiation Shielding Information Center (1994) as a part of the FGR-DOSE/DLC-167 data package.

Meteorological Data: Hourly meteorological data of 2012's from National Center of Meteorology and Seismology (NCMS) have been sampled in the dispersion calculation. Due to the atmospheric stability class was not provided in the meteorological data from NCMS, it was derived from the categorization method of modified Pasquill stability classes [17], which is described in Table 3. The meteorological data consist of wind direction, wind speed, atmospheric stability, and accumulated precipitation. The ground surface roughness length representing the desert environment was applied as 0.03 cm to calculate some parameters used in plume dispersion modeling.

Population Distribution: The recent and projected population distributions near NPP site have been obtained from local government statistics available during the period of years 2001 through 2100. The 80km radius area around the plant was divided into sixteen directions that are equivalent to a standard navigational compass rosette. This rosette was further divided into 10 "inner" rings, each with sixteen azimuthal sections. The projected population in 2050 has been applied in this assessment considering the operating reactor and assuming high population density conservatively. Usually, the range of 1.6 km (1 mile) has been applied to calculate early fatality risk in Korea and USA. However, there is no resident within 2 km in the Barakah site and the area of 4 km distance from the center point of the NPP was considered by carrying out distance-population depended sensitivity analysis. The range of 8 km was applied to the evaluation of latent cancer fatality risk.

Countermeasures: As planned in the radiological emergency response plan, countermeasures include such activities as sheltering, evacuation, and dose-dependent relocation. Sheltering and evacuation have been excluded to consider more conservative situations. Dose-dependent relocation and KI ingestion were considered.

Dose Calculation: The WinMACCS code includes five pathways in early exposure scenario: (1) direct external exposure to radioactive material in the plume (cloudshine), (2) exposure from inhalation of radionuclides in the cloud (cloud inhalation), (3) exposure to radioactive material deposited on the ground (groundshine), (4) inhalation of resuspended material (resuspension inhalation), and (5) skin dose from material deposited on the skin [18]. Dose Conversion Factors (DCFs) in Federal Guidance Report 13 (FGR-13) issued by the Environmental Protection Agency (EPA) have been applied to the dose calculation [19].

Results: Early and cancer fatalities have been calculated and then corresponding risks have been obtained with population data. Table 4 shows the results of each STC and the averaged values.

Table 2: Chemical Group of Radionuclides

Chemical Group	Radionuclides
1. Inert Gases	KR-85, Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135
2. Iodine	I-131, I-132, I-133, I-134, I-135
3. Cesium	Rb-86, Cs-134, Cs-136, Cs-137
4. Tellurium	Sb-127, Sb-129, Te-127, Te-127m, Te-129, Te-129m, Te-132, Te-131m
5. Strontium	Sr-89, Sr-90, Sr-91, Sr-92
6. Ruthenium	Co-58, Co-60, Mo-99, Tc-99m, Ru-103, Ru-105, Ru-106, Rh-105
7. Lanthanum	Y-90, Y-91, Y-92, Y-93, Zr-95, Zr-97, Nb-95, La-140, La-141, La-142, Pr-143, Nd-147, Am-241, Cm-242, Cm-244
8. Cerium	Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu-240, Pu-241
9. Barium	Ba-139, Ba-140

Table 3: Modified Pasquill Stability Classes

Wind Speed [m/s]	Daytime Incoming Solar Radiation [W/m^2]				Within 1 hr before Sunset or after Sunrise	Nighttime Cloud Amount [oktas]		
	Strong (> 600)	Moderate (300-600)	Slight (< 300)	Overcast		0-3	4-7	8
≤ 2.0	A	A-B	B	C	D	F or G	F	D
2.0-3.0	A-B	B	C	C	D	F	E	D
3.0-5.0	B	B-C	C	C	D	E	D	D
5.0-6.0	C	C-D	D	D	D	D	D	D
> 6.0	C	D	D	D	D	D	D	D

Table 4: Number and Risk of Early and Cancer Fatalities

STC	Early/Late Containment Failure	Early Fatality (4 km)		Cancer Fatality (8 km)	
		Number (Mean)	Risk (Mean)	Number (Mean)	Risk (Mean)
1	Early	0.00E+00	0.00E+00	4.58E+00	3.34E-03
2	Early	1.23E+00	8.95E-04	2.05E+01	1.50E-02
3	Early	0.00E+00	0.00E+00	3.89E+00	2.84E-03
4	Early	0.00E+00	0.00E+00	3.53E+00	2.57E-03
5	Early	5.71E-03	4.16E-06	1.17E+01	8.56E-03
6	No Failure	0.00E+00	0.00E+00	4.99E-06	3.65E-09
7	No Failure	0.00E+00	0.00E+00	9.44E-05	6.89E-08
8	Late	0.00E+00	0.00E+00	1.84E-01	1.34E-04
9	Early	0.00E+00	0.00E+00	9.18E-01	6.70E-04
10	Early	0.00E+00	0.00E+00	1.96E+00	1.43E-03
11	Early	0.00E+00	0.00E+00	5.40E+00	3.94E-03
12	Early	0.00E+00	0.00E+00	2.06E+00	1.51E-03
13	Late	0.00E+00	0.00E+00	6.81E-02	4.97E-05
14	Late	0.00E+00	0.00E+00	1.94E-01	1.42E-04
15	Late	0.00E+00	0.00E+00	1.75E-03	1.28E-06
16	Late	0.00E+00	0.00E+00	1.77E-01	1.29E-04
17	Late	0.00E+00	0.00E+00	1.73E-03	1.27E-06
Average		***1.37E-01**	***9.99E-05**	****3.68E+00**	****2.69E-03**

*the value was averaged only for early containment failure cases
**the value was averaged only for both early and late containment failure cases

3.4. Development of Risk-Informed Safety Goal Criteria

Early risk: Individual Early Risk (IER) is calculated based on data and phenomena associated with the large early release of radioactive materials, as follows [20]:

$$IER = \sum_{n=1}^{N} LERF_n \times CPEF_n = \sum_{n=1}^{N} LERF_n \times \frac{EF_n}{TP} \tag{1}$$

where,

$LERF_n$: occurence frequency of source - term release category (STRC) - n

$CPEF_n$: conditional probability of early fatality near the site in the release case of STRC - n

EF_n : the number of fatality near the site in the release case of STRC - n

TP : the number of population near the site

$CPEF_{AVG}$ can be evaluated with weight-averaged $LERF_n$.

$$CPEF_{AVG} = \frac{\sum_{n=1}^{N} CPEF_n \times LERF_n}{\sum_{n=1}^{N} LERF_n} \tag{2}$$

Then, Equation (1) can be reduced into:

$$IER = CPEF_{AVG} \times \sum_{n=1}^{N} LERF_n = CPEF_{AVG} \times LERF \tag{3}$$

where,

$$LERF = \sum_{n=1}^{N} LERF_n$$

Cancer risk: Individual Latent Risk (ILR) is calculated based on data and phenomena associated with STRC, as follows [20]:

$$ILR = \sum_{n=1}^{N} F_n \times CPLF_n = \sum_{n=1}^{N} F_n \times \frac{LF_n}{TP} \tag{4}$$

where,

F_n : occurence frequency of source - term release category (STRC) - n

$CPLF_n$: conditional probability of latent fatality near the site in the release case of STRC - n

LF_n : the number of cancer fatality near the site in the release case of STRC - n

TP : the number of population near the site

$CPLF_{AVG}$ can be evaluated with weight-averaged F_n:

$$CPLF_{AVG} = \frac{\sum_{n=1}^{N} CPLF_n \times F_n}{\sum_{n=1}^{N} F_n} \tag{5}$$

Then, Equation (4) can be reduced into:

$$ILR = CPLF_{AVG} \times \sum_{n=1}^{N} F_n = CPLF_{AVG} \times CFF \tag{6}$$

where,

$$CFF \text{ (Conditional Failure Frequency)} = \sum_{n=1}^{N} F_n$$

In Equation (6), if it is conservatively assumed that all the radioactive materials released into containment are released into the environment, that is, CCFP (Conditional Containment Failure Probability)=1, the ILR can be evaluated as follows:

$$ILR = CPLF_{AVG} \times CDF \tag{7}$$

where,

CDF : Core Damage Frequency

The early and cancer risks to public in the UAE NPP site were estimated based on the Level 2 PSA results and site-specific data described in the previous section. WinMACCS code was used for the evaluation of conditional probability of early and cancer fatality (CPEF and CPLF). The CPEF and the CPLF are then compared with the QHOs for early and cancer risks to determine the corresponding performance criteria for CDF and LERF, respectively, as follows:

$$IER = CPEF_{AVG} \times LERF < QHO_{early} \tag{8}$$

$$ILR = CPLF_{AVG} \times CDF < QHO_{cancer} \tag{9}$$

If the crude QHO values of 2.87×10^{-7}/year for early fatality and 5.92×10^{-7}/year for cancer fatality calculated in Section 3.2 and the conservative values of 9.99×10^{-5} for $CPEF_{AVG}$ and 2.69×10^{-3} for $CPLF_{AVG}$ calculated in Section 3.3 were applied, the risk-informed safety goals for CDF and LERF should less than 2.20×10^{-4}/yr and 2.87×10^{-3}/yr, respectively.

3.5. Discussions on Risk-Informed Safety Goal Criteria

In this study, conservative assumption and data were used for the evaluation of the risk-informed safety goals as follows:

- Conservative source term assumptions: the release height, thermal plume rise (see Section 3.3)
- Assumed high population density near the plant
- Assumptions of no evacuation and sheltering for the countermeasures
- Assumption of CCFP=1: usually it is required that CCFP < 0.1, especially newly constructed plants as a requirement or recommendation.

The values of the CDF (< 10^{-5}/yr) and the LRF (< 10^{-6}/yr) as probabilistic safety targets in the UAE [8] had been set lower than those considered in other countries such as USA (CDF< 10^{-4}/yr and LERF< 10^{-5}/yr; for new plants LERF< 10^{-6}/yr) and South-Korea (CDF< 10^{-4}/yr and LERF< 10^{-5}/yr) [10]. The LRF includes LERF and LLRF (Large Late Release Frequency). The large late release doesn't contribute to the early fatality risk (the risk contribution to early fatality = 0) but contributes to the cancer fatality risk, as shown in Table 4. In the $CPEF_{AVG}$ calculation, averaging only for LERF cases makes more conservative results, which is the reason why the LERF is adopted as a surrogate measure for the early fatality risk instead of the LRF. In the APR-1400 PSA, the value of LERF was estimated almost same as that of LLRF. Hence the doubled value ($2\times2.87\times10^{-3}$/yr = 5.74×10^{-3}/yr) of the risk-informed safety goal for LERF can be used for the comparison with the safety target for LRF. Finally, the probabilistic safety targets for CDF (< 10^{-5}/yr) and LRF (< 10^{-6}/yr) in the UAE [8] are concluded to be set sufficiently lower in comparison with the conservatively evaluated criteria of the CDF (2.20×10^{-4}/yr) and the doubled LERF (5.74×10^{-3}/yr) in terms of engineering judgment.

4. LIMITATIONS AND RECOMMENDATIONS

4.1. Limitations of the study

Quality of PSA: Data and information from current PSAs for APR-1400 were used in this study. The PSAs were prepared for the licensing of Barakah NPPs as construction PSAs. A lot of generic data and assumptions were used for the PSAs. As living PSAs are available with the operation of the plant, this study should be updated with quality data and assumptions.

Uncertainty and Sensitivity Analysis: Probabilistic analyses are always associated with uncertainty. To cope with uncertainty, sufficient conservatism or high quality uncertainty and sensitivity analyses are widely adopted for safety analyses in nuclear industries. Even though this study is based on sufficient conservatism for determining the safety goals, a wide range of uncertainty and sensitivity analyses was not performed in this study. If a wide range of uncertainty and sensitivity analyses is performed in this framework, more reasonable evaluation can be accomplished, which is remained as a further study.

Statistical Data for General Risk Calculation: Data used for the calculation of general risk other than NPPs operation were obtained from literature and website surveys to competent authorities of relevance. However only several years' data were collected. More well-organized data base should be developed as a further study as well.

Data and Assumptions used in Public Health Risk Assessments: Usually public health risk assessments require a huge amount of data and assumptions some of which are directly collectable and others inferable from some rational calculations or logical approaches. US Nuclear Regulatory Commission (NRC) derives the atmospheric stability class by correlating with vertical temperature gradient but some required data were not available in this study. Instead, modified Pasquill stability classes were applied. NRC use 1.6 km (1 mile) for the consideration of early fatality. However, there is no resident in 1.6 km in the Barakah site so 4.0 km distance has been chosen to calculate early fatality by the sensitivity analysis of the population weighted distance. Evacuation and sheltering was not considered conservatively and ingestion scenario was excluded in this study because of absence of farmland, few residence areas, and the lifestyle near the Barakah site.

4.2. Recommendations for Application

Limits or Objectives: due to the uncertainty and insufficient analysis scope of PSA, the risk-informed safety goals should be recommended as objectives rather than strict limits, even though some countries which have sufficient PSA experience and technologies adopt the quantitative safety goals as regulatory limits.

Future NPPs and Multi-unit Site: to get public acceptance for future reactors, the increase in risk due to the addition of new NPPs should be as low as possible. Also there have been issues regarding multi-unit sites. The total risk of a multi-unit site is not always expressed as the numerical sum of all reactors. Hence, it is desirable for future NPPs and/or multiple modular NPPs to have lower safety goals compared to those of operating NPPs. The question on "how low it should be" should be studies in a comprehensive way considering type and number of NPPs to be constructed, constructions and operating experience, quality of PSA technologies applied, and so on.

Scope of Analysis: current PSA technologies have been evolved to quantify most of the risk factors reasonably. However, in reality, risks from external events or low power and shutdown states still have large uncertainties in their analyses. Therefore, consideration should be taken into account for level of analysis technology and uncertainties.

5. CONCLUSION

In this study, a framework for establishment of risk-informed safety goals for NPPs operations in the UAE was developed and under this framework a conservative evaluation of risk-informed safety goals was performed on the basis of conservative assumptions and data. The current safety targets specified in the regulatory guideline (FANR-RG-004) in the UAE were examined to be appropriately determined with sufficient conservatism from the evaluation results. However, considering the limitations coupled with insufficient quality and scope of PSAs considered, lacks of uncertainty and sensitivity analyses, insufficient statistical data for the general risk calculation, and data and assumptions in public health risk assessments, the conservative evaluation done in this study should be considered as a first attempt and further studies should be performed to generate more reasonable evaluation on risk-informed safety goals (or targets) under the developed framework.

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