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Revising the EPZ for IRIS

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INTRODUCTION

The International Reactor Innovative and Secure (IRIS) has been identified by the United States Department of Energy (DOE) Global Nuclear Energy Partnership (GNEP) program as the example of “grid appropriate reactors” or “exportable reactors” (Reference 1). Actually, IRIS has prompted significant interest and several countries are considering its eventual deployment. One of the IRIS attractive features is the possibility that its significantly enhanced safety features allow licensing with significantly modified requirements for emergency response, by significantly reducing the size of the evacuation zone close to the plant boundary (Reference 2). This is a very desirable feature for small countries with high population density and/or where co-generation, such as water desalination or district heating, is desired and proximity of the plant to the end user is therefore necessary. IRIS does intend to pursue Nuclear Regulatory Commission (NRC) licensing for Design Certification in the US under the framework of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50 (Reference 3)), thus modification of the emergency planning zone will not be part of the IRIS licensing (Reference 4). However, because of IRIS interest in worldwide deployment, a methodology has been developed which will allow significantly reducing, and possibly even eliminating, the Emergency Planning Zone (EPZ). Such methodology is described in this report.

A review and evaluation by the NRC is requested, such that it can be considered by foreign regulatory agencies in their independent evaluation of IRIS licensing.

Current regulations relevant for Nuclear Power Plants (NPPs) and their siting require “*that adequate protective measures can and will be taken in the event of a radiological emergency.*” This requirement has been further interpreted in the USA as a requirement for emergency planning in an area with a 10 miles radius, i.e., extending far beyond the site boundary. This prescribed 10 miles radius is the same for all NPPs. Historically, it was selected to very conservatively envelop all nuclear power plant designs existing at the time the rule was prepared, but it does not include provisions to account for the significant safety improvements in plant operation and design achieved since.

The emergency planning requirement may pose a significant burden on plant owner (utility), both in the construction and in the operation phase. During construction, it may be needed to build otherwise unnecessary infrastructure (e.g., enhanced highways) to comply with the requirement. During operation, it is necessary to maintain an evacuation capability in a relatively wide area around the plant in which, for all practical purposes, any human development is frozen, this being a burden for small countries and/or areas with significant growth. Finally the fact that the off-site zone around NPP is treated in a special way sends an incorrect message to public regarding the safety of NPPs and in the unlikely event of an accident it could even induce among residents of the affected areas the “paralyzing fatalism” that is recognized to be the largest and longest lasting public health problem created by the Chernobyl accident (Reference 5).

The current advanced and safer reactor designs further reduce risk to public, and should therefore offer a possibility to reduce or eliminate some of the emergency plan and evacuation requirements. This need was identified by the International Atomic Energy Agency (IAEA) International Project On Innovative Nuclear Reactor and Fuel Cycles (INPRO), User Requirement “*The innovative nuclear reactors and fuel cycle shall not need relocation or evacuation measures outside the plant site, apart from those generic emergency measures developed for any industrial facility,*” (Reference 6) as well as by the Generation IV International Forum (GIF) (Reference 7). It is deemed possible to reduce emergency-related site requirements for advanced plants, while at the same time providing a protection to the general public equal to or better than that provided by the current generation of NPPs and current regulations.

Achieving licensing with this new objective would offer significant societal and economic benefits to the general public and plant owners/operators, including:

- Increased public acceptance of nuclear power, since nuclear plants will be treated as any other power plant;
- Reduced need for infrastructure, thus reducing cost;
- Reduced operational costs;
- Enabling co-generation, including district heating and desalination, where the plant cannot be located remotely from the intended user;
- Reduced transmission costs;
- Enabling wider choice of siting locations in countries with relatively high population density.

In summary, achieving licensing with no need for emergency plan and evacuation measures will improve viability of deploying new NPPs.

During the past 30 years, the possibility of reducing or eliminating the EPZ has been the subject of various efforts. In 1985 and 1986, the licensees for the Calvert Cliff and Seabrook power stations petitioned the NRC for a reduction of the EPZ requirement; these petitions were rejected on programmatic and technical grounds (Reference 8). During the following 20 years the NRC issued various papers dealing with the definition of emergency planning procedures for evolutionary reactors. Studies from Electric Power Research Institute (EPRI) and Nuclear Energy Institute (NEI) have directly addressed this issue. In TR-113509 (Reference 9), EPRI documented a set of probabilistic and deterministic studies to show that a significant reduction in the EPZ radius was possible for Advanced Light Water Reactors (ALWR) while maintaining the same level of safety of existing plants. In NEI-02-02 (Reference 10), NEI proposed a complete risk-informed revision to 10 CFR Part 50, in which the need and extension of the EPZ would be based on purely probabilistic considerations. Outside the U.S., similar activities have been ongoing: just to make an example, a social and technical study on the possibility of reducing the EPZ for the APR-1400 (Reference 11) was performed in Korea.

Lately, IAEA/INPRO and GIF have identified the need for licensing that would allow reducing emergency planning requirements.

The premise of this suggested approach is that there will be no change in the defense-in-depth philosophy. The total level of defense in depth will be the same or higher than currently. However, depending on the particular design characteristics the relative contribution of the various levels can be changed; being the Emergency Planning one of such levels it can be decreased in importance or substituted by another level, provided that there is no compromise whatsoever on the total overall level of defense.

The methodology presented here will combine probabilistic, deterministic, and risk management methods that would support licensing with reduced emergency planning requirements. It is articulated over the following four steps:

1. Review the licensing regulations which specified the emergency response planning for the current U.S. Light Water Reactor (LWR) plants.

2. Based on the lessons learned by previous attempts at EPZ redefinition, identify changes in the licensing approach and devise technical criteria which would be necessary if the emergency planning is to be eliminated or reduced

3. Develop an integrated methodology based on combination of the deterministic, probabilistic and risk management approach, which would enable consistent evaluation of advanced reactors, giving credit to their enhanced safety features.

4. Apply the methodology to the IRIS design.

1 REVIEW OF CURRENT LICENSING REGULATIONS CONCERNING EMERGENCY PLANNING ZONE DEFINITION

The historical perspective provided in this report will focus mostly on the U.S. situation, although some considerations on the international environment are reported to contribute at providing a technically consistent basis for defining the emergency planning requirements of nuclear installations.

1.1 EARLY REGULATION IN THE UNITED STATES

The first mention of the need to protect resident population against nuclear accident consequences dates from the 1954 Atomic Energy Act (AEA) (Reference 12), which is the foundation of what eventually became the U.S. Nuclear Regulatory Commission's regulations for commercial nuclear power plants. In this act, the U.S. Atomic Energy Commission (AEC) established its regulatory requirements for commercial nuclear power plants to ensure that "*no undue risk to public health and safety*" resulted from the licensed use of nuclear power plants. The safety features were designed to satisfy specified dose limits at various site-depending distances from the plant. As such dose limits were observed, there was no compelling reason to subject the population to evacuation procedures.

This type of general approach was followed until the early '60's when the Atomic Energy Commission published the Reactor Site Criteria (10 CFR Part 100 (Reference 13)) (1962). This regulation mentioned, as an aid in evaluating a proposed site, the duty for each applicant to "*assume a fission product release from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an Exclusion Area (EA), a Low Population Zone (LPZ) and population center distance.*" These three concepts were defined as follow:

- The EA size is such that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

- The LPZ size is such that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

- The population center distance is at least one and one-third times the distance from the reactor to the outer boundary of the LPZ. In applying 10 CFR Part 100, the boundary of the population center shall be determined upon consideration of population distribution.

It has to be noted that the mentioned subdivision of the territory surrounding the plant, and also the dose limits specified, were for licensing purposes, and for design of the safety features. In fact, the accidents examined to calculate the doses were essentially Design Basis Accidents (DBA), which determine the engineered safety features design. Moreover, being implicitly defined with the doses, the extensions of these areas are site-depending, and so they could in principle vary from plant to plant.

In the years following publication of 10 CFR Part 100, three major items would eventually contribute to define the current emergency planning approach:

1. The involvement of local, State and Federal governments in the emergency planning preparedness.

2. The expansion of the accidents considered spectrum to event more severe than DBA.

3. The introduction of the Protective Action Guide (PAG) and the Ingestion Exposure Pathway (IEP).

They are discussed below.

Off-site Agencies Involvement

In 1970, explicit requirements for plans to cope with emergencies were published in 10 CFR Part 50, (Reference 3), Appendix E. In accordance with provisions of the Atomic Energy Act of 1954, these requirements were directed to applicants for licenses to operate these facilities rather than to State or local governments. With respect to a planning basis, federal regulation 10 CFR Part 50, Appendix E not only did not provide explicit guidance as to the character or magnitude of accidental release to the environment, but also did not include any explicit references to the LPZ or other particular geographical areas other than "*within and outside the site boundary.*"

Despite these omissions, the publishing of 10 CFR Part 50, behaved as an incentive for a further approach to the current regulations. In fact, since previous guidance regarded just the licensee, it was recognized that local and State, as well as Federal authorities would have to be actually involved in the offsite response. Regulations required licensees to incorporate provisions for participation by offsite authorities or organizations whose assistance might be required, in the event of a radiological emergency, in periodic drills to test response plans. As the federal regulatory staff gained experience with these requirements, it became concerned with the abilities of State and local governments to discharge their responsibilities should the need ever arise. This concern in part gave rise to a Federal Register Notice (Reference 14) which started an interagency program for providing radiological emergency response planning guidance and related training to State and local government organizations. NRC exercised the lead role in this activity and several Federal Agencies, including the Environmental Protection Agency (EPA), participate.

Expansion of the Accident Considered Spectrum

Another substantial modification was the introduction in 1974 of the concept of "Magnitude of the Accident" in the emergency planning based criteria. In NUREG-75/111 (Reference 15) NRC stated: "*the evaluation of sites and plant designs, required testing programs, and quality assurance for the operation of such facilities all provide substantial assurance that accidents with serious consequences to the public health and safety are not likely to occur. Nevertheless, highly unlikely sequences of events are postulated and their potential consequences analyzed by the applicant in the Safety Analysis Report which accompanies each application and by the staff in its Safety Evaluation Report for each plant. The (NRC) considers that it is reasonable, for purposes of emergency planning relative to nuclear facilities, to prepare for the potential consequences of accidents of severity up to and including the most serious* *design basis accident analyzed for siting purposes."…"The (NRC) recognizes that accidents with more severe potential consequences than design basis accidents can be hypothesized. However, the probability of such accidents is exceedingly low. Emergency plans properly designed to cope with design basis accidents would also provide significant protection against more severe accidents, since such plans provide for all of the major elements and functions of emergency preparedness. An added element of confidence can be gained, however, if States and local governments assure that their plans for responding to radiological emergencies are coordinated with their plans for dealing with floods, earthquakes, or other disaster situations which might necessitate large scale displacement of people and the provision of shelter, food, medical aid, and other emergency services. Communications, traffic control, evacuation, public notification and other emergency responses will tend to be the same whether or not the emergency involves radiological considerations. The (Department of Energy's) Radiological Assistance Program (RAP), the Federal Interagency Radiological Assistance Plan (IRAP) and other Radiological Emergency Assistance Plans, which are a part of the Federal capability, provide significant additional emergency resources in the event of a serious accident.*"

PAG Limits and IEP Introduction

The dose guideline values in 10 CFR Part 100 did not constitute acceptable limits for emergency doses to the public under accident conditions, essentially for two reasons:

1. The numerical values of 25 rem whole body and 300 rem thyroid can be considered the values above which prevention of serious health effects would be the paramount concern. Good health physics practice would indicate that radiological exposures of these magnitudes should not be allowed to take place if reasonable and practical measures can prevent such exposures.

2. The assumptions on which design basis accidents studies are based are extremely conservative. Thus, the actual doses that would result from DBA are expected to be much lower than the afore-mentioned dose guidelines of 10 CFR Part 100, under most meteorological conditions. Since the NRC expressed the will to analyze a wider accident spectrum, including the most adverse meteorological conditions being present during the DBA, the resulting dose would exceed 10 CFR Part 100 dose guidelines.

Therefore, with respect to the levels at which emergency actions should be initiated, in 1975 the EPA issued, as an Agency guidance, portions of the "manual of Protective Action Guides (Reference 16) (PAG) and protective Actions for Nuclear Incidents" in order to assist public health and other governmental authorities in deciding how much of a radiation hazard to the environment constitutes a basis for initiating emergency protective actions. These guides (PAGs) are expressed in units of radiation dose (rem) and represent trigger or initiation levels, which warrant pre-selected protective actions for the public if the projected (future) dose received by an individual in the absence of a protective action exceeds the PAG. The nature of PAGs is such that they cannot be used to assure that a given level of exposure to individuals in the population is prevented; in any particular response situation, a range of doses may be experienced, principally depending on the distance from the point of release. Some of these doses may be well in excess of the PAG levels and clearly warrant the initiation of feasible protective actions. This does not mean, however, that doses above PAG levels can be prevented, or that emergency response plans should have the prevention of exposures above PAG levels as their objective.

PAGs represent only trigger levels and are not intended to represent acceptable dose levels; they are tools to be used as a decision aid in the actual response.

The PAGs are numerically defined on account of the release pathway considered: in fact, radioactive nuclides tend to contaminate the environment and the population through two major pathways:

1. The Plume Exposure Pathway (PEP): it relates to the whole body external exposure to gamma radiation from the plume and from deposited material, and for inhalation exposure from the passing radioactive plume. The time of potential exposure could range from hours to days.

2. The Ingestion Exposure Pathway (IEP): it relates to the contamination due to ingestion of contaminated foods. The time of potential exposure could range from hours to months.

PAGs referred to the PEP are expressed as a range of 1 to 5 rem whole body dose and 5 to 25 rem thyroid dose; PAGs referred to the IEP are expressed as 1.5 rem infant thyroid dose from cow's milk. These values had been set at levels below those that would produce detectable short term biological effects and at levels that would minimize long term biological effects.

1.2 THE EPZ CONCEPT IN THE UNITED STATES

Except for minor modifications, the requirements concerning the subdivision of the territory and the protection of the public were essentially unchanged till 1976, when an ad hoc Task Force of the Conference of (State) Radiation Control Program Directors passed a resolution requesting NRC to "*make a determination of the most severe accident basis for which radiological emergency response plans should be developed by offsite agencies.*"

In November 1976, a Task Force consisting of NRC and EPA representatives was convened to address this Conference request and related issues (Reference 17). Considering the three items described in Section 1.1 (i.e., involvement of local, State and Federal government, definition of accident spectrum and PAG), already implemented in separate documents, the Task Force started a deep study in order to combine these approaches and compile a unique document. The result was the publication of NUREG-0396: "Planning Basis for the Development of State and Local Government Radiological Emergency response Plans in Support of Light water Nuclear Power Plants" (Reference 18), which represented the basis of the current EPZ concept.

The NUREG-0396 main purpose was to "*provide a basis for Federal, State and local government emergency preparedness organizations to determine the appropriate degree of emergency response planning efforts in the environs of nuclear power plants." That guidance stated also that "the Task Force hopes that the guidance provided here will be used to supplement the extensive emergency planning guidance already published by NRC and EPA.*"

The Task Force took into consideration the improvements that had been suggested by previous studies supplemented by new approaches and concepts. Major recommendations of the Task Force were:

- An enlarged accidents spectrum: as mentioned before, the study was performed taking into account not only the DBA, but also more severe accidents, called class 9 events. This term refers to those accidents leading to total core melt and consequent degradation of the containment boundary and those leading to gross fuel cladding failure or partial melt with independent failures of the containment boundary. The decision to take these accidents into account came after an extended debate to establish the rationale on which the emergency planning based criteria had to be founded (i.e., on which criteria accidents had to be considered and excluded). Four were the possible choices: risk, probability, cost-effectiveness and consequences:

  - RISK: this rationale would have seemingly given a uniform basis for emergency planning, establishing a consistent basis for the emergency planning of nuclear and non-nuclear hazards. However, emergency planning for non-nuclear hazards is not based upon quantified risk analyses since risk is not generally thought in terms of probabilities and consequences, rather it is an intuitive feeling of the threat posed to the public. Moreover, nuclear power plants are unique in the fact that radiation tends to be perceived as more dangerous than other hazards, and therefore a simple mathematical risk concept would not have been accepted by the public

  - PROBABILITY: by founding the methodology on this rationale, an accident probability below which development of an emergency plan could not be justified would have been calculated. This concept is however against the public tolerance, since society tolerates much more probable non-nuclear events with similar consequences spectrum. For this reason, the probability rationale was rejected

  - COST-EFFECTIVENESS: this rationale would have implied the need to calculate what it costs to develop different levels of such a plan and the potential consequences that could be averted by that degree of development. However, absent an actual accident, it would have been very difficult to assign a dollar value to the effectiveness of the plan in terms of health effects averted: this was the main reason this rationale was rejected.

  - CONSEQUENCES: Basing the methodology on this rationale would have the great advantage, especially from the point of view of public acceptance, of being able to establish bounds on the planning effort. Further, a planning basis would have been easily stated and understood in terms of the areas, time frames and radiological characteristics that would have corresponded to the consequences from a range of possible accidents. For these reasons, consequences were chosen as the rationale basis; at the same time, in order not to consider too unlikely events, very serious consequences were tempered by probability considerations.

- The PAG limits: the Task Force accepted the principle in existing NRC and EPA guidance that acceptable values for emergency doses to the public under the actual conditions of a nuclear accident cannot be predetermined. The emergency actions taken in any individual case have to be based on the actual conditions that exist and are projected at the time of an accident. For very serious accidents, predetermined protective actions would be taken if projected doses, at any place and time during an actual accident, appeared to be at or above the applicable proposed PAGs, based on information readily available in the reactor control room, i.e., at predetermined emergency action levels. Of course, ad hoc actions, based on plant or environmental measurements, could be taken at any time.

- Two different exposure pathways: the Task Force provided separate guidance for the two exposure pathways (PEP and IEP), although a single emergency plan would have included elements common to assessing or taking protective actions for both pathways.

The whole study provided the concept of Emergency Planning Zone, designated as the area for which planning is recommended to assure that prompt and effective actions can be taken to protect the public in the event of an accident. Consistently with the coexistence of two different pathways characterized by different physic mechanisms of exposure, the EPZ is actually constituted by two areas:

- PEP area: 10 miles radius
- IEP area: 50 miles radius

The extensions of these two areas are sufficient to provide acceptable doses to the population in areas where the projected dose from design basis accidents could be expected to exceed the applicable PAGs under unfavorable atmospheric conditions. Three were the main dose criteria on which the determination of the radius was based:

- Criterion 1 – the consequences of DBA would not exceed the PAG levels outside the recommended EPZ distance;

- Criterion 2 – the consequences of less severe Class 9 accidents would not exceed the PAG levels outside the recommended EPZ distance;

- Criterion 3 – the EPZ should be of sufficient size to provide for substantial reduction in early severe health effects (injures or death) in the event of the more severe Class 9 accidents.

According to the need to make the emergency planning not site-dependent, the sizes of the two zones meet all the three criteria for any given nuclear power plant.

1.3 OTHER NATIONAL APPROACHES TO EMERGENCY PLANNING

Confirming the actuality of the EPZ subject, a seminar on Emergency & Risk Zoning around Nuclear Power Plants was held on 2005 (Reference 19) at the premises of the European Commission, Joint Research Center (JRC). Its purpose was to provide a forum for presentation and discussion of the status of emergency planning and risk assessment approaches, safety policies, as well as current and possible future requirements for emergency and risk zoning, and to consider needs for international harmonization. The aim was to help relevant stakeholders at both national and international levels to decide on the relevance of this issue on current related research and development needs.

The JRC-organized event provided a precious source of information in order to get an overall view of the probabilistic/deterministic information sources used to define risk and emergency zones around nuclear power plants in various countries. The approach to emergency planning in the European Union (EU) member States is strongly deterministic and significant differences are present among different countries. The usual approach is that a set of reference accidents have been defined (usually limited to the DBA) which is then used as the basis for setting up the emergency plans. The main disadvantage of the deterministic approach is that it analyzes only the DBA, i.e., the worst credible accident. In fact, evaluation of hazards should never be limited to the selected reference accident but should always include the complete spectrum of potential occurrences since each accident occurs differently and produces different consequences.

A non-exhaustive description of the EPZ requirements and basis in some countries is reported in the following.

Belgium

The general EPZs are associated with the following protective actions: evacuation (10 km), sheltering (10 km), stable iodine intake (20 km) and food chain control (whole country). The size of these zones has been defined taking into account a rough (presumably largely deterministic) estimation of the associated risks.

Czech Republic

The predetermined evacuation of people is performed within a 5 km internal zone around Temelín NPP and within a 10 km internal zone around Dukovany NPP. The EPZ is formed by a territory 20 km around Dukovany and 13 km around Temelín. The predetermined actions are sheltering and distribution of iodine tablets. The difference between the EPZ for Temelín and for Dukovany is mainly due to different population densities, meteorological and evacuation conditions. Czech Republic is one of the few EU Member States that focused on the usage of Level 2 Probabilistic Risk Assessment (PRA) results in a formal way as input into their emergency arrangements. Based on the Level 2 PRA results for Temelín, it was found that no sequences identified in the PRA have more serious consequences than those sequences used as a basis for the (deterministic) definition of the EPZ size. This information was used as a confirmation that the reference accident sequences had been reasonably selected.

Finland

Finnish NPP sites are surrounded by a protective zone extending to about a five kilometer’s distance from facilities within which land use restrictions are enforced. In accordance with a Ministry of the Interior Order, a nuclear power plant is surrounded by an emergency planning zone extending up to about 20 kilometers from the facility. Such a zone shall be covered by detailed rescue plans for public protection drawn up by the authorities and may not contain such populations or population centers as would render impossible the efficient implementation of rescue measures applicable to them.

Design basis for these zones was reviewed in 1984 in a statement by the Finnish Radiation and Nuclear Safety Authority to the central Rescue Service Authority (Ministry of Interior). State-of-the-art of the severe accident management and source term in an accident situation were considered. A description of possible accident scenarios and times, radioactive substances behavior at the plant and release phenomena were drawn into a perspective of possible protective actions.

France

In the French system, two emergency planning zones of 5 kilometers and 10 kilometers radii around a nuclear power plant have been defined. The emergency planning zone of 5 kilometers radius around a nuclear power plant is the zone where evacuation is pre-planned and prepared in detail. The emergency planning zone of 10 kilometers radius around a nuclear power plant is the zone where sheltering is pre-planned. Stable iodine tablets have been previously distributed in France to the population within a radius of 10 kilometers around a nuclear power plant. The two emergency planning zones provide reasonable assurance that the doses to the population in the short term would be below the different intervention levels for a spectrum of accidents and radionuclide releases, in particular for most core melt accidents. Another important consideration is that 5 and 10 kilometers are practicable distances for planning in France. There is obviously a need for flexibility in response following an accident and it is therefore essential that emergency plans take into account site-specific factors (e.g., the type of housing, whether the planned countermeasures would involve institutions such as schools and hospitals…). It is also recognized that protective actions could be extended beyond 10 kilometers if conditions warrant. Much more time would be available for emergency response beyond these distances.

As far as the defining criteria are concerned, it is recognized that the emergency plans must be able to respond effectively to accidents liable to occur at a NPP. This resulted in the definition of accident scenarios encompassing the possible consequences, with a view to determining the nature and extent of the remedial means required. The task of identifying a suitable accident scenario is difficult, since cases of real significant accidents are extremely rare, with the result being that a conservative theoretical approach is usually adopted to estimate the source terms (i.e., the quantities of radioactive materials released), calculate dispersion in the environment and finally assess the radiological impact. In summary, the definition of emergency planning zones is based on reference source terms and does not include probabilistic approach. The approach supposes nevertheless that accident scenarios, leading to more serious off-site consequences than the reference source terms, are highly improbable. Finally, in France, for the moment, level 2 PRAs do not appear as a tool for direct improvement of emergency zoning but as a tool to ensure the sufficiency of the provisions taken for population protection for a wide spectrum of severe accident situations.

Lithuania

The Lithuanian emergency plan provides means of protecting the population, their scope, terms, assignment of responsibilities, and implementation procedure. The plan is needed for organization and co-ordination of actions taken over by ministries, other State Administration institutions, county and local municipal authorities for taking protective measures for arrangement of immediate response actions, for the operative notification of neighboring countries of a nuclear accident or radiological emergency. For the accident types, emergency response takes place over two distinct areas: a sanitary protection zone and the area beyond the sanitary protection zone. Sanitary Protection Zone (SPZ) means the area surrounding the facility, which is under the immediate control of NPP in Lithuania. The area beyond the sanitary protection zone is divided into three main zones: Precautionary Action Zone (PAZ), Urgent Protective action planning Zone (UPZ), and longer-term protective action planning zone. The PAZ goal is to substantially reduce the risk of deterministic health effects of ionizing radiation before radionuclides emission into the environment. UPZ means a predefined area around the facility where a plan for urgent protective measures is made in advance. Longer term protective action planning zone means a predefined area around the facility farthest from the facility and including the urgent protective action planning zone. In this area the actions to reduce the long-term doses from deposition and ingestion should be developed in advance. These zones are roughly circular areas with NPP in the centre. The size of the zones in Lithuania has been determined by an analysis of international practices.

Slovak Republic

NPPs in the Slovak Republic are of “Soviet” VVER type; it was required by the Soviet project that NPP site should have had a three kilometers (radius) protection zone without any permanent settlement. These zones were demarcated by county hygienic-in-chief according to the territory features. Besides this three km protection zone, emergency planning zones are also defined for Slovak NPPs in relation to the maximum size of any radiation emergency that can be reasonably foreseen. The hazard area represents a circle with centre in the nuclear facility and further divided into 16 sectors (of 22.5° each). The radius is NPP-specific and ranges from the 20 km for Mochovce up to the 30 km for Bohunice, both divided into zones with radius 5 km and 10 km. In case that the boundary demarcating the hazard area interferes with an inhabited area, the whole inhabited area is considered as a hazard area. The difference in the EPZ for Bohunice and Mochovce is due to different population density, meteorology and evacuation conditions and also due to the older type of VVER reactor present at the Bohunice site.

Both probabilistic and deterministic safety analyses are applied in support of zoning, emergency preparedness and planning. Results of PRA level-2 are used for identification and selection of accident scenarios resulted in the core (fuel) damage, containment failures and release of radioactive material into environment. Deterministic safety analyses are used for the source term prediction and quantification of radioactive releases, distribution of radioactive materials in the environment, and dose calculations.

Switzerland

Following the new legal provisions introduced in 1998, protection for the public is considered in Switzerland for expected doses above 1 mSv. The current provision in this country envisions two general areas: an inner zone of around 5 km of radius, and an outer zone up to 20 km in radius. A third zone covers for the entire extension of the country.

The technical basis for the definition of the EPZ are focused on three reference scenarios assuming 100% of noble gases and 10% (reduced to 1% in 1991) of halogens released. Acceptance criteria for such reference scenarios are related to the maximum reasonable coverage of the total amount of sequences envisioned in PRA analysis, with an allowed uncovered cumulative frequency of rapid sequences or very large releases lower than 10^{-6}/yr. Different weather conditions are considered and whole body as well as thyroid doses are evaluated.

United Kingdom

For each nuclear licensed site in the UK there is a defined zone around the site – the Detailed Emergency Planning Zone (DEPZ) within which the arrangements to protect the public are planned in detail. The boundary of this zone is defined in relation to the maximum size of any radiation emergency that can be reasonably foreseen and ranges from 1 to 5 km. It is also recognized that radiation emergencies could occur that would have consequences beyond the DEPZ. The nature of the response required is more difficult to predict and will depend on a number of factors such as the characteristics of the release that has occurred and the prevailing weather conditions. To deal with this, there is a requirement that the emergency plans incorporate arrangements for "extendibility" beyond the DEPZ.

Japan

The agreement between government, municipalities and utilities regarding off-site protection measure for nuclear protection in Japan was in principle defined in 1961 in the more general framework of the Disaster Prevention Measures Law which was mainly focused on natural events such as earthquake, typhoon, etc… In June 1980, after the 1979 Three Mile Island (TMI) accident, the nuclear safety commission of Japan issued its position specifically about nuclear disaster prevention guidelines. The technical background for the definition of the nuclear disaster responses was inspired by outcome of the TMI accident, which is currently the most serious accident involving a LWR and, which, for the first time, required off-site emergency response. Since then, the reference document was regularly updated, incorporating the latest scientific knowledge. These guidelines suggest dose indicators to determine whether shelter or evacuation is required. An ad-hoc Japanese Nuclear Disaster Special Measures Law was finally issued after the 1999 uranium-processing JCO plant accident.

The EPZ is about 8 to 10 km for the facilities of commercial plants and research reactors with power levels greater than 50 MWt. The standard of EPZ is the zone with a boundary (distance from the nuclear facilities) defined to be within the lower limit of radiation exposure at the boundary (10 mSv to whole body dose and 100 mSv to thyroid).

2 CRITICAL REVIEW OF PREVIOUS ATTEMPTS OF EPZ REDEFINITION IN THE U.S.A. AND RESULTING INSIGHTS

The possibility of the EPZ reduction for advanced plants derived from two considerations:

- The very high level of safety characteristics of the Advanced Light Water Reactors (ALWR) versus the old plants;

- The fact that the prescribed emergency planning is not based upon quantified probabilities of accidents but on public perceptions of the problem and what could be done to protect health and safety: in essence, it is a matter of prudence rather than necessity.

Using simply a rational approach, these considerations would be sufficient justification, since it would be a simple matter of consistency to adjust emergency planning in relation to the type of plants being licensed. However, considering the activists opposition to nuclear power, a more prudent approach is also in order.

Following is a discussion of a number of studies dealing with ALWRs and EPZ. For reasons of completeness, first there is a brief mention of two unsuccessful requests for traditional LWRs.

2.1 REQUESTS BY LICENSEES

Two licensees petitioned early the NRC to allow for a reduction in the size of the EPZ. In 1985, the licensee for Calvert Cliffs requested exemptions and license amendments to allow for reduction in the 10-mile EPZ to 2 miles and, in 1986, applicants for the Seabrook nuclear power plant requested a waiver to allow for reduction in the 10-mile EPZ to 1 mile. The technical argument supporting these requests was that a site-specific analysis of design basis and severe-accident risks showed a decrease in these risks relative to the risks considered in NUREG-0396. In regard to the Calvert Cliffs exemption request, the NRC staff concluded that it could not consider the request because the NRC was still studying severe-accident issues (April 11, 1988, letter from S. Varga (NRC) to J. Tiernan (BG&E)). In regard to the Seabrook petition, the Atomic Safety and Licensing Board concluded that "there are a number of areas wherein it appears the Applicants had not presented full and complete results sufficient to inspire confidence that their motion deserves further consideration at this time" (ASLBP 82-471-02-02).

2.2 SECY-93-092

In the SECY-93-092 (Reference 20) document, dated April 8, 1993, the staff raised the issue: "Should advanced reactors with passive advanced design safety features be able to reduce emergency planning zones and requirements?" The staff proposed no changes to existing regulations governing Emergency Planning (EP) for advanced reactors at that time; however, it did indicate that regulatory direction would be provided at or before the start of the design certification phase so that EP implications on design could be addressed.

2.3 MINOR DOCUMENTS (1993-1995)

In a Staff Requirement Memorandum (SRM) dated July 30, 1993 (Reference 21), the Commission stated that "it is premature to reach a conclusion on emergency planning for advanced reactors.... However, the staff should remain open to suggestions to simplify the emergency planning requirements for reactors that are designed with greater safety margins. To that end, the staff should submit to the Commission recommendations for proposed technical criteria and methods to use to justify simplification of existing emergency planning requirements." The Commission further stated that "work on EP should be closely correlated with work on Accident Evaluation and Source Term, in order to avoid unnecessary conservatism. Also, the work on EP for advanced reactors should be coordinated with the approach for evolutionary and passive advanced reactors."

In response to that SRM, the staff stated in a memorandum to the Commission, dated December 22, 1993, that it would reexamine the technical basis for EP and would develop recommendations for possible simplification of EP requirements for reactors with greater safety margins.

In a memorandum of February 27, 1995, the staff informed the Commission of the progress of staff efforts to develop recommendations for possible simplification of EP requirements for reactor designs with greater safety margins. In that memorandum, the staff indicated that because design certifications of advanced reactors such as PRISM, MHTGR, and PIUS were not being pursued and because adequate design and risk assessment information for these advanced reactor designs was not available, the focus and direction of the staff's effort had changed to concentrate on the evolutionary and passive advanced LWR designs that were currently being reviewed by the staff.

2.4 SECY-97-020

In response to a NRC request, in 1997 a staff effort chaired by Hugh L. Thompson, Jr. performed an evaluation to develop technical criteria for EP for evolutionary and advanced reactor designs; SECY-97-020 (Reference 8).

The rationale upon which the EP for advanced and evolutionary reactors should be based was one of the first issues brought into question. The staff determined that the NUREG-0396 approach (consequences tempered by probability considerations) was appropriate also for this kind of plants; furthermore, rigid application of the technical criteria derived from this rationale against the evolutionary and advanced reactor designs indicated that no changes to EP requirements are warranted because the potential consequences of severe accidents associated with those plants are similar to those for current reactors.

At the same time, however, the staff recognized that "*changes to EP requirements might be warranted if the technical criteria for the EP requirements were modified to account for*:

- *The lower probability of severe accidents;*
- *The longer time period between accident initiation and release of radioactive material;*
- *Most severe accidents associated with evolutionary and passive advanced LWRs.*"

In order to justify these types of changes to the EP basis, the staff believed that three main issues would have needed to be addressed:

- The probability level, if any, below which accidents will not be considered for EP;

- The use of increased safety in one level of the defense-in-depth framework to justify reducing requirements in another level;

- The acceptance of such changes by Federal, State, and local emergency response agencies.

Because of the significant expenditure of resources that would have been required, the staff expressed its intention not to perform further studies unless a petition was received from industry.

The approach followed by the staff was essentially composed by two major parts, as explained as follow.

Part 1

This was a review of the rationale, criteria and methods that form the basis for EP for currently licensed reactor designs in NUREG-0396. The review regarded essentially:

- The basis for the determination of the size of the two areas (PEP area and IEP area), using the PAG limits.

- The time-dependent characteristics of potential releases.

- The types of radioactive materials that potentially could be released during an accident scenario.

The conclusion was a confirmation of the recommendations in NUREG-0396, without considerable modifications.

Part 2

This was the more innovative part of the entire document: an evaluation of whether improved safety features of evolutionary and passive advanced LWR designs may warrant changes in the technical criteria or methods used as the basis for the EP regulations and whether application of these criteria for the evolutionary and passive advanced LWRs indicates that changes to EP requirements are warranted.

Particularly, this part consisted in a verification of how the innovative safety features and characteristics of the advanced plants tend to influence and modify the results to which NUREG-0396 arrived, in the fields of:

- PAGs Criteria

  The staff determined the extensions of the PEP and IEP areas still applying the PAG limits, but at the same time including the improvements in safety that make the ALWR different from conventional plants. In order to determine at which distance from an advanced plant the PAG limits are met, the staff examined the three criteria as listed at the end of Section 1.2:

– Criterion 1: (the consequences of DBA would not exceed the PAG levels outside the recommended EPZ distance).

  The application of this criterion to ALWR indicated that the PAGs would not be exceeded beyond 2 miles. Rigid application of just this criterion would indicate that, for the limited study performed by the staff, the EPZ size could be reduced for evolutionary and passive advanced LWRs.

– Criterion 2: (the consequences of less severe Class 9 accidents would not exceed the PAG levels outside the recommended EPZ distance), and

– Criterion 3: (the EPZ should be of sufficient size to provide for substantial reduction in early severe health effects (injures or death) in the event of the more severe Class 9 accidents)

  The application of these two criteria to ALWR indicated that different results would be obtained depending on the approach followed:

  - If the choice of the considered accidents didn't account for probability limit cutoff, the results would be the same of NUREG-0396 (no EPZ reduction).

  - If some accident sequences were not considered, because of the low probability of their occurrence or because of the existence of design features to prevent their occurrence or mitigate their consequences, reduction in the EPZ size was possible.

- Time of Release

The time between recognition of a severe accident and the start of the release affects the time available to take action to protect the public and, therefore, affects the need for the capability to promptly notify the public of the emergency. Currently, licensees are required to notify offsite officials within 15 minutes of declaring an emergency and offsite officials need to have the capability to notify the public within about 15 minutes of receiving notification from the licensee.

The time elapsed between recognition of a severe accident and a release of radioactive material for current plants was reported to be as early as 30 minutes in NUREG-0396.

Reviewing the evolutionary and passive advanced LWR severe-accident data the staff concluded that radioactive material could be released as early as about 90 minutes after a severe accident is recognized. The 1-hour difference between current plants and advanced LWRs was not considered large enough to justify changing the requirement for prompt notification of offsite officials and the general public. However, as discussed for the EPZ size, if some accident sequences with predicted early releases of radioactive material were not applied against this criterion, due to the low probability of their occurrence or because of the existence of design features to prevent their occurrence, then perhaps the requirement for prompt public notification capability could have been changed. The staff did not fully evaluate the effect that this change may have on size of the EPZ, nor did the staff evaluate the technical and policy issues, including public acceptance, associated with this potential change in the EP basis.

- Composition and Magnitude of Release

  With regard to the composition of the release, the mixture of radionuclides for evolutionary and passive advanced LWRs is essentially the same as that on which current EP requirements are based and, therefore, no changes are needed to aspects of EP such as specifications for monitoring equipment, dose projection models, and exposure modes.

Thus, the staff arrived to the conclusion that changes to EP requirements may be warranted only if the technical criteria for EP requirements were modified to account for the lower probability of severe accidents or the longer time period between accident initiation and release of radioactive material for most severe accidents associated with evolutionary and passive advanced LWRs.

2.5 EPRI TR-113509[1]

In January 1999, the Electric Power Research Institute convened a small group to conduct an independent peer review of an Advanced Light Water Reactor Emergency Planning Technical Report. As a result, EPRI TR-113509 was published in September 1999 (Reference 9). Its main target was "to quantify the performance of the three U.S. ALWR designs, in the areas of core damage prevention, containment performance, and offsite dose, and to use the results to define an emergency planning concept cost-effectively tailored to ALWRs."

As with previous efforts, also the EPRI work was driven by the very high level of safety achieved by the U.S. ALWR designs, which would merit a new emergency planning rulemaking.

The study proceeded in three steps.

Step 1 – Identify the Utility Requirements Document (URD) (Reference 22) criteria particularly relevant to emergency planning and compare the ALWR designs with those criteria, thereby confirming detailed conformance.

The criteria of concern are:

- Containment performance: containment loads from low pressure core damage sequences shall not exceed the American Society of Mechanical Engineers (ASME) Service Level C/Unity Factored Load limits. Accident sequences will be shown not to result in loads exceeding those limits for approximately 24 hours; beyond approximately 24 hours, there shall be no uncontrolled release. Other minor containment requirements are in the URD Chapter 5, Section 6.6.2.

1. Figures and Tables of Section 2.5 are Copyright© 1999 of Electric Power Research Institute, Inc. TR-113509 Technical Aspects of ALWR Emergency Planning. Reprinted with Permission.

- Dose: the dose at 0.5 miles from the reactor due to fission product source term release from a damaged core shall not exceed the PAGs for approximately 24 hours (i.e., 1 rem for 50th meteorological conditions percentile, 5 rem for 90th meteorological conditions percentile).

The criteria and associated methodology are primarily deterministic. A supplemental PRA evaluation is also required by the URD in support of the two criteria; this reliance on deterministic criteria with PRA as a supplement is consistent with the NRC Severe Accident Policy (Reference 23) which states that safety acceptability should be based on an approach which stresses deterministic engineering analysis and judgment, complemented by PRA. Particularly, PRA is required to demonstrate that the core damage frequency is less than 10^{-5} per year and that cumulative frequency for sequences resulting in doses greater than 1 rem is less than 10^{-6} per year.

The verification of the respect of the mentioned requirements was done analyzing severe core melt accidents, which differ from DBA since they consider also ex-vessel and late in-vessel release.

EPRI also performed an in-depth study to calculate the fractions of the various nuclides released and the corresponding time of release. The results obtained don't differ much from those of conventional plants (Tables 3.12 and 3.13 of NUREG-1465 (Reference 24)): the differences are mainly in the low volatile releases. Since these differences are not expected to have a significant effect on offsite dose and since it is expected that the ALWR will have margin to dose limits, NUREG 1465 release parameters are used for the EPRI emergency planning PAG dose calculation.

As a result of Step 1, all the three ALWR analyzed satisfy the URD deterministic and probabilistic requirements before mentioned.

Step 2 – Quantify the performance of the ALWR designs, in the areas of core damage prevention, containment performance, and offsite dose, and compare that performance with emergency planning criteria currently applied to existing U.S. nuclear plants (NUREG-0396).

The comparison between the ALWR performances and the NUREG-0396 emergency planning criteria is performed drawing up the trend of the conditional probability of dose exceedance versus distance from the reactor (i.e., conditioned on the assumed occurrence of a core melt). Figure I-11 of NUREG-0396 represents this trend for conventional plants, and is reproduced in Figure 2-1.

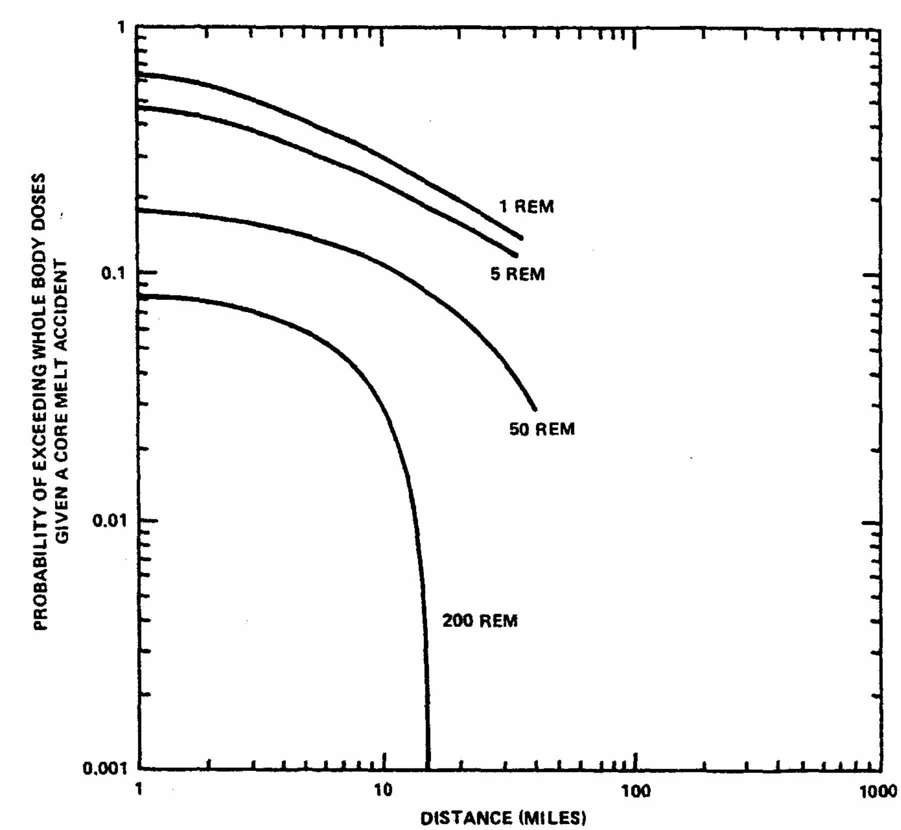

Figure 2-1 Original Figure I-11 from NUREG-0396

The NUREG-0396 estimates were re-evaluated by EPRI to establish a benchmark against the ALWR analyses. The need for a benchmark was due to various considerations:

- The code used to obtain the results listed in NUREG-0396 was an early version of CRAC, while EPRI used its successor MACCS;
- Red bone marrow dose was used to represent whole body dose since MACCS does not report "whole body dose";
- The ALWR site as defined in the URD was used in the benchmark; while it is not known how this site compares to what was used for NUREG-0396 studies, this site is relatively demanding from a meteorological standpoint in that it bounds about 80% of existing U.S. sites based on short term atmospheric distribution factors (χ/Q)

For the remaining parameters it was possible to use in the ALWR evaluation the same values used in NUREG-0396.

The results of the benchmark calculation are shown in Figure 2-2.

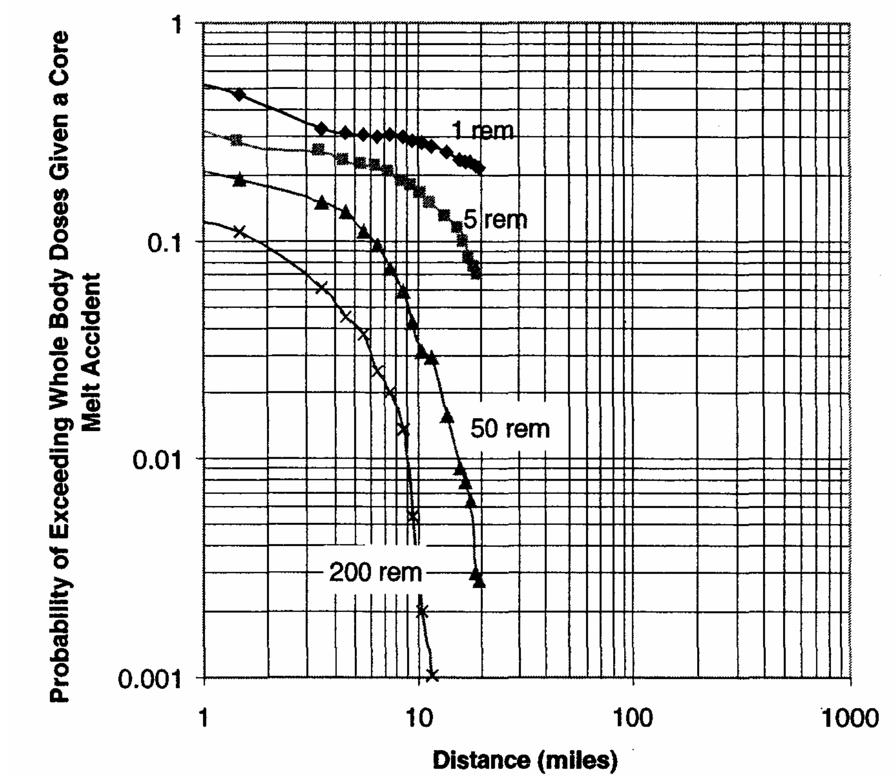

Figure 2-2 EPRI Benchmark to Reproduce Figure I-11 of NUREG-0396

As it is evident comparing the two figures, the agreement is quite good. This provides confidence that curves calculated using the ALWR source terms can be meaningfully compared with the NUREG-0396 results.

Next, changes in the emergency planning basis criteria were introduced. These changes consisted essentially in making the probability concept not simply a qualitative consideration (as done in NUREG-0396) but a quantitative factor affecting the basis criteria. In the EPRI report, in fact, tempering in the ALWR context means consideration of severe accident design features and accident management, and the resulting low probabilities, as an input to the planning rationale.

The steps in applying the tempering process are as follow:

- Use the complete set of accident sequences in the ALWR plant specific PRAs as a starting point;

- Identify those sequence classes which have probability of occurrence so low that it would be unreasonable to include the sequences in the NUREG-0396 assessment, or for which the time to the beginning of significant release provides adequate warning time. The remaining sequences are to be included in the ALWR NUREG-0396 assessment;

- Review design aspects of the low probability sequence classes to confirm the existence of design features and capabilities which would tend to support the low probability of occurrence (or long time delay) of the dominant accident sequence types which make up the sequence class.

Regarding the cutoff probability, an investigation of previous studies yielded:

- NUREG-1150 (Reference 25) used a frequency cutoff of 10^{-7} per year for PRA accident sequence progression;

- NUREG-1420 (Reference 26) discusses probability cutoff criteria for PRAs, and indicates that consequences with frequencies lower than about 10^{-7} per year "*are not meaningful for decision making*";

- Standard Review Plan (Reference 27) (Section 2.2.3) guidance specifies evaluation of potential accidents from hazards which exceed 10^{-7} per year;

- NUREG-0396, Figure I-11, has a conditional probability range down to 10^{-3} which corresponds to $\approx 10^{-7}$ per year absolute probability (since from WASH-1400 (Reference 28) the mean core damage frequency was at that time $\approx 10^{-4}$ per year)

- NUREG-1338 (Reference 29) states as part of the justification for reduced emergency planning that sequences appearing to have a frequency in the range of about 10^{-7} per year will be examined for residual risk

- Regulatory Guide 1.174 (Reference 30) specifies that, when using PRA to support decisions to modify an individual plant's licensing basis, an increase of less than 10^{-7} per year in the Large Early Release Frequency (LERF) is recognized as very small and will not prevent NRC to consider the proposed modification.

Thus, it appears that a probability cutoff of 10^{-7} would be acceptable. Actually, in TR-113509, EPRI considered another reference to cutoff probability deriving from 10 CFR 60 (Reference 31) and indicating that events which result in fatal cancer risks on the order of 10^{-8} per year do not contribute significantly to individual risk. In the case of reactor accidents with probability in the range of 10^{-7} per year and consequences which would likely not exceed several tens of rem (say 30 rem), the fatal cancer risks are of the order of:

$$(10^{-7}/\text{yr}) \cdot (5\times10^{-4} \text{ fatalities/rem}) \cdot (\approx 30 \text{ rem}) \approx 10^{-9} \text{ cancer fatalities per year}$$

that is well below the level of 10^{-8} before mentioned.

For the ALWRs however, EPRI considered the probability screen of 10^{-7} still too high, for two reasons:

1. For ALWRs there are no containment challenges above 10^{-7}: if 10^{-7} was chosen as probability screen many severe accidents resulting in containment challenges wouldn't be considered

2. The EPRI peer review report (Appendix H of EPRI TR-113509) recommended that sequences down to 3 orders of magnitude below core damage frequency be included since the NUREG-0396 Figure I-11 methodology is based on this relative frequency range (a discussion of this choice is provided in next Section 2.6).

Thus, a probability screen of 2×10^{-9} per year (i.e., 3 decades below the average ALWR core damage frequency of 2×10^{-6}) was used by EPRI in the ALWR assessment against NUREG-0396.

On the other hand, EPRI decided to take more realistic credit for mitigating factors than was generally done in the ALWR PRAs, including Reactor Coolant System (RCS) depressurization, steam generator injection, manual containment spray and aerosol retention in unisolated Steam Generator Tube Rupture (SGTR) accidents. Moreover, EPRI decided to eliminate sequences with time to beginning of release of 24 hours or longer.

Introducing the modifications just highlighted, and using the frequency, release timing, and release magnitude taken from the ALWR PRAs, EPRI drew up the conditional probability graph (shown in Figure 2-3) of exceeding various doses at different distances. Table 2-1 summarizes the ALWR conditional probabilities of exceeding various doses at 0.5 miles against the corresponding conditional probabilities from NUREG-0396 at 10 miles; it clearly appears that ALWRs have an extremely low probability of exceeding dose limits for each distance from the reactor. It can be noted that the ALWRs' probabilities at 0.5 miles are lower than the probabilities assessed for existing plants at 10 miles.

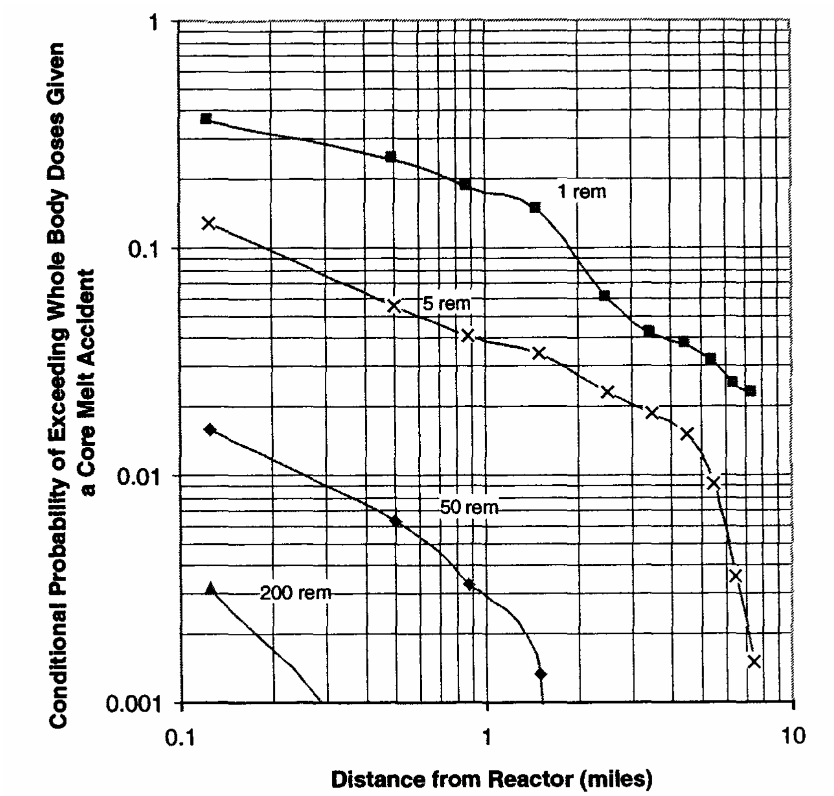

Figure 2-3 ALWR Assessment by EPRI for Comparison Against Figure I-11 of NUREG-0396

Table 2-1 EPRI Evaluated ALWR Dose Assessment Results versus NUREG-0396		
	ALWR with $2 \cdot 10^{-9}$ as cutoff frequency (0.5 mile)	**NUREG-0396 (10 miles)**
Conditional probabilities of exceeding 1 rem	0.25	0.3
Conditional probabilities of exceeding 5 rem	0.06	0.25
Conditional probabilities of exceeding 50 rem	0.006	0.1
Conditional probabilities of exceeding 200 rem	<0.001	0.01-0.001

Comparing Figures 2-1 and 2-3, it clearly appears that ALWRs have an extremely low probability of exceeding dose limits for each distance from the reactor. Table 2-1 summarizes the ALWR conditional probabilities of exceeding various doses at 0.5 miles against the corresponding conditional probabilities from NUREG-0396 at 10 miles.

It can be noted that the ALWRs' probabilities at 0.5 miles are lower than the probabilities assessed for existing plants at 10 miles.

Step 3 – Define a cost-effective, ALWR-specific emergency planning concept in which the ALWRs perform as well as currently required of existing plants in their emergency planning concept.

In the third step of TR-113509, EPRI defined four general principles to guide the development of a new emergency planning concept:

1. Emergency planning is a necessary part of the defense-in-depth philosophy of nuclear safety and should be provided for power reactors.

2. The concept and details of the emergency plan should be commensurate with the facility design, that is, with the risk associated with the specific design. The design affects the likelihood of an offsite release as well as the timing and magnitude of the release. Thus, the emergency plan for a particular class of plants should reflect the likelihood, timing, and magnitude of the offsite release for that class of plants.

3. ALWR emergency planning should reflect the experience from existing emergency planning regulations and implementation at operating plants.

4. A common framework should be considered for offsite emergency planning for non-nuclear industrial hazards and nuclear power plants. This is desirable to avoid the complexity and confusion of overlapping emergency response system and procedures, and to provide a consistent level of protection for comparable risks.

Although EPRI considered acceptable the existence of the two exposure pathways (i.e., PEP and IEP), it believed that the EPZ should consist of three areas: two related to the PEP, and the IEP. They are discussed below.

- PEP Areas

Area 1: **Response area, 0.5 mile radius**. It is the area closest to the reactor, within which a severe accident could cause radiological consequences of sufficient concern that a rapid response should be included in planning. The size chosen is such that at 0.5 mile from the reactor the following requirements are met.

- The URD requirement that the median dose for 24 hours for most core damage events is less than 1 rem Everted Dose Equivalent (EDE),
- The URD requirement that the 90th percentile dose for 24 hours for most core damage events is less than 5 rem EDE,
- The PAG limits resulting from DBA,
- The PAG limits resulting from the less severe Class 9 accidents.

Also, no early severe health effects (injures or death) resulting from the more severe Class 9 accidents are foreseen outside this zone

Area 2: **Awareness area, usually 3 miles radius (site-specific)**. This is the area beyond the response area within which the radiological effects would be smaller. EPRI introduced this area in order to provide a substantial base for expansion of response in the plume exposure pathway in the event that this is necessary. The size of the area depends upon political boundaries, the characteristics of the terrain, and the emergency planning capabilities of State and local jurisdictions. It is important to note that the current planning bases for expansion response are expressly based on nuclear hazards plans (onsite plans); instead, for ALWR, these planning bases should be based on nuclear hazards plans and on "all-hazards plans." In fact, awareness area activities would be the responsibility of the offsite agencies and would be administered as part of the all-hazards emergency plan required by the Federal Emergency Management Agency (FEMA).

- IEP area

Area 3: **Ingestion exposure pathway area, 25 miles radius**. This extension is based on maintaining the projected infant thyroid dose from cow's milk below the EPA PAG (1.5 rem).

The three areas and a wide comparison with current EPZ rules are shown in Table 2-2.

Table 2-2 Summary of Emergency Planning Zones		
	Existing Emergency Planning	ALWR Emergency Planning
Plume Exposure Pathway planning distance	10 miles	Response area: 0.5 miles
		Awareness area for possible expansion of response: site specific (usually 3 miles)
Planning basis for expansion of response	Onsite plan and offsite plan	Onsite plan and all-hazards plan (awareness area)
Ingestion Exposure Pathway planning area	50 miles	25 miles

2.6 PEER REVIEW COMMITTEE REPORT

In January 1999 an independent review group was formed in order to review the TR-113509 EPRI report and it concluded that the report can provide a reasonable revised technical basis to support decisions on Emergency Planning for ALWRs.

An important consideration by the peer review group concerned the "tempering process," i.e., allowing that some "*accident sequences can be excluded from the analysis on the basis of their frequency*." Actually, the peer review group took issue with this concept as presented, since events with lower probabilities might have significant consequences thereby representing a risk that should be evaluated. Because the exceedance curves are conditional on core damage, it is inconsistent and potentially misleading to eliminate sequences that affect the shape and magnitude of the curves over the range over which they are presented (three decades in NUREG-0396). The peer review group was asked to present the conditional probability of exceeding various dose levels with and without truncation. Table 2-3 presents the comparison between the results obtained with a cutoff frequency of 10^{-7} per year and those without truncation.

Table 2-3 Conditional Probabilities with and without Truncation: Comparison			
	NUREG-0396 (10 miles)	ALWR (0.5 miles)	ALWR No Cutoff (0.5 miles)
Conditional probabilities of exceeding 1 rem	≈0.3	≈0.1	≈0.25
Conditional probabilities of exceeding 5 rem	≈0.25	≈0.01	≈0.06
Conditional probabilities of exceeding 50 rem	≈0.12	<0.001	≈0.01
Conditional probabilities of exceeding 200 rem	≈0.01	<0.001	≈0.002

As it can be seen from the above comparison, truncation of sequences at a level of 10^{-7} per year has a significant impact on the results. However, the peer review group strongly endorsed the concept of a screening level for core damage frequency below which events are so improbable that they should not be factored into a regulatory decision process. Because the NUREG-0396 process is conditional on core damage, for consistency the ALWR analyses must include all core damage sequences that affect the results over the range where they are presented. In practice, this implies that the truncation level can be no higher than three orders of magnitude below the core melt frequency.

The marked discrepancy when using 10^{-7} as cutoff frequency, practically disappears for a cutoff frequency of $2 \cdot 10^{-9}$ which yields values very close to the case without truncation as shown by a comparison of Tables 2-1 and 2-3.

Finally, the peer review group noted that the conditional probability approach does not fully provide for a comparison of the lower risks of severe accidents estimated for the ALWRs. Accordingly, the peer review group believed that an additional comparison with the results of NUREG-0396, based on absolute probability of exceeding 1, 5, 50, 200 rem, would provide additional insights useful in consideration of revised emergency planning for the ALWRs (Appendix H of EPRI TR-113509).

2.7 NEI 02-02

In May 2001, a new vision for the nuclear industry, Vision 2020, was presented to the industry and the public. The vision supports an energy policy that would add 50,000 megawatts of nuclear generation by 2020.

In response to industry feedback on Vision 2020, NEI formed the New Plant Regulatory Framework Task Force. This task force was charged with developing a new and optional risk-informed, performance-based regulatory framework for commercial nuclear power reactors, focusing mainly on technical and operational requirements.

The resulting document, NEI 02-02 (Reference 10), includes a complete set of regulations for a new Part to Title 10 of the Code of Federal Regulations, Part 53. This new part is intended to be an alternative to 10 CFR Part 50 ("Domestic Licensing of Production and Utilization Facilities") for commercial nuclear power reactors, and, as such would be optional.

The decision to develop a completely new part rather than just amend or develop alternative Part 50 requirements is based on reducing regulatory complexity and reducing the potential for misinterpretation.

The purpose of the document is to describe the industry view of what a risk-informed, performance-based regulatory process should be for future generations of commercial nuclear power reactors, in terms of both design and operational requirements. Such a regulatory process provides for an increased focus on safety, while providing the licensee increased regulatory flexibility in meeting the regulations.

The document provides a generic process and a set of top-tier regulations that specify safety objectives, but permit flexibility in how the objectives are achieved. This is important, as the next generation of commercial nuclear power reactors may include a variety of plant designs of varying nuclear technologies.

Being a general overview, the new regulations listed in NEI 02-02 are intended to be available for use by all prospective licensees and reactor system designers, regardless of reactor design. The new part would be applicable to all types of reactor designs: light-water reactors, gas reactors, liquid metal, etc.

Differently from EPRI TR-113509, NEI 02-02 can't be considered a technical report, but rather a document reproducing a set of regulations, based essentially on qualitative considerations rather than on a quantitative approach.

Regarding explicitly the emergency preparedness, NEI 02-02 suggests a frequency cutoff of 10^{-7} per year, but no mention is made of the extension of any emergency planning area.

2.8 IAEA APPROACH

IAEA-TEC-DOC-955 (Reference 32) provides the Agency's generic assessment procedures for determining protective actions during a reactor accident. A set of public protective actions are provided based on projections and in-plume measurements.

The protective action pertaining to the Operational Intervention Level (OIL) 1 is to evacuate or provide substantial shelter for the sector, the two adjacent sectors and the sectors closer to the plant. Until evacuated, people should be instructed to stay inside with their windows closed. The basis for this OIL is an ambient dose rate in the plume of 1 mSv/h which is calculated assuming an unreduced release from a core melt accident resulting in an inhalation dose 10 times the dose from external exposure, 4 hours exposure to the plume. 50 mSv can be averted by the involved action.

These bases are compatible with what has been herein presented as the current situation in different countries around the World; nevertheless, the necessity to make the nuclear emergency planning fit with the type of plants concerned has been shared also by the IAEA. One of the main promoters of the elaboration of a new emergency planning concept was the INPRO. The objective of INPRO is to support sustainable deployment and use of nuclear technology to meet the global energy needs in the next 50 years and beyond. Particularly, INPRO would bring together both technology holders and technology users to consider jointly the actions required to achieve desired innovations in nuclear reactors and fuel cycles.

Finally, a Coordinated Research Program (CRP) has been initiated by IAEA specifically investigating the redefinition of emergency planning measures around nuclear power plants[2].

2.9 THE EUROPEAN UTILITY REQUIREMENTS (EUR) APPROACH: AN EXAMPLE OF HARMONIZATION ATTEMPT ON THE UTILITY SIDE AND ITS IMPACT ON EPZ REQUIREMENTS

An harmonization of the EPZ requirements, at least from an utility point of view, has been attempted by the European Utility Requirements (EUR) initiative, as part of a more extended effort of harmonization of the European nuclear power industry. The driving force for such an harmonization is that from a technical point of view, an ideal nuclear safety framework in Europe would be a harmonized set of

[2.

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European safety objectives and related calculation methodologies, so that different results obtained in different countries for similar plants and scenarios could be explained mainly in terms of plant-specific features and site conditions rather than different (often arbitrary) assumptions or calculation tools.

The EUR development was initiated in the early nineties by a group of European utilities participating in the development of the US ALWR URD. The objective of the group was to build upon the experience gained in the ALWR program to issue a set of requirements, in particular safety relevant ones, common to all members, in order to facilitate interaction with national Safety Authorities and allow standardized designs to be built in all participating countries. It is well recognized that this harmonization effort is purely the result of an attempt made by the utility side and that safety relevant requirements, though reflecting a common analysis and understanding of some safety issues, were never contemplated as substitutes to national Safety Authority Requirements.

The EUR aim to develop Regulatory Requirements into detailed guidance to designers so that an ALWR project designed to comply with the EUR could be licensed and built in all the different countries in Europe. In addition, the EUR define how to comply with the regulatory requirements in such a way to improve public confidence and acceptability. To this purpose the EUR introduce margins and targets toward regulatory requirements so to ease licensing and acceptability, this is done with targets values which are more demanding than regulatory limits, nevertheless being still judged to be reasonably achievable by modern well designed plants.

The EUR document (Reference 33), covering a vast spectrum of safety related requirements, also address the off-site emergency preparedness. This is done even recognizing that this aspect is probably one of the most country-specific and therefore one of the most difficult to harmonize.

Both a deterministic (i.e., dose limits) and a probabilistic set of high level requirements are proposed by the EUR. As far as the deterministic contribution is concerned, the key starting point is obviously the individual and collective dose limits to the population under accident conditions. The EUR group recognizes that the European directives give only high level guidance in this domain and leave member States to define the dose limits and, as a consequence, the actions needed. Some international organizations (i.e., IAEA and the International Commission on Radiological Protection – ICRP) provide more precise guidance. In any case the overall strategy is clear and generally accepted: compliance with the limits must be reached, with a proper level of confidence, using preventive, mitigating and protective features.

Considered the fact that plants in operation have already reduced the risk of large release for DBA and considered the attention given to severe accidents, the EUR goals have been established taking into account beyond design basis conditions, which are indicated as Design Extension Condition (DEC) in the EUR document. Actually, considering the capabilities of operating plants and the constraints of European average siting conditions, the EUR group has decided to make a very clear step forward with the objective of improving public acceptance and of reducing financial risks. Such a step has the nature of a dramatic limitation to possible releases from containment and, consequently, to the environmental impact.

Four targets have been defined to substantiate the notion of limited impact in case of severe accidents (see Sections 2.1.2.4 and 2.1.2.5 of the EUR document):

- No Emergency Protective action shall be needed beyond the site boundary, i.e., beyond 800 m from the reactor. This means that the averted Effective Committed Dose (EDC), over a period of one week following accident initiation, will remain below 50 mSv, which is the generic intervention value reported in ICRP 63 and which is adopted in the IAEA Basic Safety Standards. Practically speaking, evacuation and sheltering of people are not needed for such low values.

- No delayed action shall be taken beyond 3 km from the reactor. This means that the averted EDC over a period of thirty consecutive days following release termination remains below 30 mSv. This assures that the temporary relocation intervention level reported in the IAEA Basic Safety Standards is not reached.

- No long term action shall be required beyond site boundary. This means that the averted EDC over a period of fifty years following release termination remains below 100 mSv. Though this limit is lower than the one recommended in the IAEA Basic Safety Standards, it was considered consistent with the two above mentioned criteria by the EUR group.

- Limited economical impact linked to the restriction of foodstuff consumption. This means allowing free trading of foodstuff, provided a 5 mSv dose to individuals eating contaminated food for one year is not reached:

  - after one month following the end of the accident over a 30 km^2 area,
  - after one year following the end of the accident over a 10 km^2 area.

  More clearly, this means that foodstuff produced in areas of 30 and 10 km^2 surrounding the site could be marketed after one month and one year respectively.

As anticipated, the four goals presented above are not seen by the EUR group as a "revolution" but rather another "evolutionary" step toward power plants increasingly environmental friendliness: nothing different from what has been done in the past and nothing different from what is being done for fossil fuel electric power plants.

On the probabilistic side, ambitious goals, some of them well in line with the (INSAG 3) international mainstream, are set in subsection 2.1.2.6 of the EUR:

- The cumulated probability of sequences leading to a core melt has to be kept below 10^{-5} per reactor/year for all plant states, considering internal as well as external events;

- The cumulated probability of sequences leading to unacceptable releases (i.e., in excess of maximum allowable releases for DEC) has to be kept below 10^{-6} per reactor/year;

- The cumulated frequency of sequences leading to early containment failure or to very large releases has to be kept below 10^{-7} per reactor/year.

These requirements are meant to make negligible the challenges to containment integrity. The new advanced goals established by the EUR should be sufficient, if the same approach utilized for other industrial risks would be applied, to convince responsible authorities of no need of an emergency plan involving directly the population. Actually for most of the industrial hazards, if the probability of an health effect is less than 10^{-5}/y, the risk is considered negligible and no further action is required. A risk lower than one in one million is usually considered negligible also by courts. As seen, the EUR probabilistic goals try to exceed those targets.

The proper level of protection aiming at the compliance with the previously listed requirements is being reached on the basis of the defense-in-depth principle with its usual layers of protection:

- Design: Prevention – related requirements refer mainly to fuel elements, primary circuit and emergency cooling systems with the objective of avoiding fuel damage or melting;

- Design: Mitigation – related requirements refer mainly to containment system in order to warrant its capability to confine radioactivity, but, in the new perspective of Beyond DBA, refer also to some new emergency core or corium cooling systems;

- Operation: Safety culture and operators training – related requirements refer mainly to basic and specific training of operators, to the preparation of emergency procedures and to periodic tests;

- Planning: On site recovery and emergency actions. The EUR put particular emphasis on the capability of properly analyzing the plant conditions and reacting also using mobile or non safety grade equipments. This concept traditionally applied to fire-fighting is now extended to other conditions.

- Protection: Intervention on the population with measures defined in the off-site emergency plan. This domain usually is outside of the responsibilities of the plant owner. Nevertheless the plant owner has to provide the basis for the development of the emergency plan.

Setting the goals described before, and establishing the related requirements, the EUR have simply requested a change to the contribution that every defense layer gives to the fulfillment of the required protection level. It is very important to note that the change in role is not something happening just because of the EUR approach. It is rather part of the natural evolution from generation I to generation III NPP and will be part of the further evolution toward generation IV. Figure 2-4 tries to give a graphic and qualitative presentation of this EUR idea.

The change in height of the different layers corresponds to the changing importance of each layer in order to obtain the requested protection level, which is maintained fixed thus providing the same level of protection to the public.

First column refers to Generation I. The primary circuits were designed with lot of margins in particular on fuel power rating and on water volumes. As a consequence the designs were forgiving toward operators. The consideration given to accident conditions was basically of the “industrial safety” type. Although Emergency Core Cooling System (ECCS) have been back-fitted later, the acceptability of G-I plants relied mainly on smooth operation and protective actions.

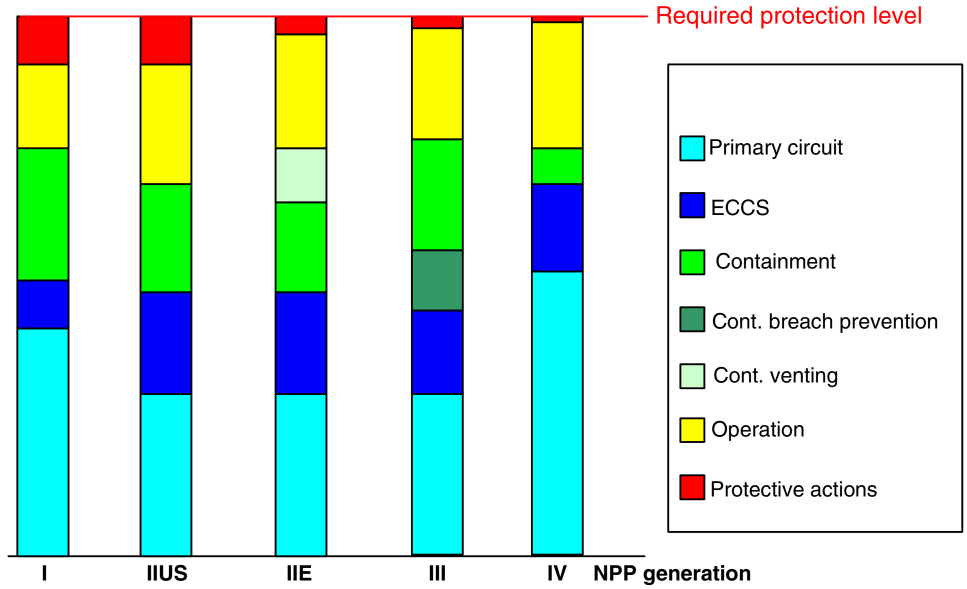

Figure 2-4 Evolution of the Relative Importance of Defense-in-depth Layers

Second column refers to Generation II in the USA. At the beginning of the '70 the US NRC issued around 70 Regulatory Guides that dramatically changed the NPP's design basis. Actually NRC had fully recognized the importance of core cooling under all possible conditions in particular in a contest of reduction of margins due to economic optimization of the designs. The 1975 Brown Ferry fire and more important the 1979 Three Mile Island core melt caused further ruling. The mitigating capability of containment was recognized, some new requirements, as for hydrogen re-combiners, were established but, at the end, greater reliance was put also on the protective actions.

Not all countries in Europe recognized immediately the implications of those new requirements, which, if accepted, would have caused to shutdown many plants because of the impossibility to comply with the American regulations. Some European countries therefore decided to give a different answer to the risk of uncontrolled large release.

Third column thus refers to Generation II in Europe. To avoid uncontrolled large release from containment some European Countries adopted the filtered containment venting concept. As the safety relief valve for pressure vessels, the venting avoids uncontrolled breaches of the containment and in addition, through filtering, reduces the release to the environment. Acting this way the need for off–site protective actions is reduced, typically there is no need for evacuation planning beyond 5 Km.

Fourth column refers to Generation III. In this case the mitigation has an increased role that can be met only if containment integrity is always preserved. Both URD and EUR have requirements aiming at warranting containment integrity under all possible accident scenarios. As a consequence all related design should have a very limited need for off-site protective actions. Nevertheless only the EUR have clearly stated the goals described above, and have stated a comprehensive set of related DEC requirements aiming at preserving containment integrity.

Fifth column is finally a tentative representation of future Generation IV. Quite clearly, in continuity with the trend described, the main safety goal of Generation IV should be to prevent core melt or, even better, fuel damage with the declared scope of reducing to the minimum the off-site protective actions.

The EUR approach is therefore just following a normal evolution of the role that the various level of protection play in the defense-in-depth philosophy.

Recognizing the As Low As Reasonably Achievable (ALARA) principle, which implies economical barriers to an oversized level of protection, the EUR approach recognizes the potentiality of focusing the major effort on the most controllable variable (i.e., levels of protection inside or very close to the plant boundary), rather than extended areas with protective actions not easily performed.

3 INTEGRATED METHODOLOGY FOR EMERGENCY PLANNING ZONE REDEFINITION

The analysis of past experiences briefly outlined in the previous section suggests the adoption of a mixed deterministic/probabilistic approach which still involves a relevant modification in the fundamental EPZ defining criteria as currently conceived (i.e., from consequences, as it is now, to risk).

The proposed methodology is based on accepted concepts such as PRA techniques and deterministic dose evaluation as used in current practice; it suggests a more complete definition of the current and accepted criteria for the EPZ by focusing on the frequency of exceeding a given dose at a given distance. The EPZ can be redefined while still maintaining the same dose (explicitly defined in the current PAG) and the same frequency (implicitly defined by the choice of a fixed distance) defined by the NRC.

The proposed methodology addresses the two conceptual weaknesses highlighted for previous efforts in the redefinition of the EPZ defining criteria:

- In the deterministic component of the methodology all the foreseen sequences are evaluated with no exclusion of severe accidents, which are obviously expected to be the limiting scenarios and cannot be removed from the analysis without infirming the completeness of the methodology. Previous attempts in the EPZ redefinition were rejected because lacking a satisfactory account of severe accidents;

- The probabilistic component is shifted from establishing a cut-off frequency to being a screening criterion of accident sequences by evaluating the frequency to overcome the dose limit at a certain distance. By means of the data provided by PRAs, such a distance can be evaluated rather then pre-set. Arbitrary selection of a value for the cut-off frequency represented the major objection against the probabilistic approach to EPZ redefinition.

Two applications of the methodology are herein described:

1. The direct application is aimed at providing a risk-informed definition of the EPZ, once the basic acceptance criteria in terms of limiting dose and limiting frequency have been provided (i.e., agreed upon with regulatory bodies);

2. The reverse application is fundamentally a method for the measure of the level of risk associated with the currently accepted EPZ size and for the existing generation of nuclear power plants. Even though risk was not retained as the main defining basis for the EPZ size in NUREG-0396, a level of risk can actually be retrieved by measuring the frequency that a pre-defined consequence is manifested at the distance from the plant which is currently adopted as the EPZ size. If measured with this approach, and on the basis of the rationale that was selected (as described in Section 1.2) for the EPZ size, the level of risk associated to the currently accepted EPZ size will also factor in the additional margin associated with the unique emotional perception of the nuclear risk. Such a risk value could then be used as the reference baseline for the definition of an EPZ for a new NPP design. The methodology suggested herein, supported by a performance-based licensing approach, would in this way allow a new NPP design to maintain the measured risk, while reducing the re-evaluation of EPZ size.

The integrated methodology herein outlined can be summarized in five steps that are described in the following sections.

3.1 DIRECT APPLICATION: EPZ DEFINITION

The probabilistic starting point of this methodology (i.e., Step 1) essentially covers the choice of the set of release scenarios to be addressed by a deterministic evaluation of the consequences.

In order to obtain this outcome, the entire spectrum of accident sequences defined through the PRA of the plant must be reviewed and re-categorized. No additional cut-off frequencies are introduced, but the same truncation level applied and accepted for the PRA development must be maintained and should reasonably guarantee to cover unlikely sequences. Similar accidents (in term of the release) could of course be lumped together to limit the analytical burden to a manageable level.

A set of release scenarios (A_i) with their related frequency f_i of occurrence is therefore the outcome of this first step of the methodology. In the following Figure 3-1 a schematic representation of the results of this step is presented where five release scenarios are obtained after the re-categorization.

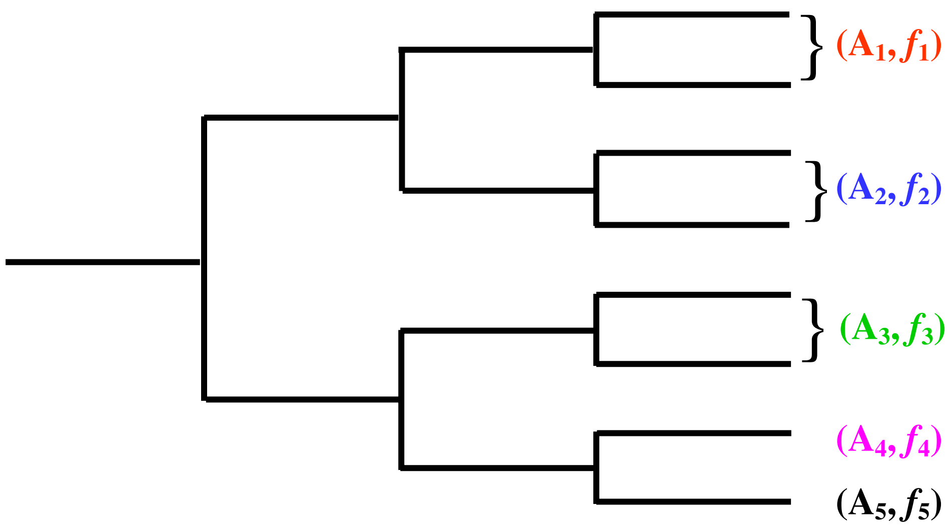

Figure 3-1 Step 1 of the Methodology: Accident Sequence Re-categorization

Once that the set of release scenarios have been identified, the second step is a deterministic evaluation of the consequences. Appropriate assumptions must be made in order to outline the scenario phenomenology; such assumptions should be based in a wider extent on best estimate, realistic models rather than on large and over-conservative safety coefficients. Using appropriate codes, the dose absorbed by an hypothetical individual located at various distances from the reactor, during the days (especially the first hours) after the onset of the accident is calculated. This calculation should be performed considering a complete set of meteorological conditions.

The final outcome of this step is a set of curves of dose equivalent (D) versus distance (x), one curve for each release scenario, A_i, regardless of the frequency of the selected accident. Figure 3-2 represents an example of the results of this second step applied to the five release scenarios hypothetically identified in Figure 3-1.

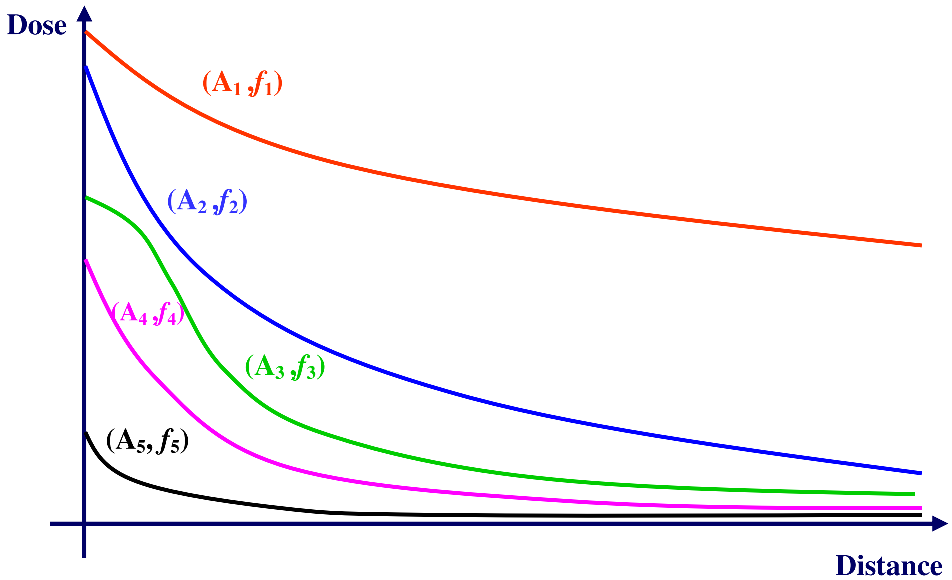

Figure 3-2 Step 2 of the Methodology: Dose versus Distance Evaluation

To be able to combine the probabilistic and the deterministic contributions, a limiting dose D^* and a limiting frequency f^* are identified in steps 3 and 4, respectively. These are two steps that require licensing considerations since the limiting values should ideally be suggested by and/or agreed with a regulatory body.

Even though these two steps will require further investigation, as far as the limiting dose, D^*, is concerned, the current mainly consequence-oriented approach for the selection of the EPZ defining criteria is felt to be able to concur in an easily identification of a value of general consensus (e.g., the PAG suggested by the US EPA).

The identification of a limiting frequency, f^*, is on the other hand could be more controversial. However, as described in Section 2.5, a value of 10^{-7} recurrently appears in various documents and it can be used as the f^* value for a first approximation of the methodology.

The aim of the direct application of this methodology is the evaluation of the frequency of exceeding a limiting dose, rather than the frequency of occurrence of some accidents. Such a frequency, as described below, is not imposed but can be evaluated applying the methodology to currently operating nuclear power plants (reverse application). In the framework of the risk-informed nature of this methodology, a modified version of the herein described methodology (i.e., a reverse application) is described in the following Section 3.2; this could be used to back-solve the limiting frequency from the current EPZ for current NPPs.

The fifth and final step is the combination of the probabilistic and deterministic contributions previously mentioned to determine the size of the EPZ. The methodology as follows: each of the curves of dose versus distance (evaluated for each Ai release scenario) is solved for the "limit dose" D* (see Figure 3-3).

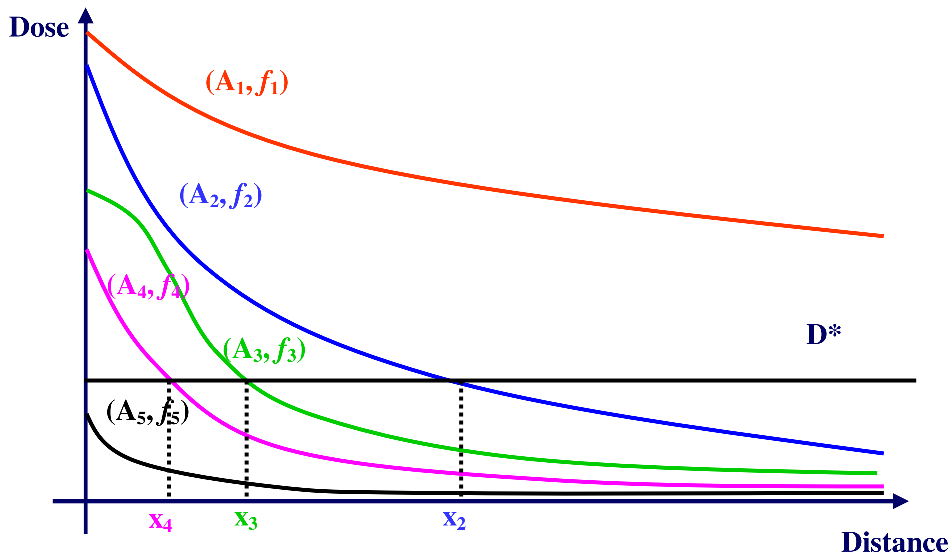

Figure 3-3 Step 5 of the Methodology: Probabilistic/Deterministic Combination (Part 1)

That is, from the dose versus distance curves the distance x_i for which the calculated dose is equal to D^* is easily identified. This is repeated for each scenario until a family of values of x_i (with $i=1$ to n, where n is the number of considered scenarios) is generated. For better clarity, it is here assumed that the accidents are ranked according to decreasing consequences (i.e., in the example, A_1 is the event with the highest associated dose and A_5 is the event with the lowest associated dose). Curve A_1 can be considered as an example of beyond DBA inducing release of high doses at virtually all distances within the considered range; the curve A_5 can be considered as an example of accident with a low consequential release (for example due to the design improvements obtained by advanced and innovative reactors) and that therefore does not play any role in the definition of the EPZ distance.

By the definition of x_i (distance at which the limit dose occurs), for each scenario A_i there would be a probability 1.0 of exceeding D^* at a distance smaller than x_i, and a probability 0.0 of exceeding D^* at a distance larger than x_i. These probabilities should then be multiplied by the PRA calculated frequencies of the occurrence of each accident so that the frequency of exceeding the D^* at a distance smaller than x_i would be, for each scenario, f_i.

Note that it can be expected that the larger x_i will be associated with the more severe accidents, which should in turn have the lower frequencies (this is not a strict requirement of this application, but it represents a reasonable expectation). With this information collected, the x_i are then ordered by decreasing values so that the frequencies of exceeding the dose limit as a function of distance can be calculated by simply considering, for each distance x_i, the contributions (i.e., the f_i) of all scenarios A_i that at the selected distance induce a released dose higher than the limiting dose. The combination is therefore as follows:

$$f_{D^*}(x_1) = f_1$$

$$f_{D^*}(x_2) = f_1 + f_2$$

...

$$f_{D^*}(x_i) = \sum_{i=1}^{n} f_i \quad \forall i \,/\, D(x_i) > D^*$$

where $f_D*(x)$ is the "Frequency of exceeding dose limit $D*$ at the distance x."

Thus an histogram of f_D* versus distance can be completed. The last remaining input to the methodology, and a critical one, is the previously identified Limit Frequency ($f*$) of exceeding the Dose Limit ($D*$) that should be used to define the associated distance determining the EPZ requirements (see Figure 3-4).

The EPZ distance will in fact be defined as the distance with a frequency equal to or greater than the given limiting frequency (e.g., x_3 in Figure 3-5, being $f_1+f_2+f_3$ the lowest summation of frequencies which is greater than the given $f*$).

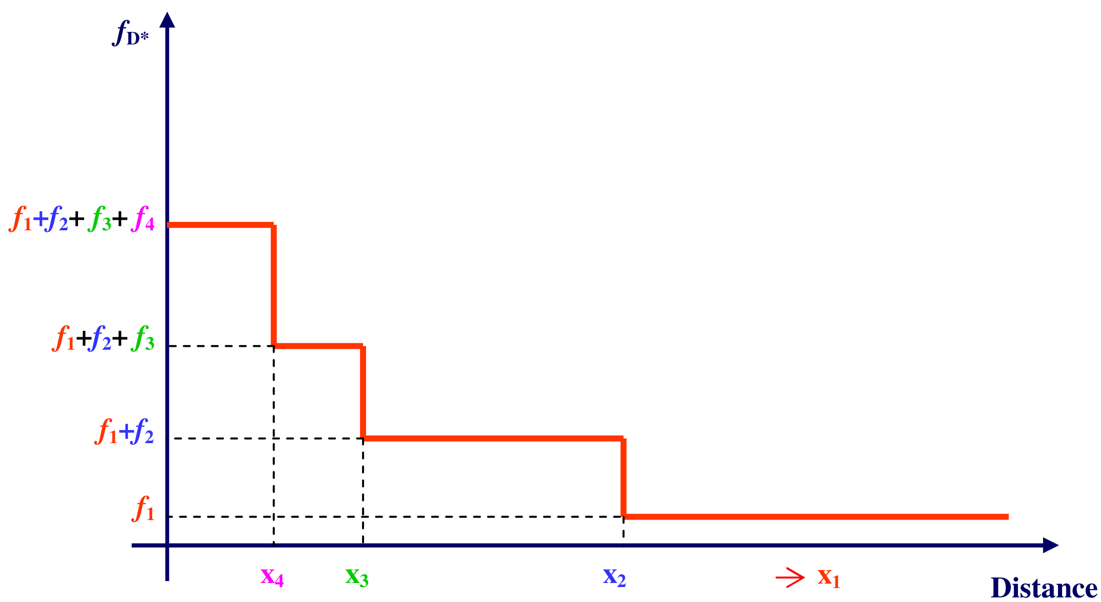

Figure 3-4 Step 5 of the Methodology: Probabilistic/Deterministic Combination (Part 2)

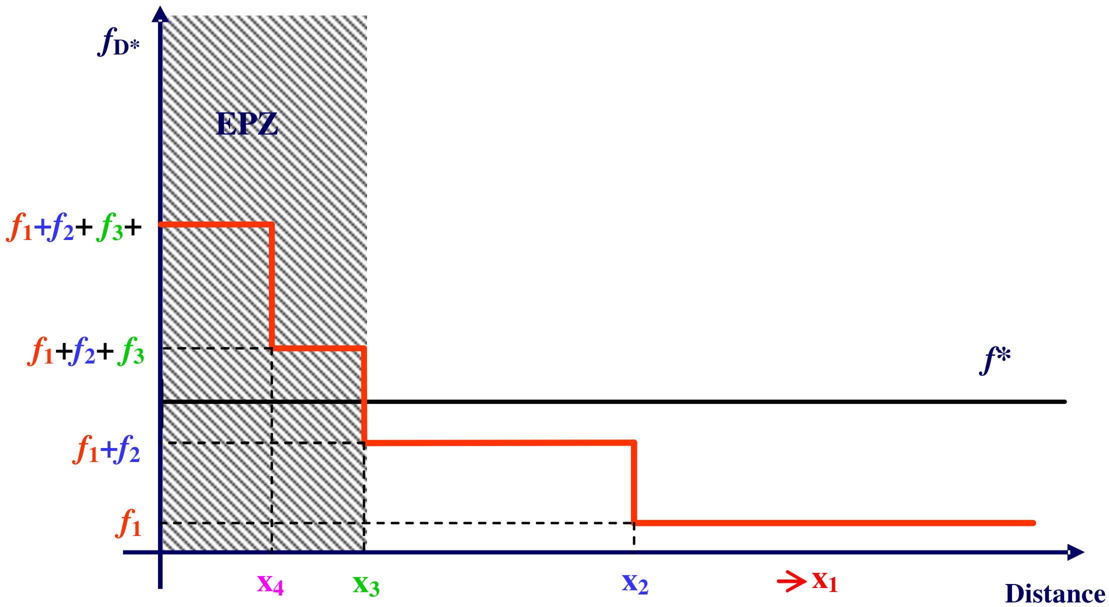

Figure 3-5 Step 5 Final Result: Risk-Informed EPZ Definition

3.2 REVERSE APPLICATION: LIMITING FREQUENCY BACKSOLVING

As previously seen, there are two critical inputs to this analysis that should be interacted upon by the applicant and the licensing authorities:

- The "Dose Limit" ($D*$) that should trigger the required implementation of emergency actions and evacuation procedures;
- The "Frequency of exceeding the Dose Limit" or limiting f_D* ($f*$) that is acceptable for defining the EPZ.

How to choose this second parameter defines this methodology as risk-informed rather than risk-based. In particular, the limiting frequency value, $f*$, can be defined in different ways, and the approach proposed herein is to analyze existing plants using a variation of the described methodology, and to identify the frequency of exceeding the dose limit that corresponds to the current EPZ regulations. Namely, by benchmarking this methodology against currently operating nuclear power plants, each correlation of dose versus distance (evaluated for each A_i accident or group of accidents for currently operating plants) is investigated with a "limiting distance" that is nothing else than the distance currently used for EPZ around NPPs. In this way, from the dose versus distance curves, the dose D_i resulting from the analyzed accident at the given distance is easily identified. This is repeated for each scenario until a family of value of D_i (with $i=1$ to n, where n is the number of considered events) is generated (see Figure 3-6).

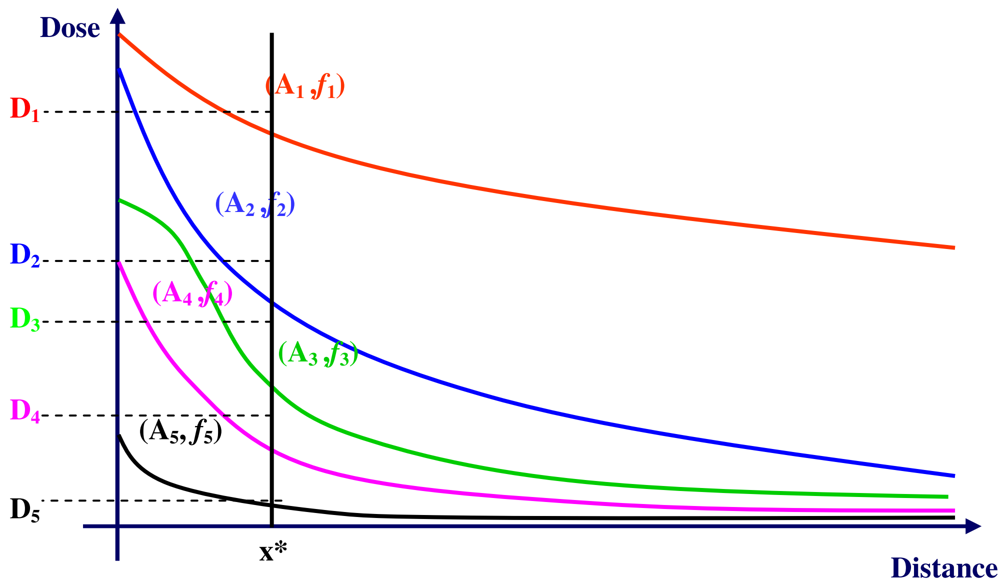

Figure 3-6 Methodology Reverse Application: Step 1

Again, the family of doses values D_i is now a set of discrete random variables with associated frequencies fi obtained from the PRA data. It is now of interest the frequency of having a dose higher than a given value (i.e., $f(D \geq D_i)$), therefore the opposite of the cumulative function (i.e., $f(D \leq D_i)$) can be constructed (see Figure 3-7).

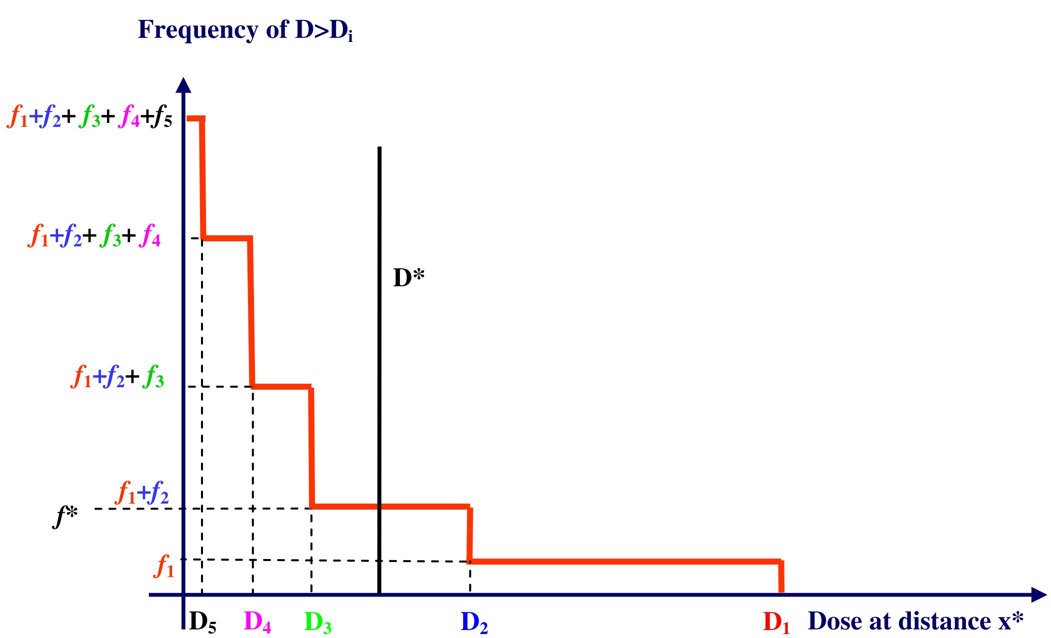

Figure 3-7 Methodology Reverse Application: Limit Frequency Back-solving

Investigating the obtained curve at the limiting dose D^*, a frequency value f^* is inferred. This value can then be used as the limiting frequency value in the definition of the EPZ distance for a new reactor. In other words, the "reverse application" of this methodology can be used to evaluate the risk currently "accepted" for NPPs. This information (i.e., risk-information) can be used to infer the adequate EPZ size for innovative and new NPPs.

4 APPLICATION TO IRIS

4.1 IRIS POTENTIALITY IN TERM OF EPZ REDUCTION

IRIS represents a good test for the proposed approach because it is characterized by an improved safety level, even beyond the advanced passive plants. Nevertheless, within the current regulatory framework, it would still have to be subjected to the same Emergency Planning Requirement. [

][a,c]

It must again be underlined that the reduction of the EPZ should not be misinterpreted as a weakening of the defense-in-depth strategy. Actually, SECY-97-020 Task Force underlined that increased safety in one level of the defense-in-depth framework must be achieved to justify reducing requirements in another level, such as the EPZ, which is the last level of the defense-in-depth strategy. The IRIS Safety-by-Design™, with the elimination of the potential for several accident scenarios (large break Loss of Coolant Accident – LOCA –, rod ejection, …) and the inherent mitigation of others (steam line breaks, small break LOCAs, …) represents the improvement in overall defense in depth that will enable a reduction in the emergency planning zone requirements.

4.2 PRELIMINARY IRIS PRA

IRIS has been primarily focused on achieving a design with innovative safety characteristics. The first line of defense in IRIS, that can be seen as a Level 0 in the defense-in-depth philosophy is the elimination by design of event initiators that could potentially lead to core damage. This is nothing more than a rational and structured application of the idea of “good engineering” as it is recommended in the IAEA safety requirements for the design of a nuclear power plant (Reference 34); in IRIS, this concept is implemented through the Safety-by-Design™ philosophy, which has been already presented in several papers (see for example (Reference 35)) and will not be repeated here.

Also, a PRA-guided, risk-informed approach to the IRIS design phase, was recognized and implemented as one of the key features to fully achieve the potential of the improved safety performance inherent in the IRIS reactor. A detailed review of the development of the IRIS preliminary PRA is out of the scope of the present document and it has been extensively reported (References 36, 37, 38, and 39); nevertheless, since the PRA is one of the bases for the risk-informed methodology previously outlined, it is briefly summarized in this section.

The starting point for the construction of a PRA model is the identification of the Initiating Events (IE). Due to the innovative safety features of IRIS, principally resulting from its integral layout, the standard list of IEs presented in reference documents, such as EPRI NP-2230 (Reference 40), NUREG/CR-3862 (Reference 41) and NUREG/CR-5750 (Reference 42), needed to be carefully reviewed in order to identify those that are still applicable to the IRIS design. The Safety-By-Design™ approach used by IRIS has in fact led to the elimination of several classical event initiators such as large break LOCAs, reactor coolant pump seals LOCAs, control rod ejection, all eliminated by the integral IRIS configuration. Other IEs, even though not eliminated, have been reduced in frequency or in their impact on the reliability of the plant. Table 4-1 summarizes all the IE categories retained for the IRIS PRA model development.

Table 4-1 IRIS Initiating Event List
LOCA from Upper Part of Vessel
LOCA from Lower Part of Vessel
Leakages
Spurious Depressurization
Vessel Ruptures
Interfacing System LOCA
Steam Generator Tube Rupture
Transients
Loss of Feedwater
Loss of Condenser
Loss of Support Systems
Loss of Offsite POWER
Isolable Steam Line Break
Not-Isolable Steam Line Break
Power Excursion
Anticipated Transients Without Scram (ATWS)

For each of the defined initiator category, the plant response to a reference challenge (i.e., a representative of the category) was evaluated with respect to the key safety functions. The response of the plant to various combinations of successes and failures of the systems supporting each safety function was evaluated to determine whether the postulated combination of successes and failures will achieve a safe, stable state or will rather reach a core damage state (due to the preliminary nature of this analysis only a few simplified plant damage states have been considered).

With the exceptions resulting from the fact that IRIS is still in the design phase, the IRIS PRA model has been developed following the guidance of the ASME Standard for PRA for NPP (Reference 43).

The ASME PRA standard defines the Core Damage (CD) as an uncovery and heatup of the reactor core to the point that prolonged oxidation and severe fuel damage is anticipated; this must involve enough of the core to cause a significant release. In the IRIS PRA this definition has been dramatically simplified by assuming CD immediately following the uncovery of the top of the active core[1].

1. To appreciate the degree of conservatism used in this preliminary evaluation the selected core damage definition is so strict that, if applied to classical loop-type pressurized water reactors, would result in all Large Break LOCA events going inevitably to core damage, since the event mitigation following the re-flooding of the core through the accumulators would not be credited.

The failure probabilities of the systems involved in the analyzed sequences were evaluated by means of a classical Fault Tree (FT) analysis. Standard modeling techniques were used to develop the FT models for the IRIS safety systems. The models include pumps, valves, heat exchangers, motive and control power, and actuation signals. Modeled failure modes include demand failures, run failures, standby failures and common cause failures, as appropriate. Since the preliminary design of the main IRIS safety systems was essentially complete, development of the models was straightforward. For the fluid and electrical support systems, where only limited design details ware available, the PRA analysts developed simplified system design diagrams based on equivalent AP1000 systems. These "PRA designs" were reviewed by the system designers to ensure that they were consistent with the designers' understanding of the intended design and operation of the system.

The IRIS Safety-by-Design™ effort included conceiving a simple design with minimal dependence on human intervention. The human errors considered in the preliminary stage are mainly post initiator human errors regarding the emergency response phase. Where the details of the design were sufficient, pre-accident human errors related to component mispositions induced by the envisioned test and maintenance strategies were also introduced. The quantification of the human error probabilities was performed consistently with the methodology outlined in NUREG/CR-1278 (Reference 44).

The IRIS PRA used generic data for quantification of the models. The primary data sources used were the EPRI PRA key assumptions and ground rules reported in the ALWR URD (Reference 22), and the database already used for the Westinghouse AP1000 PRA (Reference 45)., This information was supplemented as needed by data from the NUCLARR database (Reference 46). The multiple Greek letter approach was used for modeling bounding Common Cause Failure (CCF) with the appropriate factors again extracted mainly from the AP1000 PRA.

The IRIS PRA model is still a work in progress and it will be updated as the design is finalized; Table 4-2 summarizes the most updated IRIS internal event Core Damage Frequency (CDF) results. The current IRIS plant internal CDF has been evaluated to be as low as 2.38E-08/ry. The 5%-tile is 1.36E-09 and the 95%-tile is 7.48E-08. This yields a range factor of 7.4. A sensitivity analysis on the truncation level has been performed to ensure that the applied truncation level (i.e., 1.00E-15/ry) reasonably guarantees that all the most significant cut-sets have been retained in the evaluation.

Such a low CDF brought into the light some contributors that were usually not significant at all in the current generation of NPP, characterized by a total CDF ranging from the lower 10^{-5} to the upper 10^{-6} orders of magnitude. The clearest example is represented by the vessel rupture events. This kind of accident event, classically defined as a LOCA from the bottom of the vessel that induces a break flow beyond the capabilities of the safety injection system, is usually assumed to lead directly to CD without any credit for any mitigation strategy (i.e., conditional core damage probability equal to 1). Consistent with other PRA analysis, the suggested IEF, and therefore also the suggested associated CDF, is 1.00E-08 (Reference 45). This value is seen as a measure of the ignorance of the possible failure modes that could lead to a major damage of the vessel rather than an actual evaluation based on analyses and data.

Table 4-2 Updated IRIS PRA Results

Initiator	CDF	%
Vessel Ruptures	1.00E-08	42.00
LOCA from Upper Part of Vessel	8.55E-09	35.90
ATWS	1.86E-09	7.80
Loss of Offsite Power (LOOP)	1.11E-09	4.65
LOCA from Lower Part of Vessel	8.86E-10	3.71
Loss of Support Systems	6.10E-10	2.55
Transients	3.74E-10	1.57
Loss of Condenser	1.64E-10	0.67
Isolable Steam Line Break	1.44E-10	0.60
Spurious Depressurization	6.20E-11	0.24
Interfacing System LOCA	5.00E-11	0.21
Loss of Feedwater	2.41E-11	0.10
Not isolable Steam Line Break	2.22E-12	0.01
Power Excursion	1.20E-12	0.01
Steam Generator Tube Rupture	1.83E-13	<0.01
Leakages	1.63E-13	<0.01
Total	2.38E-08	100

As it can be seen from Table 4-2, such a pre-defined value results in reactor vessel rupture being the principal contributor to the IRIS CDF (i.e., contributing for more than 40% to the overall plant CDF). This is extremely conservative, also considering that because of the IRIS design features the vessel rupture definition is not completely applicable to IRIS (which does not have a safety injection system); moreover the capabilities of the IRIS safety systems appear to be able to manage some degree of vessel ruptures. Finally, the integral layout of IRIS, with internal control rod drive mechanism and the consequent elimination of head penetrations, eliminates one of the most feared and actually (nearly) experienced failure mechanism of the vessel: the head seal LOCA (a la Davis-Besse).

A preliminary model has also been developed to have a first order evaluation of the Large Early Release Frequency (LERF) using as a starting point the NRCs simplified LERF model from NUREG/CR-6595 (Reference 47), which is designed to take maximum advantage of the Level-1 PRA results.

Using a simplified IRIS containment event tree, four main categories have been identified for release paths in IRIS:

- CD sequences with containment bypass,
- CD sequences with containment isolation failure,
- CD sequences with containment failure at low RCS pressure,
- CD sequences with containment failure at high RCS pressure.

Because IRIS is a Pressurized Water Reactor (PWR), these four main categories are not so different from the categories historically used for standard PWRs. Nevertheless, the contributors for some of them needed to be re-evaluated because of the Safety-by-Design™ approach. A summary of the release categories and of the number of CD sequences enclosed in each category can be found in Table 4-3.

Table 4-3 IRIS Preliminary LERF Results

Category	Sequences	%	LERF
Containment Bypass	25	15.31	9.38E-11
Containment isolation failure	178	5.44	3.49E-11
CD at low RCS pressure	91	16.35	1.05E-10
CD at high RCS pressure	87	62.93	4.04E-10
		Total	6.42E-10

The IRIS LERF has been evaluated as 6.42E-10 which is significantly lower than current PWRs and around two orders of magnitude less than other advanced designs (Reference 45).

Obviously, all IRIS PRA results must be considered as preliminary and naturally affected by epistemic uncertainties, i.e., the uncertainty due to the still incomplete design or the not yet available full set of safety analyses. This last kind of uncertainty has been actively reduced during the development and especially during multiple refinements and scope extension (e.g., external events screening and qualitative analysis) of the IRIS PRA model which represents a fundamental step within the framework of the risk-informed IRIS design philosophy.

4.3 RELEASE SCENARIOS DEFINITION

The first step of the suggested EPZ-redefinition methodology is the definition of a set of release scenarios; this is done starting from an adequate grouping of the accident sequences of the PRA. Unlike the grouping of IEs performed as one of the initial steps of the IRIS PRA, the grouping to be performed in this case is focused upon the presence of common release pathways for the off-site dose evaluation, rather then on common mitigation strategies and/or plant responses. All the accident sequences analyzed in the PRA, regardless of the success or failure of the mitigation strategies, need therefore to be re-grouped according with the new criterion.

The re-grouping is performed in a double step fashion. In the initial step a set of release sequences is defined starting from the accident sequences identified in the PRA; this first step takes into consideration deterministic aspects (i.e., the presence of release pathways) as well as probabilistic aspects (i.e., containment failure probability from the IRIS preliminary LERF analysis). The PRA sequences are then quantified and the resulting frequencies combined in order to assign a frequency to each of the release sequences. In the second grouping step, conservative assumptions and simplifications are formulated to reduce the number of release sequences to a reasonably manageable number which must consider both high unlikely releases and low more-probable releases as well.

It must be underlined that the present work on EPZ redefinition methodology was performed in parallel with the continuous updating and development of the IRIS PRA model. As a result of this, and in order to have a fixed basis for the probabilistic considerations required for the purposes of the present work, the IRIS PRA model was frozen as of August 2005. All the following considerations related to the IRIS PRA model are therefore consistent with the frozen model and they do not reflect the modifications subsequently introduced. Internal event CDF for the frozen IRIS PRA model was 2.79E-08/ry.
Figure 4.1 is a schematic summary of the classification process.

Re-categorization: Step I

In the first step of the re-grouping process, the release categories addressed during the preliminary IRIS source term and dose evaluation analysis (Reference 48) are considered as starting points for the definition of a first set of release sequences. The release categories have been matched with the accident sequences (that are grouped in the corresponding Plant Damage States – PDS – categories) defined during the preliminary IRIS PRA.

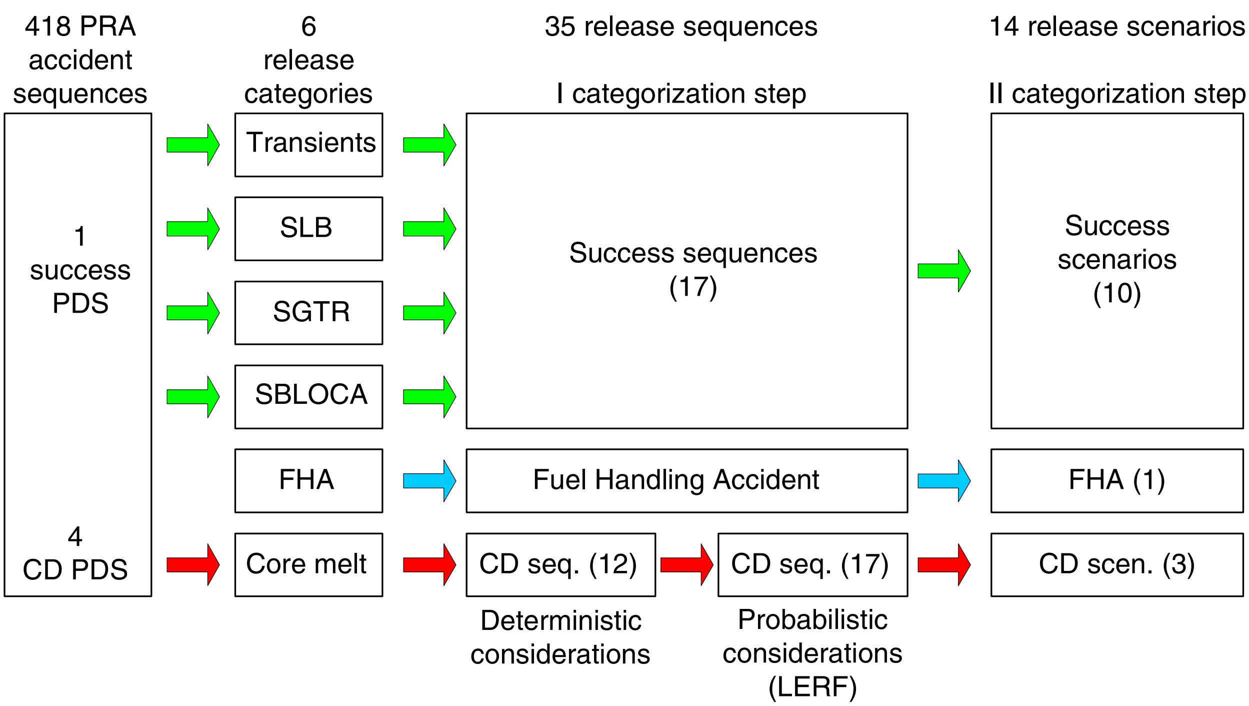

Figure 4-1 Accident Sequence Re-categorization Summary Schematic

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The grouping has been mainly based on the foreseen pathways offered to the radioactivity release following both the onsite of the accident and the actuation of the main mitigation systems; therefore, the differentiation introduced during the PRA study as a result of considerations of single systems success criteria have been removed, also lumping together accident sequences originating from different IEs.

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Success sequences are those that present the higher variability in the dose release, due to the different kind of IE originating the sequence (e.g., a successfully mitigated SGTR could potentially induce an initial higher release outside containment than a successfully mitigated LOCA); for this reason they originate the most numerous release categories and have been modeled in more details The following grouping criteria have been considered:
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Once the 14 release scenarios have been identified, they must be finally provided with the related value of frequency. In case of the release scenarios without CD, the overall frequencies are obtained by simple addition of the frequencies of all the involved accident sequences grouped in the scenario under evaluation. In case of CD scenarios some additional considerations from the preliminary LERF model developed for IRIS have been credited. [

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Table 4-4 summarizes the final list of release scenarios with their frequencies. The grouping and the subsequent definition of the release scenarios to be deterministically evaluated, consider both the deterministic and probabilistic contributions. Successfully mitigated accident sequences have been grouped mainly discerning between high frequency scenarios with low expected releases (e.g., first scenario) and lower frequency scenario with the maximum expected release (e.g., second scenario). The unique features of the IRIS integral design and Safety-by-Design™ approach have been captured by refining the grouping process for the SGTR and Secondary Line Break (SLB; i.e., feed and steam line breaks), with the result of having some release scenario, with very few accident sequences involved and a correspondent extremely low frequency.

Table 4-4 IRIS Release Scenarios

#	Release Scenario Description	Involved PRA Sequences	Overall Frequency [/ry]
1	Transients(1) successfully mitigated via MFWS	15	1.14E+00
2	ATWS successfully mitigated with OTCC	108	8.00E-07
3	SGTR successfully mitigated via MFWS	2	1.77E-04
4	SGTR successfully mitigated via EHRS	2	1.10E-05
5	SGTR successfully mitigated via OTCC	2	1.68E-11
6	Not isolated SGTR successfully mitigated via EHRS	1	1.00E-08
7	Not isolated SGTR successfully mitigated via OTCC	3	2.41E-13
8	Steam Line Break successfully mitigated via EHRS	2	9.68E-04
9	Steam Line Break successfully mitigated via OTCC	12	2.90E-08
10	Small Break LOCA successfully mitigated	25	1.02E-03
11	Early Core Melt with heat removal capability	211(2)	2.00E-08
12	Late Core Melt with heat removal capability		4.47E-10
13	Core Melt with containment failure(3)		7.05E-09
14	Fuel Handling Accidents	N/A(4)	1.00E-04

Notes:
1. Transients includes LOOP, Loss of Condenser, Loss of Support Systems, Transients with Power Excursion, Loss of main feedwater, Transients with main feedwater.
2. The frequency of CD scenarios is not directly dependant on the number of sequences involved since a weight in the frequency is introduced as a result of the Simplified LERF considerations.
3. Late and early core melt scenarios with containment failure are here treated in the same group.
4. Not included in the at-power Level-1 PRA.

The Fuel Handling Accident (FHA) scenario is obviously treated in a different way, since this is not an at-power accident sequence and it is therefore not covered by the Level-1 PRA so far developed for IRIS. A FHA is an event where a fuel assembly is mechanically damaged and gap activity is released. This accident is not normally accounted for as a significant contributor to a plant's total risk profile in modern Low Power Shut Down (LPSD) PRAs. In a recent assessment of shutdown risk performed for the PWR Owners Group (Reference 50), IEs of most interest for a low power PRA have been recognized as those involving loss of cooling capability; FHA has not been recognized as a significant initiating event for low power PRA. This is consistent with the considerations made on FHA and heavy load handling reported in

NUREG-1449 (Reference 51), where compliance of the fuel-handling equipment and procedures with specific standards and technical specifications, the shielding effect due to the fact that all fuel handling procedures are carried on under water with an high boron concentration and the capability of automatically closing all the penetrations through the fuel handling area are listed as reasons why most significant events of concern during low power operation are related with loss of cooling to the spent fuel pit.

Frequency of occurrence of FHA (even tough not in correlation with LPSD PRA modeling) was addressed many years ago in WASH-1400 (Appendix I, p. I-100, of Reference 28) which estimated a frequency of 10^{-4} event/ry (being this the frequency of events in which gap activity is actually released and not the frequency of all events in which a fuel assembly is dropped or otherwise mishandled). The same value is also reported in the more recent NUREG-0933 document (Reference 52).

4.4 RELEASE SCENARIOS DOSE EVALUATION

In the second step of the EPZ redefinition methodology, for each of the 14 release scenarios that have been previously identified a deterministic evaluation of the released dose at a set of distances of interest must be performed.

Since the main focus of the present work is the feasibility study of the methodology, the actual code used for the dose evaluation is of secondary importance. In this present analysis the code used for such evaluation is RADionuclide Transport, Removal and Dose (RADTRAD), a computer code produced by the U.S. NRC and its subcontractor Sandia National Laboratory. The purpose of this code is to predict the transport of fission products throughout the containment and attached buildings and to predict the doses to persons at various locations such as the control room, the exclusion area boundary and the LPZ. RADTRAD has been used by the NRC in generic studies.

RADTRAD is not a code that fully analyzes the severe accidents as a combination of codes like MELCOR and MAAP could do, nevertheless it is used in this first run of the methodology since the lack in the capability of the code is felt to be reasonably counterbalanced by the extremely conservative assumptions related to CD definition. RADTRAD can be considered as composed by two main modules: the first module focuses on the radionuclide transport and removal, the second module is meant to analyze the consequences of the dose that reaches a selected compartment.

RADTRAD includes models for a variety of processes that can transport and/or attenuate radionuclides. It can model sprays and natural deposition that reduce the quantity of radionuclides suspended in the containment or in other compartments. It can model the flow of radionuclides between compartments within a building, from buildings into the environment, and from the environment into the control room. These flows can pass through filters, piping, or simple air leakage. The models for flow through piping can optionally account for aerosol deposition and iodine chemical interaction. RADTRAD can also model radioactive decay and in-growth of daughters.

Additional information and details on the RADTRAD codes are not reported here and can be retrieved from NUREG/CR-6640 (Reference 53). What has been herein reported is intended to give the idea of the level of details covered in the current application of the methodology.

The underlying scheme of RADTRAD is that the user first specifies a nuclear power plant model and then specifies a scenario description. The plant model consists of time-independent information and describes the physical structure of the plant. The user specifies the compartments, the transfer pathways that connect the compartments, and information pertaining to them (e.g., compartment volumes, the existence of filters, and suppression pools). In addition, the user selects an inventory, source term and dose conversion factors. The scenario description contains time-dependent information about the state of the plant and allows the user to run several different scenarios with the same plant model. The user may limit the fission products released as part of the source term. The user is asked to select the fraction of the iodine chemical group that is elemental, organic, and aerosol. The user can also create his own design-specific source term file. For each of the transfer pathways, the user specifies the information necessary to calculate transport and retention through that pathway (e.g., the user specifies filter efficiencies). The user may also specify the locations of interest for calculating worker dose. Integrated air concentrations at the various locations is determined by the code, and the user will input the exposure time, breathing rate, and other specifics needed to calculate dose equivalents.

General practice in dose evaluation by the means of the Alternate Source Term (Reference 54) focuses on accident scenarios such as general transients, SGTR, SLB, LOCA and core melt. Each of these scenarios must be proved to be consistent with particular success criteria and are characterized by some defining assumptions that are not always the same among different cases. The preliminary IRIS Source Term and Dose Evaluation (Reference 48) has been performed according to such assumptions. Since in the herein described methodology the results from the dose evaluations are going to be combined to deduce a risk-informed value for the characteristic distance of the EPZ, the assumptions defining the various release sequences needed to be harmonized in the most extensive fashion reasonably achievable, still maintaining the differences among accident conditions.

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Each of the previously identified release scenario can be modeled through the effect superposition of different source terms pertaining to the scenario; each of the source terms (developed in accordance with the NRC R.G. 1.183 (Reference 52x)) can be then processed more than one time because of the different timing characteristics of each radionuclide group (i.e., Iodine, Alkali and Noble Gases).

The NRC RADTRAD code (Reference 53) used for the preliminary analysis requires dose location definition to evaluate the dose in three locations: control room, exclusion site boundary and LPZ. For the purpose of the EPZ redefinition methodology herein described, evaluation of the dose at the LPZ should be sufficient. Dose evaluation for the LPZ have been calculated using a combination of breathing ratios and atmospheric dispersion factors χ/Q where χ is the short term average centerline value of the ground level concentration (thus expressed in terms of Curie/m^3) and Q is the amount of material released (thus expressed in Curie/s). The atmospheric dispersion factors (thus expressed in terms of s/m^3) are therefore lumped parameters that include considerations on the timing, the elevation of the release and the meteorological condition.

For the current application of the methodology, χ/Q values for distances ranging from the fences of the reactor up to at least the distance corresponding to the current EPZ size (10 miles) are required. Table 4-6 shows the χ/Q values deduced from generic curves identified by NRC (Reference 55) and here reported in Figures 4-3 and 4-4 for distances ranging from 200 m up to 15 km.

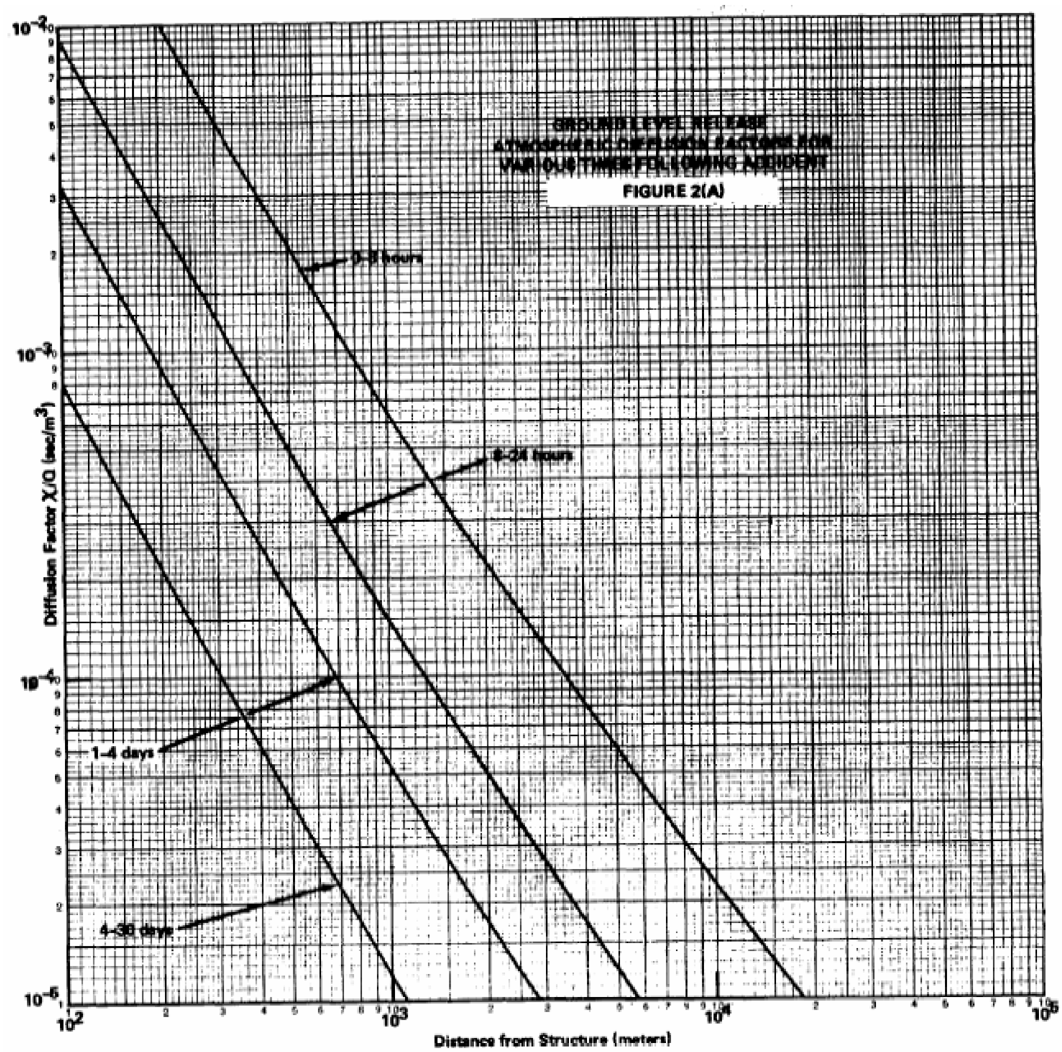

Figure 4-3 Generic Ground-level Release Atmospheric Diffusion Factors for Various Times Following an Accident (Part I)

Table 4-6 Adopted Atmospheric Dispersion Factors

Time [h]	Atmospheric Dispersion Factor χ/Q [s/m^3] for Different Distances [m]									
	200	**1000**	**2000**	**4000**	**5000**	**6000**	**8000**	**10000**	**12000**	**15000**
0	1.00E-02	6.30E-04	2.15E-04	8.00E-05	6.00E-05	4.60E-05	3.00E-05	2.15E-05	1.60E-05	1.13E-05
8	2.33E-03	1.30E-04	5.00E-05	1.60E-05	1.15E-05	9.30E-05	6.00E-06	4.40E-06	3.40E-06	1.26E-06
24	9.00E-04	5.20E-05	1.16E-05	5.30E-06	4.00E-06	3.05E-06	2.00E-06	1.20E-06	1.11E-06	8.00E-07
96	2.05E-04	1.15E-05	3.80E-06	1.16E-06	9.00E-07	6.30E-07	4.15E-07	3.05E-07	2.40E-07	1.16E-07
720	2.05E-04	1.15E-05	3.80E-06	1.16E-06	9.00E-07	6.30E-07	4.15E-07	3.05E-07	2.40E-07	1.16E-07

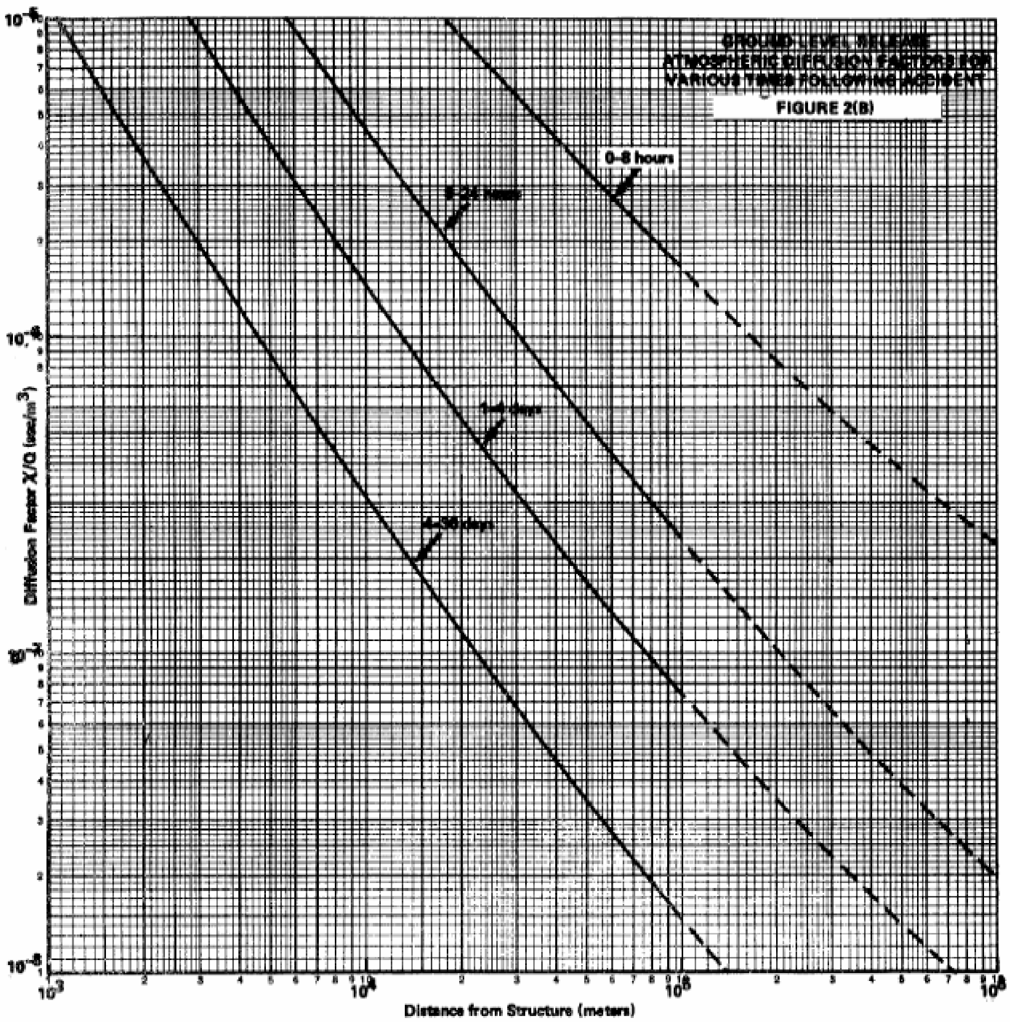

Figure 4-4 Generic Ground-level Release Atmospheric Diffusion Factors for Various Times Following an Accident (Part II)

Using the χ/Q values summarized in Table 4-6 and breathing ratios at different time after the accident deduced from comparison with dose evaluation efforts performed for other PWRs, the dose versus distance evaluation for the identified release scenarios has been performed. Such information has been completed with the information regarding the evolution in time of the accident (i.e., the estimated flow rate from one compartment to another). Data from the preliminary IRIS design (Reference 49) safety analysis (Reference 56) and from the thermal-hydraulic analysis performed to support the preliminary PRA (e.g., see Reference 57) have been extensively used. When such information was not sufficient to describe the evolution of the scenario, conservative assumptions have been adopted.

Only 12 of the 14 release scenarios identified in the previous step of the methodology have been actually modeled and evaluated for the expected released dose; the two omitted scenarios being the core melt with late containment failure and SGTR with failure of both isolation and EHRS and with successful mitigation through the OTCC strategy. The first case has not been evaluated since it is expected to result in a released dose higher than any reasonable limiting dose at all distances of interests; the second case has been merged with a not isolated SGTR mitigated via the EHRS, due to the extremely low frequency and the fact that the expected release dose is also expected to be insignificant.

The results are summarized in Figure 4-5. As expected, all the release scenarios with higher frequency (i.e., the success release scenarios) result in the lower released dose; more unlikely scenarios (i.e., core melt scenarios) result on the other hand in the higher released dose. As it can be seen, the FHA scenario has both intermediate dose release and associated frequency.

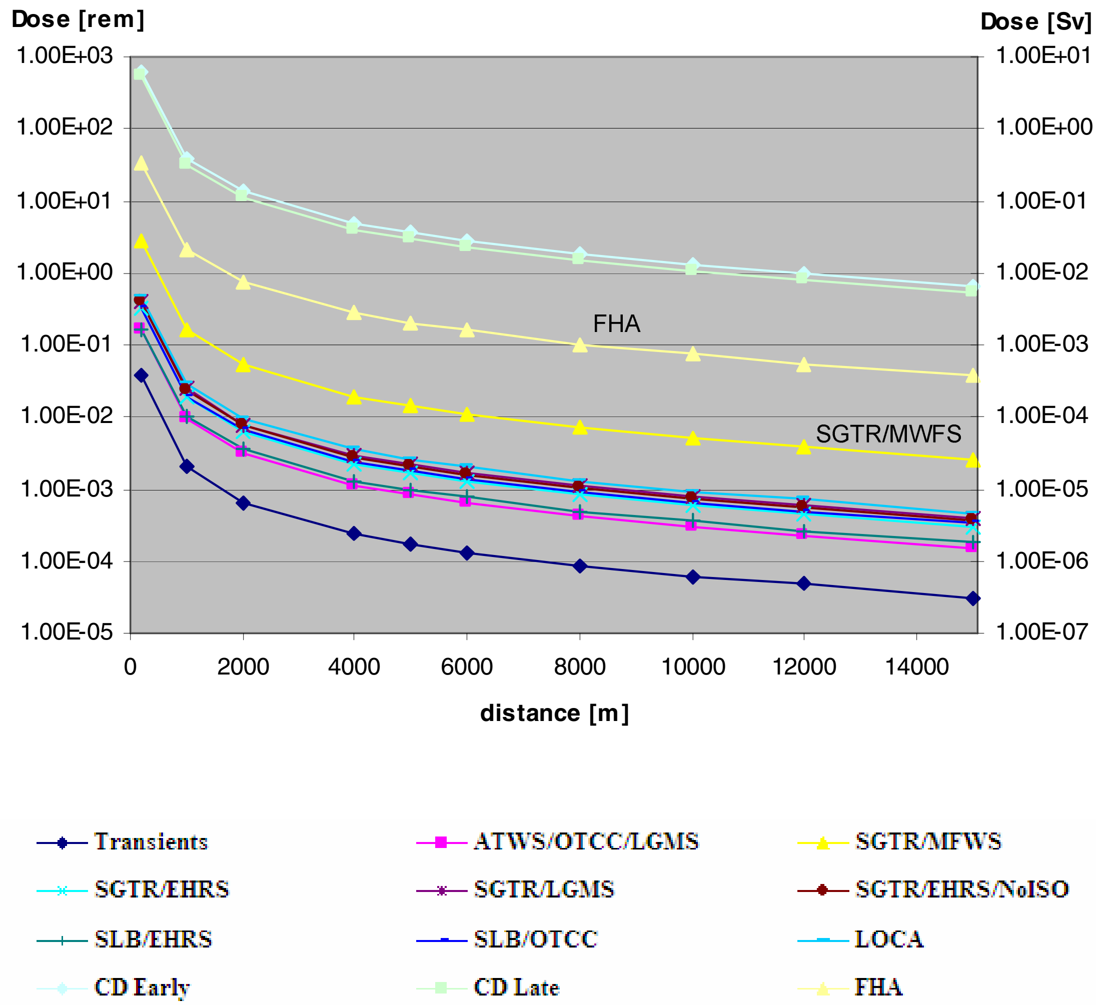

Figure 4-5 Dose versus Distance Evaluation for IRIS Release Scenarios

4.5 LIMITING DOSE AND FREQUENCY IDENTIFICATION

Steps 3 and 4 of the methodology address the identification of a limiting dose and frequency and are steps for which guidance from the regulatory body is obviously necessary. In this feasibility assessment and first application of the methodology the limiting dose D^* and the limiting frequency f^* are proposed for scoping purpose starting from available, pertinent literature. The definition of these two parameters is a sensitive issue since they define the level of safety that a regulatory body considers as appropriate for a nuclear power plant.

The U.S. NRC recently addressed a key policy issue to support near term pre-application reviews of new reactor designs (Reference 58): namely, the minimum level of safety the new plants need to meet to achieve enhanced safety. Such an issue is a central point in a more sound definition of the defense-in-depth concept and clearly appears to be a pre-requisite for the assessment of possible modifications to emergency preparedness requirements.

The position of NRC looks back at the definition of the Qualitative Health Objectives (QHOs) in the Commission's policy statements on "Safety Goals for the Operation of Nuclear Power Plants" (Reference 59). In 1986 the NRC adopted the two following qualitative safety goals for operation of nuclear power plants:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that they bear no significant additional risk to life and health;

- Societal risk to life and health from nuclear power plant 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.

In order to gauge these two qualitative safety goals (respectively addressing early and latent fatalities), the NRC introduced the two following QHOs:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accident to which members of the US population are generally exposed;

- The risk to the population in the area of nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from other causes.

These QHOs have been translated into two numerical objectives, as follows:

- The individual risk of a prompt fatality from all "other accident to which the member of the U.S. population are generally exposed," such as fatal automobile accident, etc., is about $5{\cdot}10^{-4}$ per year[3]. One-tenth of one percent of this figure implies that the individual risk of prompt fatality from a reactor accident should be less than $5{\cdot}10^{-7}$ per reactor year (/ry);

3. The individual risk is determined by dividing the number of prompt or early fatalities (societal risk) to 1 mile from the plant site boundary due to all accidents, weighted by the frequency of each accident, by the total population to 1mile and summing over all accidents.

- The sum of cancer fatality risks from all other causes is taken to be the cancer fatality rate in the U.S. which is about 1 in 500 or $2 \cdot 10^{-3}$ per year[4]. One-tenth of one percent of this implies that the risk of cancer to the population in the area near a nuclear power plant due to its operation should be limited to $2 \cdot 10^{-6}$/ry.

The above considerations, even though not currently directly concurring in the definition of the size of the off-site emergency planning zone, should be considered when identifying the limiting values for frequency and dose required for the methodology herein suggested.

The limiting dose *D** is defined considering the indication of the U.S. EPA contained in the definition of the PAG. In its more recent version (Reference 16), a PAG is defined by the U.S. EPA as the projected dose to the reference man, or other defined individual, from an unplanned release of radioactive material at which a specific protective action to reduce or avoid that dose is recommended.

The protective actions available to avoid or reduce radiation dose can be categorized as a function of exposure pathway and incident phase; for example, evacuation and sheltering are the principal protective actions for use during the early phase (also referred to as the emergency phase) i.e., the period at the beginning of a nuclear incident when immediate decisions for effective use of protective actions are required.

The PAGs do not imply an acceptable level of risk for normal (non-emergency) conditions as they do not represent the boundary between safe and unsafe conditions; rather, they are approximate levels at which the associated protective actions are justified. Furthermore, under emergency conditions, in addition to the protective actions specifically identified for application of PAGs, any other reasonable measure available should be taken to minimize radiation exposure of the general public and of emergency workers.

It is important to notice that, according to EPA, it is not appropriate to use the maximum distance where a PAG might be exceeded as the basis for establishing the boundary of the EPZ for a facility since, for example, the choice of EPZs for commercial nuclear power facilities has been based primarily on consideration of the area needed to assure an adequate planning basis for local response functions and the area is which acute health effects could occur.

The principles for the establishing of the PAG are described in Appendix B of Reference 59. Because of the variable time over which a dose may be delivered and to account for the different risk characterizing different body parts and different kind of radiations, the PAGs are expressed in term of a quantity called the “Committed Effective Dose Equivalent” (CEDE).

4. The cancer risk is determined by evaluating the number of latent cancer (societal risk) due to all accidents to a distance of 10 miles from the plant site boundary, weighted by the frequency of the accident, dividing for the total population to 10 miles, and summing over all accidents.

Four principles provide the basis for establishing the numerical values for PAGs:

1. Acute effects on health should be avoided;

2. The risk of delayed effects on health should not exceed upper bounds that are judged to be adequately protective of public health, under emergency conditions, and are reasonably achievable. It is stated that there is no dose value below which no risk can be assumed to exist.

3. PAGs should not be higher than justified on the basis of optimization of cost and the collective risk of effects on health. That is, any reduction of risk to public health achievable at acceptable cost should be carried out;

4. Regardless of the above principles, the risk to health from the protective action should not itself exceed the risk to health from the dose that would be avoided.

Detailed description of how these four basis principles have been reflected in the actual definition of the EPA PAGs is beyond the purpose of this document. In Table 4-7 the PAGs for both the early and the intermediate phase of a nuclear incident, as suggested in Reference 16, are summarized.

Table 4-7 Summary of EPA PAG

Protective Action	PAG (projected dose)	Comments
Early Phase (up to 4 days from the onset of the event)		
Evacuation (or sheltering)	1-5 rem(1)	Evacuation (or, for some situations, sheltering) should normally be initiated at 1 rem.
Administration of stable iodine	25 rem(2)	
Intermediate Phase (up to months after the accident)		
Relocate the general population	$\geq$2 rem	β dose to skin may be up to 50 times higher
Apply simple dose reduction techniques	<2 rem	
Late phase (up to years after the accident)		
Not provided in Reference 16		

Notes:
1. This is sum of the effective dose equivalent resulting from exposure to external sources and the committed effective dose equivalent incurred from all significant inhalation pathways during the early phase. Committed dose equivalents to the thyroid and to the skin may be 5 and 50 times larger, respectively
2. Committed dose equivalent to the thyroid from radioiodine

Consistently with these considerations, a limiting dose value D^* of 1 rem appears as a reasonable choice.

It must also be observed that the selected values are not specific and unique to the American reality; the IAEA indications (Reference 32) are of the same order of magnitude. The Japanese EPZ also appears to be based on a limiting dose of 1 rem: namely, EPZ around Japanese NPPs is defined with a radius ranging from 8 to 10 km and technical specifications supporting this choice are provided in terms of dose versus distance diagrams (regarding both PWR and BWR cases) obtained with different source terms and scenarios. Figure 4-6 (translated from the original Japanese version) shows the dose to distance diagram for one of the two PWR cases reported as an example in Reference 60. As can be deduced, the expected dose at the boundary of the EPZ is of the order of 1 rem.

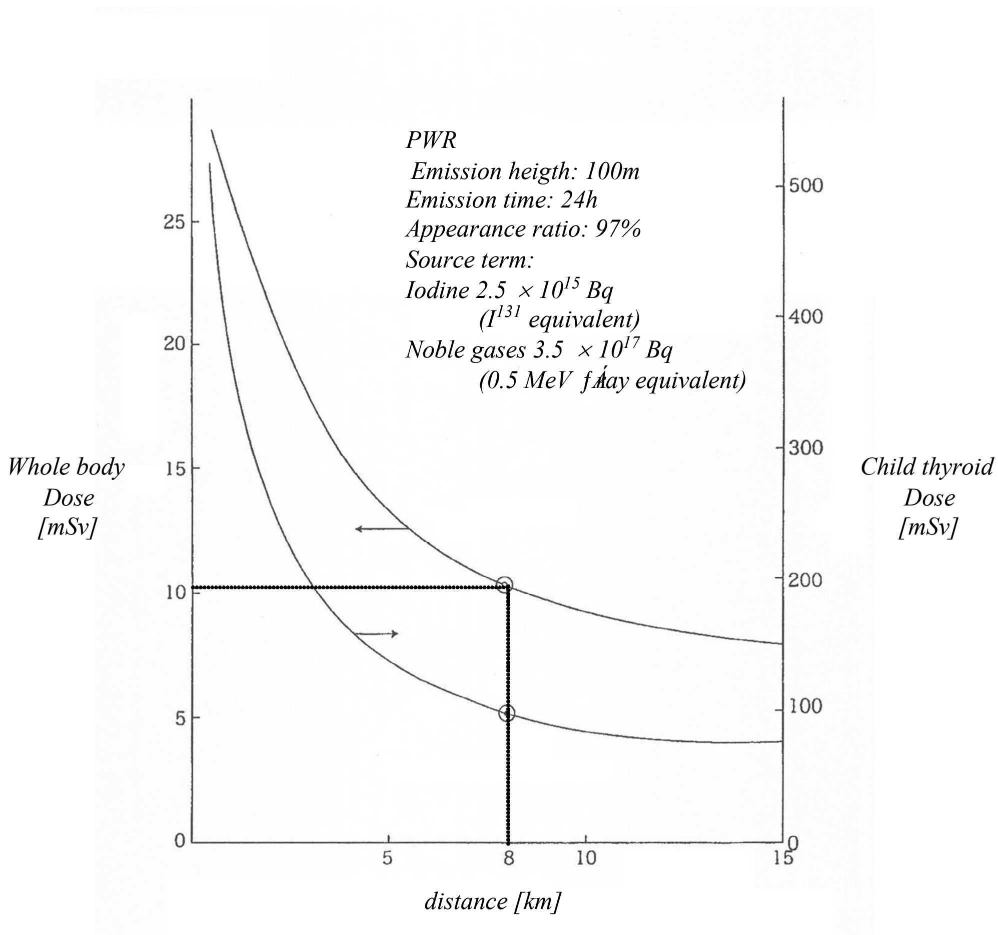

Figure 4-6 Japanese Technical Basis for EPZ Size Definition

As far as the identification of the limiting frequency *f** is concerned, which can be a more controversial matter, reliance is given on the EPRI literature study (Reference 9) presented in previous sections that identifies $1 \cdot 10^{-7}$/ry as a value of general consensus for a meaningful decision-making process.

4.6 IRIS EPZ IDENTIFICATION

The last step of the EPZ redefinition methodology is the definition of the EPZ size; this is done by investigating each dose versus distance curve evaluated for the identified release scenario in order to establish a curve giving the overall frequency of overcoming the limiting dose. Figure 4-7 shows the IRIS dose distance curves reported in Figure 4-5 but now with the limiting dose D^* superimposed; this allows the identification of a set of crossing distances x_i, which are summarized in Table 4-8.

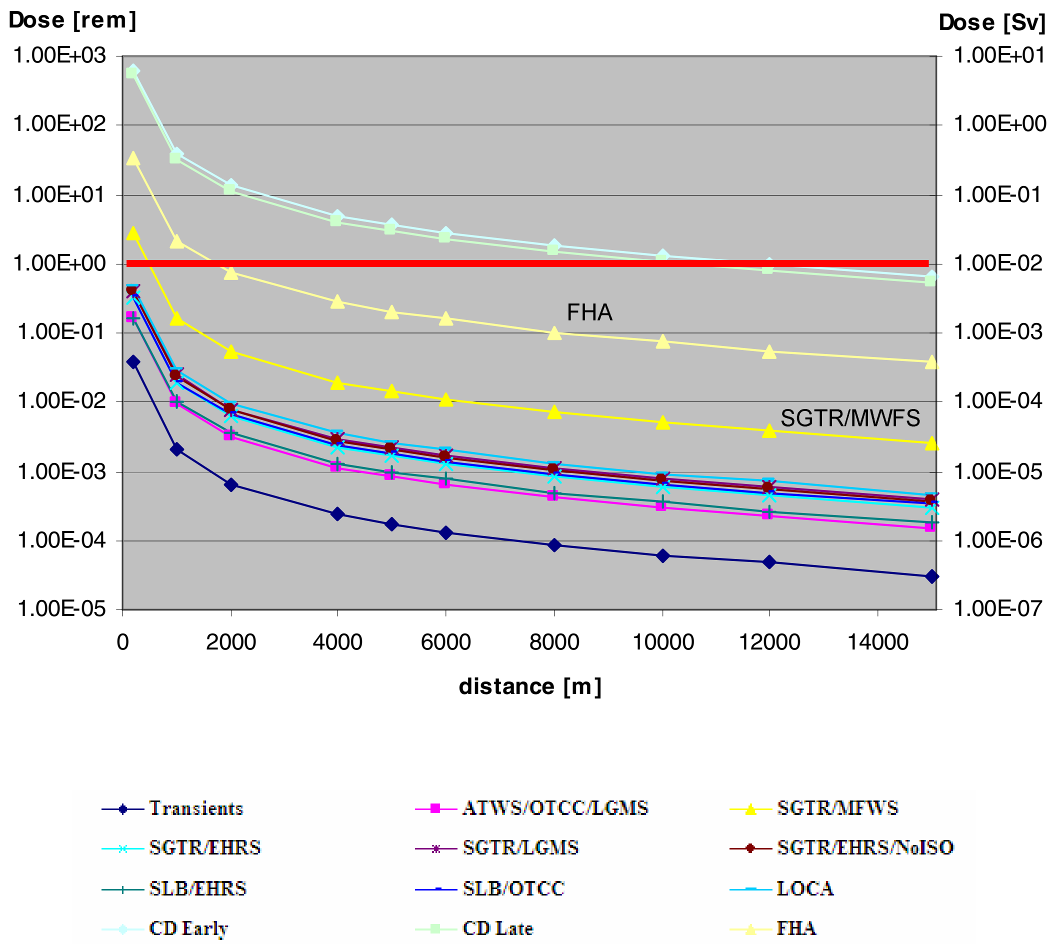

Figure 4-7 Dose versus Distance Evaluation for IRIS Release Scenarios with Superimposed *D**

The information summarized in Table 4-8 are combined to obtain the diagram of Figure 4-8, that identifies the overall frequency of overcoming the limiting dose D^* as a function of the distance from the plant. When this curve is investigated with the limiting frequency f^*, the IRIS EPZ can be identified as an area with a radius of 1800m.

Table 4-8 Crossing Distance Set

#	Release Scenario Description	Overall Frequency [/ry]	*D** Crossing Distance [m]
1	Transients successfully mitigated via MFWS	1.14E+00	200
2	ATWS successfully mitigated with OTCC	8.00E-07	200
3	SGTR successfully mitigated via MFWS	1.77E-04	600
4	SGTR successfully mitigated via EHRS	1.10E-05	200
5	SGTR successfully mitigated via OTCC	1.68E-11	200
6	Not isolated SGTR successfully mitigated via EHRS	1.00E-08	200
7	Not isolated SGTR successfully mitigated via OTCC(1)	2.41E-13	N/A
8	Steam Line Break successfully mitigated via EHRS	9.23E-04	200
9	Steam Line Break successfully mitigated via OTCC	2.91E-08	200
10	Small Break LOCA successfully mitigated	1.02E-03	200
11	Early Core Melt with heat removal capability	1.97E-08	11800
12	Late Core Melt with heat removal capability	4.52E-10	10600
13	Core Melt with containment failure(2)	6.85E-09	∞
14	Fuel Handling Accidents	1.00E-04	1800

Notes:
1. This scenario has not been evaluated and is merged with case 6.
2. This scenario has not been evaluated and an infinite distance is assumed.

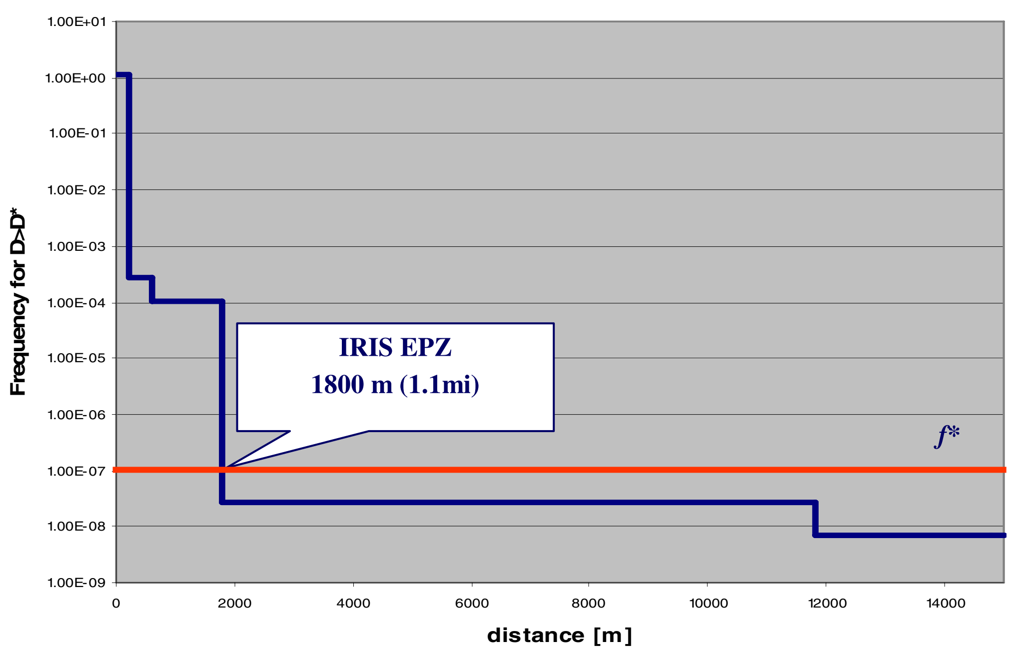

Figure 4-8 IRIS EPZ Identification

The impact of uncertainties must always be considered in the results of this first application of the EPZ redefining methodology to the IRIS plant. As previously underlined, the state of the IRIS design is not able to support an exhaustive uncertainty analysis. To have a degree of confidence in the above presented results, a set of sensitivity studies have therefore been performed for *D** and *f** which are both the most uncertain and most critical parameters in the definition of the EPZ size.

Various limiting doses *D** were investigated ranging from the base case of 1 rem down to 0.1 rem, in steps of 0.1 rem each. Varying the limiting dose *D** means changing the set of crossing distances x_i in the dose versus distance diagram for all the evaluated release scenarios

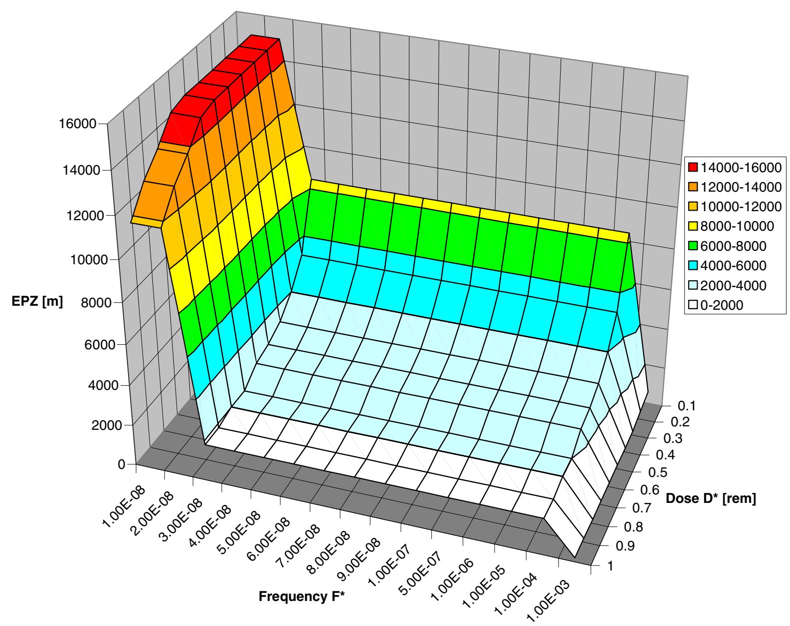

Figure 4-9 Sensitivity Analysis on *D and *f****

Similarly, various limiting frequencies f* are considered; the interval between $1{\cdot}10^{-8}$/ry and $1{\cdot}10^{-7}$/ry is investigated in steps of $0.1{\cdot}10^{-7}$/ry. The higher frequencies space has also been investigated, with a coarser approach, up to $1{\cdot}10^{-3}$/ry. Figure 4-9 summarizes the results of the sensitivity study, providing a surface that identifies an EPZ size for each combination of limiting dose and limiting frequency.

From Table 4-8 and Figure 4-9 it is apparent that the contribution of the FHA, relatively high both in released dose and in the frequency, is always the one determining the EPZ size; at the fixed limiting frequency *f**, the EPZ size as a function of the limiting dose *D** slowly changes following the FHA dose to distance diagram. If, on the other hand, the limiting dose *D** is maintained fixed, the EPZ size as a function of the limiting frequency shows a dramatic rise around $3{\cdot}10^{-8}$/ry, when also the contributions from core melt scenarios come into account, it is maintained in a stable plateau between $3{\cdot}10^{-8}$/ry and $1{\cdot}10^{-3}$/ry and than drops to 200m.

From the performed sensitivity analysis it appears clear that the reference case is on a relatively stable plateau on the surface identified in Figure 4-9. Being reasonably far from any areas with significant gradients could be an indication of the soundness of the obtained results, especially considering the extremely conservative assumptions both in the probabilistic and deterministic parts of the analysis.

Due to the significant importance of the FHA, some additional considerations have been made on this scenario. As seen in Section 4.3, the IEF pertaining to a FHA is given in the WASH-1400 document (Reference 28, Appendix I, Section 5.3) as 1.00E-04/ry. This was estimated considering that accidents that can occur during refueling result either in mechanical damage to the fuel or in inhibiting heat transfer from the spent fuel being handled. Thus:

- Mechanical damage: the probability of release due to mechanical damage will likely be dominated by crane failures. Using a crane failure probability of 3.00E-06/operating hour and assuming 100 hours of crane operation per refueling leads to a prediction of the probability of clad failure due to mechanical damage of the order of 10^{-4}/year.

- Inhibition of heat transfer: calculations indicate that even if a fuel assembly is completely withdrawn from the refueling canal or spent fuel pool, air convection cooling is adequate to prevent fuel melting. In addition, plants are designed in such a manner that it is physically impossible to completely withdraw a fuel assembly from the water using normal refueling equipment. As a result of these considerations the contribution to the FHA IEF coming from inhibition of heat transfer can be ignored.

In accordance with the previous considerations, issue 192 of NUREG-0933 (Reference 52) suggested an IEF for a FHA event of 1.00E-04/ry. The two determinant factors in the estimation of the FHA IEF appear therefore to be the hourly crane failure probability and the hours of crane operation per refueling. The combination of these two factors results in a accident probability per refueling that must be then combined with the fuel cycle (i.e., with the number of refueling per year) in order to obtain the IEF. In this extremely simplified approach, the longer fuel cycle envisioned for the IRIS could result in a reduction of the FHA IEF. The FHA IEF for the straight burn (4 years) IRIS fuel cycle is:

$$\text{FHA IEF} = (3.00\text{E-}6/\text{hr}) \cdot (1.00\text{E+}02\ \text{hrs})/4\ \text{years} = 7.50\text{E-}05/\text{y}$$

Such value must be considered only as a scoping value and must be further discussed and studied in more depth, especially considering that the reference values are dating back to a 1975 document such as the WASH-1400.

The very conservative evaluation of Figure 4-8 does not credit any form of filtering of the postulated release; in PRA terms, the FHA scenario so far considered can be summarized in an ET with probability 1.0 of unavailability of any form of filtering during the refueling procedure. As previously mentioned, NUREG 1499 lists a comprehensive set of features which need to be in place during refueling procedures (e.g., filters, automatic isolation capability of the refueling area). If filtering availability is credited, a new probability must be introduced.

Qualitative considerations to determine filters unavailability probability are:

- The filter failure mode involved in this scenario is only the filter misplacement (or the lack of filters at all), since filter clogging has a beneficial effect during a FHA;
- Filters misplacement (or lack of filters in selected positions) has few envisioned mechanical causes, the main cause for such an event being, reasonably, an human error during the filter placement before the actual initiation of the fuel handling phase;
- A strict refueling procedure can be envisioned which put confirmation of filter positioning as a conditional step for the initiation of the fuel handling procedure; moreover, a set of indicators and alarms can be assumed to enforce such a procedure;
- A human error related to the positioning of the filters is reasonably not related with the human error currently envisioned as the cause of the initiation of the FHA scenario since the latter is focused on a very specific task (i.e., the handling of the fuel with a polar crane).

Crediting a failure probability for the filtering capability, a new scenario is made possible, where the filters are available during the refueling procedure. The FHA scenario is then split into two contributors (i.e., FHA with filtering and FHA without filtering); the deterministic evaluation of the newly defined scenario is performed assuming filtering performance deduced from existing power plants analysis and is summarized in Figure 4-10.

A quantitative evaluation of the filtering capability failure probability is not part of the herein presented study and is currently not performed for IRIS due to early phase in the definition of IRIS refueling procedures and auxiliary building layout with its relationship with the IRIS containment. For the purpose of the current analysis, a simple parametric analysis has been performed with the probability of filtering failure ranging from 1.00E-01 to 1.00E-04 which is within the range of other IRIS safety systems failure probabilities (e.g., the EHRS).

The updated frequency of overcoming the limiting dose D* as a function of the distance from the plant is presented in Figure 4-11 for the different filtering capability failure probability failures.

With the current limiting frequency, a filtering capability failure probability comprised between 1.00E-03 and 1.00E-04 is the threshold below which the IRIS EPZ can be reduced as low as 800m in radius.

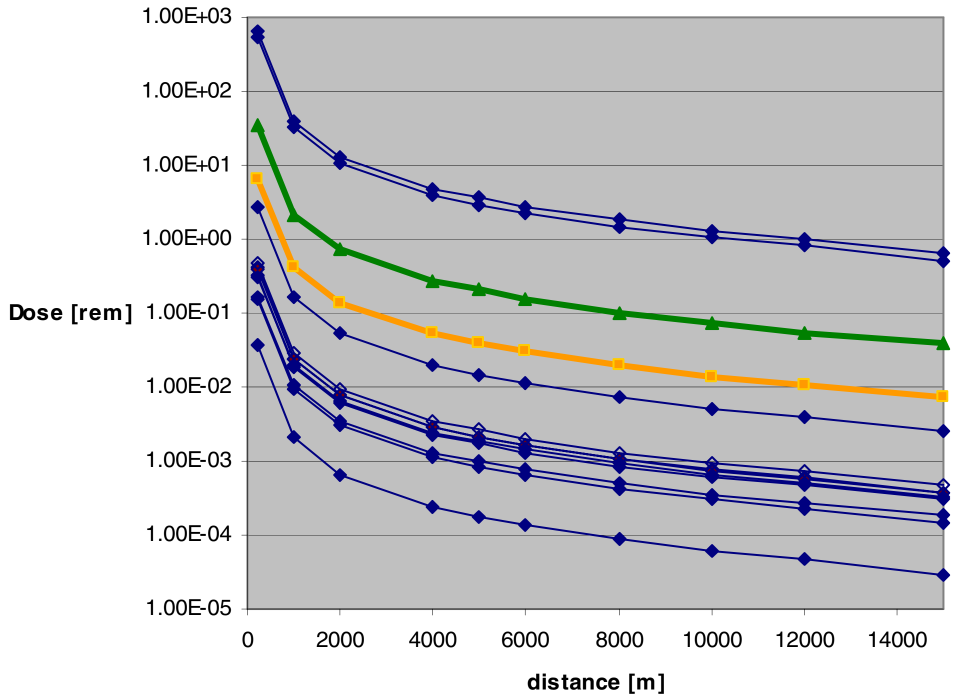

Figure 4-10 New Dose versus Distance Evaluation – The original FHA scenario (in green) is split into two contributions; the yellow one is the new contribution depicting a FHA with successful filtering while the green is the old FHA scenario without filtering capability.

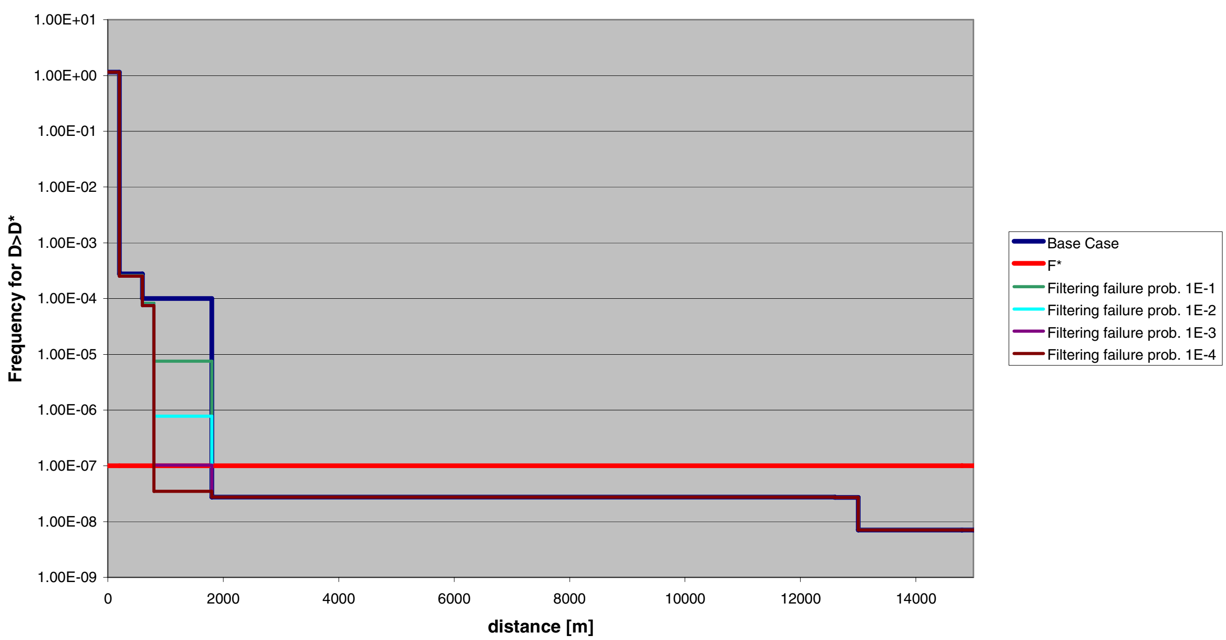

Figure 4-11 Frequency of a Released Dose Higher than the Limiting Dose as a Function of the Distance – Base case refers to the results reported in Figure 4-8 – With a limiting frequency of 1.00E-07, the EPZ can be reduced up to 800m when the probability of failure of filtering capability is 1.00E-03<pFHA<1.00E-04.

4.7 IRIS RISK-INFORMED DESIGN AND EPZ REDEFINITION

The results herein presented for a possible IRIS EPZ definition must be considered preliminary and will be updated as the design progresses. They are nevertheless indicative of the effectiveness of the designing philosophy (i.e., the Safety-by-Design™ approach) which has been the basis of this project from its inception.

These results must also be considered in the framework of the high degree of conservatism adopted for some of the most significant assumptions, such as:

[

]a,c

5 CONCLUSIONS

5.1 CONCEPT AND APPLICATION

The quality of PRA techniques adopted as the main supporting tool for risk-informed applications has been continuously increasing over the past decades. A full power internal events PRA standard has been developed by ASME and endorsed by the U.S. NRC (Reference 62) as the basis for evaluating the quality of PRA models and assess their applicability for use in a broad spectrum of risk-informed applications. As part of this continuous effort, a combined ANS/ASME PRA standard extending to external events and internal fire is being finalized[1] and expected to be endorsed as well by the NRC[2]. Low Power PRA and Level-2 PRA standards are currently being developed or planned. The maturity of the techniques involved in the risk-informed approach and concept suggests now the possibility of considering the extension of the range of potential risk-informed applications to the last level of the defense in depth philosophy: the off-site emergency preparedness.

In the present document the current bases for the definition of the Emergency Planning Zone have been reviewed with the intent of re-introducing the concept of risk, previously ruled out due to technical considerations but especially due to public concern and distorted risk perception. A conceptual methodology is then presented, which would allow relating the size of the Emergency Planning Zone to the safety performance of a plant design, thus recognizing the enhancement in safety attained by new plant designs during the last thirty years (i.e., since when the basis for emergency planning have been defined).

The methodology, which allows for a bridge (i.e., applicable to a relatively early design phase) towards the use of a full scope Level-3 PRA as the reference supporting tool in the definition and sizing of Emergency Planning around nuclear power plant, builds on the fundamentals of the concept of risk: i.e., a potentially complete probabilistic approach to the entire spectrum of accident scenarios and the deterministic evaluation of consequences through dose and dispersion analysis. The simplified approach used for the test bed herein investigated (the IRIS reactor) was geared towards a feasibility and conceptual test of the methodology rather than towards the details of the analysis implementation. Nevertheless, the very preliminary results show the potential for a significant reduction in the size of the EPZ for a small/medium nuclear plant like IRIS.

5.2 QUALITATIVE IMPACT OF THE EPZ REDEFINITION

Having so far underlined the potential for EPZ reduction in term of the risk to the population (that is actually maintained equal to current PWR), a semi-qualitative example is here reported in order to outline the potential benefit that the EPZ reduction could represent in term of the burden that an utility has to carry for satisfying the current requirements. This example is applied to an Italian site for a general understanding of the potentiality of the methodology. Figure 5-1 identifies the EPZ size pertaining to an IRIS reactor hypothetically built on the site of the Caorso NPP (currently under decommissioning), in northern Italy.

1. The combined ANS/ASME PRA standard is expected to be published in the first half of 2009.
2. Revision 2 of U.S. NRC RG 1.200 is expected to be released for public comments in the first half of 2009.

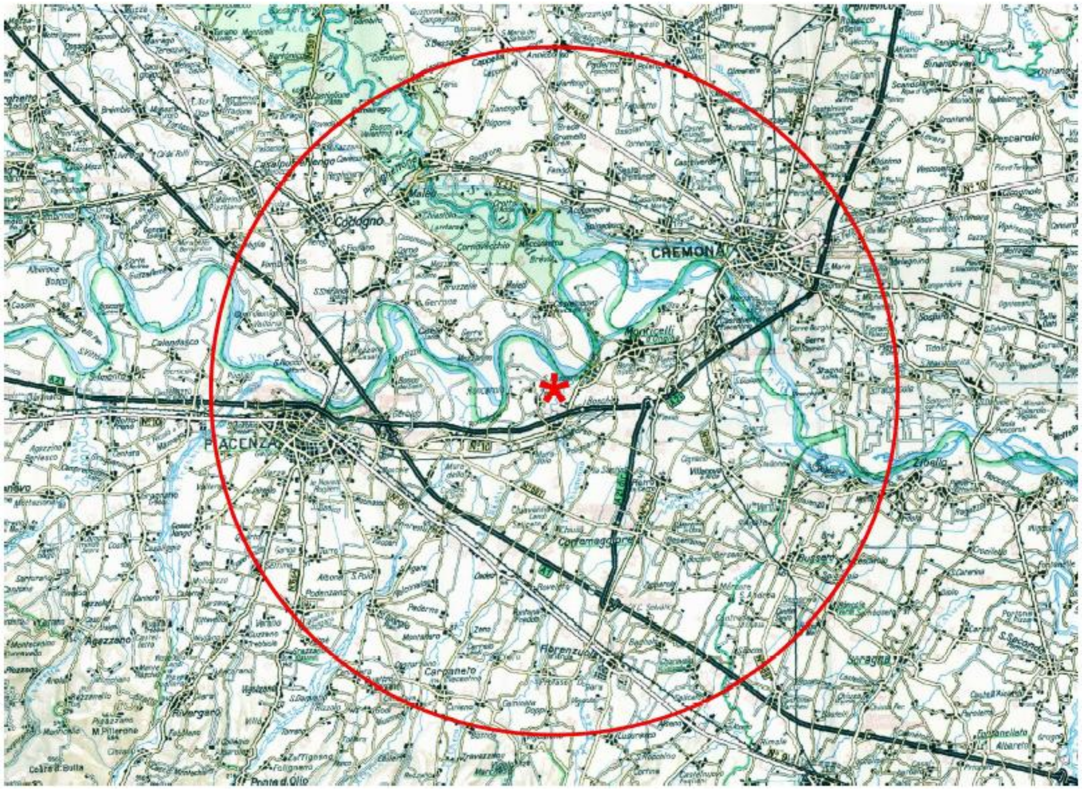

Figure 5-1 Hypothetical EPZ for an IRIS Located on the Caorso Site in Accordance with Current NRC Requirements

The red line identifies the EPZ size in accordance with the current NRC requirements (i.e., 10 miles).

Figure 5-2 reports the IRIS EPZ relative to the Caorso site as identified by the herein outlined methodology, which would be reduced to slightly less than 2 km (base case) or even 1 km (in case the effect of the FHA can be further reduced).

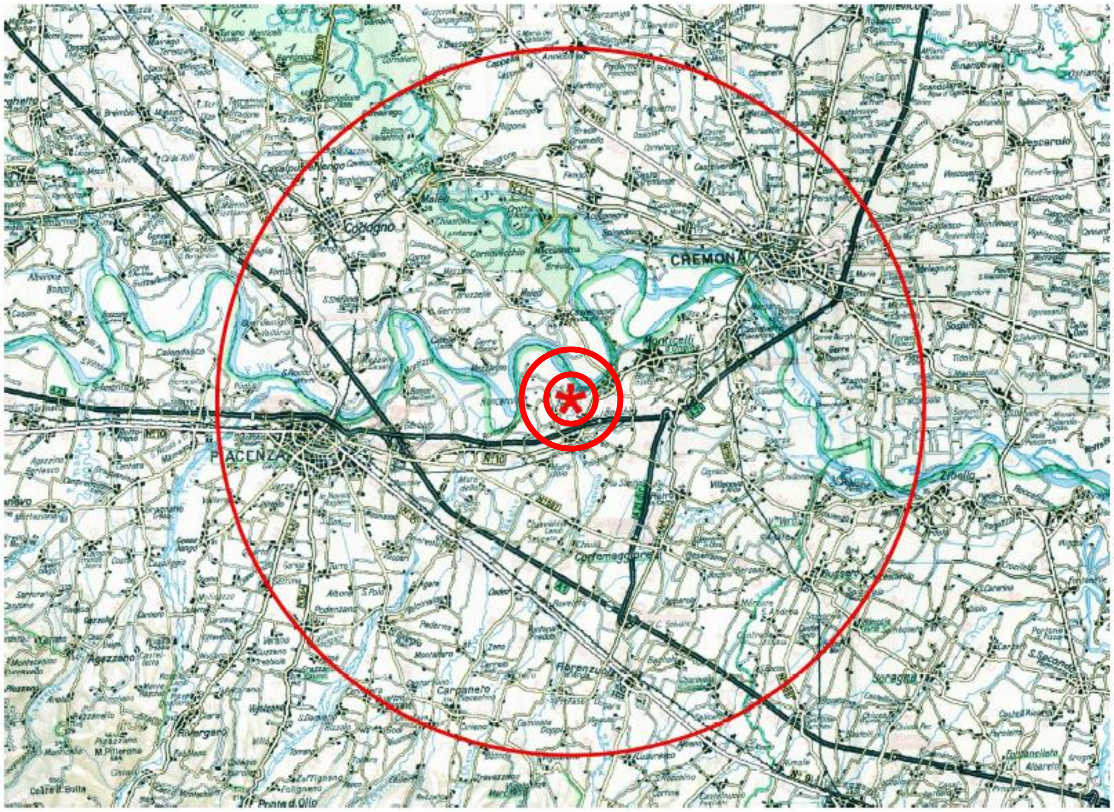

Figure 5-2 Hypothetical EPZ for an IRIS Located on the Caorso Site – Potential Reduction including or excluding FHA scenarios

The outer circle identifies the EPZ size in accordance with the current NRC requirements (i.e., 10 miles). The two inner circles indicate the IRIS EPZ relative to the Caorso site as identified by the herein outlined methodology, which would be reduced to slightly less than 2 km (base case) or even 1 km (in case the effect of the FHA can be further reduced).

A detailed description of the practical aspects involved in the enforcement of the EPZ requirements is beyond the scope of the current work and can be retrieved in the already quoted IAEA coordinated research project on EPZ for small and modular reactor of which this study constitutes only the initial and methodological part. Even without entering in the details, the beneficial impact on the economics of the hypothetical utility managing the "Caorso IRIS NPP" is easily understandable noticing the two relatively big population centers of Piacenza and Cremona (with up to 180,000 people in these two cities alone) being now excluded by the newly defined EPZ.

While the benefit for such a reduction for the utility and the nuclear industry is apparent, the main benefit for the final stakeholder (i.e., the public) is a reduced impact of the presence of the NPP from under the economical and social points of view, due to the increase in safety and a corresponding reduction of the burden associated with outside emergency planning.

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