STAFF MEMBER

NRC Comments

NEI

ESP-6
Radiological Consequence Assessment of Design Basis Accidents For Preparation of Early Site Permit Environmental Report

Presentation to the NRC

December 5, 2002

NEI

INDUSTRY APPROACH

design-basis

- Applicants for Early Site Permits will evaluate a spectrum of representative accidents to assess the environmental radiological consequences associated with the alternative reactor technologies being considered for an ESP site.
- Selection of the accidents is based on the accidents identified in NRC regulatory guidance.
- Applicants will use the 50 percentile post-accident site dispersion χ/Q factors at the exclusion and low population zone boundaries to perform the assessments.

  for certified reactor designs
- The radiological consequences of the selected design basis accidents will be assessed using the activity released to the environs as provided in the standard safety analysis reports or as specified by the reactor vendor for non-certified reactor designs.

  source term
- The released activities account for the reactor core and accident mitigation features in the reactor vendor's standard plant designs for certified reactor designs or as specified by the reactor vendor for non-certified reactor designs.

SOURCE TERMS AND ACTIVITY RELEASES

- The reactor technologies use different source terms and approaches to define the post accident activity releases.
- The Advanced Boiling Water Reactor's (ABWR) source term ~~is~~ and releases are based on TID-14844.
- The AP-1000 Pressurized Water Reactor's source term and accident approaches are based on the Alternate Source Term (AST) in accordance with Regulatory Guide 1.183. AP 600 and/or AP 1000 source terms and releases bound the limiting accident release for the IRIS advanced reactor.
- Source terms activity releases associated with the limiting DBA are identified by the vendors of the non-LWR advanced reactor designs. These designs include: GT-MHR, PBMR, and the ACR-700.
- the ABWR source term may or may not bound the ESBWR or SWR-1000 reactors.

SELECTION OF DESIGN BASIS ACCIDENTS

- A representative set of design basis accidents is selected based on applicable regulatory guidance for assessing the radiological consequences of accidents in the ESP ER is provided in Regulatory Guide 1.183 and NUREG 1555.

- Current USNRC guidance focuses on LWRs. It does not specifically address the other reactor concepts considered in the ESP effort.

- Reasonably representative and bounding DBAs events for the BWR, PWR and non-LWR designs have been chosen

SELECTION OF DESIGN BASIS ACCIDENTS
Comparison of LWR Accidents Addressed by RG 1.183

- ABWR

> **RG 1.183 Listed Accidents Analyzed in Certification Package:**

Loss of Coolant Accident (LOCA): TID Source Term used

Fuel Handling Accident (FHA): RG 1.25 assumptions used

Main Steam Line Break (MSLB)

Rod Drop: No Fuel or Clad Damage – No Radiological Consequences

> **Other Accidents Analyzed in Certification Package But Not in RG 1.183:**

MSIV Closure

Small Line (Instrument) Break Outside Containment

RWCU Break Outside Containment

Offgas System Failure

SELECTION OF DESIGN BASIS ACCIDENTS
Comparison of LWR Accidents Addressed by RG 1.183

- PWR

- **RG 1.183 Listed Accidents Analyzed in Certification Package:**

  LOCA

  Steam Generator Tube Rupture (SGTR)

  Reactor Coolant Pump (RCP) Locked Rotor

  Rod Ejection

  Main Steam Line Break

- **Other Accidents Analyzed in Certification Package But Not in RG 1.183:**

  Small Line Break Outside Containment: Sample

  Feedwater Line Break: No radiological consequence presented – bounded by steam line break

SELECTION OF DESIGN BASIS ACCIDENTS

❑ Non-LWR Design Basis Accidents

- The post accident releases from the PBMR and GT-MHR are mechanistic releases and limited to the source terms for one of the multi-module station installations.

and activity releases,

- Best available information indicates that the ACR-700 large break LOCA has the largest source term and hence the greater potential for offsite dose consequences of the non-LWR plants.

- The ACR-700's large break LOCA accident sequence is included in the set of design basis accidents as a benchmark for the non-LWR reactor technologies.

SELECTION OF ENVIRONMENTAL REPORT DESIGN BASIS ACCIDENTS

- NUREG 1555 provides the guidance for evaluating the radiological consequences of accidents in the Environmental Report.
- NUREG 1555, Section 7.1, Listed DBAs are:
  - Main Steam Line Failure Outside Containment (PWR)
  - Feedwater System Pipe Breaks Inside/Outside Containment (PWR)
  - RCP Pump Rotor Seizure on
  - RCP Shaft Break
  - Control Rod Drop Accident (BWR)
  - Failure of Small Lines Carrying Primary Coolant Outside Containment
  - Steam Generator Tube Failure (PWR)
  - LOCA: Including Containment Leakage Contribution

    LOCA: Leakage from ESF Components Outside Containment

    LOCA: Leakage From MSIV Leakage Control System (BWR)
  - Fuel Handling Accident

EVALUATION OF RADIOLOGICAL CONSEQUENCES

- The results of the evaluation should demonstrate that the offsite doses are less than values given in 10CFR100 to provide an adequate level of protection to the public.

- Doses for the representative DBAs are evaluated at the ESP Exclusion Area Boundary (EAB) and Low Population Zone (LPZ).

- The evaluations use 50 percentile accident X/Qs. The X/Qs are determined using Regulatory Guide 1.145 methods with on-site meteorological data.

- Activities released to environs are time-dependent and based on conservative vendor's standard SAR analyses for certified reactor designs or based on information provided by reactor vendors for non-certified reactor designs.

Example Comparison
ESP Environmental Report Limiting Offsite Dose Consequences

Design Basis Accident	Reactor	Doses (REM TEDE) EAB	LPZ
❑ Loss of Coolant Accident	AP-1000	**1.5E+00**	2.6E-01
	ABWR	2.3E-01	7.6E-01
	ACR-700	7.0E-01	**1.1E+00**
❑ Fuel Handling Accident	AP-1000	**1.4E-01**	1.5E-02
	ABWR	8.0E-02	9.8**E-03**
❑ Main Steam Line Break			
➢ Pre-existing Iodine Spike	AP-1000	4.2E-02	**1.3E-02**
➢ Accident-initiated Iodine Spike		**4.7E-02**	**5.0E-02**
➢ Pre-existing Iodine Spike	ABWR	**6.8E-02**	6.5E-03
➢ Max Equilibrium Iodine Activity		3.4E-03	3.3E-04

Example Comparison
ESP Environmental Report Limiting ESP Offsite Dose Consequences

Design Basis Accident	**Reactor**	**Doses (REM TEDE) EAB**	**LPZ**
Steam Generator Tube Rupture	AP-1000		
➢ Pre-existing Iodine Spike		1.8E-01	8.8E-03
➢ Accident-initiated Iodine Spike		8.9E-02	6.6E-03
❑ Reactor Coolant Pump Locker Rotor	AP-1000	1.5E-01	1.5E-02
❑ Control Rod Ejection Accident	AP-1000	1.8E-01	4.5E-02
❑ Small Line Break	AP-1000	**7.7E-02**	**3.3E-03**
	ABWR	3.0E-03	5.7E-04

Summary

- ESPs will reflect that bounding EAB and LPZ doses are
  - < 2 (EAB) rem TEDE
  - < 1 (LPZ) rem TEDE
- For COL, environmental impact of DBAs is resolved provided the actual impact (calculated dose) is not significantly greater than the bounding dose reflected in the ESP