An Overview of NRC Fission Product Research

Presentation at the 27th Nuclear Air Cleaning and Treatment Conference

Jason Schaperow
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

September 23, 2002

Recent NRC Fission Product Research

Applicability of revised source term for high burnup and MOX fuels

Address regulatory issues for high burnup and MOX fuels

Design basis consequence assessment for dry cask storage

More realistic design basis analysis

Severe accident consequence assessment

Risk-inform regulations for decommissioning reactors

Risk-inform regulations for dry cask storage

Applicability of Revised Source Term for High Burnup and MOX Fuels

Overview

Objective

Assess applicability of revised source term to high burnup and MOX fuels

Approach

Hold a series of expert panel meetings, including experts who developed basis for the revised source term

Results

Experts suggested source term values for high burnup and MOX fuels, identified issues, and recommended research

Revised Source Term

Source term is the fission product release into containment atmosphere which is available for release to the environment

RES published revised source term (aka alternative source term) in NUREG-1465 in 1995

More realistic than earlier TID-14844 source term

Aerosol except for 5% of iodine which is vapor

Four-phase release: gap, early in-vessel, ex-vessel, late in-vessel

A few differences in release timing and magnitude between PWR and BWR (main difference is release timing for I)

Regulatory Applications

Gap and early in-vessel phases of revised source term used for LOCA design basis accident analyses

- **Exclusion Area Boundary, Low Population Zone, and control room doses**
- **containment isolation valve closure time (start time of gap release)**
- **integrated dose used to qualify equipment in containment**
- **post accident shielding, sampling, and access**
- **hydrogen generated by radiolytic decomposition of water**

All four phases of source term may be used for severe accident risk assessment

Regulatory Applications (cont.)

Revised source term being implemented voluntarily because of safety and cost benefits

License amendments issued:	**Perry**	**Fort Calhoun**
	Grand Gulf	**Three Mile Island 1**
	Indian Point 2	**Hope Creek**
	Duane Arnold	**Surry 1 & 2**
	Crystal River	

Applications under review:	**Oyster Creek**
	Brunswick
	Columbia (WNP2)
	Oconee 1, 2, & 3
	Kewaunee

Approach

Held a series of expert panel meetings (Sep 2001 - Feb 2002)

Panel members were requested to judge applicability of each aspect of the revised source term, and if judged not applicable, to propose alternative

As part of this effort, panel members...

- **considered recent data from international tests**
- **discussed physical phenomena affecting source term for high burnup and MOX fuels**
- **identified and prioritized source term research**

Approach (cont.)

Panel of International Experts

Bernard Clement (IPSN, France)
James Gieseke (consultant)
Thomas Kress (consultant)
David Leaver (Polestar Applied Technology)
Dana Powers (Sandia National Laboratories)

Others

Principal Investigator: Mohsen Khatib-Rahbar (Energy Research)
Panel Facilitator: Brent Boyack (Los Alamos National Laboratory)
Consultant: Hossein Nourbakhsh (Energy and Environmental Science)

Applicability of Revised Source Term for High Burnup Fuel

Panel assessment based on:

Maximum assembly burnup of 75 Gwd/t

Core average burnup of 50 GWd/t

Zirlo cladding (PWR), Zircaloy cladding (BWR)

Low pressure scenario (minimizes RCS retention)

Table 3.1

Results of Panel Assessment for High Burnup Fuel

Physical/chemical forms expected to be applicable.

Only small changes in release-phase duration and release fraction expected.

Burnup-independent issues identified based on recent tests

- **potential for enhanced Te release**
- **continued uncertainty in releases of noble metals, Ce, La groups**
- **recent data suggests subdividing noble metals, Ce, La groups**

Related issues

- **BWR power uprates**
- **BWR fuel design**

Tellurium Release

Revised source term specifies early in-vessel Te release of 0.05

> **ORNL tests indicate Te gets sequestered in the Sn in Zircaloy cladding and not released until high fraction of cladding is oxidized**

More recent French tests (VERCORS, PHEBUS-FP) indicate that Te release could be similar to I (i.e., 0.30)

For PWRs, this was a contentious issue among panel members.

For BWRs, panel members specified release fractions similar to revised source term

> **BWR zircaloy fuel channels tend to limit cladding oxidation**

Other Source Term Issues Related to High Burnup

BWR Power Uprates

One expert saw no basis for significant effect on fission product release

Another expert stated that flux-profile flattening associated with power uprates could increase the release rate for the outer assemblies.

BWR Fuel Design

NUREG-1465 specifies a different source term for a BWR than a PWR

Characteristics of more recent BWR fuel rod designs are closer to PWR fuel rod characteristics (e.g., pellet diameter, cladding thickness)

Panel indicated that similar rod designs tend to result in similar source terms.

Applicability of Revised Source Term for MOX Fuel

Panel assessment based on:

- **Using MOX in PWR (about ½ of core)**
- **Typical MOX assembly burnup of 42 GWd/t**
- **M5 cladding**
- **Low pressure scenario (minimizes RCS retention)**

Results of Panel Assessment for MOX Fuel

Physical/chemical forms expected to be applicable.

Only small changes in release-phase duration and noble gas, I, and Cs release fractions expected.

Same Te issue as for high burnup fuel.

Some of the experts did not recommend release fractions for Ba/Sr, noble metals, cerium, and lanthanum groups, because of the lack of test data.

- **Only data was a VERCORS test result for Cs with an arbitrary scale on the y-axis**

Panel-Recommended Research

High Priority Research

Validate severe-accident analysis codes against recent source term tests

Investigate in-vessel core degradation following vessel failure (air ingress)

Acquire any available data on fission product releases for high burnup and MOX fuels

Perform fission product release tests for high burnup fuels using modern cladding designs (Zirlo and M5)

Perform revaporization tests

Panel also recommended several medium and low priority source term research efforts.

Status

Panel report completed June 2002.

Report being slightly revised to address a few final comments from panel members

Results of expert panel assessment to be used to help address reactor safety issues

- **applications for high burnup and MOX fuels**
- **severe accident risk assessment**
- **other applications (e.g., vulnerability assessment)**

Design Basis Consequence Assessment for Dry Cask Storage

Overview

Object of the analysis

Provide more realistic quantification, with uncertainty bounds, of offsite doses associated with dry storage cask leakage

Summary of approach

Used RADTRAD code with isotopic inventories for spent fuel after 5 years of decay to calculate individual offsite dose

- **Focus of more realistic modeling was aerosol deposition in cask**

Conclusion

Modeling aerosol deposition in cask reduces dose by a factor of 400

- **provides rough estimate of λ**
- **based on dimensions and T-H conditions for a reactor containment**

NUREG/CR-6189 adjusted for cask dimensions

- **provide insight into effect of dimensions (containment vs. cask)**
- **not directly applicable, because do not know how much deposition due to each mechanism**

Stand-alone calculation of settling using distributions for aerosol density, diameter, and shape factor from reactor accident studies

- **ignores additional deposition due to thermophoresis**

Stand-alone calculation of settling using distributions for aerosol density, diameter, and shape factor for a spent fuel cask

- **provides best estimate of λ**

Doses for Accident Conditions

Case	Deposition Modeling	TEDE for an Individual at the Site Boundary (mrem)		
		lower bound	best estimate	upper bound
1	none	N/A	N/A	44
2a,2b,2c	NUREG/CR-6189	.037	.059	.097
3a,3b,3c	NUREG/CR-6189 with cask dimensions	.0088	.014	.024
4a,4b,4c	settling only, based on reactor containment conditions	.027	.077	.35
5a,5b,5c	settling only, based on cask conditions	.031	.096	.24

Comparison with MELCOR Results

MELCOR accident analyses performed for TN-125 cask with a 4 mm^2 hole

(SAND98-1171/7, *Data and Methods for the Assessment of the Risks Associated with Maritime Transport of Radioactive Materials, Results of the SeaRAM Program Studies*, May 1998).

Calculated accident dose using RADTRAD for HI-STORM using MELCOR-predicted deposition rate constants from the TN-125 cask study.

RADTRAD accident doses using MELCOR-predicted deposition rate constants were .070 to .11 mrem.

RADTRAD accident dose using settling rate constant (Case 5b) was .096 mrem.

Excellent agreement because settling was dominant mechanism in MELCOR analyses.

Comparison with Direct Shine Dose

Pathway	TEDE (mrem)

	Accident Conditions (30 days)	Normal Conditions (365 days)
Direct Shine[1]	**60**	**700**
Leakage	**44**	**<4.2[2]**
Leakage with deposition	**.096**	**<.012[2]**

[1] Direct shine dose is for 2x5 array at a distance of 100 meters (Figure 5.1.3 of HI-STORM SAR).

[2] Dose is for 10 casks.

Possible Future Work

Quantify dose reduction from deposition mechanisms other than settling using a integrated, mechanistic accident analysis code such as MELCOR.

Surface	Area (m^2)	Orientation
confinement floor	2.37	upward facing
confinement ceiling	2.37	downward facing
confinement wall	24.7	vertical
basket	189.6	vertical
outside of fuel channel	149.5	vertical
inside of fuel channel	149.5	vertical
fuel cladding	680.0	vertical

Estimate the uncertainty in offsite dose due to variability in the weather.