Public Meeting
ALTERNATIVE SOURCE TERM REGULATORY GUIDANCE (DG-1081; SRP15.0.1)
February 15, 2000

Name	Organization	Phone	E-Mail
JERRY BURFORD	ENTERGY	601-368-5755	fburfor@entergy.com
Mike Withrow	Entergy	601/437-8247	mwithro@entergy.com
Greg Broadbent	Entergy	601/437-6224	gbroadb@entergy.com
Patricia Campbell	Winston & Strawn	202-371-5828	pcampbel@winston.com
Michelle Hart	NRC/NRR/SPSB	301-415-1265	mlh3@nrc.gov
[illegible]	NRC/NRR	415-2829	[illegible]@nrc.gov
Jim Grover	Westinghouse	412-374-5585	groverjl@westinghouse.com
DUC NGUYEN	NRC/NRR	301-415-3202	dtn1@nrc.gov
TERRY HEAMES	ITSC	505 872 10[illegible]	THEAMES@ITSC.COM
Tom Mscisz	PECO Nuclear	610 640 6875	TMSCISZ@peco-energy.com
Bill Hopkins	Bechtel	301 417 3341	WHOPKINS@BECHTEL.COM
Althea Wyche	Bechtel	(301) 417-4458	awyche@bechtel.com
[illegible]	NRC/RES	301-415-6236	[illegible]
Bryan Carroll	Duke Energy	704 382 5945	bccarrol@duke-energy.com
JUAN M. CAJIGAS	APPLIED ANALYSIS CORP.	(610) 775-0272	JMCAJIGAS@APPLIED-ANALYSIS.COM
MIKE RADOVANSKY	GPU NUCLEAR	973-316-7190	MRADOVANSKY@GPU.COM
Ron Andersen	Applied Analysis	610 798-1200	ReAndersen@[illegible]
John Hayes	NRC	301 415-3167	JJH@NRC.GOV
Muffet Chatterton	NRC	301-415-2889	MSC1@NRC.GOV
Tom Milton	Southern Nuclear	205/992-7867	TMMILTON@southernco.com
MICHAEL SNODDERLY	NRC	(301) 415-2047	MRS1@NRC.GOV
Jason Schaperow	NRC	(301) 415-5907	jhs1@nrc.gov

Attachment 1

Public Meeting
ALTERNATIVE SOURCE TERM REGULATORY GUIDANCE (DG-1081; SRP15.0.1)
February 15, 2000

Name	Organization	Phone	E-Mail
Mark Reinhart	NRR/DSSA/SPSB	301 ~~[illegible]~~ 415 1185	fmr@nrc.gov
Steve LaVie	NRR/DSSA/SPSB	301 415-1081	sfl @ NRC.gov
Kurt Cozens	NEI	~~301~~ 202-739-8085	
Stan Rittenbusch	ABB - CENP	860-285-5206	stanley.e.rittenbusch @us.abb.com
Bill Hopkins	Bechtel	301-417-3341	WHOPKINS@Bechtel.com
Sreela Ferguson	SWEC	617 589 8073	Sreela.ferguson@stoneweb.com
MARK Blumberg	NRC	301-415-1083	wmb1@nrc.gov

AGENDA

NRC/NEI SOURCE TERM TASK FORCE MEETING

FEBRUARY 15, 2000

Item	Topic	Responsible Party
1.	Welcome and Opening Remarks	NRC and NEI
2.	Gap Fission Product Content	Stan Ritterbusch
3.	Duration of Accident	Sreela Fergunson
4.	Prior Design Basis	Sreela Fergunson
5.	Discussion of Iodine Chemical Form in Gap	Bill Hopkins
6.	Discussion of Full versus Selective Implementation Requirements and Guidance	NRC staff
7.	Meeting Review and Wrap-up	NRC and NEI
8.	Adjourn	

Attachment 2

PRIOR DESIGN BASIS

NEI

PRIOR DESIGN BASIS

- **DG 1081, Section 5.1.4 statement** : *"Prior design basis that are unrelated or unaffected by the AST may continue as facility design basis"*
- Appendices to DG1081however provide safety analyses guidance *beyond* AST
- DG 1081 should be updated to clarify that if unrelated with AST, *prior design basis assumptions take precedence* over guidance in Appendices to DG 1081.

NEI

PRIOR DESIGN BASIS

- **Industry Recommendation**: Add the following additional statement to DG 1081 Section 5.1.4 after : "Prior design basis that are unrelated or unaffected by the AST may continue as facility design basis"

"This includes the continued use/acceptability of site specific models/assumptions unaffected by the AST, which were previously accepted by the staff, even though they may be different from those identified by the staff and listed in DG1081 and its Appendices as appropriate and prudent for use in safety analyses.. This includes but is not be limited to assumptions with respect to single failure, passive failure, amount of ESF leakage, iodine spiking, etc.

Duration of Accident - LPZ/CR Analyses

Duration of Accident - LPZ/CR Analyses

- **Concern**
  - DG 1081 does not consistently define what constitutes a reasonable accident duration for several of the DBAs.
  - For example, the time duration for the LOCA is not addressed in DG 1081. The 30 day duration currently used can only be traced to TID 14844 which is not a valid reference for AST.
  - Table 6 of DG 1081 should be updated to summarize / co-relate allowable dose limits to expected accident durations
  - Industry recommendation on accident durations consistent with current guidance for traditional source terms (i.e.;based on SRPs/TID).

Duration of Accident - LPZ/CR analyses

Recommendation - Update Table 6 to include duration.

Accident	Accident Duration
LOCA	30 days unless demonstrated shorter by plant design
BWR MSLB	2 hrs unless demonstrated shorter by plant design
BWR Rod drop	24 hrs unless demonstrated shorter by plant design
PWR SGTR	Until shutdown cooling can remove all decay heat
PWR MSLB	Unaffected SGs : Until shutdown cooling can remove all decay heat Affected SGs : Until primary coolant temp. reaches 212F
PWR LR	Until shutdown cooling can remove all decay heat
PWR REA	Containment Scenario : 30 days Secondary Side release : Until shutdown cooling can remove all decay heat
FHA	2 hrs

Duration of Accident - EQ

Duration of Accident - EQ

- **Concerns**
  - No existing regulatory guidance on what constitutes a reasonable accident duration for equipment qualification purposes.
  - Varied time frames (from couple of months to 1 year) used by licensees, and accepted by staff
  - No basis/reason provided for the differences in the expected duration of the same accident (e.g. LOCA) from site to site.
  - No credit given to the fact that additional/backup equipment can be brought on site and utilized for maintenance of safe shutdown as long *as a reasonable amount of time has passed since accident initiation*.

Duration of Accident - EQ

- **Industry Recommendation**
  - Distinction should be made between the mitigation and recovery phase of an accident.
  - The mitigation phase is the time immediately after the event when existing plant design/response has to be dependent on to mitigate the event
  - The recovery phase is the period after the mitigation phase during which additional cleanup/recovery equipment can be brought on site, as needed, and credited for maintenance of safe shutdown .
  - Safety related equipment should be *qualified for the mitigation* phase.
  - The duration of the "mitigation phase" should be consistent with the accident duration for LPZ/CR analyses, e.g. 30 days for the LOCA

NEI

Duration of Accident - EQ

- **Basis for Industry Recommendation**
  - TMI experience has indicated that an entire safety related RHR system, (including associated structures for housing the referenced equipment) was installed/operable within seven days of the event.
  - As part of NRC's effort on rebaselining plants for use of AST, NUREG/CR 5313 concluded that *EQ* applications associated with *long term operability are not risk significant.* "

NEI

Comments on DG-1081 Gap Fission Product Content

February 15, 2000

Agenda

- Basis for commenting
- Percent of fuel rod inventory that is released to the gap during normal operation
- Radial power peaking factor used in predicting the quantity of fission products generated in the fuel pellet column

Note: Fission product content in fuel-cladding gap
= (total fuel rod content) x (percent released from pellet)
[fission product decay is credited for the fuel handling accident.]

Basis for Commenting

- Respond to staff efforts to promote more frequent and open communications with the industry and public
- Provide staff with technical bases to support changes to DG-1081
- Provide an approach for non-LOCA accidents:
  - conservative relative to data
  - follows source term philosophy of being physically consistent
- Facilitate industry implementation of the alternate source term

NEI

Percent Fuel Rod Inventory Released - Overview

- DG-1081:

I-131	12%
Kr-85	15%
Other iodines	10%
Other noble gases	10%
Alkali metals	10%

NEI

Percent Fuel Rod Inventory Released - Overview....

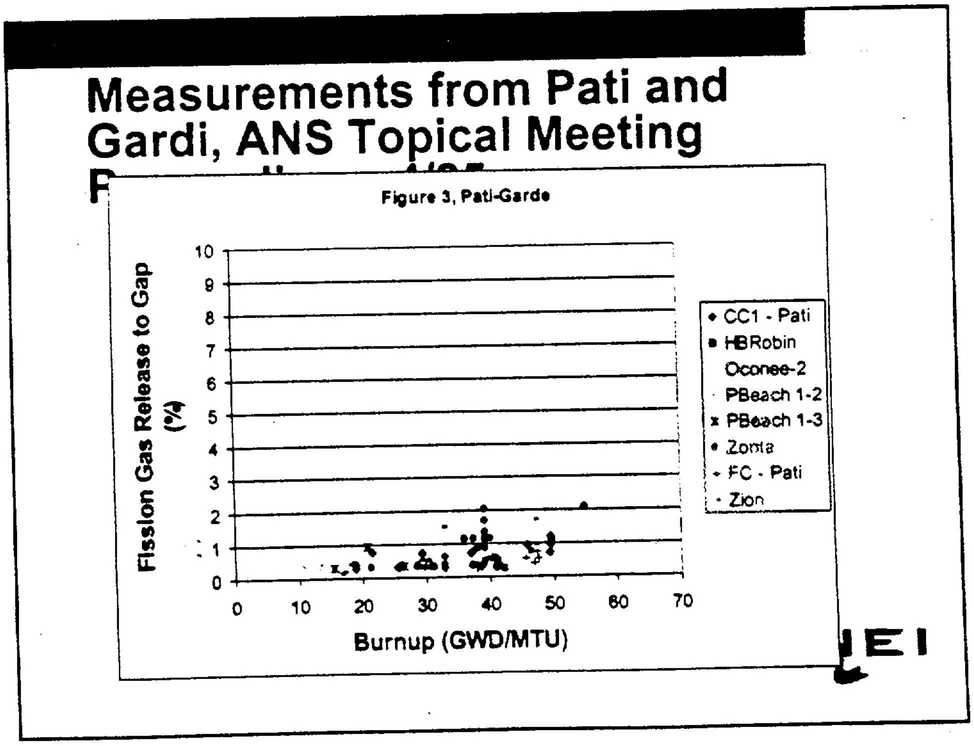
- NRC position is based on FRAPCON-3 analysis; however,
  - details of the FRAPCON-3 analysis input assumptions are not available to the public
  - results appear to be excessively conservative relative to measurements from actual fuel rods
- Industry proposes a model (very conservative) that is based on actual measurements for a number of fuel designs over a wide range of burnup

---

Industry Observations

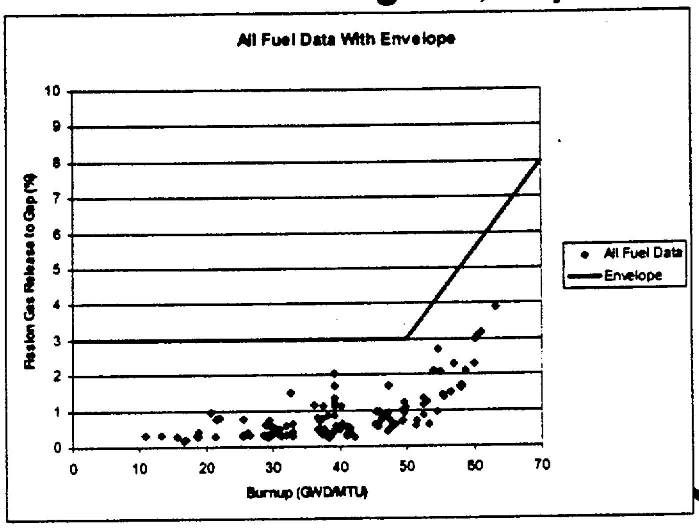
- The ANS 5.4 model, used traditionally, is overly conservative due to its exponential burnup-dependent multiplier, based on:
  - few data points
  - plutonium/uranium mixed oxide fuel
- The NRC FRAPCON-3 model and analysis have not provided to industry
- High-burnup gap content measurements from actual fuel rods are available
- Fuel rod behavior for the control rod ejection event is being addressed in NRC's high-burnup fuel research and evaluation program

This Figure was inadvertently presented during the meeting but is considered proprietary by EPRI and has been redacted.



Measurements from Pati and Gardi, ANS Topical Meeting P

Composite Data From Both Reports With Bounding Envelope



Proposed Fission Product Inventory in Gap for Non-LOCA Events, No Heatup

Rod Average Burnup (MWD/MTU)	Industry Proposed Gap Fraction for Use in Design Basis Analysis – No Heatup (%)
0	3
20,000	3
30,000	3
40,000	3
50,000	3
60,000	5.5
62,000	6.0
70,000	8.0
75,000	9.25

Fuel Heatup

- Some transients (e.g., excess load, small break LOCA) may result in fuel pellet heatup during the event
- NUREG-1465 supporting analysis (in NUREG/CR-4881) showed that heatup (10 minutes at 1200C) during a LOCA could result in an additional 2% of fission products released from the pellet over 10 hours time
- For non-LOCA events with fuel heatup, conservatively assume that an additional 2% is instantaneously released to the gap

Proposed Fission Product Inventory in Gap for Non-LOCA Events

Rod Average Burnup (MWD/MTU)	Industry Proposed Gap Fraction for Use in Design Basis Analysis – No Heatup (%)	Industry Proposed Gap Fraction for Use in Design Basis Analysis – With Heatup (%)
0	3	5
20,000	3	5
30,000	3	5
40,000	3	5
50,000	3	5
60,000	5.5	7.5
62,000	6.0	8.0
70,000	8.0	10.0
75,000	9.25	11.25

Application to Specific Failed Fuel Analyses

- Fuel predicted to fail during an accident may have various burnups, depending on their particular power histories
- It is proposed that gap percentages can be applied to individual fuel rods or assemblies if specific burnup calculations have been performed
- If such burnup calculations have not been performed, the assumption of gap percentage should assume that the failed fuel is at the maximum licensed core burnup (highest gap percentage) within the limits of the core design.

NEI

Radial Power Peaking Factor for Total Fission Product Generation - Overview

- Fission product content in fuel-cladding gap = (total fuel rod content) x (percent released from pellet)
- Total fuel rod content = (core average fuel rod content) x (radial power peaking factor)
- NRC proposes using the maximum power peaking factor for all burnup levels
  - industry believes this is excessively conservative
- Industry proposes using a power peaking factor that bounds actual analytical results

NEI

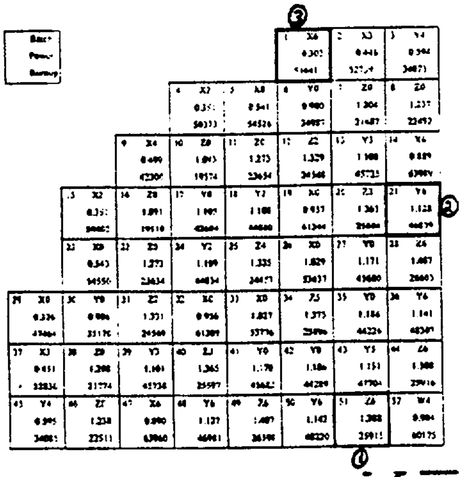
Power Peaking Factor for Core Inventory Calculations

- Core-wide average-rod fission product inventories from a code such as ORIGEN-II are multiplied by a radial power peaking factor to obtain the inventory of fuel rods predicted to fail
- The DG-1081 assumption that the failed fuel rods are at the maximum power level is excessively conservative because fuel assembly power decreases at high burnup
  - uranium is depleted
  - fuel assemblies are moved to low-power regions of the core as they accumulate burnup

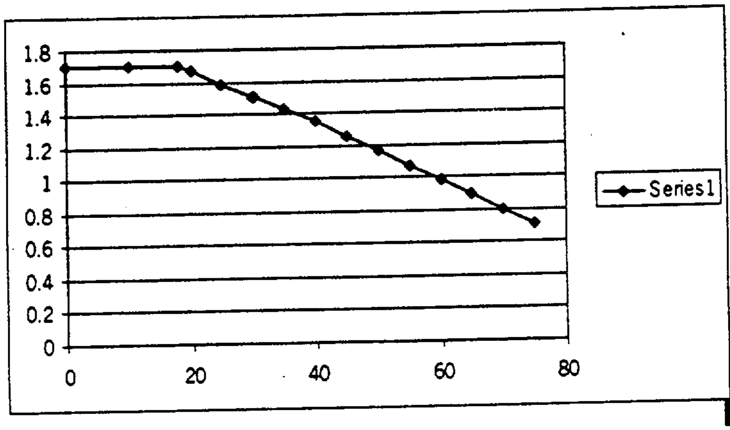
---

Decreasing Peaking Factor With Increasing Burnup

Movement of fuel assembly from high-power region of core to low-power region as burnup increases:



Example of Bounding Envelope of Fuel Rod Power Vs. Burnup



Power Peaking Factor for Core Inventory Calculations....

- Proposal:
    - For low-burnup fuel (<30,000 MWD/MTU), use the radial power peaking factor from the Core Operating Limits Report (COLR) Power Peaking Factor for Core Inventory Calculations
    - For fuel with higher burnups, use a radial power peaking factor based on the bounding power history envelope associated with the fuel cycle design being used
    - if a radial power peaking factor from a power history envelope is not available, use the low-burnup value from the COLR

Conclusions

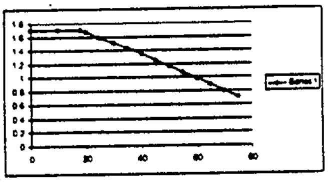
- Replace DG-1081 Table 3 with:

Rod Average Burnup (MWD/MTU)	Industry Proposed Gap Fraction for Use in Design Basis Analyses - No Heatup (%)	Industry Proposed Gap Fraction for Use in Design Basis Analyses - With Heatup (%)
0	3	3
20,000	3	5
30,000	3	5
40,000	3	5
50,000	3	5
60,000	5.5	7.5
62,000	6.0	8.0
70,000	8.0	10.0
75,000	9.25	11.25

- allow application of fuel-specific burnup levels

- allow application to low-burnup fuel in control rod ejection analyses

- Allow credit for burnup effects on radial power peaking factor:



NEI

PWR Gap Fractions

	Core Average	Peak Rod
Kr-85	2.9	7.9
I-131	1.9	6.8
Xe-133	0.6	2.8
Cs-137	4.9	11.1

- 62 GWD/MTU
- FRAPCON-3 with Massih model
- 3-cycle irradiation model
- Does not include fabrication and modeling uncertainties. Does not include all possible normal power transients. If considered, adjusted results could be 1.8 - 2.5 times higher

UNITED STATES NUCLEAR REGULATORY COMMISSION

Attachment 4

BWR Gap Fractions

	Core Average	Peak Rod
Kr-85	2.9	7.9
I-131	1.5	4.1
Xe-133	0.6	1.6
Cs-137	3.4	9.3

- 62 GWD/MTU
- FRAPCON-3 with Massih model
- 3-cycle irradiation model
- Does not include fabrication and modeling uncertainties. Does not include all possible normal power transients. If considered, adjusted results could be 1.8 - 2.5 times higher

UNITED STATES NUCLEAR REGULATORY COMMISSION

Specific DG-1081 "Gap Fraction" Revisions
DRAFT – 2/15/2000

Comment 1: Section 3.1

The draft guide states that, "For non-LOCA events, the appropriate radial peaking factor from the facility's core operating limits report (COLR) should be applied." For many events this would be an appropriate approach since, in general, the fuel rods that would be damaged in a postulated accident involving a reactor transient would be those at a high power level.

However, it is well known that the relative power, and thus the radial power peak in a fuel rod, decreases as burnup increases. This is an expected phenomenon since power production must necessarily decrease as fissionable material is consumed. The assumption of a radial peaking factor based on the COLR report therefore should not be a requirement for all analyses since it may be an inappropriate assumption. For example, in the fuel handling accident, the damaged fuel may have been operating at a low power level, far below the peaking factor identified in the COLR – this is particularly true for high burnup fuel. Considering that high burnup fuel is expected to have higher fission product gap fractions than lower burnup fuel (see comment #2 that follows), the application of the high radial peaking factor to high burnup fuel results in an unreasonable level of conservatism.

It is recommended that the above quoted sentence from the draft regulatory guide be replaced with:

> For events in which only a fraction of the core is damaged, an appropriately conservative radial peaking factor should be applied to the damaged fuel. For fuel with low burnup (i.e., ≤30,000MWD/Mtu), the radial peaking factor from the facility's core operating limits report (COLR) should be applied. For fuel with a moderate or high level of burnup, the radial peaking factor may be reduced from the COLR value based on the bounding power history curve associated with the fuel design being used.

Specific DG-1081 "Gap Fraction" Revisions
DRAFT – 2/15/2000

Comment: Section 3.2

The specification that the gap fractions in Table 3 should be used for all non-LOCA accidents is excessively conservative. While footnote 10 states that, "The fractions shown in Table 3 are consistent with available data for extended burnup fuel (based on the limiting assembly).", the validity of this statement is not evident. The data obtained from fuel rods removed from power reactors support lower gap fractions than those in Table 3.

Additionally, the use of a limiting assembly basis for the determination of the gap fractions results in the inherent assumption that any fuel damaged in a postulated accident is high burnup fuel. This is appropriate only if the level of burnup in the damaged fuel has not been ascertained. This approach does not allow for application of information on the fuel burnup associated with the fuel that would be damaged in a postulated accident.

It is requested that DG-1081 be revised to remove the paragraph preceding Table 3:

> For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

In its place, the following is suggested:

> For events other than the LOCA with core melt, the fractions of the core inventory assumed to be in the gap for the various radionuclides are dependent on the level of fuel burnup in the damaged fuel rods. The gap fractions for noble gases, iodines, and alkali metals should be as given in Table 3. This table addresses the current upper level for licensed operation of 62,000 MWD/Mtu for the lead rod burnup and also addresses the potential for future increases in burnup that may be permitted as fuel designs change. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

Table 3

Fraction of Fuel Fission Product Inventory in Gap

Burnup (MWD/Mtu)	Fraction
0 – 50,000	0.0300
55,000	0.0425
60,000	0.0550
62,000	0.0600
70,000	0.0800
75,000	0.0925

> These gap fractions are applicable for fuel damage in accidents for which there is no significant fuel heatup transient (e.g., fuel handling accident, steam line break, steam generator tube rupture, locked rotor). If a transient has a significant fuel heatup transient (e.g., small break LOCA), then an additional two percent of the activity in the damaged rods should be assumed to be released – this is the same as specified in NUREG-1465 for the gap release phase of the large break LOCA that proceeds to core melt.

Specific DG-1081 "Gap Fraction" Revisions
DRAFT – 2/15/2000

If an applicant chooses not to determine the burnup associated with the fuel damaged in a postulated accident, the analysis should assume that all of the damaged fuel is at the maximum licensed core burnup as is appropriate within the limits of the core design (e.g., if 50% of the core is projected to be damaged and there is no more than 30% of the core that would be above 50,000 MWD/Mtu burnup, then the remaining 20% of the core that is damaged could use the 3% gap fraction).

An exception is made for reactivity insertion accidents (rod ejection for the PWR and rod drop for the BWR) because of uncertainties associated with these events and how high burnup fuel will respond during the transient. For the reactivity insertion accidents, the gap fractions for any rods having burnup in excess of 40,000 MWD/Mtu (the NRC's current definition of high burnup fuel) should use the gap fractions in Table 4. The gap fraction of 3% can be used for fuel rods having burnups ≤40,000 MWD/Mtu (consistent with Table 3).

Table 4
High Burnup Fuel in a Reactivity Insertion Accident
Fraction of Fuel Fission Product Inventory in Gap

Nuclide	Fraction
I-131	0.12
Kr-85	0.15
Other Noble Gases	0.10
Other Iodines	0.10
Alkali Metals	0.10

It is noted that the gap fractions here identified in Table 4 are those from the current Table 3 of DG-1081. The above suggestion to use these values does not mean that these are necessarily appropriate. It is industry's understanding that these gap fractions are still being reviewed and that they may be decreasing. With the addition of the new Table 4, the subsequent tables would require renumbering and appropriate corrections made elsewhere for proper referencing of the tables.

A more complete discussion of the arguments supporting the above change to DG-1081 is provided in Appendix A.

Fission Product Content in the Fuel Rod Gap
DRAFT - Rev. 6, 2/15/2000

Introduction

The NRC alternate source term (AST) report (NUREG-1465) [1] states that for LOCAs an appropriate value for noble gas and halogen fission product content in the fuel rod gap would be 5% (3% initial release and an additional 2% due to heatup), based on a review of previous research and analysis. Furthermore, it was noted in NUREG-1465 that a value of 3% could be used for events for which fuel cooling was maintained (e.g., the fuel handling accident (FHA) or a LOCA in which core cooling is maintained). For the System 80+ design certification program, a value of 5% was used for LOCA and all non-LOCAs. For the AP600 design certification program, a value of 5% was used for the LOCA, but a value of only 3.6% was assumed for the non-LOCA events. The value of 3.6% was derived by multiplying the 3% value by a factor of 1.2 to account for high-burnup effects.

In the NRC's ongoing effort to allow the use of the AST for design basis accident (DBA) analysis of operating reactors, NRC staff proposes in Table 3 of draft Regulatory Guide DG-1081 that the NUREG-1465 gap fractions be used for LOCA, but that the following, more conservative, assumptions for fission products in the fuel rod gap be used for non-LOCA events:

I-131	12%
Kr-85	15%
Other iodines	10%
Other noble gases	10%
Alkali metals	10%

* It is our understanding that recent research by an NRC subcontractor indicates that a lower number such as 8% may be appropriate.

It is industry's understanding based on a review of NUREG-1465 and on recent discussions with NRC staff that this increase in gap fractions has been proposed for non-LOCA events for two main reasons: (1) concern about recent test data on gap release in reactivity insertion events, and (2) concern about increased gap release for fuel irradiated beyond 40,000 MWd/MTU.

While acknowledging the NRC concerns (see further discussion below), industry believes that the formulation in NUREG-1465 for non-LOCA DBAs is still generally applicable, and that the values proposed in Draft Regulatory Guide DG-1081 are excessively conservative (except possibly for reactivity insertion accidents). The NRC's concerns and associated industry proposed alternatives to DG-1081 are addressed below.

Reactivity Insertion Events

Industry recognizes that design basis Reactivity Insertion Accidents (RIAs) (i.e., PWR rod ejection or BWR rod drop) present the potential for power excursions and associated rapid change in local fuel and cladding conditions (e.g., fuel temperature, cladding stress and strain, fuel rod pressure). The unique nature of these transients and their localized behavior may warrant the assumption of a fission gas release fraction that is greater than that assumed in the radiological consequences analysis of other non-LOCA events. Experimental simulations at both the French CABRI and Japanese NSRR facilities, have produced results that indicate a potential for significant fission gas releases from high-burnup fuel during RIA events. This research is continuing and involves the participation of international organizations, as well as NRC and EPRI, through its Robust Fuel Program. Until sufficient information becomes available to resolve this issue, industry recommends retaining the values currently proposed in DG-1081 (see the above table) for high-burnup fuel damaged in RIA events. Industry expects to reassess the proposed high-burnup fuel gap fraction values for RIA events when the results and interpretation of the ongoing research becomes more conclusive. It is recommended that DG-1081 be revised to indicate that the Table 3 gap fractions should be applied to the high-burnup (>40,000 MWD/MTU) fuel damaged in the event but that the fuel not defined as having high burnup may use the gap fractions as defined in the following sections. Further, DG-1081 could state that if the burnup of damaged fuel rods is not determined, all damaged fuel rods should be considered as high-burnup fuel.

Gap Fraction vs. Burnup

One of the issues which bears on gap release for high-burnup fuel is how gap fraction changes with increasing burnup. The discussion in this section applies directly to accidents in which long-term cooling is maintained (e.g., the fuel handling accident, steam generator tube rupture, or steam line break). For in-core, non-LOCA events in which long-term cooling is not maintained, the gap fractions defined in this section may need to be increased to reflect the additional releases associated with fuel heating (depending on the extent of fuel pellet heating). The increase in gap activity releases due to fuel heating is discussed below in the section entitled, "Increase in Gap Fraction from Post-Accident Heating".

Industry recognizes that gap fraction can increase with increasing burnup and believes that a reasonably conservative estimate of the fission products in the fuel rod gap can be made by bounding the measurements of the percent fission gas release in fuel rods taken from operating reactors. Recent measurements of volatile fission product content in the fuel rod gap for high-burnup fuel have been published by the Electric Power Research Institute [2], and similar data have been presented in other reports [3, 4]. The data cover a range of fuel rod designs and fuel designers, and burnups range from about 20,000 MWD/MTU to about 64,000 MWD/MTU.

The combined data of references [2-4] show that fission product release is less than 1.0% up to a burnup of about 30,000 MWd/MTU. As the burnup increases the gap fission product content increases to about 2% at 40,000 MWd/MTU. At about 50,000 MWD/MTU the gap fraction increases with burnup at a rate of about 2.3% per 10,000 MWd/MTU per reference [2]. In the table below, the middle column shows an envelope of the data, for which it is assumed that the fission gas release increases at a rate of 2.3% per 10,000 MWd/MTU. The right-hand column of the table shows the industry recommended gap fractions for use in analyzing non-LOCA events, including significant margin to the reference data.

The reference data have been extrapolated to 75,000 MWd/MTU in order to encompass the burnup range which industry anticipates could be utilized over the next decade or so. This is a modest extrapolation of the above-referenced data. The 62,000 MWD/Mtu burnup data point is included in the table below since this is currently the maximum licensed burnup for operating plants.

Rod Average Burnup (MWD/MTU)	Envelope of Measured Fission Gas Release (%)	Industry Proposed Gap Fraction for Use in Design Basis Analysis (%)
0	0.0	3
20,000	1.0	3
30,000	1.0	3
40,000	2.0	3
50,000	2.0	3
60,000	3.5	5.5
62,000	3.8	6.0
70,000	5.0	8.0
75,000	5.8	9.25

Accordingly, industry proposes that DG-1081 be changed to specify that for events which may have damaged fuel but do not have a fuel heatup, licensees should utilize the gap fractions as a function of burnup as specified in the right-hand column from the table above.

Increase in Gap Fraction from Post-Accident Heating

The following discussion applies to in-core events in which there is significant fuel heatup or long-term cooling is not maintained (excluding the high-burnup fuel rods damaged in a reactivity insertion accident which are discussed above).

It is recognized that if fuel experiences heatup due to a transient, some additional fission gas may be released from the pellet to the reactor coolant through the failed cladding. This was explicitly addressed in NUREG-1465 for a LOCA. It is also true for non-LOCA events; however, the degree of heatup and corresponding fission gas release is a function of the particular event being analyzed. For some events there is little or no fuel heatup (e.g., fuel handling accident, locked rotor, steam generator tube rupture, main steam line break) and, hence, there would be no transient fission gas release from the fuel pellets.

The fraction of fission product activity that would be released due to holding fuel at a temperature of 1200 °C (2192 °F) for a period of ten minutes was modeled and reported in reference [5] with the determination that 2.8% of the krypton would be released, less than 1.0% of the xenon would be released, and less than 0.1% of the iodine and cesium would be released.

Industry proposes that release of fission products to the fuel clad gap during non-LOCA events be addressed as follows: if it is known or demonstrated that there is little or no fuel heatup, no transient fission product release would be assumed. However, if it is expected that some sustained fuel heatup would occur, then the assumed transient fission product release would be the same as that identified in NUREG-1465 for the design basis LOCA.That is, an additional 2% of the fuel rod fission gas, iodines, and cesiums would be assumed to enter the fuel rod gap and be available for release from the damaged rods.

References

1. L. Soffer et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February, 1995.
2. G. Smith et al., "Hot Cell Examination of Extended Burnup Fuel from Calvert Cliffs-1," EPRI report TR-103302-V2, July 1994.
3. "Extension of the 1-Pin Burnup Limit to 65 MWd/kgU for ABB PWR Fuel with OPTIN™ Cladding," ABB report CENPD-388-P, Figure 2.2.2.2-1.
4. S. R. Pati et al., "Fission Gas Release from PWR Fuel Rods at Extended Burnups," Proceedings of the American Nuclear Society Topical Meeting on Light Water Reactor Fuel Performance, Orlando, April 21 – 24, 1985, Vol. 2, Pg 4 – 19.
5. H. P. Nourbakhsh et al., "Fission Product Release Characteristics into Containment Under Design Basis and Severe Accident Conditions," NUREG/CR-4881, March, 1988.