November 17, 2015

MEMORANDUM TO: Timothy J. McGinty, Director
Division of Safety Systems
Office of Nuclear Reactor Regulation

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SUBJECT: SAFETY ASSESSMENT: FUEL FRAGMENTATION, RELOCATION, AND DISPERSAL DURING DESIGN-BASIS ACCIDENTS

The purpose of this memorandum is to provide an assessment of the potential safety impacts of Fuel pellet Fragmentation, Relocation, and Dispersal on operating reactors. Specifically, the Safety Assessment (SA) documents the following: (1) under what accident conditions fuel rods become susceptible to fuel fragmentation and dispersal, (2) the extent of condition (i.e., number of susceptible fuel rods), (3) potential safety concerns, and (4) SA for the entire operating fleet. Conclusions from this SA will inform the staff's response to the Commission's direction and dictate future regulatory action and speed at which those actions are taken. This SA focuses on the non-Loss-Of-Coolant Accident (LOCA) Design-Basis Accidents. A separate SA is being prepared to address the potential consequences of fuel dispersal under LOCA conditions.

Enclosure:
Safety Assessment: Fuel Fragmentation, Relocation
and Dispersal During Design-Basis Accidents

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SAFETY ASSESSMENT
FUEL FRAGMENTATION, RELOCATION, AND DISPERSAL
DURING DESIGN-BASIS ACCIDENTS

1.0 INTRODUCTION

The purpose of this enclosure is to provide an assessment of the potential safety impacts of Fuel pellet Fragmentation, Relocation, and Dispersal (FFRD) on operating reactors. Specifically, the Safety Assessment (SA) documents the following: (1) under what accident conditions fuel rods become susceptible to fuel fragmentation and dispersal, (2) the extent of condition (i.e., number of susceptible fuel rods), (3) potential safety concerns, and (4) SA for the entire operating fleet. Conclusions from this SA will inform the staff's response to the Commission's direction and dictate future regulatory action and speed at which those actions are taken. This SA focuses on the non-Loss-Of-Coolant Accident (LOCA) Design-Basis Accidents (DBAs). A separate SA is being prepared to address the potential consequences of FFRD under LOCA conditions.

Section 2 of this report describes prior Commission communication and direction. Section 3 provides some background on this phenomena, including under what accident conditions fuel rods become susceptible to fuel fragmentation and dispersal, and describes applicable regulations and guidance. Section 4 describes potential safety concerns. Section 5 identifies the applicable non-LOCA DBAs. Section 6 documents the extent of condition and potential impacts on the operating fleet. Appendix A describes differences in the design and licensing bases between LOCA and non-LOCA DBAs.

2.0 COMMISSION DIRECTION

Recognizing the potential intersection between 50.46c implementation and the emerging fuel fragmentation and dispersal research, U.S. Nuclear Regulatory Commission (NRC) staff sent a Commissioner's Assistant (CA) note up in parallel with the 50.46c proposed rule. The CA note described the state-of-knowledge, as well as potential impacts, of fuel fragmentation and dispersal. In response, the Commission's Staff Requirements Memorandum (SRM) to the proposed Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46c Emergency Core Cooling System (ECCS) performance rule (SRM-SECY-12-0034, "Proposed Rulemaking – 10 CFR 50.46c: ECCS Performance During LOCA's (RIN 3150-AH42)," dated January 7, 2013, in the (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13007A478) included the following direction to the staff:

> "The staff should complete its research on fuel fragmentation, relocation, and dispersal, and incorporate any necessary changes before requesting Commission approval of the draft final rule. The staff should inform the Commission if this action is not practicable or has unintended consequences. In this case, the staff should provide the Commission an information paper containing additional details of the anticipated research into fuel fragmentation, the staff's best judgment of the impact the results of that research could have on

ENCLOSURE

the proposed rule, and the staff's best estimate of when final conclusions may be drawn from this work. In addition, this paper should clearly and specifically indicate which elements of the proposed rule, if any, should be deferred pending completion of fragmentation research and which elements, if any, could proceed to implementation without concern that they will be revised based on the anticipated research."

3.0 BACKGROUND

During normal power operation, fuel pellets undergo microscopic and macroscopic changes (e.g., accumulation of fission products, changes in grain size and porosity, pellet cracking). In addition, fuel rod internal pressure builds with prolonged irradiation due to the release of fission gas from the pellets into the free space within the fuel rod (e.g., plenum). Many of these exposure related phenomena will likely increase the susceptibility of fuel rods to fuel fragmentation, relocation, and dispersal. Fuel rod design (e.g., cladding alloy, plenum volume, fuel pellet composition) and operating power history may also play an important role.

The following questions and answers provide background on FFRD.

How has fuel design and utilization evolved in the past 25 years?

Beginning in the 1980's, the NRC staff began approving high burnup License Amendment Requests (LARs). For this discussion, high burnup is defined as a rod average burnup greater than 45 GWd/Metric Ton Uranium (MTU) and a peak pellet exposure greater than 50 GWd/MTU. Current approved license burnup limits range from 60 – 62 GWd/MTU rod average burnup and 70 GWd/MTU peak pellet exposure.

Approval of high burnup LARs, coupled with advancements in computational methods and economic drivers, lead to innovations in fuel design and fuel utilization. A few examples are provided below.

- Improvements in computational methods have resulted in advanced fuel designs and core loading patterns, including axial and radial ^{235}U and burnable poison loading patterns. These changes promote a flatter radial power distribution and a larger percentage of the fuel rods with burnup near the peak burnup rod.

- Longer operating cycles, coupled with more efficient fuel management, have resulted in a significant increase (~10 GWd/MTU) in assembly average discharge exposure.

- Longer operating cycles and power uprates have increased batch-average ^{235}U enrichment and core average Linear Heat Generation Rate which promotes higher fuel temperatures (stored energy) and more Fission Gas Release (FGR).

- More aggressive fuel management and introduction of Integral Fuel Burnable Absorbers promote an increase in rod internal pressure.

- Fuel rod cladding thickness has decreased to accommodate longer cycles (i.e., more uranium) or changes in bundle lattice design (e.g., 8x8 to 9x9 to 10x10).

As described below, many of these innovations in fuel design and utilization increase the likelihood of fuel fragmentation and dispersal during postulated DBAs.

How have regulations and guidance changed to address more aggressive fuel utilization?

The innovations in fuel design and utilization described above have outpaced our regulations and guidance. To address modern fuel design and utilization, the staff updated it's guidance for the review of fuel system designs in the Standard Review Plan (SRP). The following updates were included in SRP Section 4.2, "Fuel System Design," Revision 3 (March 2007):

- New guidance related to Boiling Water Reactor (BWR) shadow corrosion-induced channel distortion and control blade interference (Section II.1.A.v). This new guidance necessitated by longer, higher power BWR operating cycles which require more frequent, deeper control blade insertion to hold down excess reactivity.

- New guidance related to cladding lift-off, hydride reorientation, and Departure from Nucleate Boiling (DNB) propagation (Section II.1.A.vi). This new guidance necessitated by higher fuel rod burnup and associated higher rod internal gas pressure.

- New guidance related to defining mechanical and nuclear lifetimes for control rod/blade designs (Section II.1.A.viii). This new guidance necessitated by longer, higher power BWR operating cycles which require more frequent, deeper control blade insertion to hold down excess reactivity.

- Revised guidance related to defining Anticipated Operational Occurrence (AOO) overpower cladding strain failure threshold (Section II.1.B.vi). This revised guidance necessitated by longer, higher power operating cycles and associated cladding corrosion and hydrogen uptake.

- Revised guidance related to steady-state fission product inventory (Section II.1.C.ix). This revised guidance necessitated by longer, higher power operating cycles. Revised radionuclide inventories are presented in Diesel Generator -1199 (draft revision to Regulatory Guide 1.183).

- Revised guidance related to high temperature cladding failure thresholds during Reactivity-Initiated Accidents (RIAs) (Appendix B, Section B). This revised guidance necessitated by higher fuel rod burnup and associated higher rod internal gas pressure.

- New guidance related to Pellet-Cladding Mechanical Interaction (PCMI) cladding failure thresholds during RIAs (Appendix B, Section B). This new guidance necessitated by longer, higher power operating cycles and associated cladding corrosion and hydrogen uptake.

- Revised guidance related to core coolability criteria during RIAs (Appendix B, Section C). This revised guidance necessitated by higher fuel rod burnup.

- Revised guidance related to fission product inventory (i.e., transient FGR) during RIAs (Appendix B, Section D). This revised guidance necessitated by higher fuel rod burnup.

10 CFR 50.46 prescribes ECCS performance requirements during a postulated LOCA. Rulemaking to this regulation is currently ongoing (i.e., 50.46c) to address newly identified degradation mechanisms associated with higher fuel rod burnup (i.e., cladding inner surface oxygen ingress) and longer, higher power operating cycles and associated cladding corrosion and hydrogen uptake (i.e., hydrogen-enhanced beta-layer embrittlement).

FFRD is similar to the above examples in that it's another case of regulations and guidance trying to catch up to fuel utilization. A regulatory framework is needed to address these newly discovered phenomena.

Are fuel fragmentation and dispersal real phenomena or an experimental artifact?

The authenticity of fuel fragmentation and dispersal has been vetted by several international organizations and it is widely agreed that it is not an experimental artifact.

As described in NUREG-2121, "Fuel Fragmentation, Relocation, and Dispersal During the LOCA," the review of over 90 LOCA test results performed in eight different programs over the last 35 years prompted the staff to conclude that fragmentation, relocation, and dispersal of fuel could not be precluded as possible phenomena during a LOCA.

Halden in-pile and Studsvik out-of-pile integral LOCA testing conducted on irradiated fuel rod segments has shown that fuel rods that experience cladding ballooning and rupture are likely to exhibit fuel pellet fragmentation. Several variables, including balloon strains, rupture opening, and fuel rod exposure, affect how fragmented fuel particles may relocate within the enlarged ballooned and ruptured region and/or disperse outside of the fuel rod.

In addition, in-pile rapid power excursion testing conducted at the Power Burst Facility, Nuclear Safety Research Reactor (NSRR), and CABRI research reactor has reported fuel pellet fragmentation and dispersal.

Under what conditions could fuel fragmentation and dispersal occur?

Fuel dispersal is possible during <u>any</u> DBA which experiences cladding failure. There are likely three different initiators for fuel fragmentation under accident conditions: (1) rapid power excursion, (2) fuel rod balloon/rupture, and (3) mechanical loading. Each are discussed below.

<u>Rapid Power Excursion</u>:

Rapid thermal expansion of the fuel pellet and entrapped fission gas bubbles prompt fuel pellet fragmentation. Upon fuel rod cladding failure due to PCMI, the constraint provided by the cladding is lost and fuel fragments may be dispersed into the coolant. Figure 3-1 shows the degree of cladding failure during rapid power excursion tests. Note that these fuel rods

experienced fuel enthalpies below the allowable maximum criterion of 230 cal/g. Hence, significant PCMI cladding failure is permissible under existing regulatory guidance.

Figure 3-2 illustrates reported fuel dispersal as a function of burnup, pulse width, and peak fuel enthalpy. Figure 3-3 illustrates reported fuel dispersal as a function of peak fuel enthalpy and test rod segment burnup. Examination of this figure reveals that extensive fuel dispersal (up to 100%) has been reported on test segments exposed to power pulses below the existing regulatory limit of 230 cal/g (red line). Hence, fuel dispersal is possible under existing regulatory guidance.

Fuel Rod Balloon / Rupture:

Rod internal pressure increases with exposure as more fission gas is released from the pellet to the void volume. Rod internal pressure begins at approximately 800 psi and may increase up to 3000 psi by end of life.

During any DBA which experiences cladding dry-out conditions (e.g., Pressurized-Water Reactor (PWR) DNB due to flow reduction or local power increase), cladding temperatures will increase, cladding yield strength will decrease, and if rod internal pressure exceeds Reactor Coolant System (RCS) pressure, then the fuel rod is vulnerable to ballooning and potentially rupture. As the fuel rod cladding begins to balloon, it tears apart the outer radius of the pellet which was bonded to the cladding Inner Diameter (high burnup only); it removes the constraint that the cladding provided to the fuel pellet, allowing fission gas bubbles along the pellet grain boundary to expand (i.e., grain boundary separation); and providing a transport path within the fuel rod for fragmented pieces. The fragmentation of the fuel pellet releases fission gas (previously trapped within the pellet) which further increases rod internal pressure. Upon burst, the sudden depressurization blows fuel particles within the balloon region out into the RCS. Fuel above the burst opening within the balloon region is also susceptible to exit through the rupture opening. The sudden depressurization may cause further expansion of trapped fission gas bubbles and grain boundary separation.

Figure 3-4 illustrates the extent of fuel cracking (before test) and fragmentation (after test) on medium and high burnup fuel rod segments exposed to LOCA conditions at the Halden research reactor. Significant pellet fragmentation is observable in the post-test stage.

Figure 3-5 illustrates the degree of cladding damage, extent of cladding strain, and amount of dispersed fuel for four medium to high burnup fuel rod segments exposed to LOCA conditions at the Studsvik hot-cell facilities. Research suggests that the extent of fuel stack fragmentation is related to the extent of the balloon strain. Furthermore, only a minimal amount of cladding strain (i.e. 5% diametral) is needed to promote fragmentation.

Figure 3-6 shows dispersed fuel fragments collected following LOCA testing. Examination of this figure suggests that fuel fragmentation size distribution decreases with increasing pellet exposure.

These research findings are applicable to fuel rods under any DBA condition which results in fuel rod ballooning and rupture.

Mechanical Loading:

Mechanical loads due to postulated transportation accidents or fuel mishandling (e.g., fuel assembly drop) may also prompt fuel pellet fragmentation, especially within the porous high burnup rim region of the pellet. However, experimental evidence on mechanically-induced fuel fragmentation is limited.

What is the Current Regulatory Framework and Guidance with respect to FFRD?

Current regulatory framework with respect to FFRD is built around a philosophy of defense-in-depth. Defense-in-depth requires multiple safety barriers exist to mitigate the consequences of postulated DBAs and prevent the release of significant amounts of fission products. The three fission product barriers most are familiar with are: fuel rod cladding, RCS pressure boundary, and containment. In fact, there are four barriers. The fuel pellet maintains greater than 95% of the fission products during normal operation and is the 1st barrier to prevent release of fission products.

During normal operation, only a small percentage of the fission gas is released from the fuel pellet to the rod void volume. Experiments have shown that a large fraction of the trapped fission gas is released upon pellet fragmentation. In addition, solid fission products as well as fissile material may be dispersed into the RCS upon rupture. Existing regulations and guidance do not account for the loss of this 1st fission product barrier or its impact on radiological source term, coolable geometry, etc.

RG 1.183 and 1.195 provide acceptable radiological sources terms and guidance for calculating on-site and off-site doses during DBAs. Table 3 of these guides lists fission product gap inventories. These gap inventories are based on normal operation, diffusion-based fission gas transportation and release and do not account for any fragmentation-induced FGR.

10 CFR 50 Appendix A, General Design Criteria (GDC), Number 10, Reactor Design GDC-10, precludes fuel cladding failure during normal operation and AOOs. Hence, GDC-10 ensures that fuel fragmentation and dispersal do not occur during normal operation and AOOs. GDC-10 does not apply to DBAs.

GDC-27 and GDC-28 require that reactivity control systems and limits on reactivity ensure that the RCS pressure boundary (barrier for release of fission products) is not damaged and a coolable core geometry is maintained. For RIAs (e.g., BWR control rod drop, PWR Control Rod Ejection (CRE)), empirically-based limits on predicted fuel enthalpy protect against gross fuel rod failure and molten Fuel-to-Coolant Interaction (FCI). These limitations, which are translated into fuel management restrictions (e.g., maximum control rod worth), provide assurance that a coolable geometry (i.e., fuel rod bundle array) is maintained and fuel dispersal is limited in order to minimize any damaging pressure pulse due to FCI. Hence, existing guidance already acknowledges safety concerns related to fuel fragmentation and dispersal and provides appropriate limits to prevent catastrophic damage. Existing limits on fuel enthalpy (230 cal/g) only limit fuel fragmentation and dispersal during less likely transient scenarios involving the highest reactivity worth control rod. No regulatory limits or guidance address fuel fragmentation and dispersal (due to rod balloon/rupture) during more likely transient scenarios involving lower worth or partially inserted control rods.

GDC-35 requires that an ECCS be designed to prevent a loss of coolable core geometry and limit the generation of combustible gas. 10 CFR 50.46 prescribes limits on peak cladding temperature (2200 ºF) and maximum local oxidation (17% ECR) which preclude a loss of coolable geometry due to gross brittle cladding failure. However, ECCS performance demonstration calculations do not quantify the population of fuel rods which experience balloon/burst or the magnitude of fuel which may be dispersed. It is also worth noting that fuel rod balloon and rupture may occur very early in the transient scenario such that ECCS actions may not be capable of limiting the amount of dispersed fuel.

Westinghouse PWRs have been licensed with a peak cladding temperature limit of 2700 ºF during non-LOCA DBAs which precludes a loss of coolable geometry due to gross brittle cladding failure. However, no limits exist to preclude a loss of coolable geometry due to gross ductile cladding failure (i.e., rod balloon/burst).

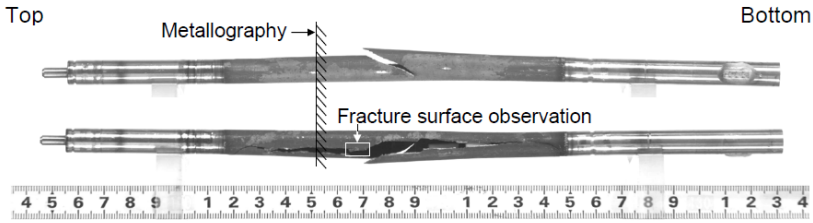
In conclusion, existing regulations and guidance were designed to protect against fuel fragmentation and dispersal under RIA-type rapid power excursions and protect against fuel dispersal (not fragmentation, but dispersal of fuel pellets) due to gross brittle cladding failure during LOCA and non-LOCA DBAs. However, more aggressive fuel utilization, including high burnup effects, exacerbate fuel fragmentation and dispersal phenomena and may necessitate further revision to regulatory guidance.

Figure 3-1: Fuel Cladding Failure During Rapid Power Excursion Testing

Source: Tomoyuki Sugiyama, Japan Atomic Energy Agency, "PCMI failure of high burnup fuels under RIA conditions," Fuel Safety Research Meeting 2007, Tokai, Japan, May 16-17, 2007.

Results of test LS-1

- Peak fuel enthalpy: 126 cal/g (tentative)
- PCMI failure at 60 cal/g (tentative)
- Large axial crack (no interaction with TC welding spots)
- All pellets were fragmented and ejected
- Water column movement was detected



Rod appearance

6

Tests VA-1 and -2

Fuel	- Irradiated at Vandellos #2 (PWR), Spain - 17x17 type, UO_2, 4.5% initial enrichment
Test conditions	- Reactivity: $4.6 - Coolant: room temperature, 0.1MPa - No cladding surface thermocouples
Results	- PCMI failure - Pellet fragmentation and ejection

	cladding	burnup	oxide	hydrogen	enthalpy at failure
VA-1	MDA	78 GWd/t	73 μm	660 ppm	61 cal/g
VA-2	ZIRLO	79 GWd/t	70 μm	760 ppm	55 cal/g

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Figure 3-2: Reported Fuel Dispersal During Rapid Power Excursion Testing
(Solid diamonds dispersed fuel, Open circles reported no dispersal, Green diamonds failed end cap weld)

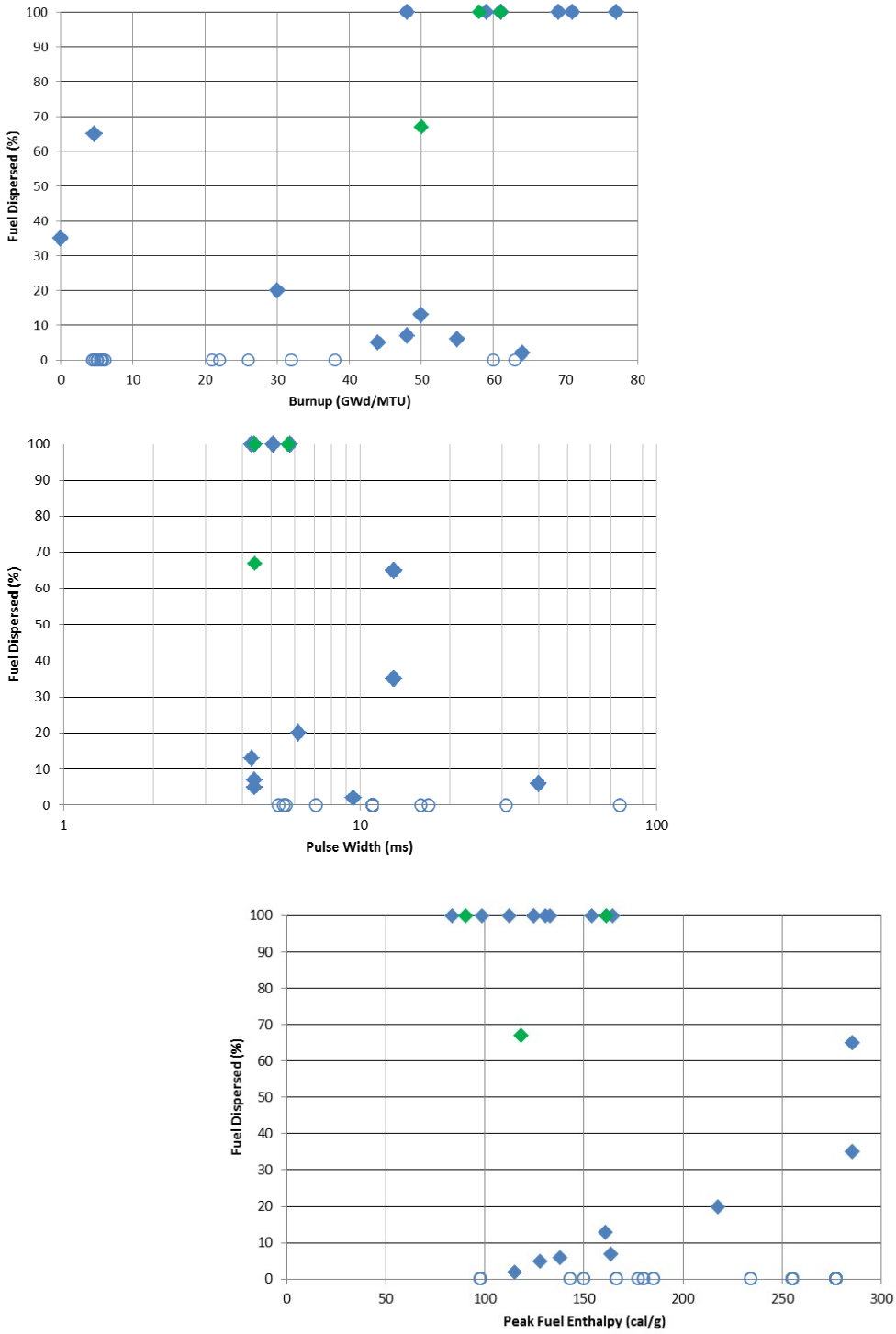


Figure 3-3: Sensitivity of Fuel Dispersal to Peak Fuel Enthalpy and Exposure

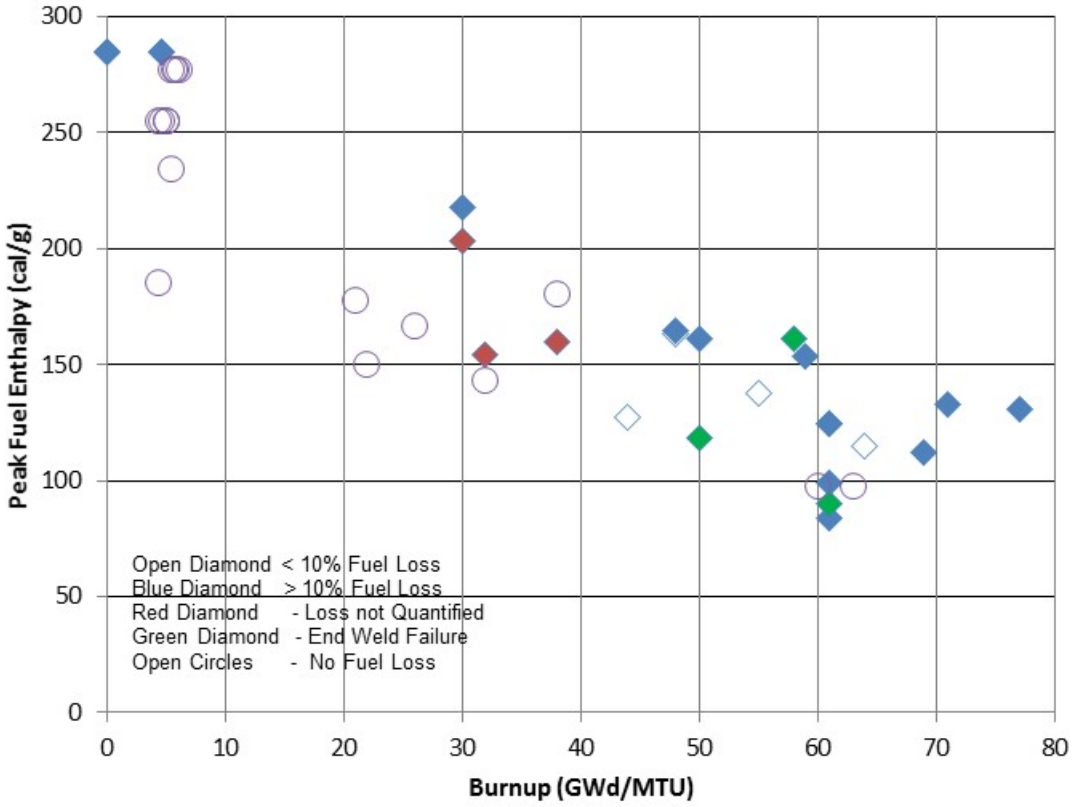


Figure 3-4: Fuel Fragmentation During Halden LOCA Tests

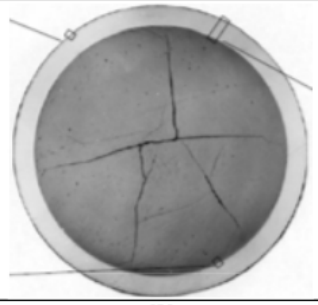
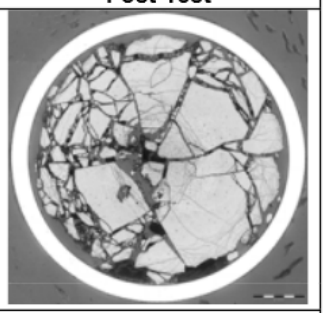
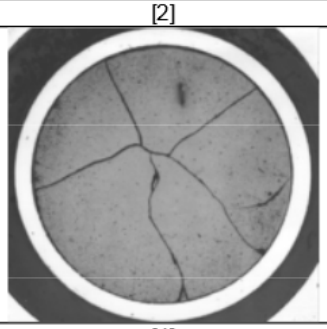
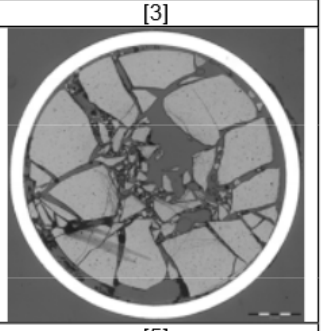
Test/Laboratory	Pre-Test	Post-Test
IFA-650.3, Halden 82 GWd/MTU, segment average		
Reference	[2]	[3]
IFA-650.7, Halden 44.3 GWd/MTU, segment average		
Reference	[4]	[5]

Figure 3-5: Fuel Fragmentation During Studsvik LOCA Tests

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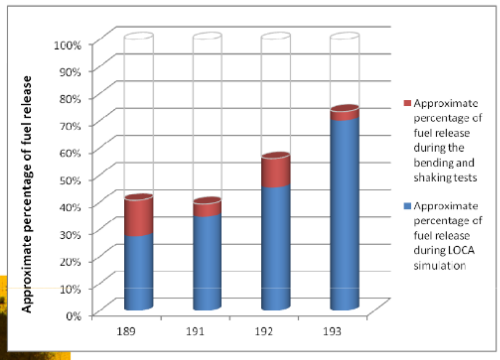
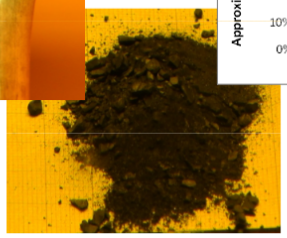
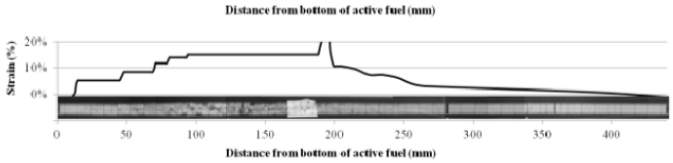
2010-2011: Tests in NRC's LOCA program at Studsvik also exhibit fuel dispersal



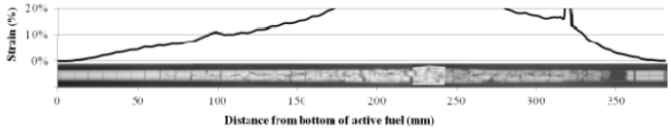


- Single, 30 centimeter long fuel rod segments
- Segments taken from high burnup rods, ~70 GWd/MTU that had been commercially irradiated in the US
- Tested in a specially designed capsule in a hot-cell facility (not in a reactor).
- Subjected to test conditions designed to maximize ballooning.
- Wire probe and mass balance showed major loss of fuel material after LOCA testing

5



IFA-650.11
VVER
56GWd/MTU, segment average



IFA-650.12
BWR
72GWd/MTU, segment average

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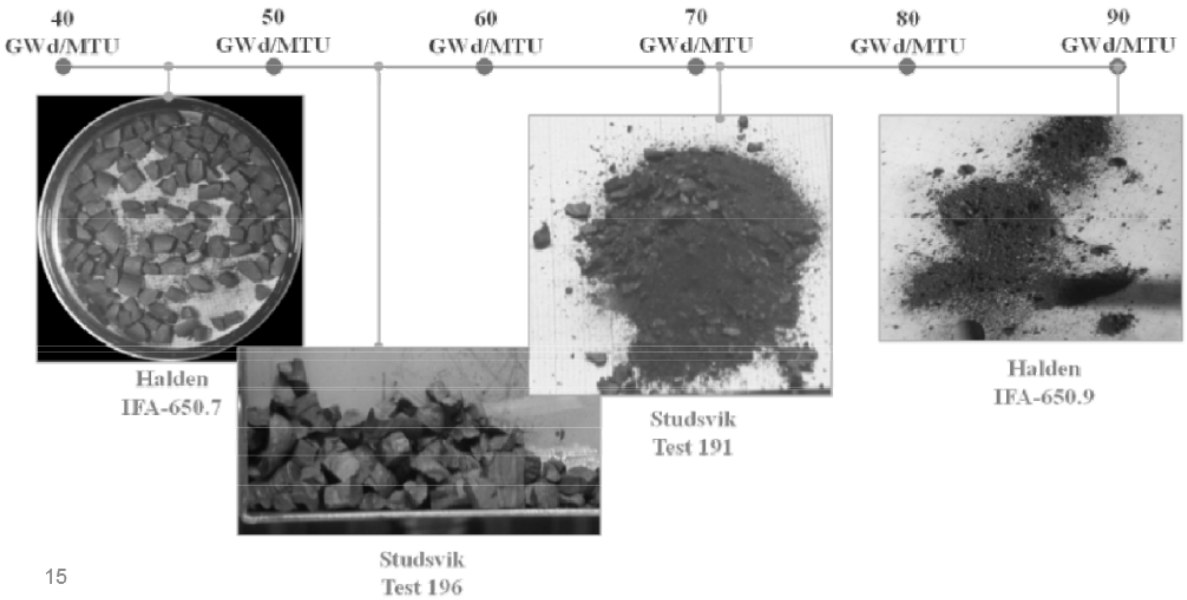
Figure 3-6: Fuel Fragmentation Size Distribution



4.0 POTENTIAL SAFETY CONCERNS

Fuel fragmentation and dispersal into the reactor coolant (and potentially beyond) introduces safety concerns beyond those currently analyzed and adds a significant level of complexity into safety-related systems' performance demonstrations.

Radiological consequences:

Non-LOCA DBA radiological consequences analyses employ explicit source terms based on accident progression, mitigating actions of safety-related Systems, Structures, and components (SSCs), and estimated fuel damage. Fragmentation-induced FGR introduces a new component to the accident source term (beyond the pre-existing gap inventory). This additional component potentially increases the source term which promotes higher on-site and off-site dose. Figure 4-1 provides measured fragmentation-induced FGR following rapid power excursion tests on irradiated fuel rod segments. Fragmentation-induced FGR due to balloon/rupture has been reported, but data is limited. Transient FGR due to mechanical loading (e.g., drop accident) has not been quantified.

In addition, dispersed and deposited fuel particles create local radiological "hot spots" within RCS, Containment, and Shutdown Cooling Systems (SDCs) which may not be accounted for in existing dose calculations.

Demonstrated compliance and performance of safety-related SSCs:

Mitigating actions from safety-related Reactor Protection System (RPS) and Engineered Safety Feature Actuation System (ESFAS) limit fuel damage and ensure acceptable consequences during postulated DBAs. Demonstrated compliance and performance of the integral plant response to postulated DBAs is captured in the Updated Final Safety Analysis Report (UFSAR). The capability of the safety-related SSCs to perform their intended functions does not consider the detrimental effects of fuel fragmentation and dispersal. Design calculations are all based on a known core geometry (i.e., fuel stack within cladding, rods within bundle array). Fuel dispersal creates unknown quantities and locations of heat sources which may alter accident progression and impact the performance of these safety-related SSCs.

Fuel-Coolant Interaction:

High enthalpy fuel particles dispersed into the reactor coolant will rapidly release their stored energy. The rate of release depends on many factors including particle size distribution and coolant quality. This FCI generates steam resulting in a pressure pulse which may challenge the integrity of the fuel bundle, reactor internals, and RCS pressure boundary. In addition, any rapid infusion of thermal energy into the reactor coolant will alter local thermal-hydraulic conditions (impacting DNBR/ Critical Power Ratio calculations) and may affect the overall accident progression.

Long-Term Cooling:

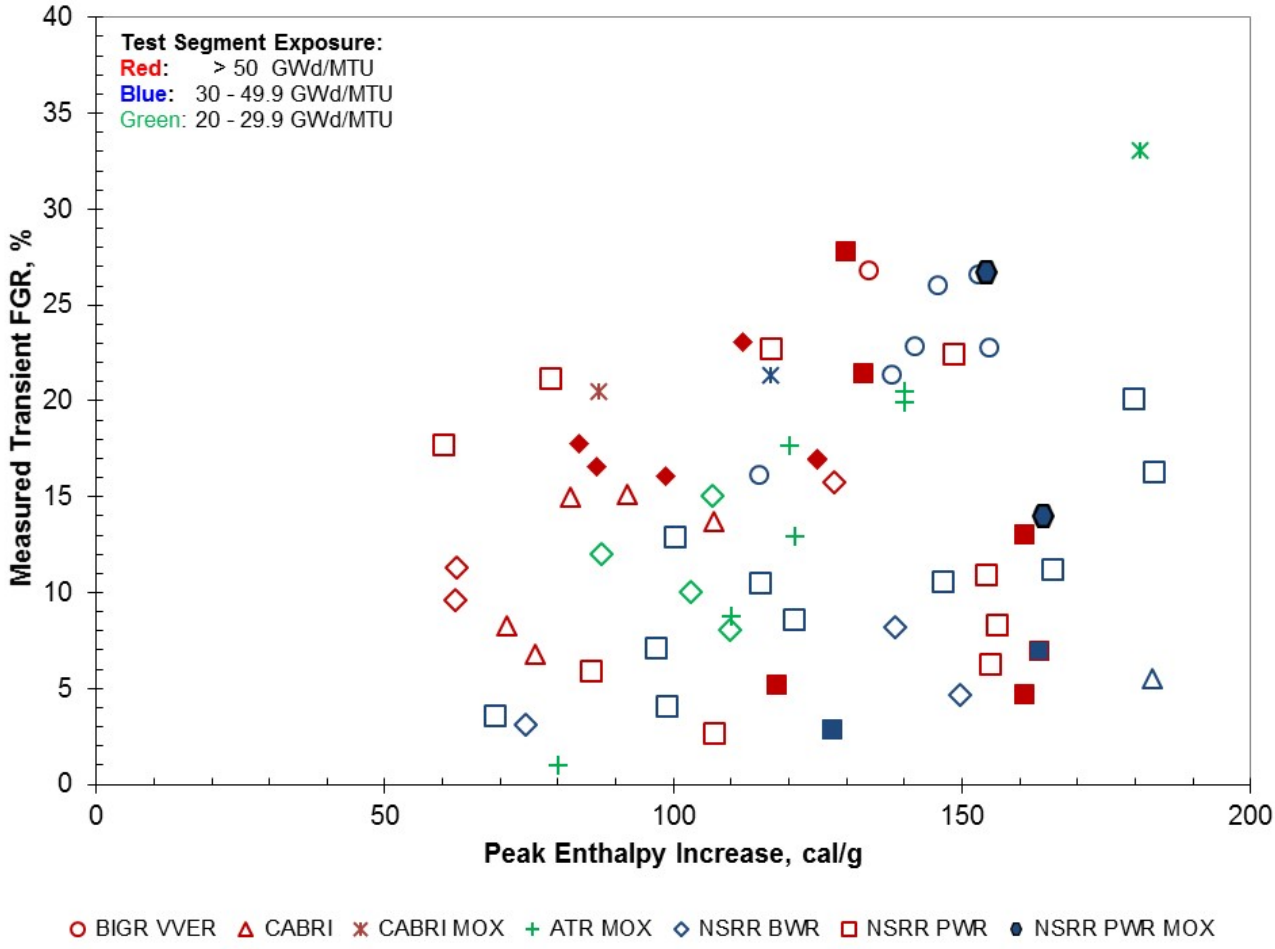
The ability of the SDC to perform its intended design function, long-term decay heat removal, must be demonstrated. The performance demonstration would need to address the long-term

decay heat removal of both the intact portion of the core and the dispersed fuel particles. Fuel transport and deposition depend on many variables including fuel particle size distribution, coolant mass flow and quality, and geometry. A sufficient empirical database does not exist to develop detailed analytical models nor quantify uncertainties associated with these variables. As a result, demonstrating long-term cooling in the presence of any significant amount of fuel dispersal would be difficult.

Equipment Qualification:

For the PWR CRE accident, a break in the control rod drive mechanism housing is the initiating event. Fuel particles blown out of the RCS via this leakage path may impact operability of instrumentation and cables important to accident mitigation and post-accident monitoring. For the remaining DBAs, deposited fuel particles create local thermal and radiological “hot spots” within the RCS and the SDC.

Figure 4-1: Fragmentation-Induced FGR Under RIA Conditions



5.0 NON-LOCA DESIGN-BASIS ACCIDENTS

Classified as Condition IV events, the following postulated DBAs are allowed to predict fuel cladding failure provided criteria associated with coolable geometry and radiological consequences are satisfied.

- PWR main steam line break (MSLB) accident
- PWR single reactor coolant pump (RCP) Locked Rotor (LR) or Sheared Shaft (SS)
- PWR CRE accident
- BWR Control Rod Drop Accident (CRDA)
- BWR MSLB (fuel failure allowed, but not reported)

For non-LOCA DBAs, fuel rod cladding failure mechanisms include (1) ballooning and rupture, (2) oxidation and embrittlement, (3) hydrogen-enhanced pellet-to-cladding mechanical interaction (PCMI), and (4) fuel melting induced PCMI. The first two failure mechanisms are identical to those experienced during a LOCA. Table 5-1 identifies the potential fuel rod failure mechanisms for each DBA.

Table 5-1: Non-LOCA DBA Fuel Failure Mechanisms

DBA	Balloon/ Rupture	Oxidation	PCMI	Fuel Melt
PWR MSLB	✓	✓		✓
PWR RCP LR/SS	✓	✓		
PWR CRE	✓	✓	✓	✓
BWR CRDA	✓	✓	✓	✓
BWR MSLB				

In order to experience fuel rod balloon and rupture, both elevated cladding temperature and a positive differential pressure across the cladding wall (i.e., rod internal pressure above system pressure) must co-exist. During a postulated large break LOCA RCS pressure drops dramatically. As such, the entire population of fuel rods will exhibit a positive differential pressure. During Non-LOCA DBAs, the RCS pressure boundary remains intact and only a small population of the fuel rods will exhibit a positive differential pressure.

For DBAs which experience cladding dryout conditions leading to elevated cladding temperature, fuel rod ballooning, and rupture, the expected susceptibility to fuel fragmentation and dispersal may be similar to those observed in the LOCA empirical database. As such, variables that influence fuel fragmentation and dispersal under LOCA conditions (e.g., cladding strain, burnup) may also be applicable to these non-LOCA DBAs.

Cladding dryout transient conditions leading to cladding perforations due to excessive local oxidation are not likely to promote fuel fragmentation or dispersal. Furthermore, gross brittle failure of the cladding due to oxygen embrittlement is avoided by limiting cladding time-at-temperature. For most non-LOCA transients, the time duration under these extreme conditions is limited to just a few seconds.

The last two failure mechanisms are predicted during over-power scenarios. Several non-LOCA DBAs experience a large power (and fuel temperature) excursion, which may further contribute to fuel fragmentation and dispersal. Additional variables related to fuel design, fuel management (as it relates to predicted core physics parameters such as rod worth, shutdown margin, MTC), and Technical Specification (TS) Limiting Conditions of Operation may play a role with respect to the degree of fragmentation and dispersal predicted under these over-power/over-temperature scenarios.

6.0 SAFETY ASSESSMENT

The potential impact of fuel fragmentation and dispersal on accident progression, predicted fuel failure, and radiological consequences is described below. For each event, this SA described the following:

1. Initiating event, mitigating actions, and sequence of events
2. Amount of fuel susceptible to fragmentation and dispersal
3. Safety margins to relevant criteria (e.g., dose, RCS integrity)

To support this SA, the staff reviewed every plant UFSAR and collected the following relevant information for each of the DBAs described in Section 5:

- Predicted core damage
- Core damage assumed in dose calculations
- Predicted onsite and offsite dose consequences
- Allowable onsite and offsite dose consequences

The information obtained from the UFSARs was used to quantify the number of fuel rods susceptible to fuel fragmentation and dispersal, amount of dispersed fuel, and potential impacts of dispersed fuel on accident progression and predicted consequences for each postulated DBA at each operating reactor. The compilation and statistical evaluation of this data are contained in the following database. Due to the sensitive nature of the information contained in the database, it is not publically available in ADAMS.

DBA_Fuel_Failure_Dose_Database_r0.xlsx (ADAMS ML15314A124)

6.1 PWR Main Steam Line Break

A description of the MSLB initiating event, inputs and assumptions, sequence of events, mitigating SSCs, and predicted consequences is provided in each licensee's UFSAR. The following event description was obtained from Chapter 15.1.3.1.2 of the San Onofre UFSAR and is representative of many PWRs.

> The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator, and subsequently the RCS, which results in a reduction of the reactor coolant temperature and pressure. In the presence of a negative Moderator Temperature Coefficient (MTC), this cooldown causes an increase in core reactivity. The reactor trips which may occur due to a steam line

break, assuming no loss of offsite ac power, are low steam generator pressure, low steam generator water level, high linear power level, and high containment pressure. For cases that assume a concurrent loss of offsite ac power, a reactor trip may also be caused by a steam generator ΔP low flow trip or by a low DNBR trip initiated by the core protection calculators. For any reactor trip, the control rod assembly of maximum worth is conservatively assumed to be held in the fully withdrawn position. In all cases, a low steam generator pressure signal would also initiate a Main Steam Isolation Signal which begins closure of the Main Steam Isolation Valves (MSIVs) and Main Feedwater Isolation Valves (MFIVs). The reduction of the RCS pressure empties the pressurizer and initiates a Safety Injection Actuation Signal (SIAS). The emptying of the steam generator associated with the ruptured steam line and the initiation of safety injection boron causes the core reactivity to decrease. A parametric review of the single failures that could occur during the Steam Line Break transient has determined that the failure of one of the High-Pressure Safety Injection (HPSI) pumps to start subsequent to the SIAS has the most adverse effect. Consequently, one HPSI pump is assumed to fail. The operator, via the appropriate emergency procedure, may initiate plant cooldown by manual control of the atmospheric steam dump valves, or the MSIV bypass valves associated with the intact steam generator, anytime after reactor trip occurs. This analysis assumes operator action is delayed until 30 minutes after first indication of the event. The plant is then cooled to 350F at which point shutdown cooling is initiated.

The MSLB accident progression and fuel damage is dictated by break size, break location, Steam Generator (SG) liquid inventory, MTC, shutdown margin, stuck control rod worth, initial thermal margin, and RPS/ESFAS actuations.

Differences in plant design result in a wide range of predicted performance during the postulated MSLB accident. For example, the large liquid inventory of a 2-loop Combustion Engineering U-tube SG design promotes substantially more cool-down than the smaller 4-loop Westinghouse U-tube SG design or once-through Babcock and Wilcox SG design. Because of these physical differences, plant licensing bases are also different. For example, Westinghouse and B&W plants do not predict any fuel failures during the postulated MSLB accident. These differences are illustrated in Figure 6.1-1 which shows the distribution in fuel failure for the MSLB event for the entire PWR fleet. Note that the fuel failure amounts presented in Figure 6.1-1 represent bounding values assumed in the offsite and onsite dose calculations. Examination of this figure reveals that a majority of the PWR fleet (55 of 65 reactors) do not predict any fuel failure. A few of the remaining PWRs predict no fuel failure; however, assume an amount in their licensing basis dose calculations. For the remaining few PWRs which predict fuel damage, the amount of fuel susceptible to fragmentation and dispersal is insignificant due to the following reasons:

- During the entire period of the accident (until steam flow in the faulted loop is halted by MSIV closure or SG dryout), RCS coolant temperature and pressure decrease. Figure 6.1-2 illustrates the trend in RCS pressure for a large U-tube SG design. Fuel damage may occur during the initial, pre-trip power excursion or during the subsequent return-to-criticality. Fuel damage during the initial, pre-trip power excursion is limited to the hottest fuel rods in the core. Rod internal pressure in these fuel rods is likely below system pressure during this initial portion of the transient (approx. 1900 - 2100 psia).

Hence, any fuel rods which experience cladding DNB conditions would not have the positive differential pressure necessary to experience balloon and rupture.

- Fuel cladding failure is presumed if predicted DNBR drops below the 95/95 DNBR Specified Acceptable Fuel Design Limit (SAFDL). Given the conservative nature of these calculations and minimal time in DNB (3-5 seconds), it is unlikely that any fuel damage would be experienced should an actual event occur.

- The power excursion experienced during the initial, pre-trip portion of the MSLB is benign compared with the prompt critical power excursion experienced under RIA conditions (for which dispersal was reported, Figure 3-2). Based on the limited power excursion, fuel fragmentation and dispersal due to this mechanism is not anticipated.

- In the event of a return-to-criticality, RCS pressure would be low (600 – 1200 psi) and expose a larger population of fuel rods to positive differential pressure. In addition, the time in DNB may be substantially longer than during the pre-trip portion of the event. These two factors make fuel rods more susceptible to balloon and rupture. However, without the loss of shutdown margin and large local power peaking factors beneath the assumed stuck out control rod, fuel rod damage would be avoided. Considering that design calculations demonstrate control rod insertion even with the combined applied loads of a coincident safe shutdown earthquake and LOCA, the assumption that the highest worth control rod does not insert is judged unnecessary for this SA. Hence, fuel damage is unlikely to occur. Furthermore, based upon review of current UFSARs, no plants predict fuel damage during the return-to-criticality portion of the MSLB accident (even with stuck out control rod).

Based on the above, the amount of fuel susceptible to fuel fragmentation and dispersal is insignificant for the MSLB accident. Hence, there are no detrimental effects of dispersed fuel on accident progression, systems' performance, and long-term cooling.

With respect to radiological source term, the amount of fragmentation-induced FGR is expected to be small, if any, due to the minimum quantity of predicted core damage and burnup distribution of failed rods. Only a small quantity of fragmentation-induced FGR is possible from low burnup fuel rods due to (1) retention of fission gas within larger size fragments at low burnup and (2) limited quantity (moles) of fission gas at low burnup.

Guidance for MSLB dose calculations is provided in Appendix E of RG 1.183. For the majority of the PWR fleet which do not experience fuel damage, dose calculations are based on assumed coolant activity levels corresponding to a Pre-existing Iodine Spike (PIS) and accident generated iodine spike (GIS). Iodine spiking is postulated due to allowable operation with damaged fuel (i.e., pre-existing failures). TSs on allowable RCS activity (typically 1.0 μCi/gm DE I-131) limit operation to a very small population of damage fuel rods. TSs also limit associated iodine spiking to values assumed in dose calculations. Fuel fragmentation (and fragmentation-induced FGR) is unlikely to occur in the limited population of pre-existing failed fuel rods during a MSLB event. Power excursion is limited and no positive differential pressure would exist in fuel rods with breached cladding.

Dose calculations for plants which predict or assume fuel damage are based on conservative source terms, including bounding quantities of fuel damage, maximum power peaking factors, and end-of-life exposure. Figure 6.1-3 illustrates the proximity of the existing PWR fleet to their respective allowable radiological consequences. None of the existing fleet report predicted offsite doses within 50% of allowable limits. With respect to control room habitability, 6 plants report predicted doses within 50% of allowable. However, all of these plants currently predict zero fuel damage during the MSLB accident. Hence, actual control room doses would be significantly lower than reported (based on assumed fuel damage) for the postulated accident.

In conclusion, a vast majority of the PWR fleet predicts zero fuel damage during the postulated MSLB accident. For the remaining fleet, the amount of dispersed fuel particles and fragmentation-induced FGR are insignificant for the MSLB accident. All PWRs exhibit ample margin to allowable radiological consequences for the MSLB accident.

Figure 6.1-1: PWR MSLB Distribution of Predicted Fuel Failure

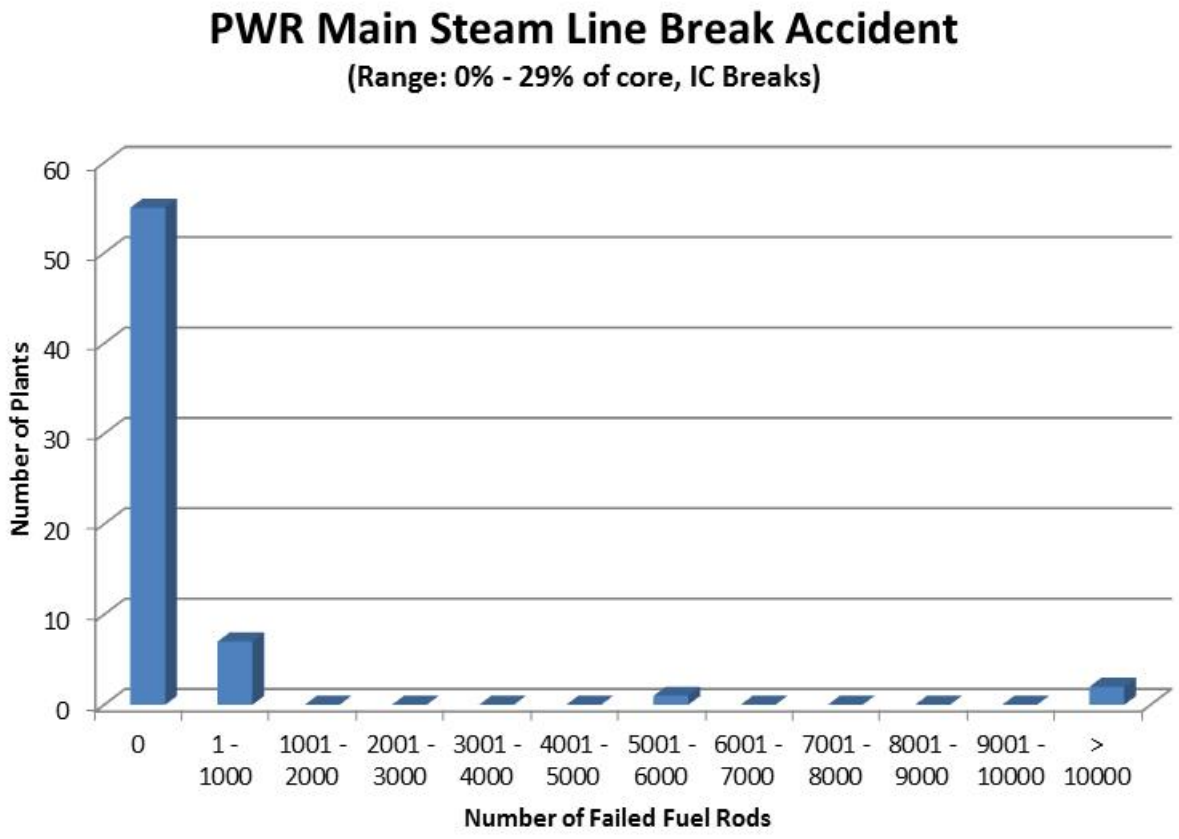


Figure 6.1-2: PWR MSLB RCS Pressure

(Source: PVNGS UFSAR Figure 15.1.5-6)

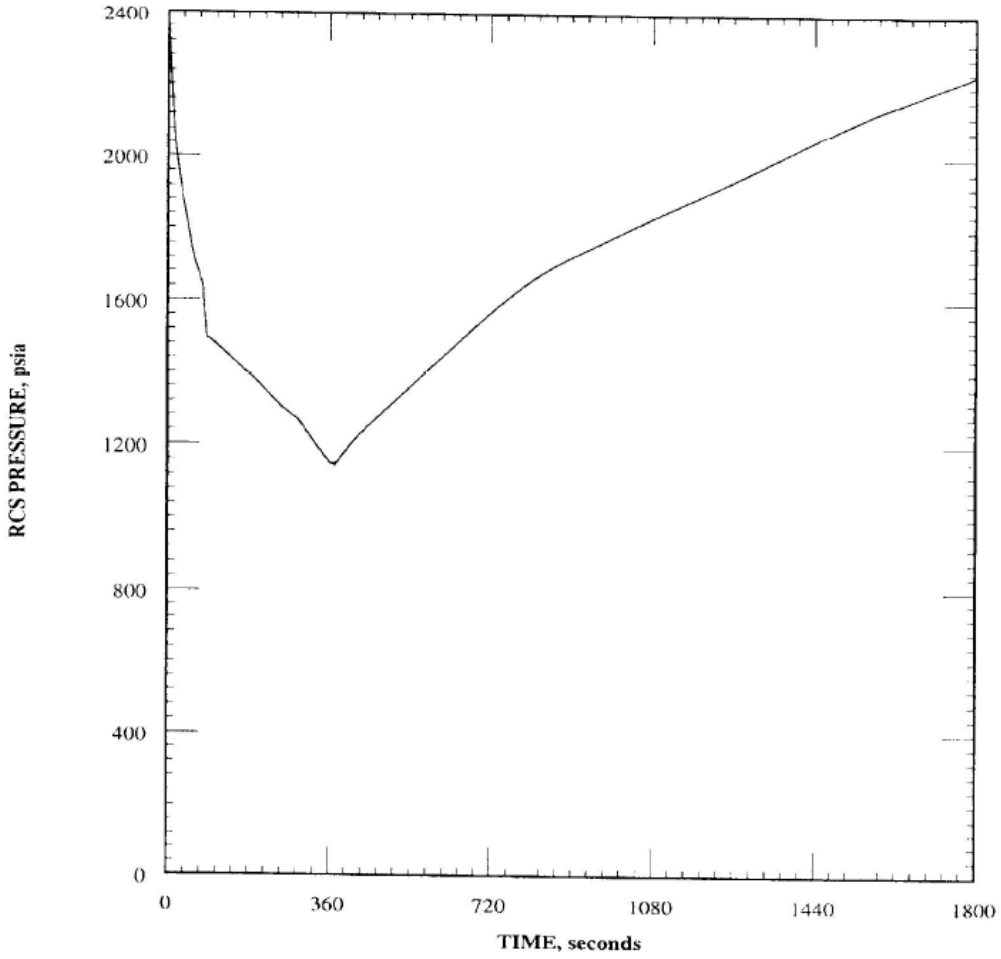
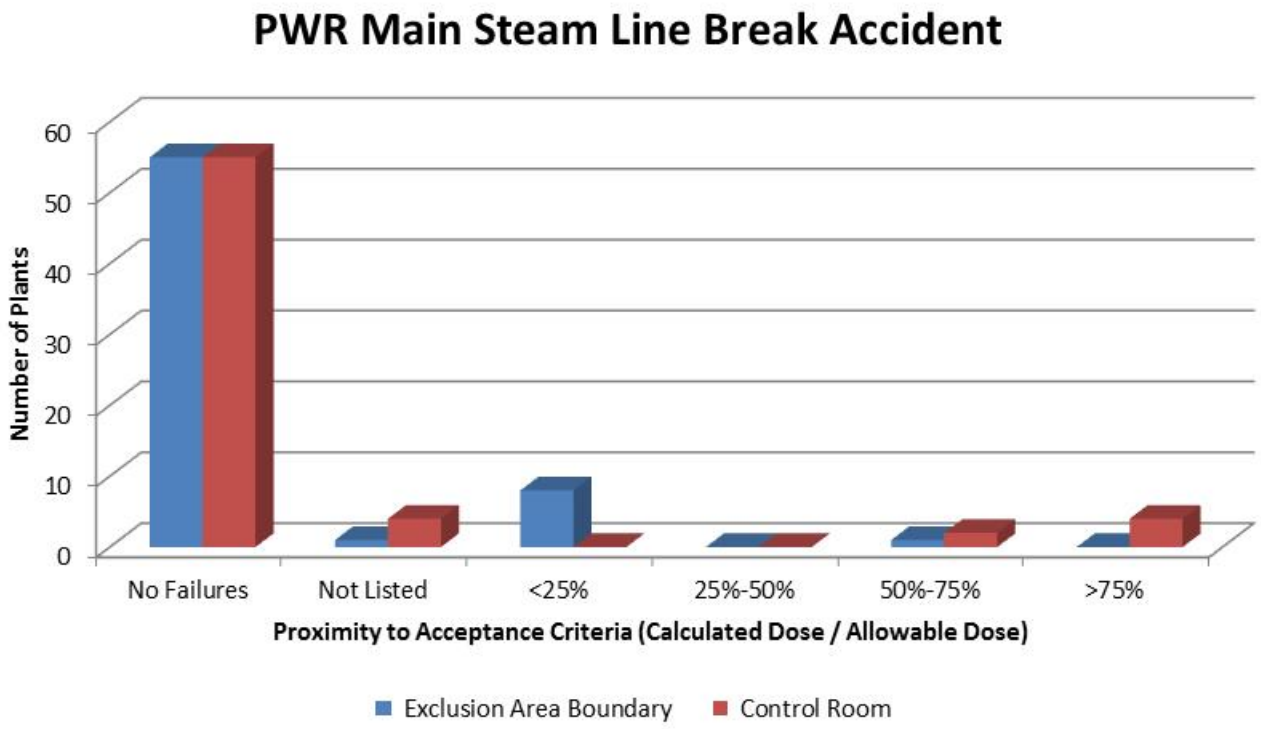


Figure 6.1-3: PWR MSLB Proximity to Dose Limits



6.2 PWR RCP Locked Rotor / Sheared Shaft

A description of the RCP Sheared Shaft initiating event, inputs and assumptions, sequence of events, mitigating SSCs, and predicted consequences is provided in each licensee's UFSAR. The following event description was obtained from Chapter 15.3.3.2 of the San Onofre UFSAR and is representative of many PWRs.

> Following the shearing of a RCP shaft, the core flowrate rapidly decreases to the value that would occur with only three RCPs operating. The reduction in coolant flowrate causes an increase in the average coolant temperature in the core and may produce a DNB in some portions of the core. A reactor trip is generated when the rapid flow reduction across the steam generator in the affected loop decreases the delta-pressure below the trip setpoint. The reactor trip produces an automatic turbine trip. Following turbine trip, offsite power is available to provide ac power to the auxiliaries. Due to the turbine trip, the turbine bypass valves would normally open. If the steam bypass control system is in the manual mode and no credit is taken for immediate operator action, the steam generator safety valves will open to relieve steam and provide an ultimate heat sink for the NSSS. The operator can initiate a controlled system cooldown using the turbine bypass valves any time after reactor trip.

The RCP LR/SS accident progression and fuel damage is dictated by the rate of flow coastdown, MTC, initial thermal margin, and RPS/ESFAS actuations.

Figure 6.2-1 shows the distribution in fuel failure for the RCP LR/SS event for the entire PWR fleet. Note that the fuel failure amounts presented in Figure 6.2-1 represent bounding values assumed in the offsite and onsite dose calculations and are not necessarily those predicted in the actual design calculations. A majority of UFSARs do not provide a predicted number of failed rods, instead only report the value used in the dose calculation. Of those plants that provided a specific value, 23 reactors (35% of fleet) reported zero fuel damage.

For the following reasons, the amount of fuel susceptible to fuel fragmentation and dispersal is likely insignificant:

- As shown in Figure 6.2-2, RCS pressure increases during the LR/SS accident. Fuel damage during the asymmetric flow coastdown is limited to lower burnup, higher power fuel rods. Rod internal pressure in these fuel rods is likely below system pressure (2200+ psia). Higher burnup fuel rods with higher rod internal pressure operate at lower power and possess higher initial thermal margin (i.e., operate further from DNBR SAFDL) and will not experience DNB prior to reactor trip. Hence, any fuel rods which experience DNB conditions would not have the positive differential pressure necessary to experience balloon and rupture.

- Fuel cladding failure is presumed if predicted DNBR drops below the 95/95 DNBR SAFDL. Given the conservative nature of these calculations and minimal time in DNB (3-5 seconds), it is unlikely that any fuel damage would be experienced should an actual event occur.

- No power excursion is experienced during the LR/SS event (due to negative MTC). Hence, the type of fuel fragmentation and dispersal reported under rapid power excursion conditions is not applicable.

Based on the above, the amount of fuel susceptible to fuel fragmentation and dispersal is insignificant for the LR/SS accident. Hence, there are no detrimental effects of dispersed fuel on accident progression, systems' performance, and long-term cooling.

With respect to radiological source term, the amount of fragmentation-induced FGR is expected to be small, if any, due to the minimum quantity of predicted core damage and burnup distribution of failed rods. Only a small quantity of fragmentation-induced FGR is possible from low burnup fuel rods due to (1) retention of fission gas within larger size fragments at low burnup and (2) limited quantity (moles) of fission gas at low burnup. Higher burnup fuel rods are unlikely to fail and therefore not a concern with respect to radiological source term.

Dose calculations for plants which predict <u>or assume</u> fuel damage are based on conservative source terms, including bounding quantities of fuel damage, maximum power peaking factors, and end-of-life exposure. Figure 6.2-3 illustrates the proximity of the existing PWR fleet to their respective allowable radiological consequences. While several plants report doses within 25% of allowable consequences, no significant increase in source term is expected as a result of fuel fragmentation-induced FGR and existing reported consequences remain bounding.

In conclusion, the amount of dispersed fuel particles and fragmentation-induced FGR are insignificant during the PWR LR/SS accident. All PWRs exhibit margin to allowable radiological consequences for the PWR LR/SS accident.

Figure 6.2-1: Distribution of Predicted Fuel Failure

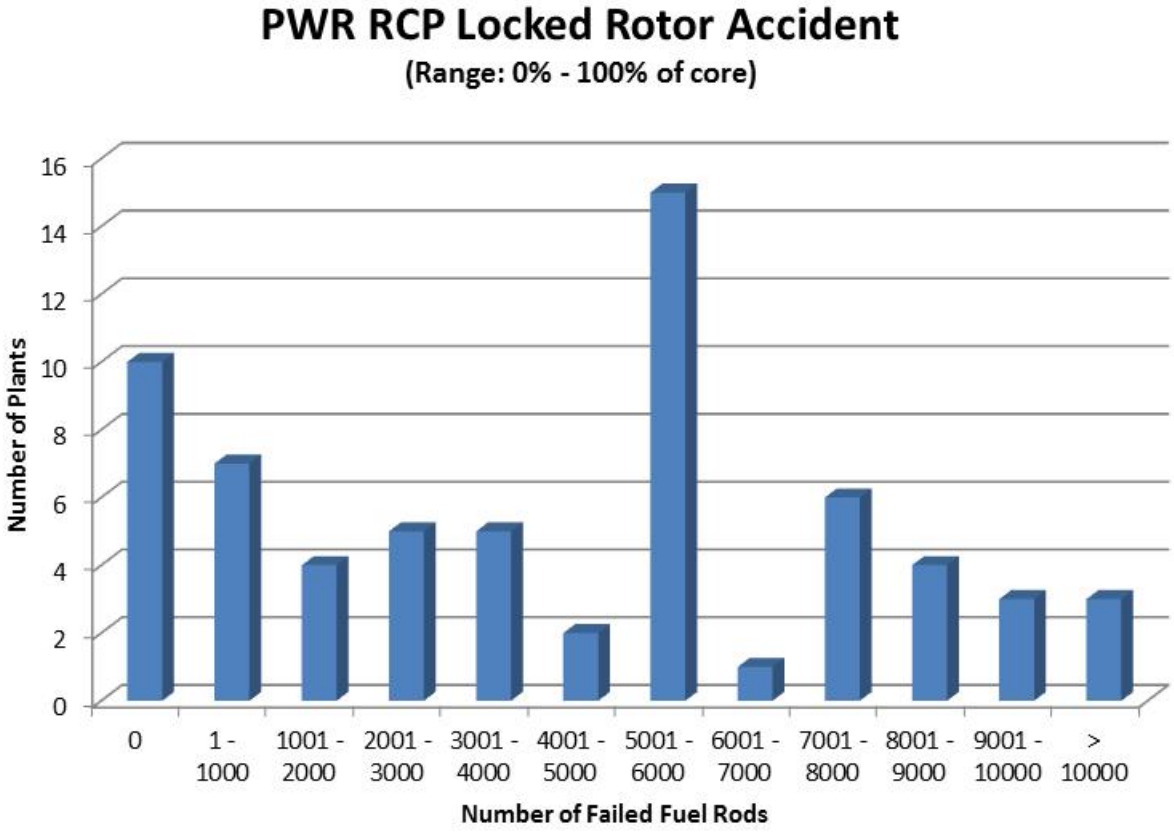


Figure 6.2-2: PWR RCP Locked Rotor RCS Pressure

(Source: Byron / Braidwood UFSAR Figure 15.3-6)

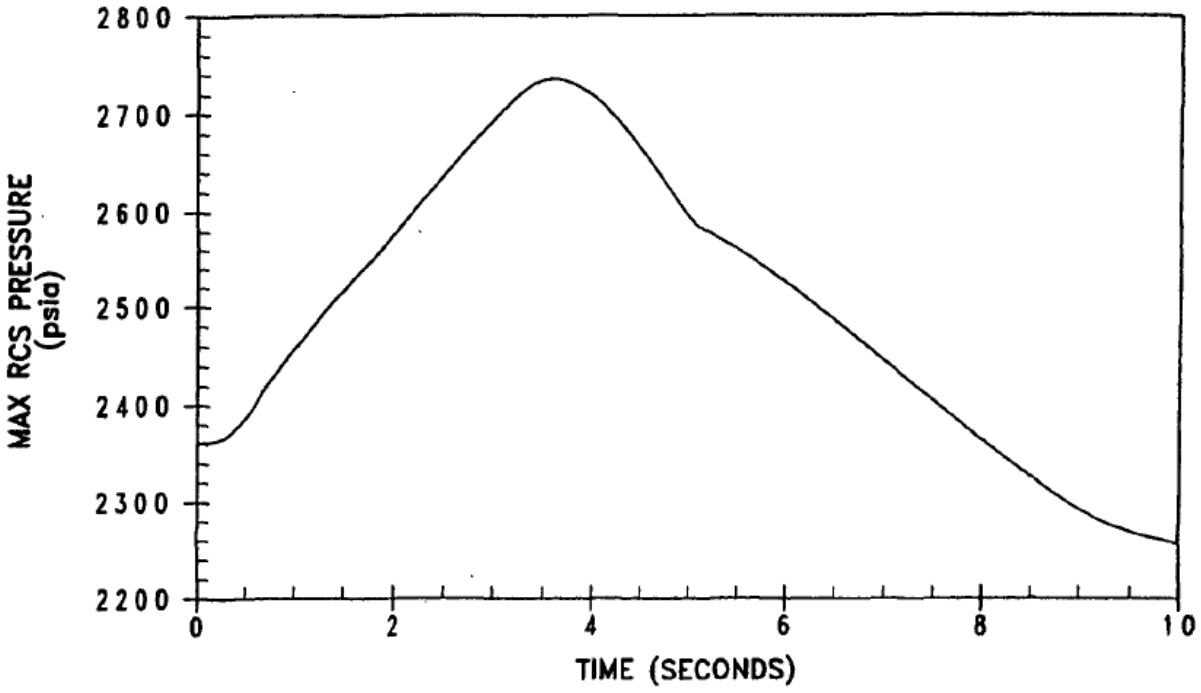
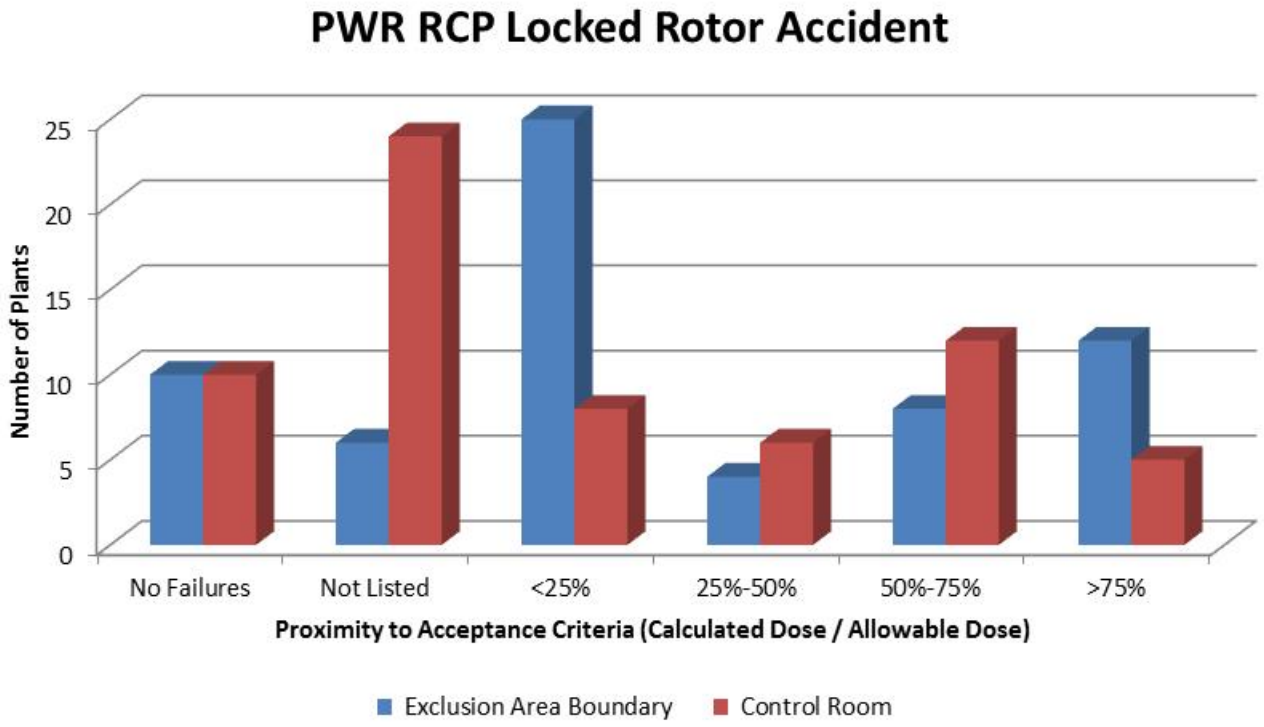


Figure 6.2-3: Proximity to Dose Limits



6.3 PWR Control Rod Ejection

A description of the CRE initiating event, inputs and assumptions, sequence of events, mitigating SSCs, and predicted consequences is provided in each licensee's UFSAR. The following event description was obtained from Chapter 15.4.3.2 of the San Onofre UFSAR and is representative of many PWRs.

> For a Control Element Assembly (CEA) ejection to occur, a mechanical failure of the Control Element Drive Mechanism (CEDM) housing or of the CEDM nozzle must be postulated such that the reactor coolant system pressure ejects the CEA and drive shaft to the fully withdrawn position. For this analysis, it was assumed that a complete and instantaneous circumferential rupture of the CEDM housing or of the CEDM nozzle resulted in the ejection of a CEA.
>
> The transient behavior following a CEA ejection accident is as follows. The reactor core power rises rapidly for a short period of time. This increase is terminated by Doppler feedback (predominant at zero power) or delayed neutron effects (predominant at full power). Following this, a reactor shutdown is initiated by a reactor trip on high neutron power and the power transient is terminated. The potential for fuel damage from a CEA ejection is minimized by restrictions on CEA patterns and/or power dependent insertion limits (PDIL) during operation, and by a reactor trip, all of which combine to limit the fuel enthalpy, fuel and clad temperatures, and RCS pressure during the transient to acceptable values.
>
> The consequences of the loss of coolant resulting from the RCS rupture are similar to those for small RCS breaks as discussed in subsection 6.3.3. Protection from a postulated missile resulting from a CEA ejection is discussed in section 3.5.

The CRE accident progression and fuel damage is dictated by ejected rod worth, reactivity kinetics (βeff), fuel temperature coefficient (Doppler), MTC, initial thermal margin, and RPS/ESFAS actuations.

For the purpose of this discussion, the CRE accident will be divided into two categories: (1) high worth prompt power excursion terminated by Doppler feedback (i.e., high ejected worth) and (2) non-prompt power excursion terminated by reactor trip (i.e., low ejected worth). For the first scenario, existing guidance defines a peak radial average fuel enthalpy limit to preserve the rod bundle array (i.e., coolable geometry) and limit fuel dispersal and molten FCI. Since its inception within RG 1.77 (dated May 1974), the coolability limit on peak radial average fuel enthalpy, 280 cal/g, has evolved to account for burnup effects. Westinghouse plants employ a coolability limit of 200 cal/g. SRP-4.2 Appendix B provides a two tier coolability criteria: (1) peak radial average fuel enthalpy must remain below 230 cal/g and (2) peak fuel temperature must remain below incipient fuel melting conditions. The no melt criterion effectively reduces the fuel enthalpy limit below 230 cal/g, decreasing with increasing exposure.

As shown in Figure 3-3, significant fuel dispersal has been reported during rapid power excursion tests at fuel enthalpies below the coolability criteria discussed above. However, the

amount of fuel fragmentation and dispersal is likely limited for the following reasons:

- High ejected rod worths are achievable only in heavily rodded core configurations. TSs limit control rod insertion as a function of power (power dependent insertion limits, PDIL). Heavily rodded core configurations are only allowable at lower power levels. Given that PWRs generally operate at full power (i.e., baseload) with all control rods out of the core, the probability of a reactivity-initiated accident is extremely low. Low worth power shaping rods may be inserted at power conditions; however, ejection of these rods would produce only a benign power transient.

- Ejection of a single control rod promotes a rapid power excursion in the adjacent, surrounding fuel. For high worth scenarios involving heavily rodded configurations, only a small region of the core experiences a significant power excursion. Hence, core damage and any possible fuel dispersal limited.

- Existing regulatory guidance limits peak fuel enthalpy to maintain a coolable geometry (i.e., loss of fuel bundle array) and minimize fuel dispersal. A majority of the plants employ a 200 cal/g limit on peak fuel enthalpy for irradiated fuel.

- In response to the latest results from international RIA test programs (e.g. NSRR, CABRI, IGR, and BIGR), the NRC completed an assessment of currently operating reactors. Research Information Letter 0401 (ADAMS Accession No. ML040920189) compiled all of the RIA test results, performed best-estimate 3D neutronics analyses for a range of LWR conditions, and concluded that the control rod worth needed to reach the cladding failure threshold were beyond expected values. Without cladding failure, coolable geometry is ensured and energetic FCI is avoided. Hence, application of 3D analytical methods is likely to demonstrate no fragmentation and dispersal.

- Figure 3-3 suggests that fuel dispersal of high burnup fuel rods may occur at a fuel enthalpy of approximately 175 cal/g and decreasing with increasing exposure. The 3D analytical results discussed above suggest that fuel enthalpy would remain well below these values. Furthermore, higher burnup fuel has less reactivity leading to lower ejected rod worth and a more benign power excursion.

For the low worth, non-prompt rod ejection scenario, fuel fragmentation and dispersal may result from post-DNB fuel rod balloon and rupture. This failure mode may produce more global core damage and is limiting with respect to the number of fuel rod failures input to the onsite and offsite dose calculations. Figure 6.3-1 shows the distribution in fuel failure for the CRE event for the entire PWR fleet. Note that the fuel failure amounts presented in Figure 6.3-1 represent bounding values assumed in the dose calculations and are not necessarily those predicted in the actual design calculations. A majority of UFSARs do not provide a predicted number of failed rods, instead only report the value used in the dose calculation.

For the following reasons, the amount of fuel susceptible to fuel fragmentation and dispersal (due to balloon and rupture) is likely insignificant:

- With the exception of very low worth power shaping rods in B&W reactors, PWRs generally operate with all-rods-out. Coupled with the low probability of failure for a CEDM housing, the probability of a CRE accident is extremely low.

- As shown in Figure 6.3-2, RCS pressure increases during the CRE accident. Fuel damage is limited to lower burnup, higher power fuel rods. Rod internal pressure in these fuel rods is likely below system pressure (2200+ psia). Higher burnup fuel rods with higher rod internal pressure operate at lower power and possess higher initial thermal margin (i.e., operate further from DNBR SAFDL) and will not experience DNB prior to reactor trip. Hence, any fuel rods which experience post-DNB conditions would not have the positive differential pressure necessary to experience balloon and rupture.

- Fuel cladding failure is presumed if predicted DNBR drops below the 95/95 DNBR SAFDL. Given the conservative nature of these calculations and minimal time in DNB (3-5 seconds), it is unlikely that any fuel damage would be experienced should an actual event occur.

Based on the above, the amount of fuel susceptible to fuel fragmentation and dispersal is minimal for the CRE accident. Hence, there are no detrimental effects of dispersed fuel on accident progression, systems' performance, and long-term cooling.

With respect to radiological source term, the amount of fragmentation-induced FGR is expected to be small due to the following reasons:

- Amount of fragmentation-induced FGR due to balloon and rupture is insignificant due to minimum quantity of predicted core damage and burnup distribution of failed rods. Only a small quantity of fragmentation-induced FGR is possible from low burnup fuel rods due to (1) retention of fission gas within larger size fragments at low burnup and (2) limited quantity (moles) of fission gas at low burnup.

- Amount of fragmentation-induced FGR due to rapid power excursion is expected to be limited due to minimum core damage. Figure 4-1 shows measured fragmentation-induced FGR as a function of increase in local fuel enthalpy. Examination of this figure reveals that fragmentation-induced FGR was reported on specimens exposed to an increase in local fuel enthalpy of greater than 60 cal/g. CRE is a localized event. As such, only a small portion of the core experiences a rapid power excursion. Furthermore, as described above, application of improved 3D analytical methods will greatly reduce predicted core damage and fuel enthalpy.

Dose calculations for plants which predict fuel damage are based on conservative source terms, including bounding quantities of fuel damage, maximum power peaking factors, and end-of-life exposure. Figure 6.3-3 illustrates the proximity of the existing PWR fleet to their respective allowable radiological consequences. While several plants report doses within 25% of allowable

consequences, no significant increase in source term is expected as a result of fuel fragmentation-induced FGR and existing reported consequences remain bounding.

In conclusion, the amount of dispersed fuel particles and fragmentation-induced FGR are minimal during the PWR CRE accident. All PWRs exhibit margin to allowable radiological consequences for the PWR CRE accident.

Figure 6.3-1: Distribution of Predicted Fuel Failure

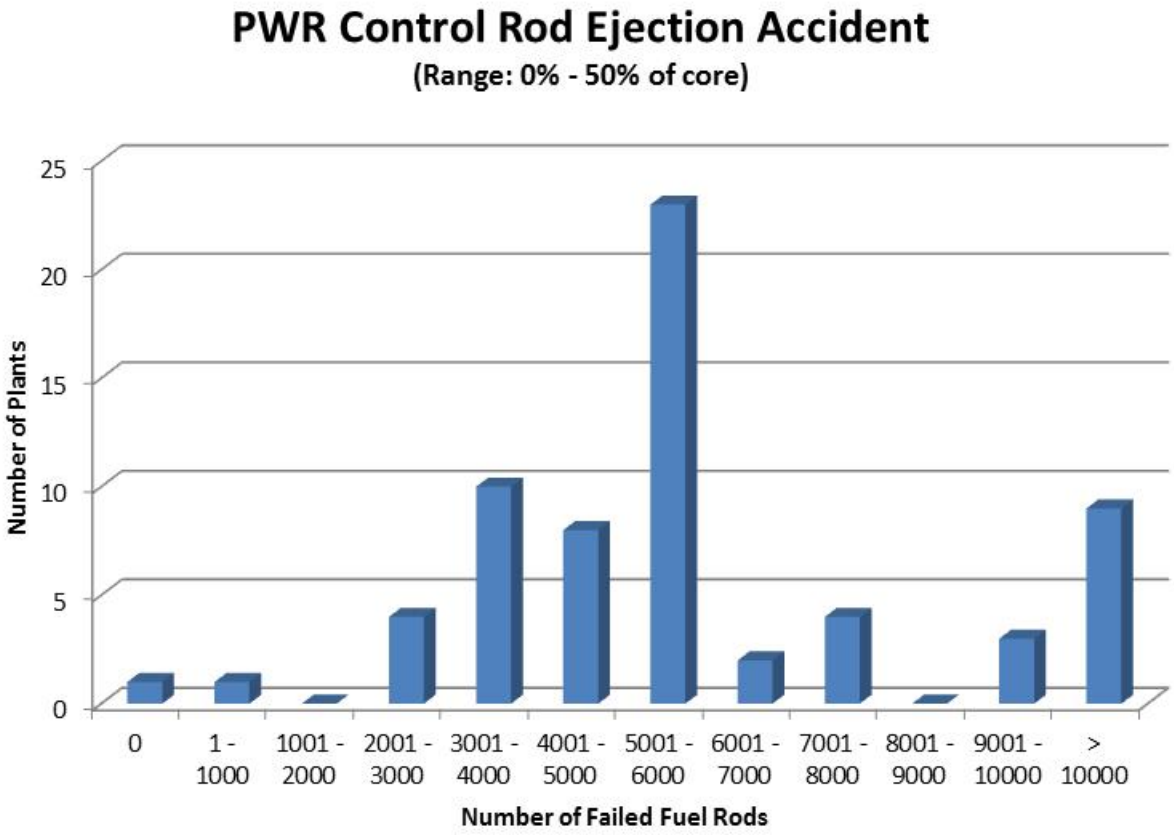


Figure 6.3-2: PWR CRE RCS Pressure

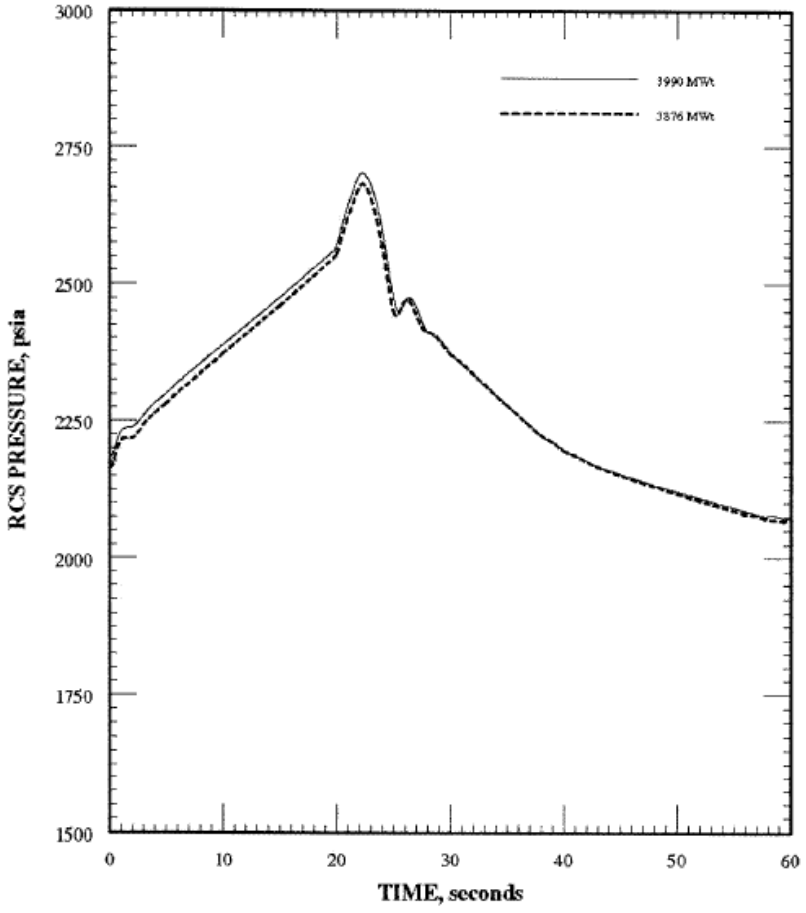
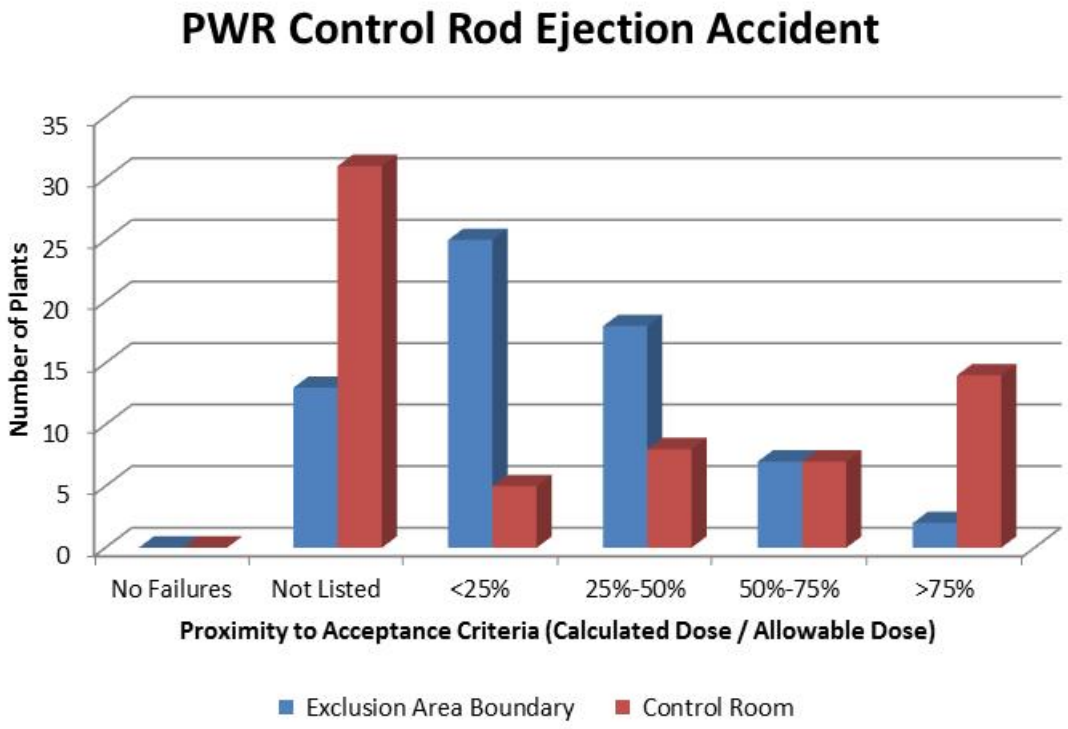


Figure 6.3-3: Proximity to Dose Limits



6.4 BWR Control Rod Drop Accident

A description of the CRDA initiating event, inputs and assumptions, sequence of events, mitigating SSCs, and predicted consequences is provided in each licensee's UFSAR. The following event description was obtained from Chapter 15.4.9 of the Clinton UFSAR and is representative of many BWRs.

> The control rod drop accident is the result of a postulated event in which a highest worth control rod, within the constraints of the banked position RCIS, drops from the fully inserted or intermediate position in the core. The highest worth rod becomes decoupled from its drive mechanism. The mechanism is withdrawn but the decoupled control rod is assumed to be stuck in place. At a later moment, the control rod suddenly falls free and drops to the control rod drive position. This results in the removal of large negative reactivity from the core and results in a localized power excursion.
>
> The unlikely set of circumstances, referenced above, makes possible the rapid removal of a control rod. The dropping of the rod results in high reactivity in a small region of the core. For large, loosely coupled cores, this would result in a highly peaked power distribution and subsequent operation of shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion.
>
> The Rod Control and Information System (RCIS) limits the worth of any control rod which could be dropped by regulating the withdrawal sequence. This system prevents the movement of an out-of-sequence rod in the 100 to 75% rod density range, and from the 75% rod density point to the preset power level the RCIS will only allow bank position mode rod withdrawals or insertions. The banked position mode of this system is described in Reference 2 for a typical BWR.
>
> The RCIS used redundant input to provide absolute assurance on control rod drive position. If either of the diverse input were to fail the other would provide the necessary information. The termination of this excursion is accomplished by automatic safety features of inherent shutdown mechanisms. Therefore, no operator action during the excursion is required. Although other normal plant instrumentation and controls are assumed to function, no credit for their operation is taken in the analysis of this event.

As described below in an excerpt from the LaSalle UFSAR (Section 15.4.9.3), the BWR CRDA is analyzed from the hot-standby condition assuming the highest worth control rod. The reactor goes on a positive period and the initial power increase is terminated by Doppler feedback within 1 second. A reactor trip signal on high power (120% ARPM) is generated within 1 second and the resulting scram terminates the event by 5 seconds.

> At the time of the control rod drop accident, the core is assumed to be at an operating cycle point which results in the highest control rod worth. The core is also assumed to contain no xenon, to be in a hot-standby condition, and to have the control rods in sequence A and be near critical. The assumption to remove xenon, which competes well for neutron absorptions, increases the fractional absorptions, or worth, of the control rods.

Figure 6.4-1 shows the distribution in fuel failure for the CRDA event for the entire BWR fleet. Note that the fuel failure amounts presented in Figure 6.4-1 represent bounding values assumed in the dose calculations and are not necessarily those predicted in the actual design calculations. A majority of UFSARs do not provide a predicted number of failed rods, and instead only report the value used in the dose calculation.

For the CRDA, a fuel rod cladding failure threshold of 170 cal/g (radial average fuel enthalpy) is used to assess fuel damage and resulting onsite and offsite doses. Failure mechanisms reported at rapid power test programs for this threshold include ductile failure (balloon and rupture) and oxidation embrittlement. However, applicants are allowed to predict fuel enthalpy above the 170 cal/g cladding failure threshold up to the 230 cal/g coolability criteria. As shown in Figure 3-3, significant fuel dispersal has been reported during rapid power excursion tests at fuel enthalpies below the coolability criteria discussed above. However, the amount of fuel fragmentation and dispersal is likely limited for the following reasons:

- Application of improved 3D analytical methods will dramatically reduce predicted fuel enthalpy and minimize the number of rods which experience cladding failure.

- CRDA power excursion is limited to a small region of the core. As shown on Figure 6.4-1, bounding fuel failure quantities range from 1.2% - 3.5% of the core. Of those rods, only a portion of the fuel stack will experience the power excursion.

- Figure 3-3 suggests that fuel dispersal of high burnup fuel rods may occur at a fuel enthalpy of approximately 175 cal/g and decreasing with increasing exposure. The 3D analytical results discussed above suggest that fuel enthalpy would remain well below these values. Furthermore, higher burnup fuel has less reactivity leading to lower ejected rod worth and a more benign power excursion.

Based on the above, the amount of fuel susceptible to fuel fragmentation and dispersal is minimal for the CRDA. Hence, there are no detrimental effects of dispersed fuel on accident progression, systems' performance, and long-term cooling.

With respect to radiological source term, the amount of fragmentation-induced FGR is expected to be small due to the following reasons:

- Amount of fragmentation-induced FGR due to balloon and rupture is insignificant due to minimum quantity of predicted core damage and burnup distribution of failed rods. Only a small quantity of fragmentation-induced FGR is possible from low burnup fuel rods due to (1) retention of fission gas within larger size fragments at low burnup and (2) limited quantity (moles) of fission gas at low burnup.

- Amount of fragmentation-induced FGR due to rapid power excursion expected to be limited due to minimum core damage. Figure 4-1 shows measured fragmentation-induced FGR as a function of increase in local fuel enthalpy. Examination of this figure reveals that fragmentation-induced FGR was reported on specimens exposed to an increase in local fuel enthalpy of greater than 60 cal/g. CRDA is a localized event. As such, only a small portion of the core experiences a rapid power excursion.

Furthermore, as described above, application of improved 3D analytical methods will greatly reduce predicted core damage and fuel enthalpy.

Dose calculations for plants which predict fuel damage are based on conservative source terms, including bounding quantities of fuel damage, maximum power peaking factors, and end-of-life exposure. Figure 6.4-2 illustrates the proximity of the existing BWR fleet to their respective allowable radiological consequences. While several plants report doses within 25% of allowable consequences, no significant increase in source term is expected as a result of fuel fragmentation-induced FGR and existing reported consequences remain bounding.

In conclusion, the amount of dispersed fuel particles and fragmentation-induced FGR are minimal during the BWR CRDA. All BWRs exhibit margin to allowable radiological consequences for the BWR CRDA.

Figure 6.4-1: Distribution of Predicted Fuel Failure

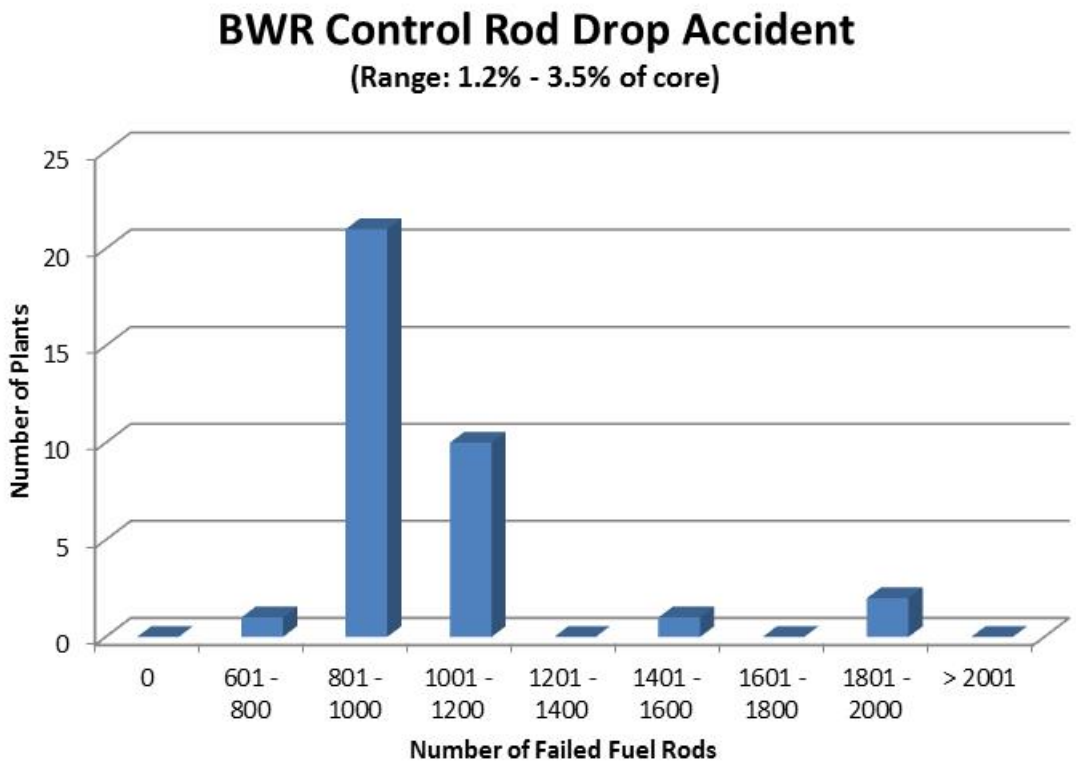
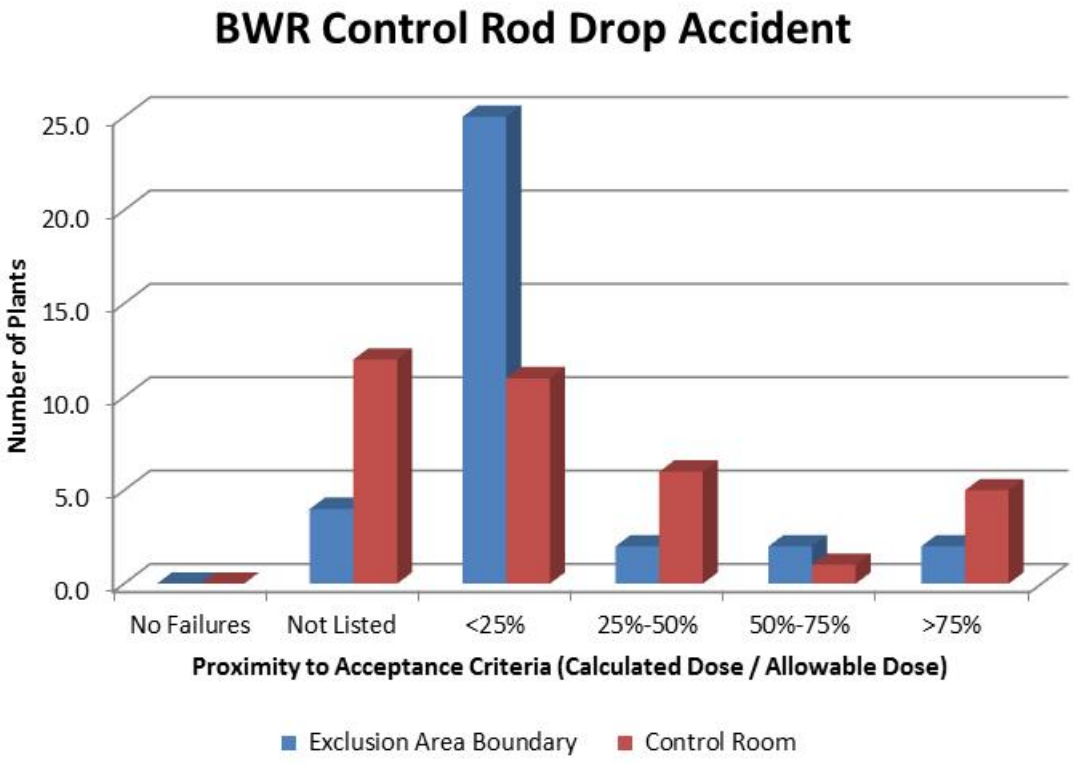


Figure 6.4-2: Proximity to Dose Limits



6.5 Fuel Handling Accident

A description of the Fuel Handling Accident (FHA) initiating event, inputs and assumptions, sequence of events, mitigating SSCs, and predicted consequences is provided in each licensee's UFSAR. The following event description was obtained from Chapter 15.7.4 of the Clinton UFSAR and is representative of many plants.

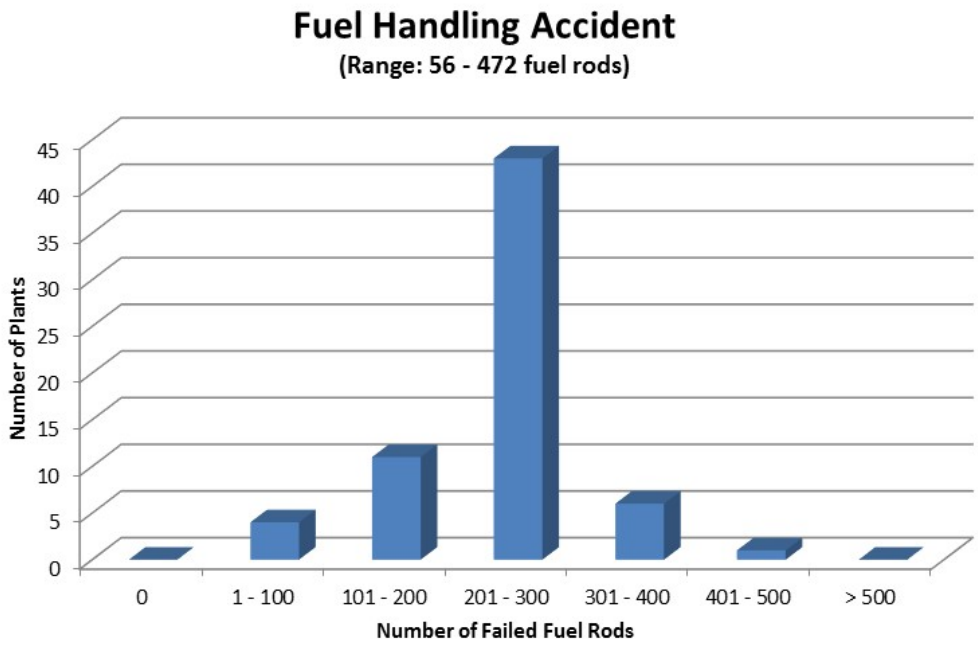
> The FHA is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism resulting in the dropping of a raised fuel assembly onto stored fuel bundles. A variety of events which qualify for the class of accidents termed "fuel handling accidents" has been investigated. These included considerations for containment upper pool refueling operations as well as fuel building-pool activities. The accident which produces the largest number of failed spent fuel rods is the drop of a spent fuel bundle onto the reactor core when the reactor vessel head is off.

Due to time restrictions associated with fuel movement, the FHA may only occur days after reactor shutdown. As such, there is no fuel or cladding temperature transient resulting from these accidents. The only postulated fuel fragmentation mechanism is the impact loads. No data on fragmentation-induced FGR exists for FHA prototypical conditions. As such, no assessment is provided in this report.

For completeness, Figure 6.5-1 shows the distribution in fuel failure used to assess radiological consequences for the FHA. Figure 6.5-2 illustrates the proximity of the existing fleet to their respective allowable radiological consequences.

Figure 6.5-1: Distribution of Predicted Fuel Failure

PWR



BWR

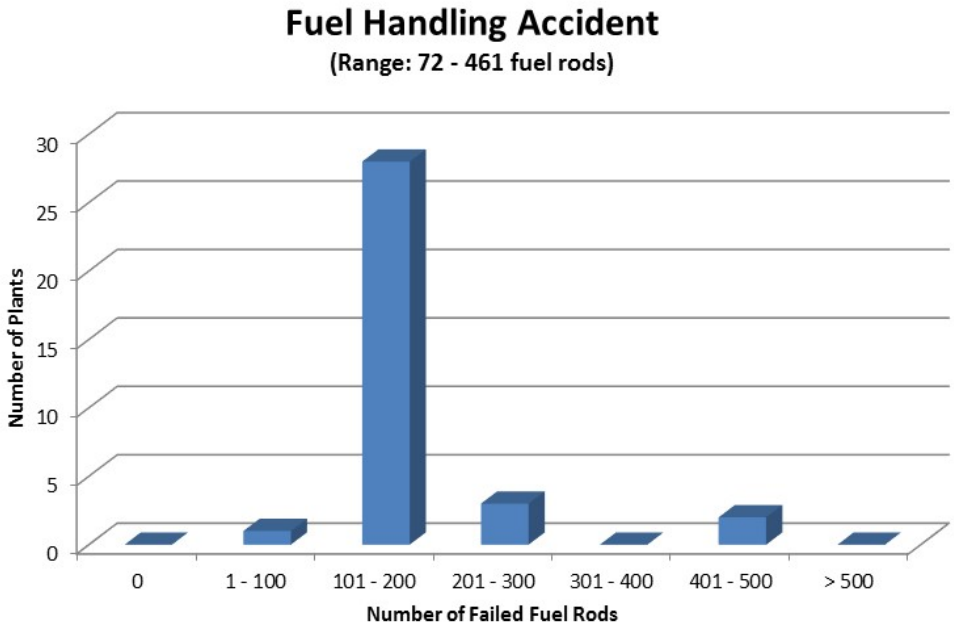
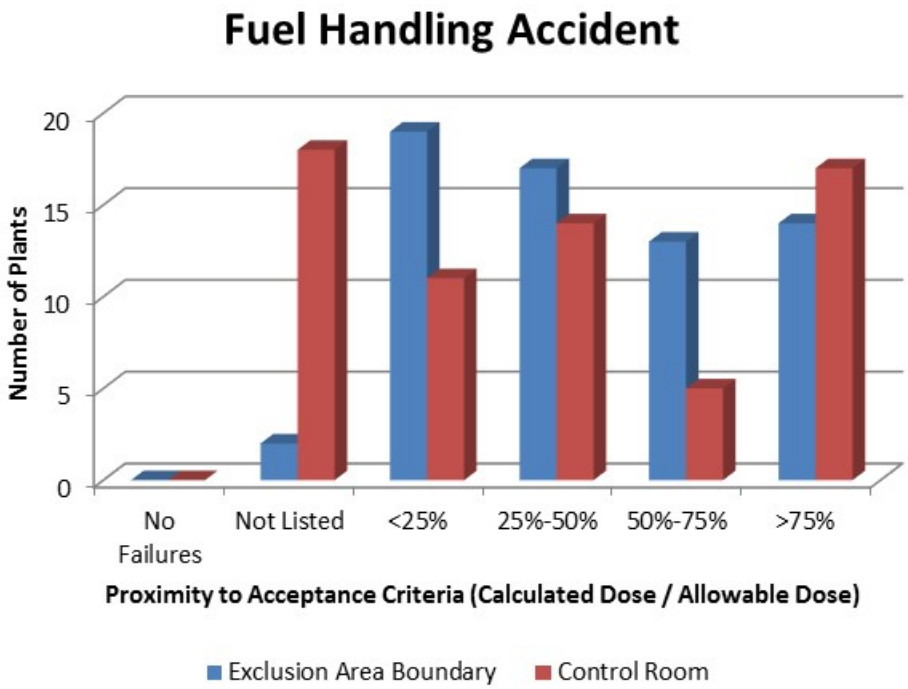
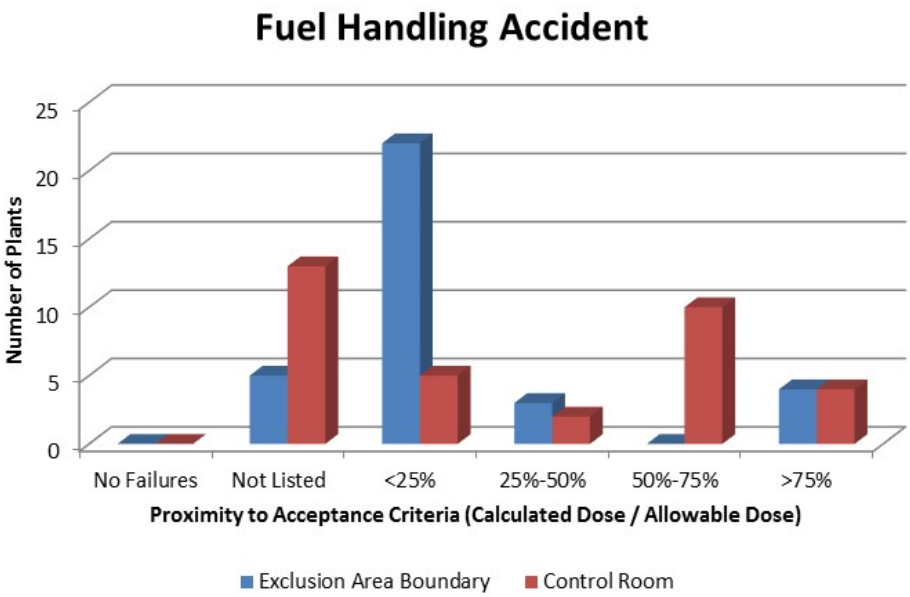


Figure 6.5-2: Proximity to Dose Limits

PWR



BWR



7.0 CONCLUSIONS

As described above, the population of failed fuel rods along with the portion of fuel volume within each of these failed rods susceptible to fragmentation is expected to be small. As a result, the impact of dispersed fuel on accident progression and mitigation is expected to be small. Similarly, the amount of fragmentation-induced FGR is expected to be minimal. Current radiological consequences are based on conservative source terms (i.e., estimates of failed fuel, burnup, and peaking) and remain bounding and within regulatory limits. As such, no imminent safety concern exists with respect to FFRD for non-LOCA DBAs for the existing fleet.

With respect to FHA, no data on fragmentation-induced FGR exists for prototypical accident conditions. As such, no assessment is provided in this report.

Appendix A:

LOCA Versus Non-LOCA DBAs

There is a common misconception that LOCA radiological consequences bound those associated with non-LOCA DBAs. The purpose of this appendix is to describe differences in the design and licensing bases between LOCA and non-LOCA DBAs in order to resolve this misconception.

The Maximum Hypothetical Accident (MHA) is the DBA with respect to containment building design (i.e., leakage, shielding) and reactor siting (10 CFR 100). The radiological consequences for the MHA are based on a conservative source term which assumes core wide damage and significant fuel melt (See Tables 1 and 2 of RG 1.183). For most licensees, there is no explicit LOCA radiological consequence calculated or reported. Instead, the MHA radiological consequences are reported to bound offsite doses for the LOCA. However, due to differences in design and licensing bases, the MHA radiological consequences do not necessarily bound non-LOCA DBA radiological consequences.

As described below, fuel fragmentation and dispersal under non-LOCA DBAs has the potential to alter the accident progression, the response of designed to mitigate the consequences, and the predicted radiological consequences.

1. The accident progression and engineered safety features designed to mitigate the consequences for the postulated LOCA and non-LOCA DBAs are different. The potential impact of fuel fragmentation and dispersal on the ability of these mitigating SSCs to perform their intended functions in a timely manner must be evaluated. With the possible exception of MSLB, the ECCS system is not engaged during non-LOCA DBAs. As described below, the “design basis” of these SSCs is their response to specific DBAs.

    a. PWR DBAs rely upon the rapid response of the RPS to initiate a control rod scram to limit the number of failed fuel rods. Specifically, the locked rotor/sheared shaft (LR/SS) analysis relies on the low shaft speed and low flow (SG ΔP) trip functions; MSLB relies on the low SG pressure, high containment pressure, and high power trip functions, and CRE relies on the high power trip function. If fuel fragmentation impacts FGR, then the timing or setpoints for these trip functions may need to be changed to maintain acceptable radiological consequences.

    b. PWR DBAs rely on the rapid response of primary safety valves (PSVs) and Main Steam Safety Valves to limit system pressure to within acceptable levels. If fuel dispersal impacts energy deposition into the RCS, then the setpoints for these safety valves may need to be changed to limit the pressure excursion.

    c. PWR DBAs rely upon Atmospheric Dump Valves (ADVs) to cool and depressurize the primary and secondary systems. If fuel dispersal impacts energy deposition into the RCS, then the timing and amount of ADV steaming (input to dose calculation) may change.

d. PWR DBAs rely upon the SDC for long-term decay heat removal. The ability of this system to cool dispersed fuel fragments and remove long-term decay heat is difficult to demonstrate.

2. Plants report explicit radiological consequences for both onsite and offsite locations for several different non-LOCA DBA. Unlike the treatment of LOCA, the plant's licensing basis does not specify that non-LOCA DBA radiological consequences are bounded by the MHA.

3. Allowable radiological consequences may differ between LOCA/MHA and non-LOCA DBAs. For example, CRE radiological consequences must be well within (25%) of 10 CFR Part 100 guidelines or 10 CFR 50.67 limits, as applicable. Whereas, LOCA/MHA radiological consequences are allowable up to 100% of these limits.

4. For the LOCA/MHA dose assessment, bounding core inventory release fractions (e.g., 100% release of nobles) are used, which are independent of the accident progression with respect to predicted fuel damage or FGR. In contrast, non-LOCA DBA dose assessments credit (1) the response of SSCs designed to mitigate the consequences of the accident, (2) a limited number of failed fuel rods, and (3) a limited release fraction (e.g., 10% release of nobles).

   a. Experimental data shows that pellet fragmentation may be associated with a release of trapped fission products. Additional transient FGR resulting from fuel fragmentation will have no impact on predicted LOCA/MHA dose consequences; however, may impact predicted consequences for non-LOCA DBAs.

   b. Fuel fragmentation and dispersal may alter the timing and rate of energy deposition into the reactor coolant which may result in cladding failure in adjacent fuel rods previously not expected to experience DNB (i.e., DNB propagation). Additional fuel rod failures would increase the accident source term for non-LOCA DBAs.

5. For the LOCA/MHA dose assessment, the primary release path for the core activity to the environment is via containment leakage following containment isolation. Plant TSs limit allowable containment leakage (e.g., 1% volume/day) such that the activity release is small (relative to core wide inventory). For non-LOCA DBAs, the primary release paths are (1) via SG leakage and through ADVs or outside containment steam line break for PWRs and (2) via the turbine and condenser for BWRs. Non-LOCA DBA dose calculations may be more limiting than LOCA/MHA even though the predicted overall RCS activity is smaller because of differences in release paths. Atmospheric dispersion factors and proximity to important features (e.g., control room ventilation) is also important for these different release paths.

6. For the PWR CRE and BWR CRDA, the integrity of the RCS pressure boundary is challenged by the large power pulse. GDC 28 (10 CFR 50 Appendix A) requires that the;

   > effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2)

sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core.

Fuel fragmentation and dispersal may alter the timing and rate of energy deposition into the reactor coolant which may further challenge the integrity of the pressure boundary. For the postulated LOCA, the integrity of the RCS pressure boundary is already compromised and therefore less of a concern.

7. Due to differences in core thermal-hydraulic conditions, the magnitude and timing of energy deposition due to fuel dispersal and associated FCI would be different between LOCA and non-LOCA DBAs. For the same reasons, fuel particle transport and deposition would also differ.