Screening Analysis for the Proposed Generic Issue on Dispersal of Fuel Particles during a Loss-of-Coolant Accident

SUMMARY

This report documents the screening analysis for the proposed generic issue (GI) related to dispersal of fuel particles from ruptured fuel rods during a loss-of-coolant accident (LOCA). The proposed GI was submitted to the Generic Issues Program on October 6, 2011, by the Division of Systems Analysis (DSA) in the Office of Nuclear Regulatory Research (RES) to evaluate the possibility of fuel dispersal into the core from ballooned and ruptured fuel rods during a LOCA and its potential adverse effects on accident progression and radiological activity levels. The Generic Issues Program staff completed the acceptance review of the proposed GI on October 21, 2011, and determined that the proposed GI, which was designated as PRO-GI-010, "Dispersal of Fuel Particles during a LOCA," warranted further processing in the screening stage of the Generic Issues Program. The acceptance review of the proposed GI is available under Agencywide Documents Access and Management System (ADAMS) Accession No. ML112910156.

The purpose of the screening stage is to assess proposed GIs against seven screening criteria described in U.S. Nuclear Regulatory Commission (NRC) Management Directive (MD) 6.4, "Generic Issues Program," dated November 17, 2009 [1]. Based on the assessment of the proposed GI against the seven screening criteria, the Generic Issues Program staff determines if the issue should proceed to the next stage or if the issue should exit the Generic Issues Program. A proposed GI would not be further processed under the Generic Issues Program if it does not meet all of these criteria. Because a graded approach is used as an issue is processed through each stage of the Generic Issues Program, the staff's effort in collecting information and assessing the proposed GI in the screening stage is more rigorous in comparison with the previous stage of acceptance review.

The Generic Issues Program staff completed the screening assessment for the proposed GI in accordance with MD 6.4. A GI Review Panel was formed for the proposed GI to review the screening analysis and make a recommendation regarding further processing in the Generic Issues Program. The GI Review Panel concluded that the proposed GI will not move on to the safety/risk assessment stage of the generic issue process because the proposed GI did not pass one of the seven GI screening criteria. Specifically, the proposed GI did not pass the third criterion which states, "The issue cannot be readily addressed through other regulatory programs and processes; existing regulations, policies, or guidance; or voluntary industry initiatives." The rulemaking process was initiated to address issues associated with separate LOCA phenomenon (i.e. embrittlement and breakaway oxidation) and will consider the need for rule language changes to address fuel fragmentation during a LOCA, and therefore the third criterion is no longer met.

Management Directive 6.4 states, "A proposed GI or a GI that does not meet any of these criteria at any time will not be processed further by the GIP." This issue no longer warrants further analysis under the Generic Issues Program. The recommendation for not processing further on the proposed issue will be submitted to the RES Office Director for endorsement.

The screening analysis for the proposed GI documented in this report provides a description of the issue and a discussion of the existing regulatory framework related to fuel fragmentation, relocation and dispersal. Moreover, the screening analysis includes a literature search, which reviews related experiments conducted at several test facilities. Finally, this report describes assessment of the proposed GI against seven screening criteria described in MD 6.4 [1] and recommends no further processing of the proposed GI.

BACKGROUND

Issue Description

During normal operation, oxide fuel pellets develop many cracks because of thermal stresses. In addition, the fuel pellet microstructure experiences significant evolution of its grain and pore structure and this evolution increases with burnup. During anticipated operational occurrences where fuel rod ballooning takes place, some fragmented fuel particles located above the ballooned region of a fuel rod may thus relocate into the enlarged volume of the balloon under the influence of gravity and pressure differences. This effect was first noticed in 1980 in reactor tests in the United States and Germany. More recent tests at Argonne National Laboratory (ANL), the Halden Reactor Project in Norway, and Studsvik Laboratory in Sweden have confirmed fuel fragmentation and relocation. The consequence of fuel relocation is an increase in heat generation in the ballooned region with corresponding increases in cladding temperature and oxidation compared with an undeformed length of the fuel rod.

Loss of fuel particles through the rupture opening in the ballooned region of a fuel rod was not expected based on any research performed before ANL experiments. When integral LOCA tests were completed at ANL on high-burnup boiling-water reactor (BWR) rods with a local burnup of 64 GWd/MTU, a small amount of fuel loss, about the quantity of one fuel pellet, was noticed. Because the amount of material was small, this observation was not thought to be important. However, in April 2006, a LOCA test, IFA-650.4, was run in the Halden reactor on a fuel rod segment with a very high local burnup of 91.5 GWd/MTU. Results from this test showed gross loss of fuel material from above the rupture opening. Online instrumentation indicated that this fuel loss occurred during the temperature transient rather than after the test was over. The pellet cracking, grain and pore structure evolutions coincident with high burnup may enhance the ability of fuel material to flow freely under the influence of gravity and pressure differences out of the rupture opening. Thus, the loss of large amounts of material in the Halden tests, as compared to the lower burnup fuel rods used in ANL tests, and the even lower burnup rods used in earlier German tests, may be consistent with the effects of high burnup operation on fuel material. As part of NRC's integral LOCA program, four integral LOCA tests were run at Studsvik Laboratory with high burnup fuel rods. These fuel rods had a rod average burnup near 70 GWd/MTU. In these tests, significant fuel loss was observed. The burnup of the rods tested at Studsvik is still above the current NRC burnup limit of 62 GWd/MTU, however, it is closer to the limit than that of the Halden experiments, and therefore generated greater concern than the findings of IFA-650.4. In addition, the fuel fragmentation size of the dispersed fuel could be readily observed in the Studsvik tests, and the fuel fragments were observed to be fine.

Potential consequences of fuel fragmentation, relocation and dispersal could include a change in axial linear heat generation and temperature profile, fuel coolant interaction, hydraulic and mechanical effects (i.e., flow blockage) of fuel material in the reactor coolant system, and

radiological consequences. Of all of these effects, flow blockage caused by rods ballooning in the same axial regions and change in axial linear heat generation and temperature profile may occur without rupture of the cladding; the other effects require rupture of the fuel rod cladding.

The staff of RES/DSA conducted a review of past research programs for observations related to the phenomenon of fuel fragmentation, relocation and dispersal [2]. The goal of this review was to determine whether these phenomena occur during a LOCA, and whether they were or should be incorporated into the criteria used to evaluate the acceptability of emergency core cooling systems (ECCSs). Trends or new observations identified by the review were used to characterize the likelihood of fuel fragmentation, fuel relocation and fuel dispersal under LOCA conditions. DSA staff concluded that fragmentation, relocation and dispersal of fuel could not be precluded as possible phenomena during a LOCA and submitted the issue of fuel dispersal as a proposed GI to the Generic Issues Program for further review.

Regulatory History

Historical information in this section has been selectively extracted from *The History of LOCA Embrittlement Criteria* [3] by Hache and Chung.

In 1967, a U.S. Atomic Energy Commission (AEC) Advisory Task Force on Power Reactor Emergency Cooling, appointed to provide additional assurance that substantial meltdown is prevented by core cooling systems, concluded that:

> *The analysis of (a LOCA) requires that the core be maintained in place and essentially intact to preserve the heat-transfer area and coolant-flow geometry. Without preservation of heat-transfer area and coolant-flow geometry, fuel-element melting and core disassembly would be expected…. Continuity of emergency core cooling must be maintained after termination of the temperature transient for an indefinite period until the heat generation decays to an insignificant level, or until disposition of the core is made [4].*

This rationale made it clear that it is important to preserve both the heat transfer area and the coolant flow geometry, not only during the short-term portion of the core temperature transient but also for long term.

Consistent with the conclusions of this Task Force, the AEC promulgated General Design Criteria [5] such that "fuel and clad damage that could interfere with continued effective core cooling is prevented." At the same time, the AEC also promulgated Interim Acceptance Criteria [6] for ECCS for Light Water Reactors (LWRs).

These criteria were subjected to a Rulemaking Hearing in 1973, and the proceedings were well documented in the Journal of Nuclear Safety in 1974 [7][8]. The peak cladding temperature (PCT) and maximum oxidation limits, now found in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46(b), "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors" [9] were established during this hearing process.

During the Rulemaking Hearing in 1973, the following remarks were made part of the formal record:

The purpose of these first two criteria [peak cladding temperature and maximum oxidation] is to ensure that the Zircaloy cladding would remain sufficiently intact to retain the UO_2 fuel pellets in their separate fuel rods and therefore remain in an easily coolable array. Conservative calculations indicate that during the postulated LOCA, the cladding of many of the fuel rods would swell and burst locally with a longitudinal split. The split cladding would remain in one piece if it were not too heavily oxidized, and would still restrain the UO_2 pellets [10].

In further discussion on "coolable geometry," the Atomic Energy Commission noted that:

If there were no emergency core cooling after a LOCA, the core would probably eventually fuse together into a large mass with insufficient external surface area to allow the fission product heat generated within it to be transferred away. Intermediate steps in arriving at such a state might be the oxidation and melting of the Zircaloy cladding, allowing the uranium dioxide fuel pellets to fall together into a heap that would be difficult to cool [10].

Examination of the historical record indicates that, at the time the cladding criteria were developed, there was no expectation of fuel loss – or at least no expectation of significant fuel loss – during a successfully mitigated LOCA.

By 1984, NRC had classified the effect of fuel relocation into the enlarged volume of the balloon as GI-92, "Fuel Crumbling during LOCA," [11][12] which was later given a low priority based on compensating conservatisms in Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" [13]. As explained in the next section of this document, GI-92 was dropped from further pursuit in 1998 [11].

In 2004, axial fuel relocation was the central issue in a hearing proceeding about the insertion of four lead test assemblies of mixed plutonium-uranium oxide (mixed-oxide) fuel in the Catawba Nuclear Station. The contention in the proceeding claimed that axial fuel relocation would be worse in mixed-oxide fuel than in standard uranium dioxide fuel. The staff successfully argued that there would be no significant difference; however, the testimony did not deny the importance of the axial fuel relocation itself.

In 2008, RES issued Research Information Letter, RIL-0801 [14], which discussed research findings in the area of high burnup fuel performance under postulated LOCA events and provided a technical basis for revising the LOCA cladding embrittlement criteria found in 10 CFR 50.46 [9]. With the results of Halden's IFA-650.4 available, axial fuel relocation and the loss of fuel particles through a rupture opening were discussed. The RIL noted that additional integral testing was being performed and suggested these tests should provide more definitive information on fuel loss during a LOCA with high-burnup fuel. Regarding the issue of fuel particles being dispersed through a rupture opening in the fuel cladding, RIL-0801 [14] concluded that the current NRC burnup limit of 62 GWd/MTU (average for the peak rod) was "probably low enough to prevent significant fuel loss during a LOCA."

Related Generic Issues

Experiments conducted at several test facilities before 1984 showed that irradiated fuel could fragment (crumble) into small pieces during a LOCA and may relocate axially and set into ballooned regions. Results of these experiments are discussed in the next section of this report. In 1984, NRC classified this effect as GI-92 [11][12]. Some evaluation of this effect was made for NRC by EG&G [12]. In the evaluation of this effect, the focus was on the consequence of an increase in heat generation in the ballooned region with corresponding increase in cladding temperature and oxidation compared with an unballooned length of the fuel rod.

At the time of the initial evaluation of this issue in July 1984, the existing ECCS performance analysis codes did not account for fuel settling into ballooned regions. Thus, the lack of inclusion of this effect was non-conservative. However, the EG&G study [12] concluded that known conservatisms in Appendix K [13] would more than offset this effect, and the issue was therefore given a low priority. This determination meant that there was insufficient risk-based justification for starting a major reconsideration of existing ECCS Appendix K performance analyses.

By the early 1990's, there were ongoing efforts to develop and license ECCS performance models that were more realistic (and consequently less conservative) than the models in use at the time of the evaluation in July 1984. As summarized in NUREG-0933, "Resolution of Generic Safety Issues," [11] with the move towards best estimate models, it was no longer valid to conclude that the effects of fuel crumbling and settling into ballooned regions could necessarily be neglected in any new, more realistic models. Instead, it was expected that these effects would be appropriately addressed in the calculations. Moreover, it was determined that a separate generic issue on fuel crumbling was not necessary; such work was best done within the scope of the review of the new analytical methodology. Thus, the issue was given a low priority (see Appendix C of NUREG-0933 [11]). In NUREG/CR-5382 [15], it was concluded that consideration of a 20-year license renewal period did not change this priority. Further prioritization in 1998, using the conversion factor of $2,000/man-rem approved by the Commission in September 1995, resulted in an impact/value ratio (R) of $50,000/man-rem, which placed the issue in the "drop" category.

On the topic of fuel relocation, RIL-0801 noted this history of GI-92, and stated:

> After the best estimate option was added to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," those conservatisms were no longer guaranteed and fuel relocation was again prioritized. This time it was classified as "drop," but the evaluation that led to this classification may not have been adequate [11][16]. That evaluation appears to have overlooked the possibility that rapid cladding embrittlement would occur at the assumed cladding temperature of 1427°C (2600 °F) and that embrittled fuel rods might collapse. NRC and Halden programs are performing additional testing to resolve this issue and the resolution of GI-92 will be documented when the testing is completed.

Summary of Experimental Results

To assess the risk and safety significance of fuel dispersal during a LOCA, the Generic Issues Program staff reviewed a wide range of historical data, gathered by RES staff, from eight

different research programs performed over the last 35 years [2]. This section provides a short description of each experimental program. Moreover, a summary of findings for each experiment related to the risk of fuel dispersal is presented. Although the proposed GI does not directly raise the issue of fuel fragmentation and relocation and its consequences, these phenomena are also discussed where data is available. The proposed GI was limited to fuel dispersal and its consequences because fuel fragmentation and relocation were previously known. The phenomenon of fuel relocation was the subject of a previous GI.

1) Power Burst Facility at the Idaho National Energy Laboratory

Four transient tests were conducted in the Power Burst Facility (PBF) at the Idaho National Energy Laboratory (INEL) between 1978 and 1983 [17][18][19]. The fuel design was typical pressurized-water reactor (PWR) 15x15 array rods with Zircaloy-4 cladding but 9.6% enriched UO_2 fuel, which is higher than the 5% limit currently used for LWR fuel in the U.S. The PBF experimental program was designed to investigate rod ballooning and failure in the event of a LOCA, thus the main test parameters were burnup and rod fill pressure. A number of tests were designed to cause rupture at different temperatures. From the review of data obtained from PBF experiments that included tests ranging in burnup from 0 to 17.66 GWd/MTU, staff observed the phenomena of fuel fragmentation and relocation in a number of experiments, which appeared to increase with the fuel burnup. Fuel loss was observed in one instance, LOC-3 rod 2 (burnup of 15.96 GWd/MTU). A combination of relocation and burnup resulted in fuel melting near the centerline in that instance. For this instance, the magnitude of fuel loss or dimensions of the rod rupture were not reported.

2) FR-2 – Karlsruhe

Concurrently with the PBF test program in the United States, the FR-2 program was conducted in Germany by the Karlsruhe Institute of Technology (KfK) [20][21][22]. A total of 39 fueled rods were tested in the FR-2 reactor between 1978 and 1983. PWR fuel rods with UO_2 fuel enriched to 4.9% and clad with Zircaloy-4 were re-fabricated into test rods about 1 meter in length, with 50 cm active fuel length. The initial re-fabricated rod fill pressure was 43.5 psi, but the rods were irradiated to different burnups (from 0 to 35 GWd/MTU) resulting in a wide range of rod internal pressures at the beginning of the LOCA transient. The staff concluded from the FR-2 experiments that fuel fragmentation occurred early in fuel rod life, and tended to increase with the fuel burnup. Additionally, extensive fuel relocation occurred in ballooned rods where the diametral strain exceeded 8%. Although fuel dispersal might have occurred when fuel fragments were smaller than the opening in ruptured rods, no fuel loss was observed in these experiments.

3) National Research Universal - Pacific Northwest Laboratory

The research program conducted by Pacific Northwest Laboratory (PNL) at the National Research Universal (NRU) reactor in Canada was the first series of tests performed on a bundle of rods instead of a single rod [23][24][25][26]. The objective of these tests was to study the thermal-hydraulic and thermo-mechanical behavior (deformation, flow blockage) within a bundle of 32 full-length PWR rods. The test rods were fueled with fresh UO_2 pellets but were preconditioned prior to the LOCA transients by power cycling to full power, which caused the fuel pellets to fracture. The power cycling to full power would simulate the pellet fracturing that occurs very early in life. Four thermo-mechanical tests were conducted and labeled MT-1 through MT-4. The test sequence consisted of a hold at 375°C in steam, followed by cutting off steam to generate the LOCA transient, then quench and scram. The test rods all went through a ballooning phase, and 6/11, 8/11, 12/12 and 8/11 rods ruptured in tests MT-1, MT-2, MT-3 and MT-4 respectively. The staff concluded from the PNL/NRU experiments that in a bundle of rods, ballooning appeared to occur such that all the balloons were co-planar. Furthermore, ballooning was mostly prevented in the length of the fuel rod beneath a grid spacer. In other words, grid spacers appeared to “pin” rod ballooning, potentially limiting fuel relocation to the

span between grid spacers. Fuel dispersal was not documented for test MT-2, although a possibility exists that dispersal occurred. MT-4 showed extensively fragmented pellets and large rupture openings. For MT-4 test, the ruptured region of some rods was empty, likely indicating the fuel dispersal through the rupture opening.

4) PHEBUS-LOCA – IRSN (France)

The PHEBUS-LOCA program performed by IRSN consists of an in-pile LOCA program with integral tests on Zircaloy-4 clad PWR fuel rods in a 5 by 5 bundle configuration [27][28][29][30]. Starting from an initial state representative of PWR nominal conditions, the test transient aimed at reproducing a hypothetical large break LOCA transient from initial blowdown until the final quench of the test device. The test train consisted of a bundle of 25 5x5 PWR type rods containing fresh UO_2 fuel, maintained by 4 Inconel spacer grids. The PHEBUS-LOCA experimental program was conducted from 1980 to 1984. The staff concluded from the PHEBUS-LOCA experiments that axial relocation of very low burnup fuel, which showed minimal fragmentation, was unlikely, even in the event of relatively large cladding diametral strains. Radial fuel relocation may nonetheless occur under these conditions. No fuel loss was observed in these experiments.

5) FLASH Tests –CEA (France)

Five tests were carried out in the FLASH facility of the SILOE reactor at CEA/Grenoble [31][32]. The main purpose of the tests was to study the fission products release during a LOCA transient. These tests were performed on a pre-conditioned fresh fuel rod and a high burnup fuel rod. The PWR 17x17 type test rods with an active length of 30 cm were centered in an unheated shroud tube located on the reactor periphery, which induced large azimuthal temperature differences. The staff concluded from the results of the FLASH experiments that axial fuel relocation of very low burnup pre-conditioned fuel is unlikely even for large cladding diametral strains. However, axial and radial fuel relocation can occur for diametral strains as low as 16% and burnups as low as 35 GWd/MTU even with very small rupture openings that mostly prevent fuel dispersal. No fuel loss was observed in these experiments.

6) Argonne National Laboratory LOCA Test Program

Four single-rod LOCA tests were performed at ANL using high burnup BWR rods (56 GWd/MTU rod average, 64 GWd/MTU local burnup) [33]. These tests used external heating of the rods in a steam environment. The temperature was ramped to 1200°C, then held for a pre-determined period of time, and ramped down to 800°C, at which point quench occurred. The ANL program was focused on the details of cladding oxidation, hydriding, and ductility, rather than on the fuel behavior. Furthermore, methods used to “freeze” the fuel particles in place (e.g., epoxy) to characterize the cladding prevented characterization of fuel particle size distribution. The staff concluded from the ANL experiments that fine fuel fragmentation is likely to facilitate fuel axial relocation, as well as fuel dispersal in the form of fine particles that can easily migrate to the rupture opening and escape into the coolant. Although it was evident that a fine dust of fuel particles were expelled from the rods during the ramp to 1200°C, the amount of fuel dispersal was not measured in the ANL LOCA test.

7) Halden Boiling Water Reactor LOCA Test Series

The Organization for Economic Co-operation and Development (OECD) Halden Reactor Project is a series of in-pile experiments performed in the Halden Boiling Water Reactor (HBWR), a heavy-water cooled and moderated reactor. Two LOCA programs were run in the Halden reactor in the early 1980s [34][35]. These programs were mostly aimed at the study of thermal-hydraulic and thermo-mechanical phenomena, respectively. No information was found regarding the phenomena of fragmentation, relocation or dispersal of fuel. The ongoing Halden experiments (2003 - present) are single pin tests and focus on effects that are different from those studied in out-of-reactor tests. A prototypical bounding LOCA transient does not exist, and Halden project participants recommended that the test conditions be selected to meet the following two primary objectives: (1) Maximize the balloon size to promote fuel relocation, and to evaluate its possible effect on cladding temperature and oxidation, and (2) Investigate the extent (if any) of "secondary transient hydriding at high temperature" on the inner side of the cladding around the rupture region in presence of pellet-cladding bonding layer.

The new Halden experiments were deemed necessary because industry trends to high burnup fuel design and introduction of new cladding materials have generated a need to re-examine and verify the validity of the safety criteria for loss of coolant accidents. High burnup fuel rods irradiated in commercial reactors have been used in this series of tests. To date, 13 LOCA tests have been conducted in HBWR [36][37][38][39][40][41][42][43][44][45][46].

The phenomenon of fuel fragmentation is consistently noted in the Halden LOCA tests. The degree of fragmentation (the resulting particle size) is not uniform. These experiments show that fuel fragmentation is more pronounced in some regions of the fuel (e.g., the rim region) than others, and in those cases, both large and small particles may result. Moreover, a small rupture opening may be sufficient to prevent significant fuel dispersal, but smaller fuel particles may still escape. The lack of cladding restraint and the presence of a large rupture opening are sufficient to result in extensive dispersal of fuel pellet fragments – particularly in high burnup fuel. Finally, staff concluded that fuel fragmentation and relocation is not limited to the rim region.

The Nuclear Energy Agency Working Group on Fuel Safety (WGFS) was asked by the Committee on the Safety of Nuclear Installations (CSNI) to evaluate how the Halden LOCA tests affect regulation. The summary report of the WGFS members' evaluation concludes that "[w]hile the tests have clearly identified the [fuel relocation and dispersal] phenomen[a], they also raise questions. The recommendations therefore include gaining more insight through more experimental work, in particular to perform test on fuel segments with burnup representative of bounding industrial situations." The report further states that "[a]t the current burnup limit licensed in the US (62MWd/kgU rod average), it is believed that any fuel dispersal would be minimal. Fuel relocation and dispersal will need to be addressed prior [to] approving increases in licensed burnup." [47][47]

8) Studsvik LOCA Test Program

The LOCA program at Studsvik recently completed 6 single-rod integral LOCA tests using the same overall procedures used in the ANL LOCA program. The first four integral LOCA test segments were taken from irradiated rods with a rod average burnup of ≈70 GWD/MTU and the last two segments were taken from irradiated rods with a rod average burnup of ≈55 GWD/MTU. In each of these tests, a pressurized, high burnup, fueled rod segment was subjected to a

temperature transient in a steam environment through ballooning and rupture. The first and fifth tests were terminated just after rupture, and the other tests each experienced some degree of high temperature oxidation, followed by quench. In the first four tests at Studsvik, significant fuel loss through the rupture opening was observed during the LOCA transient and additional

fuel was lost when the fuel rod was "shaken" in a later part of the experiment[1]. In the last two tests, no fuel loss through the rupture opening was observed during the LOCA transient, however significant fuel loss was observed when the severed rod was "shaken" in a later part of the experiment (It should be noted that the rupture opening in the last two tests was significantly smaller than those of the first four test, and the rupture openings in the last two tests were smaller than the measured fuel fragment size of those tests). In the first four tests, the fuel was finely fragmented and all fragments measured less than four millimeters. The mass of fuel fragments from these tests was approximately evenly distributed between the size groups separated by the six sieves used. In contrast, the fuel in the last two tests was not finely fragmented and the fragments measured predominately larger than 4 millimeters.

The staff concluded from the Studsvik LOCA tests that very fine fuel fragmentation can occur at high burnups around 70 GWd/MTU, such that a significant portion of the fuel resembles a fine powder. Under conditions of extensive fragmentation of fuel, relocation can occur for diametral strains as low as 5%. Fuel relocation was generally observed for diametral stains in the 4-12% range. As mentioned earlier, fuels that demonstrated significant fuel dispersal during these experiments have burnups greater than the licensed limits; hence, the significant loss of fuel in the Studsvik LOCA test program does not necessarily represent the fuel dispersal phenomena for fuel burnup under the licensed limit.

SCREENING ANALYSIS

The Generic Issues Program staff reviewed the proposed GI against seven screening criteria described in MD 6.4. The proposed GI is evaluated in the context of design basis accidents. The discussion under each criterion below provides the screening analysis for the proposed GI.

1. The issue affects public health and safety, the common defense and security, or the environment.

The purpose of this criterion is to eliminate issues not directly involving or affecting safety or security. Figures A1 through A3 in the RES Office Instruction TEC-002, "Procedures for Processing Generic Issues," dated September 29, 2010 [48] provides guidance based on risk regarding whether a reactor issue should continue in the program or exit it. These figures are derived from the criteria for assessing safety enhancement issues by addressing the safety goal analysis as described in Section 3 of NUREG/BR-0058, Rev. 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission." [49] These figures may be applied at anytime

[1] Following the LOCA simulation in the experiments at Studsvik each fuel rod was subjected to a four-point bend test (4PBT), to measure the residual mechanical behavior of the ballooned and ruptured region. After the 4PBT, a shake test was performed to determine the mobility of fuel particles that remained in the fuel rod. The shake test consisted of an inversion of the two halves of the broken fuel rod followed by minor shaking to dislodge any loose fuel particles. This test was conducted approximately two days after the LOCA simulation.

during the Generic Issues Program process if the information is known. For some issues, it may not be possible to use probabilistic risk assessment methods and tools either because of their limitations or because they may not have been developed. In cases where probabilistic tools and methods are not useful, the decision to accept the issue in the Generic Issues Program is generally based on more qualitative elements linked to NRC's strategic plan and expert judgment. In general, only those issues that represent credible threats to NRC's strategic and performance goals and measures, unless current regulatory programs are changed, meet this criterion.
The main safety concern of the proposed GI is the potential for loss of coolable geometry due to fuel dispersal occurring at temperatures and oxidation levels much lower than limited by safety criteria. Furthermore, the proposed GI could potentially affect public health and safety if fuel dispersal is found to adversely impact the calculated radiological consequences of a LOCA or to aggravate accident progression. Although many of the experiments explained earlier were deliberately designed and carried out with conditions that would emphasize the occurrence of certain phenomena, concerns raised by the fact that fuel dispersal could occur upon ballooning and burst at cladding temperatures as low as 800°C and thus lower than the temperature limited by the current limit should be addressed.

While experiments described above provide valuable information regarding the phenomena of fuel fragmentation, relocation and dispersal, a number of questions need to be answered to assess the risk of fuel dispersal quantitatively. These questions include the likelihood of fuel dispersal for lower burnup fuel, parameters and variables that affect this likelihood, quantity of the dispersed fuel and effects of dispersed fuel on safety equipment necessary to mitigate the accident. Therefore, probabilistic risk assessment methods cannot be used to quantify the risk with available information. Nevertheless, the increase in the large early release frequency or core damage frequency will not be significant because of a very low likelihood of the occurrence of conditions that lead to fuel dispersal, such as LOCA with at least partial core uncovery in combination with rod internal pressures and PCT, resulting in cladding rupture, sufficient cladding diametral expansion (ballooning) to allow fuel relocation and fuel fragments smaller than the cladding rupture opening.

Regardless of the probabilistic risk numbers, in the event of fuel dispersal during a LOCA, two of the three barriers to prevent the release of radioactive material, namely the fuel cladding and the reactor vessel, would be lost and release of physical fuel material and threat to coolable geometry could challenge the third barrier (containment) due to an increased pressure and heat generation. Loss of these barriers could present a credible threat to NRC's performance goals and measures. Therefore, the proposed GI meets this criterion.

2. The issue applies to two or more facilities and/or licensees/certificate holders or holders of other regulatory approvals.

The purpose of this criterion is to eliminate site-specific issues that are handled under other NRC processes. Because all reactors in the US fleet must consider LOCA in their design basis, the issue of fuel dispersal during LOCA could affect all reactors. Therefore, the proposed issue meets this criterion.

3. The issue cannot be readily addressed through other regulatory programs and processes; existing regulations, policies, or guidance; or voluntary industry initiatives.

The Generic Issues Program facilitates the staff's identification of an efficient mechanism for addressing a regulatory issue. Once another mechanism (regulatory program or process or voluntary industry initiative) to address the issue is identified and agreed upon by the appropriate regulatory office, the issue is transferred to the regulatory office and may exit the Generic Issues Program.

Fuel dispersal during a LOCA will be considered during rulemaking for the proposed 10 CFR 50.46c. If it is determined that fuel dispersal is not appropriate to be addressed by the proposed new rule, the staff will address any alternative regulatory actions needed in the SECY for the rulemaking, as directed by the Staff Requirements Memo (SRM) SECY-12-0034. Therefore, the proposed GI does not meet this criterion.

4. The issue can be resolved by new or revised regulation, policy, or guidance.

The Generic Issues Program is intended to provide a way to "fix" identified potential weaknesses and deficiencies in existing safety requirements and guidance. An issue exits the program when a fix is identified and agreed upon that is within the staff's ability to implement (e.g., guidance change). An issue also exits the Generic Issues Program if the staff determines that either no change is needed or that a change cannot be justified under backfit provisions. The proposed GI has been communicated to the Office of Nuclear Reactor Regulation and Office of New Reactors. In addition, the staff concluded that the consequences of fuel fragmentation and dispersal are not likely to result in an imminent safety hazard. This conclusion was made in consideration of the anticipated low number of fuel rods expected to burst and the conservative manner in which the radiological consequences for a postulated LOCA are calculated. The proposed GI may be resolved by corrective actions such as demonstrating that the consequences of fuel dispersal are not safety significant, preventing fuel rod cladding ruptures during a LOCA, or demonstrating that current analysis assumptions are bounding for fuel dispersal. These corrective actions could be implemented by new or revised regulation, policy or guidance. Therefore, the proposed GI meets this criterion.

5. The issue's risk or safety significance can be adequately determined (i.e., it does not involve phenomena or other uncertainties that would require long-term studies and/or experimental research to establish the risk or safety significance).

The purpose of this criterion is to eliminate those issues requiring long-term studies. Current state of understanding for this issue is that fuel fragmentation is a recognized phenomenon that may be related to a variety of factors, including burnup, fuel design, and transient history. Research findings suggest that higher fuel burnup may promote smaller fragmentation size, which would lead to a higher degree of transportability of fuel fragments. The findings also suggest that higher rod internal pressure may promote lower rupture temperatures and larger rupture openings, which in turn would lead to a larger population of ruptured fuel rods susceptible to fuel dispersal. Both rod average burnup and rod internal pressure are significantly higher today than 20 years ago.

Recent experimental tests using fuel of 70 GWd/MTU burnup (near the current NRC burnup limit of 62 GWd/MTU) showed significant fuel loss. The transition from small fuel loss to significant fuel loss around this range and its sensitivities are unclear, but is reasonable to assume that larger amounts of fuel dispersal would likely have a greater negative impact on the ECCS than lower amounts of fuel loss. Although the current 10 CFR 50.46 and proposed draft 10 CFR 50.46c rule recognize and accommodate fuel rod rupture by limiting peak cladding temperature and oxidation to reduce further degradation within the rupture region, neither the current nor proposed regulation will prevent ballooning and rupture and therefore may not guarantee the retention of fragmented fuel within the fuel rod. Therefore, rulemaking will consider the need for rule language changes to address fuel dispersal in the proposed 10 CFR 50.46c rule.

As described in the “Background” section of this document, the potential consequences of concern include:

a) A change in the axial linear heat generation and temperature profile
b) Fuel coolant interaction
c) Hydraulic and mechanical effects (i.e., flow blockage) of fuel material in the RCS
d) Radiological consequences (i.e., to operators) given the increased radiation levels

In NUREG-2121, “Fuel Fragmentation, Relocation, and Dispersal during the Loss-of-Coolant Accident,” dated March 31, 2012 (ADAMS Accession No. ML12090A018), the RES staff concluded that the consequences of fuel fragmentation and dispersal are not likely to result in an imminent safety hazard at the current limits. Nevertheless, the staff also concluded in this report that more research and detailed analyses are required to understand the sensitivities of these phenomena and their potential significance. Moreover, it appears that industry is intent on increasing burnup limits and licensing new fuel types, which may exacerbate this problem, if approved.

Although the staff, in conducting the Safety/Risk Assessment stage of the Generic Issues Program, may identify additional alternative approaches for determining safety significance, the GI Review Panel considers the following approaches as acceptable means to determine the significance of the above potential consequences, and recommends due consideration of these approaches to preclude additional long-term research in determining safety/risk significance:.

a) Bounding calculations to determine if regulatory requirements are met for design basis accidents
b) Reference to previous staff technical work (e.g., GI-191 results that ECCS sump screens are properly sized to prevent particles that could interfere with ECCS operation)
c) Realistic calculations (e.g., estimates of radiological conditions to demonstrate that radiation protection measures are capable of allowing access for operators during design basis accidents).

NRC regulatory research may be needed to understand the relationship and degree of influence of the various potential factors on the likelihood and severity of this phenomenon as a result of these calculations or for cases of higher burnups or changing fuel designs. The GI Review Panel found sufficient information to determine that the safety significance of the issue should be further explored in the next stage of the Generic Issues Program, considering approaches mentioned above. Therefore, the proposed GI meets this criterion.

6. The issue is well defined, discrete, and technical.

The purpose of this criterion is to keep the scope of a GI from inadvertently growing and to ensure that matters extraneous to the issue are excluded such that the issue remains manageable. In general, related issues or topics may be proposed as a single GI and will likely be separated and undergo individual processing and screening in the Generic Issues Program. The proposed GI has been clearly identified and explained in the submittal to the Generic Issues Program and in this document. Moreover, the proposed GI has been discussed in reports such as OECD NEA/CSNI/R(2010)5, “Safety Significance of the HALDEN IFA-650 LOCA Test Results.” [47] Therefore, the proposed GI meets this criterion.

7. Resolution of the issue may potentially involve review, analysis, or action by the affected licensees, certificate holders, or holders of other regulatory approvals.

The criterion is in keeping with the intent of the Generic Issues Program to address potential weaknesses and deficiencies in existing regulations and guidance affecting safety and security. If it becomes apparent that no licensee action will be needed, then further assessment under the Generic Issues Program is not needed and the issue exits the Generic Issues Program. For the proposed GI, licensees may be asked to take into account fuel dispersal in LOCA safety analyses or demonstrate that the consequences of fuel dispersal are not safety significant. Therefore, the proposed GI meets this criterion.

CONCLUSION

The GI Review Panel concludes that the proposed GI on dispersal of fuel particles during a LOCA does not meet the criteria specified in MD 6.4 and the issue does not warrant further analysis under the NRC Generic Issues Program. It will be addressed as part of rulemaking for the proposed 10 CFR 50.46c. If it is determined by NRR staff that fuel dispersal is not appropriate to be addressed by the proposed new rule, the NRR staff will document the basis in their rulemaking package responding to the Commission's SRM and return the issue to the Generic Issues Program.

REFERENCES

[1] Management Directive 6.4, "Generic Issues Program," November 17, 2009.

[2] NUREG-2121, "Fuel Fragmentation, Relocation, and Dispersal During the Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, March 2012, ML12090A018

[3] G. Hache and H. M. Chung, "The History of LOCA Embrittlement Criteria", Proceedings of the Twenty-Eighth Water Reactor Safety Information Meeting, Bethesda, Maryland, USA, NUREG/CP-0172, pp. 205-237, October 23-25, 2000. USNRC ADAMS Accession Number ML011370559.

[4] "Report of Advisory Task Force on Power Reactor Emergency Cooling", TID-24226, 1967.

[5] "General Design Criteria for Nuclear Power Plants", U.S. Code of Federal Regulations, Title 10, Part 50, Appendix A, 20 February 1971 (amended).

[6] "Interim Acceptance Criteria for Emergency Core-Cooling Systems for Light-Water Power Reactors", U.S. Federal Register 36 (125), pp. 12247-12250, 29 June 1971.

[7] W.B. Cottrell, "ECCS Rule-Making Hearing", Journal of Nuclear Safety, Volume 15, pp. 30-55 (1974).

[8] "New Acceptance Criteria for Emergency Core-Cooling Systems of Light-Water-Cooled Nuclear Power Reactors", Journal of Nuclear Safety, Volume 15, pp. 173-184 (1974).

[9] “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, U.S. Code of Federal Regulations”, Title 10, Part 50, Section 46, January 1974 (amended).

[10] Atomic Energy Commission Rule-Making Hearing, Opinion of the Commission, Docket RM-50-1, 28 December 1973.

[11] NUREG-0933, “Resolution of Generic Safety Issues,” U.S. Nuclear Regulatory Commission.

[12] Memorandum for T. Speis from R. Mattson, "Fuel Crumbling During LOCA," February 2, 1983.

[13] Appendix K, “ECCS Evaluation Models,” to U.S. Code of Federal Regulations, Title 10, Part 50, “Domestic Licensing of Production and Utilization Facilities.”

[14] U.S. Nuclear Regulatory Commission Research Information Letter 0801, “Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46”, Washington DC, May 30, 2008. USNRC ADAMS Accession Number ML081350225.

[15] NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," U.S. Nuclear Regulatory Commission, December 1991. USNRC ADAMS Accession Number ML072500168.

[16] R. Meyer, NRC memorandum to J. Flack, “Update on Generic Issue 92: Fuel Crumbling During LOCA,” February 8, 2001, USNRC ADAMS Accession no. ML010390163.

[17] T.F. Cook, “An Evaluation of Fuel Rod Behavior during Test LOC-11”, NUREG/CR-0590, March 1979.

[18] J.M. Broughton et al., “PBF LOCA Test Series, Test LOC-3 and LOC-5 Fuel Behavior Report”, NUREG/CR-2073, June 1981.

[19] J.M. Broughton et al., “PBF LOCA Test LOC-6 Fuel Behavior Report”, NUREG/CR-3184, April 1983.

[20] E.H. Karb et al., “LWR Fuel Rod Behavior in the FR2 in-Pile Tests Simulating the Heatup Phase of a LOCA”, KfK-3346, March 1983, ISSN 0303-4003.

[21] E.H. Karb et al., “KfK in-Pile Tests on LWR Fuel Rod Behavior during the Heatup Phase of a LOCA”, KfK-3028, October 1980, ISSN 0303-4003.

[22] L.J. Siefken, 7th SMIRT, vol. C, paper 2, Chicago, August 1983.

[23] G.E. Russcher et al., “LOCA Simulation in the NRU Reactor: Materials Test-1”, NUREG/CR-2152, October 1981. USNRC ADAMS Accession Number ML083360603.

[24] J.O. Barner et al., “Materials Test-2 LOCA Simulation in the NRU Reactor”, NUREG/CR-2509, March 1982. USNRC ADAMS Accession Number ML101970158.

[25] C.L. Mohr et al., “LOCA Simulation in the National Research Universal Reactor Program: Data Report for the Third Materials Experiment (MT-3)”, NUREG/CR-2528, April 1983. USNRC ADAMS Accession Number ML083360587.

[26] C.L. Wilson et al., “LOCA Simulation in NRU Program: Data Report for the Fourth Materials Experiment (MT-4)”, NUREG/CR-3272, July 1983. USNRC ADAMS Accession Number ML101960140.

[27] B. Adroguier, “Synthèse de l’Essai PHEBUS 215 P”, Rapport Technique IPSN/DERS/SAEREL-2/84, July 1984.

[28] E. Scott de Martinville et al., “Interprétation de l’Essai 215 R”, Note Technique SAEREL-176/87, Note PHEBUS-77/87, October 1987.

[29] Drosik et al, “Interprétation de l’Essai PHEBUS 218”, Note Technique SAEREL-154/87, Note PHEBUS-86/88, June 1988.

[30] H. Rigat et al, “Interprétation de l’Essai PHEBUS 219”, Note Technique SEMAR-04/88, Note PHEBUS-87/88, May 1988.

[31] M. Bruet et al, “FLASH Experiments in SILOE Reactor: Fuel Rod Behavior During LOCA Tests”, OECD-CSNI/NEA Experts Meeting on Water Reactor Fuel Safety and Fission Product Release in Off-Normal and Accident Conditions”, Riso, Denmark, 16-20 May 1983

[32] M. Bruet et al., “High Burnup Fuel Behavior During a LOCA Type Accident: the FLASH-5 Experiment”, IAEA Technical Committee Meeting on Behavior of Core Material and Fission Products Release in Accident Conditions in LWRs, Cadarache, France, 16-20 March 1982.

[33] M. Billone et al., “Cladding Embrittlement During Postulated Loss-of-Coolant Accidents”, NUREG/CR-6967, July 2008. USNRC ADAMS Accession Number ML082130389.

[34] K. Svanholm, et al., “Halden Reactors IFA-511.2 and IFA-54X: Experimental series under adverse cooling conditions”, Experimental Thermal and Fluid Science, Volume 11, 1995.

[35] T.J. Haste, “Conclusions from the IFA-54X series to compare the ballooning response of nuclear and electrically heated PWR fuel rods in the Halden reactor”, Halden Project Seminar on High Burn-up Fuel Performance Topics, Fredrikstad, Norway, 1987.

[36] V. Lestinen, E. Kolstad and W. Wiesenack, “LOCA testing at Halden, trial runs in IFA-650”, 2003 Nuclear Safety Research Conference, Washington, DC, USA, NRC, pp. 299-309, 2004. USNRC ADAMS Accession Number ML042050210.

[37] V. Lestinen, “LOCA Testing at Halden, First Experiment IFA-650.1”, OECD Halden Reactor Project report HWR-762, March 2004.

[38] M. Ek, “Minutes of the LOCA Workshop Meeting”, LOCA Workshop Meeting, Halden, Norway, 2005.

[39] M. Ek, “LOCA Testing at Halden, the Third Experiment IFA-650.3”, OECD Halden Reactor Project report HWR-785, October 2005.

[40] L. Kekkonen, “LOCA Testing at Halden, the Fourth Experiment IFA-650.4”, OECD Halden Reactor Project report HWR-838, January 2007.

[41] L. Kekkonen, “LOCA Testing at Halden, the PWR Experiment IFA-650.5”, OECD Halden Reactor Project report HWR-839, January 2007.

[42] L. Kekkonen, “LOCA Testing at Halden, the VVER Experiment IFA-650.6”, OECD Halden Reactor Project report HWR-870, June 2007.

[43] R. Josek, “LOCA Testing at Halden, the BWR Experiment IFA-650.7”, OECD Halden Reactor Project report HWR-906, June 2008.

[44] F. Bole du Chomont, “LOCA Testing at Halden, the Eighth Experiment IFA-650.8”, OECD Halden Reactor Project report HWR-916, October 2009.

[45] F. Bole du Chomont, “LOCA Testing at Halden, the Ninth Experiment IFA-650.9”, OECD Halden Reactor Project report HWR-917, October 2009.

[46] A. Lavoil, “LOCA Testing at Halden, the Tenth Experiment IFA-650.10”, OECD Halden Reactor Project report HWR-974, December 2010.

[47] “Safety Significance of the Halden IFA-650 LOCA Test Results”, OECD Nuclear Energy Agency/ Committee on the Safety of Nuclear Installations Report NEA/CSNI/R(2010)5, November 2010.

[48] RES Office Instruction TEC-002, “Procedures for Processing Generic Issues,” dated September 29, 2010.

[49] NUREG/BR-0058, Rev. 4, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission.”