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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO TASK INTERFACE AGREEMENT 99-03

REGARDING POTENTIAL NONCONSERVATIVE ASSUMPTIONS

FOR FUEL-HANDLING ACCIDENT

MCGUIRE NUCLEAR STATION

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated March 23, 1999, Region II expressed concern that the assumptions used in assessing the radiological consequences of a postulated fuel-handling accident (FHA) at the McGuire Nuclear Station may be nonconservative. This concern was identified during resident inspector followup of fuel handling activities at McGuire and was documented as Unresolved Item 50-369, 370/97-10-01, "Radiological Consequences of a Fuel Handling Accident Involving High-burnup Fuel."

The three questions posed by Region II follow:

1. If pellets fragment upon impact during a dropped assembly accident, what additional amount of fission gas trapped in the fuel pellets could be released to the environment? Would the increase be significant in terms of the assumptions related to the amount of activity released in a postulated McGuire dropped-assembly accident?

2. Does Regulatory Guide (RG) 1.25 need to be updated to reflect current information on the high-burnup fuel?

3. If the RG 1.25 values are nonconservative, is the magnitude of the nonconservatism sufficient to call into question the capability of the licensee's spent fuel pool ventilation system (ESF [engineered safety features] grade) to perform its design function of limiting offsite doses?

Region II also identified two questions that would be addressed under the agency's Program Plan for High-Burnup Fuel:

4. Are the release fractions and internal rod pressures calculated (in-reactor effects) using ORIGEN or TACO3 codes conservative for high-burnup fuel in view of the recent NRC and industry work/findings issues for high-burnup fuel? How does this compare to FRAPCON (NRC) models?

Attachment

5. Have adequate structural calculations and experiments been conducted to evaluate mechanical loads (from impact) on high-burnup fuel pellets?

The Probabilistic Safety Assessment Branch (SPSB) has addressed Questions 2 and 3, and part of Question 4. The Reactor Systems Branch (SRXB) has responded to Questions 1 and 5, and part of Question 4.

2.0 BACKGROUND

Safety Guide 25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Ref. 1), was issued in 1972. FHAs are analyzed to assess the risk to public health and safety resulting from the operation of the facility and to demonstrate compliance with various regulatory requirements (Ref. 2 and 3). An illustrative accident sequence consists of the dropping of a fuel assembly during refueling operations, resulting in the breaching of the fuel cladding, release of a portion of the volatile fission gases from the damaged fuel rods, transport of soluble and insoluble gases through the water of the spent fuel pool, absorption of soluble gases in the water, and release to and transport through the environment. Although supplemented by Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," of NUREG-0800 (Ref. 4), this safety guide has never been updated.

The technical bases for many of the assumptions are contained in a 1971 Atomic Energy Commission (AEC) staff paper by G. Burley (Ref. 5). This paper was based, in part, on experimental work performed by the Westinghouse Electric Company that was documented in a proposed 1970 topical report WCAP-7518-L (Ref. 6).

The Burley paper established the following conditions with regard to the specification of isotopic releases:

1. Peak linear power density of 20.5 kw/ft for the highest power assembly discharged

2. Maximum center-line operating fuel temperature less than 4500°F for this assembly

3. Average burnup for the peak assembly of 25 GWD/MTU[1] or less (this corresponds to a peak local burnup of about 45 GWD/MTU)

The iodine decontamination factor (DF) specified in the guide is primarily a function of bubble contact time, which is, in turn, a function of release pressure and pool depth. The guide established the following conditions associated with the DF:

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[1] There are several expressions for the level of fuel burnup. In the interest of clarity, the more typical expressions are defined in this footnote: *Batch average* – the arithmetic mean of the burnup of all fuel assemblies in the quantity of fuel to be replaced at a single fuel replacement outage; *peak assembly average* –the burnup of all fuel rods in the peak assembly averaged (arithmetic mean) over the assembly; *peak rod average* – the burnup of the peak fuel rod averaged (arithmetic mean) over the length of the rod; and *peak rod local* – the maximum burnup of the peak fuel rod.

4. Minimum water depth (above fuel) of 23 feet

5. Maximum release pressure of 1200 psig

These five conditions, appropriate for the fuel designs that existed when the guide was issued, are specified as footnotes to the safety guide. However, current fuel designs are being licensed to burnup levels as high as 62 GWD/MTU. These fuel designs have been approved by the Office of Nuclear Reactor Regulation (NRR) staff and staff evaluation reports (SERs) have been issued for these designs. These SERs (Ref. 7) typically indicate that there was sufficient conservatism in the assumptions of the regulatory guide to allow operation at higher burnup. There are indications that requests will be made for even higher burnup levels.

On February 29, 1988, the NRC published an environmental assessment (Ref. 8) and a finding of no significant hazard with regard to the use of extended-burnup fuel in commercial light-water reactors. A study of the potential environmental impacts of increasing the batch average burnup from 33 to 50 GWD/MTU (with a peak rod average burn-up of 60 GWD/MTU) evaluated the impact on the isotopic release assumptions of Safety Guide 25. This study was documented in "Assessment of the Use of Extended-burnup Fuel in Light Water Power Reactors," NUREG/CR-5009 (Ref. 9). An effort sponsored by NRR is currently underway to evaluate the environmental impact of an increase in burnup from 60 to 62 GWD/MTU, thus updating the conclusions of NUREG/CR-5009. The results of this study will be used in determining the need for future regulatory action on Safety Guide 25.

On March 25, 1997, the NRC staff briefed the Commission on a broad range of high-burnup fuel issues. Following this meeting, the NRC staff was directed to prepare an agency program plan for high-burnup fuel (Ref. 10). The program plan addresses nine issues:

1. cladding integrity and fuel design limits
2. control rod insertion problems
3. criteria and analysis for reactivity accidents
4. criteria and analysis for loss-of-coolant accidents (LOCAs)
5. criteria and analysis for boiling-water reactor power oscillations (ATWS)
6. fuel rod and neutronic computer codes for analysis
7. source term and core-melt progression
8. transportation and dry storage
9. high enrichments (greater than 5%)

These issues were identified on the basis of observed operational problems, experimental results from test programs, and an understanding of basic phenomena. The program plan is focused on confirming safety at the currently approved burnup levels and on those aspects that pose the greatest risk to the public. This plan does not address potential revisions to Safety Guide 25.

The NRC staff has briefed the ACRS on high-burnup issues, most recently on March 11, 1999, (Ref. 11). The gap release fraction[2] associated with an FHA was specifically addressed in this presentation.

3.0 EVALUATION

The NRR staff provides the following responses to the subject questions:

> Question 1 If pellets fragment upon impact during a dropped assembly accident, what additional amount of fission gas trapped in the fuel pellets could be released to the environment? Would the increase be significant in terms of the assumptions related to the amount of activity released in a postulated McGuire dropped-assembly accident?

If the pellets fragment upon impact during a dropped assembly accident, the amount of additional fission gas released to the environment is not known. We are not aware of any experiments or calculations that have been done to answer this question. As stated in the answers to questions 2 and 3, the risk significance of the postulated fuel handling event is considered to be low. Furthermore, typical analyses of the fuel handling event assume that all the kinetic energy in the dropped assembly is dissipated in the breaking of the fuel cladding. No credit is given for dissipation of the kinetic energy by the fuel assembly structures, by the spent fuel racks, or by buoyancy or hydrodynamic drag forces. Since the amount of kinetic energy available to be dissipated is a fixed amount, only a very small number of fuel pellets could be fragmented if all of the kinetic energy went into fragmenting the fuel pellets and breaking the associated fuel cladding. Thus, we consider this situation to be bounded.

> Question 2 Does RG 1.25 need to be updated to reflect current information on the high-burnup fuel?

Consideration will be given to updating RG 1.25 (Safety Guide 25) to reflect extended-burnup fuel and current NRR staff analytical practice. There is an effort underway within NRR to make 10 CFR Part 50 risk-informed. A possible outcome of this effort is that the current regulatory requirements regarding the analysis of an FHA may be revised or perhaps eliminated because of the relatively low risk significance of this event. Additional bases are provided in the Appendix.

> Question 3 If the RG 1.25 values are non-conservative, is the magnitude of the nonconservatism sufficient to call into question the capability of the licensee's spent fuel pool ventilation system (ESF grade) to perform its design function of limiting offsite doses?

The NRR staff notes that spent fuel pool ventilation systems are not always credited in FHA analyses. As such, the following response addresses the more generic question of whether there is adequate public protection of FHAs. The NRR staff has identified uncertainties

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[2] The gap release fraction is the fraction of the fission products that migrate from the fuel matrix to the region between the fuel pellets and fuel cladding during normal operation and that would be available for immediate release in the event of clad damage.

associated with the gap release fractions specified in RG 1.25 with regard to extended-burnup. The magnitude of the conservatism or nonconservatism cannot be readily quantified. The NRR staff expects increases, if any, to be limited to 20 to 50 percent of the current values. The NRR staff has concluded that the increased rod pressures associated with extended-burnup fuel can be expected to decrease the value of the iodine DF. However, the NRR staff believes that the iodine DF value of 100 provided in RG 1.25 has sufficient margin to compensate for the increases in rod gas pressures at current allowable burnup levels and for the expected increases in gap release fractions. Conservatisms in the assessment of the amount of fuel damage provide additional margin. Design basis FHAs are not considered to have a high risk significance. On the basis of these findings, the staff concludes that there is reasonable assurance that adequate protection of the public from the effects of design basis FHAs involving fuel with peak rod average burnups as high as 62 GWD/MTU will continue. Additional bases are provided in the Appendix.

> <u>Question 4</u> Are the release fractions and internal rod pressures calculated (in-reactor effects) using ORIGEN or TACO3 codes conservative for high-burnup fuel in view of the recent NRC and industry work/findings issues for high-burnup fuel? How does this compare to FRAPCON (NRC) models?

The gap release fractions specified in Safety Guide 25 were not determined using ORIGEN or TACO3. The conservatism of the release fractions was addressed in the response to Question 2. The core inventory, against which the gap release fractions are applied, is determined using ORIGEN and similar codes. Although the nuclides of interest in DBA analyses reach equilibrium values early in a fuel cycle, extended-burnups can affect the core inventory. A substantial fraction of the energy produced during the final fuel cycle may derive from Pu-239. The most significant difference in terms of radiological analyses is a 27 percent greater I-131 yield from Pu-239 fission as compared to that for U-235 fission (Ref. 5). This concern can be resolved by reassessing the core inventory using ORIGEN-II or similar codes with the current licensed values of core power, burnup, and enrichment. Additional bases are provided in the Appendix.

Calculations using FRAPCON-3 were performed by the staff to assess the adequacy of the licensee evaluation of fission gas release and internal rod pressure. Information provided by the licensee was used to develop the FRAPCON model to ensure that the staff and the licensee were evaluating the same fuel using the same assumptions. The staff's evaluation predicts an internal rod pressure of 907 psia which is significantly smaller than the 1200 psia predicted using TACO3. This trend is consistent with the staff's previous evaluations of TACO3. FRAPCON does not have the capability to predict Iodine release fractions, but it is the staff's position that the ANS 5.4 model should provide conservative assessments of iodine gap fractions.

> <u>Question 5</u> Have adequate structural calculations and experiments been conducted to evaluate mechanical loads (from impact) on high-burnup fuel pellets?

We do not know of any specific structural calculations or experiments that have been done to evaluate the mechanical loads from impact on the high burnup fuel pellets themselves. Typical analyses assume that all the kinetic energy is dissipated in breaking of the fuel clad. No credit is given for dissipation of kinetic energy by the fuel structures other than the fuel rods (e.g., upper and lower nozzle, spring clips, etc.), by the spent fuel racks, or by buoyancy or hydrodynamic drag effects.

Therefore, the NRR staff concludes that adequate conservatism is recognized in the assumptions for FHA.

4.0 REFERENCES

1. USNRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," Safety Guide 25.

2. USNRC, Criterion 60--*Control of releases of radioactive materials to the environment*, Title 10 Code of Federal Regulations Part 50, Appendix A, GDC-60.

3. USNRC, Criterion 61–*Fuel storage and handling and radioactivity control*, Title 10 Code of Federal Regulations Part 50, Appendix A, GDC-61.

4. USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800.

5. Burley, G., "Evaluation of Fission Product Release and Transport," 1971 (Accession 8402080322).

6. Bell, M. J., et al, "Topical Report: Radiological Consequences of a Fuel Handling Accident," Westinghouse Electric Corporation, WCAP-7518-L, June 1970 (attachment to accession 9804290400).

7. For example, "Acceptance for Referencing of Licensing Topical Report BAW-10153P, Extended-Burnup Evaluation," Dec 3, 1985; "Safety Evaluation Report on Westinghouse Electric Corporation Extended-Burnup Topical Report - WCAP-10125 (Proprietary)," May 1985 (to 60 GWD/MTU rod average); "Issuance of Amendment No. 153 to Facility Operating License No. NPF-1, Trojan Nuclear Power Plant," May 24, 1989.

8. USNRC, "Extended-Burnup Fuel Use in Commercial LWRs; Environmental Assessment and Finding of No Significant Impact," 53 *Federal Register* 6040, February 29, 1988.

9. USNRC, "Assessment of the Use of Extended-Burnup Fuel in Light Water Power Reactors," NUREG/CR-5009, 1988.

10. USNRC, letter from L. J. Callan to Commissioners, "Agency Program Plan for High-Burnup Fuel," July 6, 1998.

11. Meyer, R. O., "Status of NRC Research Activities on High-Burnup Fuel," presentation before the ACRS, March 11, 1999.

12. Besmann, T. M. and T. B. Lindemer, "Chemical Thermodynamics of the System Cs-U-Zr-H-I-O in the Light Water Reactor Fuel-Cladding Gap," *Nuclear Technology* vol 40 297(1978).

13. Collins, J. L., et al., "Fission Product Iodine and Cesium Release Behavior Under Light Water Reactor Accident Conditions," *Nuclear Technology* vol 81 78(1987).

14. Taylor, J. H., Babcock & Wilcox, letter to D. H. Moran, USNRC, dated June 21, 1983.

15. USNRC, "Design, Testing, and Maintenance Criteria for Postaccident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Regulatory Guide 1.52.

16. USNRC, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145.

17. ICRP, "Report of Committee II on Permissible Dose for Internal Radiation," ICRP Publication 2, 1959.

18. ICRP, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, 1978.

19. Brooks, W. L., et al., "Further Additional Supplemental Testimony on Contention I.D.2" (in the matter of Long Island Lighting Company et al., Jamesport Nuclear Power Station).

APPENDIX

TASK INTERFACE AGREEMENT 99-03

POTENTIAL NONCONSERVATIVE ASSUMPTIONS FOR FUEL-HANDLING ACCIDENT

MCGUIRE NUCLEAR STATION

DOCKET NOS. 50-369 AND 50-370

In preparing the response to Task Interface Agreement 99-03, the NRR staff reviewed Safety Guide 25, the Burley paper, WCAP-7158-L, NUREG/CR-5009, and other documents. The significant analysis assumptions provided in the safety guide are (1) gap release fractions, (2) iodine species, (3) pool decontamination factors (DFs), (4) ventilation system filtration efficiency, (5) atmospheric diffusion, and (6) dose calculation.

Gap Release Fractions

The gap release fractions specified in Safety Guide 25 were developed in the Burley paper. Burley determined core inventory as a function of the thermal power generated (Ci/MWt) and used a temperature-dependent diffusivity value to determine the transport of the noble gas and iodine fission products from the fuel to the gap region. The I-131 gap release fraction was estimated to be 8.6 percent (rounded to 10 percent in Safety Guide 25). For Kr-85, the fraction was estimated to be 30 percent. With the exception of Kr-85, the gap release fractions for the remaining nuclides were all less than that for I-131. The NRR staff makes the following observations regarding the gap release fractions.

1. Burley identified two mechanisms that could reduce the activity available from release from the damaged fuel, but neither was credited in his analysis. First, the gas phase concentration of iodine would be limited by its vapor pressure. Burley noted that the vapor pressure curve for iodine rises very steeply in the temperature range just above 170°F (the design temperature 100 hours after shutdown). As such, a small change in temperature would yield a large change in the gas concentration. Nonetheless, this could be a conservatism in those cases in which the temperature was kept at less than this point on this vapor pressure curve.

   Second, Burley noted that a substantial fraction of the gap activity could be in chemical combination at the inside surface of the clad and not be available for immediate release. The reaction of Zircaloy with elemental iodine is temperature dependent and goes further toward completion at low temperatures—a condition consistent with fuel handling. Burley stated that this mechanism was implicitly addressed since his analysis had assumed equal diffusivity for iodines and noble gases when it is known that the iodine diffusion coefficients are higher. However, the reaction between Cs and I_2 is favored over that for Zr and I_2 (Ref. 12).

2. The gap release fractions in the Burley paper were based, in part, on a peak power of 20.5 kw/ft, a temperature distribution with a peak temperature of 4500°F and a peak local burnup of 45 GWD/MTU (25 GWD/MTU peak assembly average) in the peak power

assembly in the entire core. The Burley paper and the WCAP report also considered a case involving the peak power assembly in the region to be discharged. The first case (i.e., the Safety Guide 25 case) was considered to represent upper limit values, while the second case was thought to provide a more realistic estimate of the release from an FHA. The gap release fractions for the discharge region case were shown to be approximately 50 percent less, a conservatism. However, this conservatism would not exist in the case of a full core offload as currently performed by some licensees.

3. NUREG/CR-5009 evaluated the potential impact of increasing the batch average burnup from 33 GWD/MTU to 50 GWD/MTU, with a peak rod average burnup of 60 GWD/MTU. The evaluation showed increased gap release fractions with increasing burnup. With the exception of I-131, the resulting gap release fractions were less than those specified in Safety Guide 25. The evaluation indicated a gap release fraction of 12 percent for I-131, approximately 20 percent higher than the 10 percent fraction specified in Safety Guide 25. The NRR staff has used the 12 percent I-131 gap release fraction in reviewing licensee submittals involving FHAs. However, not all licensees with extended-burnup fuel have re-analyzed the FHA for the increased I-131 gap release fraction.*

   NRR has recently sponsored two contractor analyses pertaining to the impact of extended-burnup fuel. The results of these efforts are currently being evaluated by the NRR staff and have not yet been published. However, preliminary results point to an increase in the gap release fractions. Further regulatory action will depend on the final results and conclusions of these two efforts.

4. The NRC staff presented the results of FRAPCON-3 analyses of fuel with a peak average rod burnup of 65 GWD/MTU in an ACRS meeting on March 11, 1999. The maximum long-lived gap release fraction was determined to be 11.1 percent. The corresponding I-131 gap release fraction would be about 9.5 percent. This fraction is less than that specified in Safety Guide 25 and NUREG/CR-5009.

5. The NRR staff reviewed several papers on fuel performance that were part of conference[§] proceedings. In general, papers discussing direct measurements of gap activity in rods irradiated up to 50-60 GWD/MTU reported gap release fractions less than the 10 percent specified in Safety Guide 25, and many were in the range of 2 to 3 percent. These results suggest a conservative bias in the models used to assess fission transport in fuel.

<u>Iodine Decontamination Factors and Iodine Species</u>

The iodine DF of 100 specified in Safety Guide 25 was developed in the Burley paper using data from experiments performed by Westinghouse and documented in WCAP-7518-L.

Westinghouse conducted an experimental test program to evaluate the extent of iodine removal from gas released from a damaged irradiated fuel assembly by the water in the spent fuel pool.

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*Staff SERs prepared for the vendor topical reports for fuel upgrades typically concluded that there would be no significant impact on the previously analyzed FHA doses because of conservatisms in the FHA analysis methods and assumptions. Some licensees based their license amendments, in part, on these conclusions.

§ For example, "International Light Water Reactor Fuel Performance," 1991, 1994, and 1995

This test program involved small-scale tests with iodine and carbon dioxide in an inert carrier and large-scale tests with carbon dioxide gas released from a test rig submerged in a deep pool. The test apparatus was designed to simulate the instantaneous gas release from a sheared-off 14 x 14 fuel assembly. Bubble rise time and DF were measured for various combinations of bubble size and gas pressure. Since the full-scale tests did not measure iodine DF directly, Westinghouse analytically correlated the small-scale tests results with those obtained from the full-scale tests to estimate an iodine DF. Westinghouse estimated an expected iodine DF of 760 for a 26-foot pool and applied a factor of 66 percent to obtain a conservative iodine DF of 500 for a 26-foot pool.* The Burley paper indicates that Westinghouse reevaluated its results using a more detailed mass transfer analysis and reported[§] iodine DFs of 3200 and 720 for bubble diameters of 1.21 and 1.37, respectively.

It can be inferred from the Burley report that the AEC staff had concerns about the analytical method used by Westinghouse. The WCAP report was never approved for use in licensing submittals. The AEC staff performed a parametric analysis using the WCAP report bubble rise times for a pool depth of 23 feet, varying both the bubble diameter (1.20 to 2.50 cm) and the iodine partition factor. The AEC staff qualitatively selected a central value on the axis between the most and the least conservative values, such that the effective DF considering both elemental and organic iodine would be 100 (elemental DF = 133). Since the DF for organic iodine is assumed to be unity because of its low solubility, the Safety Guide 25 assumption that 0.25 percent of the gap iodine activity is organic effectively limits the overall DF to 400 regardless of the value of the inorganic iodine DF.

1. The Westinghouse test program did not measure the iodine DF in the full-scale pool but instead, derived it analytically from the carbon dioxide DF measured in the full-scale and small-scale tests and the iodine DF measured in the small-scale tests. On the basis of the brief description in the Burley paper, the AEC calculated the pool DF using a theoretical treatment of iodine mass transfer without reference to the DFs measured by Westinghouse. The NRR staff has concluded that the Safety Guide 25 DF is likely over conservative.

2. The bubble rise tests were performed using sheared-off tubes in a vertical alignment. This is a conservative arrangement for determining bubble rise time since the initial bubble velocity from the upward-directed, open-ended fuel tubes is enhanced. Although fuel damage such as that modeled is possible, administrative controls over the movement of heavy loads and the configuration of structures and equipment in the path of irradiated fuel during movement reduce the likelihood of its occurring. A more likely situation would involve the dropping of a fuel assembly to an oblique or horizontal position in which the gas jet might not be upward-directed.

The bubble rise tests involved tubes with internal diameters comparable to those of current fuel rod designs. This is a conservative simulation in that actual fuel rods contain pellets that significantly reduce the free cross-sectional area available for gas flow. In extended-burnup

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* Using the raw test data documented in the WCAP report, the NRR staff determined iodine DFs for a 23-foot pool of 464 (expected) and 300 (conservative) for the current evaluation. The WCAP report values incorporated some data smoothing on intermediate results.

§ Additional documentation of these data could not be located.

fuel, expansion of the pellet further reduces the space between the pellet surface and the inner surface of the clad. As such, there would be resistance to the expansion of the gases in the gap region. Initial bubble sizes would be less than those observed in the tests. The mass transfer is more efficient for smaller bubbles.

1. The WCAP report and the Burley paper both assume that the inorganic iodine released from the damaged fuel is in the form of elemental iodine. Current insights on fission product behavior indicate that the predominant chemical form would be CsI. Collins et al. reported on experiments to establish the amounts of fission products and the chemical species released from the gap of fuel rods irradiated in commercial reactors. The maximum test fuel temperature (2200°F) was selected to be less than that associated with fuel melting such that only the gap inventory would be measured. The test results indicated that neither elemental cesium or elemental iodine was released from the gap. Other experimental evidence suggests the formation of CsI. The authors report 91.2 percent CsI, 8.6 percent particulate iodine, and 0.27 percent elemental iodine in the limiting case (Ref. 13). The DF for CsI would be greater than that estimated for elemental iodine because of its enhanced solubility in water.

2. In LOCA analyses, dissociation of CsI in containment sumps is minimal if pH>7 is maintained. If the pH is less, evolution of elemental iodine is assumed. Since the spent fuel pH ranges between 4 and 5, evolution of elemental iodine could be expected. However, the thermal and hydraulic conditions in the spent fuel pool are substantially less energetic than those in a containment sump during LOCA conditions. It is reasonable to assume that much of this dissociated iodine would be retained by the pool. Burley concluded that re-evolution of elemental iodine from the pool water would contribute less than 10 to 20 percent of the long-term thyroid dose.

3. The Burley paper postulates that methane from impurities in the fuel will react with elemental iodine to form methyl iodide. This reaction would compete with the reactions that create CsI and ZrI_2. Given the small amount of impurities present in comparison to the large amounts of Zircaloy and cesium present, the assumed methyl iodide fraction of 0.25 percent is likely overstated. A reduction in the methyl iodide fraction would increase the overall pool DF since the methyl iodide DF is assumed to be unity.

4. The maximum fuel rod pressure modeled in the pool DF tests was limited to 1200 psig. Rod pressures associated with extended-burnup fuel can be expected to range to 1800 psig and higher. Figure 3-9 in the WCAP report plots the pool DF versus rod pressure, showing a trend of decreasing DF with increasing pressure. Although this plot shows a straight line fit to the data, the limited data suggest that a curvilinear fit might show a decreasing slope. This possibility suggests that the pool DF may not change as rapidly with increasing pressure. It has been suggested that the higher rod pressure causes a higher velocity, more energetic discharge that will tend to breakup the gas jet into smaller bubbles. This result is consistent with the discussion in the Burley paper regarding instability in large bubbles that results in their breakup into smaller bubbles. The low density of the gas relative to the pool water is expected to cause the gas jet to decelerate rapidly and reach the terminal rise velocity of the created smaller bubbles within a few feet of the break location. Given the expected smaller bubbles at higher pressures and the reduced terminal velocity of the smaller bubbles, the bubble rise time may be slightly longer (Ref. 14). Although this hypothesis can not be readily quantified, it

would provide a possible explanation of the suggested reduction in the slope of the DF versus pressure plot.

Ventilation System Filtration Efficiency

Safety Guide 25 specifies a charcoal filter removal efficiency of 90 percent for inorganic species and 70 percent for organic species. These assumptions are not affected by extended-burnup. Although these assumptions have been used in some early licensee analyses, the removal efficiencies of RG 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants" (Ref. 15), have been used in the more recent NRR staff reviews. This guide provides removal efficiencies that are a function of filter system design and periodic testing criteria. Removal efficiencies of 95 percent and higher for both inorganic and organic species are achievable within the guidance provided.

Atmospheric Diffusion

The Safety Guide 25 guidance on acceptable methods for assessing atmospheric diffusion has been superseded by the guidance in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (Ref. 16), and Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases," of the SRP (Ref. 4). This guidance is not affected by extended-burnup.

Dose Calculation

The Safety Guide 25 dose calculation guidance continues to be acceptable. The iodine dose conversion factors (DCFs) specified in the guide were based on the methodology of ICRP Publication 2, "Report of Committee II on Permissible Dose for Internal Radiation" (Ref. 17). The iodine DCFs in the later ICRP Publication 30, "Limits for Intakes of radionuclides by Workers" (Ref. 18), result in postulated doses that are approximately 30 percent less than the doses calculated with the ICRP-2 DCFs. This is a conservatism for those licensees that have not updated the DCFs used in their design basis analyses.

Amount of Fuel Damage

Safety Guide 25 does not provide guidance on the number of fuel rods assumed to be damaged. The NRC staff has traditionally assumed that all of the rods in the dropped assembly are damaged. Some licensees have submitted and received approval for a more mechanistic treatment of the number of rods assumed to fail. Traditional assessments of the number of rods damaged included significant conservatisms that lead to a substantial over-estimate of the number of damaged rods that could result from a dropped fuel assembly. Typical analyses assume that all the kinetic energy in the dropped rod is dissipated only in the breaking of fuel clad. No credit is given for dissipation of kinetic energy by the fuel structures other than the fuel rods (e.g., upper and lower nozzle, spring clips, fuel pellets, etc.), by spent fuel pool racks, or by buoyancy or hydrodynamic drag effects (Ref. 19). While extended-burnup operations could conceivably reduce the strength of the fuel clad, fuel vendors and licensees are required to demonstrate that the fuel structural performance continues to meet regulatory criteria.

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