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January 10, 2001

Mr. Samuel J. Collins
Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-5 E7
Washington, DC 20555-0001

Dear Mr. Collins:

The industry has performed an evaluation of the behavior of spent fuel stored in spent fuel pools following a catastrophic event, i.e., a beyond design basis seismic event that drains and so damages the spent fuel pool that replacement of cooling water is impossible. The industry has expressed its views on several occasions regarding the questionable value of focusing resources on the consequences of such an unlikely event. Nevertheless, if the NRC intends to base its regulations on this event, it is imperative that it be characterized using realistic phenomenology.

I am enclosing a technical report prepared by Dr. Robert Henry of Fauske and Associates. Dr. Henry's report evaluates the phenomenology and consequent releases of fission products with results that differ substantially from those postulated by the Advisory Committee on Reactor Safeguards (ACRS). The substance of this report was presented to the ACRS on November 2, 2000.

I urge you to consider the views expressed in the attached technical report and incorporate them in finalizing the NRC report on the risk of spent fuel pools. If I can be of any assistance to you or your staff please call me or Lynnette Hendricks (202 739-8109, or, LXH@NEI.org).

Sincerely,

Ralph E. Beedle

Enclosure

c: Dr. Dana Powers, Chairman of ACRS
   Michael Tschiltz, OCM /RAM
   Thomas Hiltz, OCM/GJD
   Patrick Castleman, OCM/NJD
   Jim Beall, OCM/EXM
   Brian McCabe, OCM/JSM

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2.0 *SPENT FUEL POOL EVALUATIONS*

Table 1 gives guidance for such evaluations to assure that the results are useful. In particular, the results should address those issues relevant to both operating plants and those being decommissioned to enable the operating organization and the NRC to use the results in the most productive manner.

Table 1: Approach to Evaluations
• All evaluations should use a mechanically identified failure condition.
• Evaluations should assess the results of potential recovery actions consistent with the postulated accident initiator.
• Evaluations should consider all mechanisms for cooling and for energy generation, including the influence of water vaporization in the lower regions of the pool as well as natural circulation of air.

When assessing the response of fuel assemblies to the loss of water inventory, the elements listed in Table 2 need to be considered. The first and second elements are the most important. The third element identifies those possible open channels within individual fuel assemblies where control rods may have been inserted for PWR designs. Figure 1 illustrates such a configuration and the possibility that openings exist or perhaps are only partially filled with spent control rod assemblies. The configuration will depend on the plant specific utilization of the spent fuel pool. However, if these openings are available, they should be part of the evaluation since they are regions of comparatively low frictional pressure drop and therefore permit more local flow.

Table 2: Focus for Analytical Models
• Spent fuel pool is at atmospheric pressure.
• Flow within the fuel assemblies is laminar, i.e., resistances are well characterized by standard representations.
• Opening in individual fuel assemblies are influential flow paths and should be considered.
• The fuel assembly distribution within the pool does not matter for those accident conditions where the water inventory decreases below the top of fuel until the water is at about 70% of the fuel assembly height. The fuel assembly distribution would matter in the multi-dimensional flow pattern that would develop at lower water levels, i.e. if a thermal plume would develop.

The fourth element addresses the extent of cooling for partially uncovered fuel assemblies. The analysis used here is from the evaluations performed in the Technical Basis Report (TBR) for Severe Accident Management Guidance (SAMG) (Henry, 1992). Specifically, the analyses in Section Z of Volume 2 of the TBR show that when considering a condition like that illustrated in Figure 2, the peak cladding temperature can be approximated by the equation listed in Table 3. The assumptions for this evaluation are included in Table 3. Figure 3 is taken from the TBR and shows the results of this calculation, which is (for practical purposes) independent of the decay power in the fuel assemblies. For this steady-state representation, the peak cladding temperature remains below levels where significant oxidation can occur as long as approximately 70% of the fuel is covered. Note that this evaluation and does not characterize the time required for the fuel temperature to increase to the levels shown in Figure 3. With the low decay power in the fuel and the effective heat removal, this would occur over many hours at the very minimum. However, the important element is that sufficient heat removal can be achieved even with part of the fuel uncovered.

Table 3: Estimation of Peak Cladding Temperature for Assumed Accident Conditions Where the Top of the Fuel is Uncovered -- Assumptions --
1. The process is assumed to progress quasi-steady manner.
2. Steam and water are the only fluids in the core.
3. The inlet water is at the saturation temperature T_{sat}.
4. The decay power (Q_D) is constant along the fuel pin length.
5. The collapsed water level (y) can be used to represent the covered portion of the fuel assemblies.
6. The cladding temperatures remain low enough that the energy released by Zircaloy oxidation is an insignificant fraction of the decay power.
7. This results in $$T_o - T_{sat} = \left[\frac{1-y}{y}\right]\frac{h_{fg}}{c_{pg}}$$ where h_{fg} is the latent heat of vaporization for water and c_{pg} is specific heat at constant pressure for steam.

Figure 3: Quasi-Steady Cladding Temperature for a
Partially Uncovered Group of Fuel Assemblies
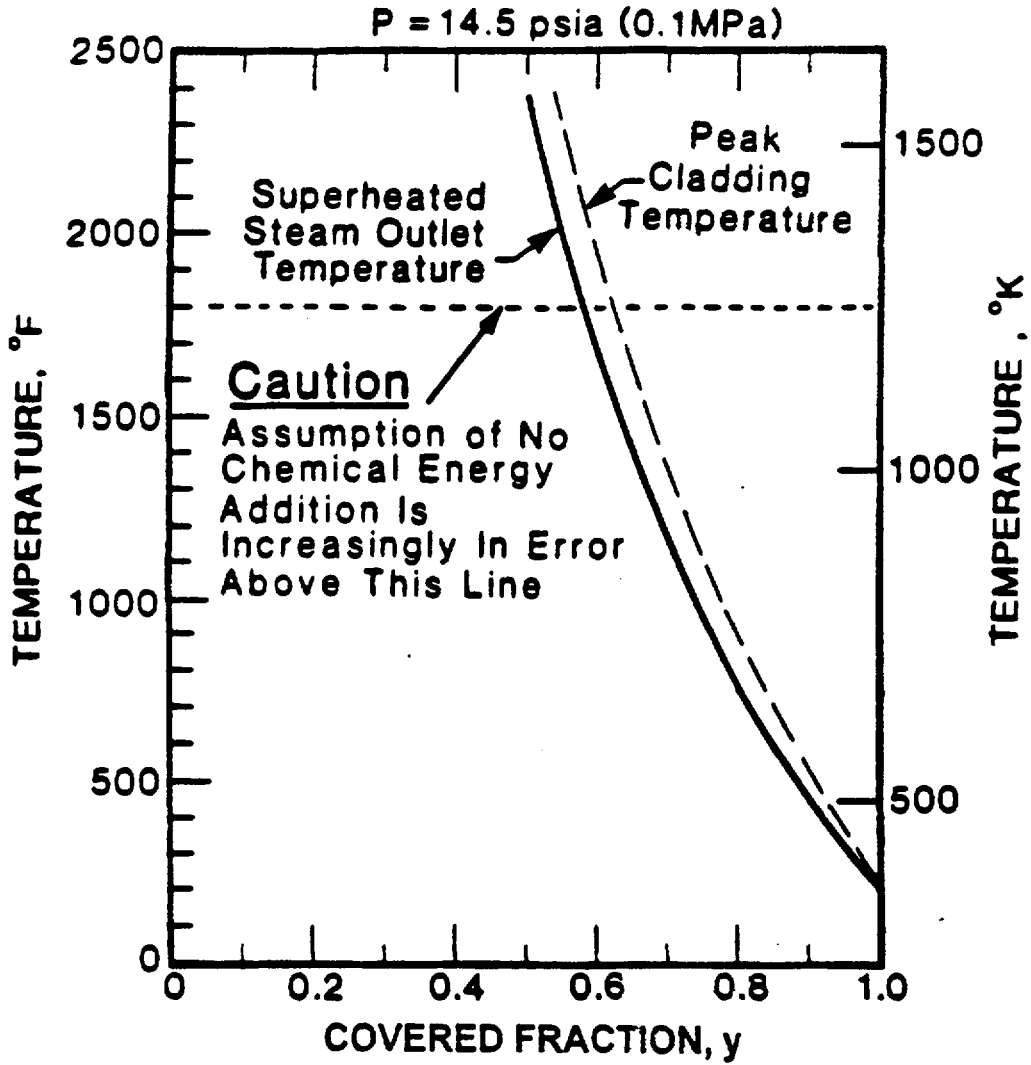
the acceleration of gravity, L_h is the length over which the effective density difference occurs and $\Delta \rho$ is the density difference in the gas flow through the core. These density differences are due to temperature differences as well as changes in the gas mixture as a result of oxidation in the fuel pin cladding, i.e. changing air to nitrogen. The laminar flow friction factor can be expressed by

$$f = \frac{64}{N_{Re}} \tag{2}$$

where N_{Re} is the Reynolds number for the flow defined by

$$N_{Re} = \frac{U D_h}{\nu} \tag{3}$$

with ν being the gas kinematic viscosity.

Considering that the gas density continuously changes along the fluid path, the density difference that is responsible for the flow can be approximated as half of the maximum density difference which would be derived from the maximum temperature difference. Hence, this density difference can be expressed by

$$\Delta \rho = \frac{\Delta \rho_{max}}{2} \tag{4}$$

with the heat removal rate for the gaseous flow described by

$$\dot{Q}_R = \bar{\rho} A_F U c_p \Delta T_{max} \tag{5}$$

Substituting and solving for the maximum temperature difference in the flow results in

release even though the inadequate removal of decay power was the initiating condition that caused the cladding to reach temperatures where such rapid oxidation could occur.

Substituting the parameters given in Table 4, the temperature for the gas flow through the bundle would eventually reach values in excess of 1200 K and significant oxidation would occur between the Zircaloy cladding and the air flowing through the bundle. Given these conditions, the chemical energy release would dominate the power generation within the spent fuel bundles and as a result, the heatup, melting and relocation of material would be expected to be similar to the condition for the "at power" evaluations that have been performed in the past. In this regard, the heatup rate of the fuel pin and cladding is limited by the flow rate of air circulated into the bundle to supply oxygen for the oxidation process. For the "at power" evaluations, the limitation is one of steam starvation since this is the oxygen supply to sustain the oxidation process. For the assumed condition of a completely drained spent fuel pool the heatup would be limited by air starvation since this is the major oxygen source for cladding oxidation.

Conceptually, the general characterization of the fuel heatup, oxidation of the fuel pin cladding, clad melting and relocation as well as liquefaction of the uranium dioxide by the unreacted molten Zircaloy would be expected to be similar to the fuel damage configuration observed for the TMI-2 accident. For the spent fuel pool, the fuel pin heatup would be limited by the oxygen supply to the cladding and the fuel pin configuration. Those configurations presenting the most extensive surface area would more fully support the oxidation process. With the natural circulation being driven by the temperature differences between the incoming gas and that in the reaction zone, the oxygen supply rate would initially be increased by the higher cladding temperatures caused by oxidation. Subsequently, the cladding surface area would rapidly decrease as the materials reach melting and liquefaction temperatures and relocate downward into a more compacted configuration. This relocation diminishes the oxidation effects in two ways: (a) the available surface area for oxidation decreases, and (b) the open areas to support natural circulation gas flow decrease as the fuel geometry becomes more pin compacted. At some point the oxidation is self-limiting and results in a highly compacted configuration that supports essentially no continued oxidation.

further oxidation of either or both of the uranium dioxide and fission products retained in the fuel matrix in the absence of cladding or after complete oxidation of the cladding. However, these do not represent the influence of an oxidation behavior in the presence of zirconium, a very reactive metal. Furthermore, such experiments were performed at very small scale and are not capable of demonstrating the geometry changes and the consequential changes of the gaseous flow paths through the fuel pin region that were observed in the CODEX experiments and the TMI-2 accident. The scale of the experiment is essential in demonstrating this composite result of core degradation.

For the fission product release experiments reported by Barrand et al., the results are particularly relevant to evaluations of ruthenium release in the spent fuel pool. Specifically these irradiated fuel studies included the Zircaloy sheath (cladding). The authors state that:

"Rapid Cs release was delayed after the addition or ingress (e.g., test H04) of oxidizing atmosphere until oxidation of the sheath was complete."

Two paragraphs later they add:

"The solid fraction of fission-product Ru would have been present almost entirely in the metallic state at the temperatures and $P(O_2)$ values in the HCE3 experiment, and release probably occurred by oxidation to RuO_x gaseous species. As observed previously for fragment samples, release of Ru in test H02 (and probably also in tests H01 and H03) began a significant length of time (> 2000 s) after oxidative release of Cs began. This delay was due to the competition among Zircaloy, UO_2 and Ru for oxygen. Test H02 showed the largest percentage release of Ru from any complete Zircaloy-sheathed sample tested at CRL, due to the high test temperature and the comparatively long exposure to air."

These observations are consistent with those of the CODEX experiments which demonstrate the controlling role of unreacted Zircaloy in determining the chemical state of both the ruthenium and the uranium dioxide.

release rate information developed by Iglesias et al. for the ruthenium release rate in air at this temperature, the release rate would be less than 0.01%; a small rate of release. Consequently, the impact of this limited amount of material on the top of the debris would need to be evaluated in terms of the amount of material and the temperatures that could be achieved with the upper regions of the debris pile considering energy loss to the environment by convection and thermal radiation. While this can result in some exposed fuel, the mass of fuel involved is substantially less than the total mass of material. In fact, based on the experience in the TMI-2 accident, it is expected that this would be no more than 20% of the core material with a limited release rate, less than 0.01% per minute. This translates to a value of 0.002% per minute based on the total core inventory.

$$\frac{.01\%}{min} \cdot \frac{60\ min}{hr} \cdot \frac{24\ hr}{day} = \frac{14.4\%}{day}$$

$$\frac{.002\%}{min} \cdot \frac{60\ m}{h} \cdot \frac{24\ h}{d} = \frac{2.9\%}{day}$$

temperature would limit the fission product releases. Preliminary estimates show that this temperature would be in the range of 800-850°K which results in a very slow release rate of ruthenium in air based upon the published experimental information.

6. Considering the composite behavior of the fuel assembly response and the potential slow release of debris from declad fuel pins on top of the debris, the release rates of ruthenium would be limited by the unreacted zirconium in the molten region and by the low surface temperatures for any overlying solidified debris. Hence, the technical basis for evaluations of the public risk resulting from these very low probability events should be consistent with these observations. Of particular note here are the observations of:

- Iglesias et al. (1990) that "Ruthenium release did not start until the sample stoichiometry was close to its equilibrium value and, hence, oxygen was available for the less thermodynamically favorable reaction of oxidizing ruthenium."

- The CODEX experiments (Mathus et al., 1999) that "There was no evidence at all any transformation of the UO_2 to higher oxides" for the oxidation of Zircaloy clad UO_2 overheated in air. This shows that essentially all the zirconium must be oxidized for the ruthenium to begin oxidizing.

- Barrand et al. (1999) summarized the results of the HCE3 fission product release experiment from Zircaloy sheathed CANDU fuel as:

    "The release rates of Kr, Xe, I and Cs were very low before complete sheath oxidation; their peak release rates in steam after complete sheath oxidation were not very temperature-dependent. Ru release was delayed

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