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Mr. James A. Norberg, Special Assistant
Division of Engineering and System Technology
Office Of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Small Break LOCA Position Paper

Reference: J. H. Taylor to Carl Berlinger, "ECCS Methodology for Licensing Catawba and McGuire Reloads,"JHT/88-128, July 25, 1988.

Dear Mr. Norberg:

The attached paper summarizes the B&W Fuel Company (BWFC) position concerning small break loss of coolant accidents (SBLOCA). The essence of the letter is that the existing SBLOCA calculations contained in the McGuire and Catawba FSARs remain valid and bounding for upcoming reload cores which will contain fuel manufactured by BWFC. The BWFC SBLOCA evaluation model was provided in BAW-10168P which was enclosed with the reference letter.

Very truly yours,
C. J. McPhatter
for
J. H. Taylor
Manager, Licensing Services

cc: R. B. Borsum
T. L. Baldwin
Gene Hsii, NRC
R. C. Jones, NRC

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SBLOCA POSITION PAPER

The Babcock & Wilcox Fuel Company (BWFC) has been awarded the refueling contract for the Duke Power Company McGuire and Catawba nuclear steam supply system (NSSS) units which were designed and licensed by the Westinghouse Corporation. In order to license these units, a spectrum of large and small break loss-of-coolant accidents (LOCAs) were performed by Westinghouse and documented in chapter 15 of the plant final safety analysis report (FSAR). The BWFC will reanalyze the LBLOCA analyses because fuel design can affect these evaluations. The SBLOCA analyses will not be reanalyzed. The present SBLOCA evaluations, as performed by Westinghouse, will remain the basis for the plant license. Two major reasons substantiate the validity of this statement: 1) the current SBLOCA results are not the most limiting or severe LOCA cases, and 2) SBLOCA evaluations are unaffected by the design differences between B&W and Westinghouse fuel assemblies. These reasons will be discussed further in the following paragraphs.

The first item which justifies the position of the BWFC deals with the current SBLOCA evaluation results. The Westinghouse calculations of SBLOCA accidents for the McGuire and Catawba units are not the limiting LOCAs as predicted by the NOTRUMP and LOCTA-IV computer codes. The calculated results documented in the current McGuire and Catawba FSARs predict peak SBLOCA cladding temperatures less than 1500 F. All parameters are well within the Acceptance Criteria limits of 10 CFR 50.46 as documented in the respective FSARs. Even wide variations in SBLOCA results would not cause the SBLOCA to be limiting. Thus, considerable margins exist such that variations in the SBLOCA results would not alter either the plant technical specifications or operating procedures.

The second item which supports the BWFC position is rooted in the progression of any potentially limiting SBLOCA transient. SBLOCAs are generally characterized by five specific transient phases: 1) subcooled depressurization, 2) pump/loop flow coast-down and natural circulation, 3) loop draining, 4) boiling pot, and 5) long term cooling. An examination of these periods in the evolution of the transient will demonstrate that the reload fuel design differences would cause a negligible impact on the current SBLOCA evaluation results.

The worst case SBLOCA scenario begins with a subcooled reactor coolant system (RCS) depressurization until the primary system pressure matches the initial hot leg temperature saturation pressure. During this depressurization phase the low pressure reactor trip, ECCS injection, and reactor coolant pump trip signals are achieved. Tripping of the pumps begins the pump and loop flow coast-down period.

Following reactor scram, the core power generation declines rapidly. The initial forced and residual loop flow, both prior to and during the pump coast-down, provide a continuous coupling between the steam generators and the core. This coupling allows the core decay heat and initial stored energy in the fuel to be rejected via the steam generator. The pump coast-down and natural circulation flows during this period are sufficient to prevent CHF from occurring in the core. As a result, the fuel pins are cooled toward the quasi-steady temperature distribution required to simply conduct and convect the decay heat energy out of the pins. These pin temperatures approach the RCS saturation temperature. Loss of continuous loop flow marks the end of this period.

The third phase in the transient is referred to as the loop draining period. During this period the system reaches a quiescent state in which the core decay heat, leak, pumped ECCS

injection, and residual steam generator heat transfer interact to control the formation and response of steam-water mixture levels within the RCS. The system inventory distribution is a strong function of the system geometry and break location. RCS liquid inventory will continue to decrease until component mixture levels provide a continuous vent path for core steam production. Relief of core steam production allows the RCS to further depressurize and enter the boiling pot mode.

As noted, the course of events which demarcate the end of the loop draining and onset of core boil-off are governed by the break location. For hot leg breaks, the continuous core steam venting path is readily established. A significant system inventory loss is required to establish the vent path for steam generator downstream piping breaks. However, the maximum inventory loss, and therefore, the most severe of all SBLOCAs occurs in the cold leg pump discharge breaks. In these breaks, liquid inventory is lost until the primary levels descend to the spill-under elevation at the low point in the cold leg pump suction piping. This liquid trap or loop seal must be cleared of liquid to establish the steam venting path to the leak. Since the loop seal elevation is located slightly above the middle of the core, the core collapsed level will be depressed by the manometric pressure balance imposed by the RCS geometry. Once the loop seals clear, the steam venting path is established and the residual liquid inventory in the pump discharge and downcomer regions drains into the core region.

The onset of the boiling pot mode is generally coincident with the beginning of a final saturated depressurization. Voiding at the break increases the leak volumetric flow rate which ultimately depressurizes the system until the accumulator fill pressure is reached or the pumped ECCS injection matches core boil-off. During this period the reactor vessel mixture levels descend into the core heated region. Pin temperature excursions

in the upper elevation are maximized by the assumption of a bounding, core outlet skewed, peaking. During these heat-ups, the cladding may swell and even rupture if the temperature approaches 1500 F. However, as long as the cladding temperature remains below 1800 F, the occurrence of rupture is treated conservatively by not including the effects of rupture in the evaluation.

Swelling and rupture produce three primary effects on the temperature calculation. First, the cladding expansion increases the fuel pin gap thus allowing a momentary cooling of the clad. This condition is temporary, however, it delays the temperature excursion which results in a lower peak clad temperature because the decay heat level has decreased slightly. Secondly, the rupture may divert flow out of the channel. The evaluation model uses only average channel flows to cool the hot channel. This flow is significantly lower than that expected in the hot channel. Thus, the effects of rupture-induced flow diversion are already conservatively bounded by the modeling. Finally, the inside of the cladding is exposed to metal-water reaction which creates a new heat source. The metal-water reaction is exponential with the cladding temperature, becoming significant relative to decay heat levels as the temperature approaches 1800 F. Below 1800 F the metal-water heating is a small fraction of the decay heat. The extra heat from the inside of the cladding need not be considered to conservatively evaluate the cladding temperature so long as that temperature remains below 1800 F.

The temperature excursions are arrested as the combined ECCS flows exceed the core decay heat level and final core refill begins. The suppression of core steam production further depressurizes the RCS, and thus increases the ECCS injection flow and hastens core refill. Eventually the RCS system will be depressurized to the containment pressure and the core will be refilled. At this point the start of a long term cooling

configuration has been established and the transient is mitigated.

SBLOCA transients are dominated by system behavior and core decay heat levels. Minor differences between the BWFC fuel reload assemblies and the current Westinghouse assemblies exist; however, they do not alter the approved SBLOCA evaluations. The BWFC and Westinghouse assemblies are different in the following areas: irrecoverable pressure drops across the assemblies, initial fuel temperatures, initial pin internal gas pressure, and the axial power profile. The impact of each of these items will be evaluated with respect to the controlling aspects of the SBLOCA transient scenario which was previously described.

The BWFC fuel assemblies have irrecoverable pressure drops which are within ±2 psi of the Westinghouse assembly value. The associated change in overall loop pressure drop associated with the BWFC reload could therefore modify the initial 100% forced flow rate by a negligible ±1%. Since the steady-state core power and the loop flows are nearly identical to those used by the current evaluation model, the controlling hot leg initial temperature is also effectively unchanged. The maximum hot leg temperature variation will be within 1 F. This difference will be further diminished by steam generator heat removal. As a result the controlling parameters in the initial subcooled depressurization phase of the SBLOCA will be unaltered. The reactor trip signal and pump trips will occur at the same time in the transient.

The impact of the fuel bundle resistance changes will be even less during the pump coast-down and natural circulation phase because the flows during this phase are much reduced over those occurring earlier in the transient. Significant margins exist such that CHF will not be exceeded. All of the initial stored energy in the fuel will still be transferred to and removed by

the steam generators. Therefore, core resistance variations [illegible] not change the fuel thermal transient or impact the existing evaluations.

Changes in the initial fuel temperature add or subtract overall energy from the RCS. The initial fuel energy is removed from the fuel pin during the reactor coolant pump coast-down phase and rejected from the system via the steam generators. Therefore, the initial fuel enthalpy of operation has virtually no impact beyond the loop coast-down period. The core energy content during the loop draining and boiling pot mode will be identical to the current licensing base.

The fuel pin internal gas fill pressures are similar to the Westinghouse values, but may differ slightly. The internal gas pressure could affect the fuel-cladding gap dimensions and rupture time. However, the fuel temperatures approach the system saturation temperature within a fraction of a minute following reactor trip. The impact of gap differences at low temperatures is negligible. Since the SBLOCA temperatures peak at approximately 1500 F, the impact of a rupture or difference in rupture timing would be negligible.

As a final point, the allowable local power levels, core peaking, for core elevations at or above 8 feet will not be altered because of the loading of BWFC fuel. Thus, the axial power profile used by Westinghouse in the SBLOCA analyses remains bounding. This ensures that the thermal load imposed on the fuel during a temperature excursion remains conservatively modelled. The thermal results, cladding temperatures, for the present evaluations are therefore conservative for BWFC fuel.

In summary, the core resistance variations will not affect the loop flows such that the controlling hot leg temperature or CHF points are altered. The steam generator heat removal rate during

the flow coast-down period will compensate for any initial fuel stored energy fluctuations. All controlling parameters in the phases following the pump coast-down and natural circulation phase will be unchanged. Therefore, since the overall RCS geometry, initial operating conditions, licensed power, and governing phenomena are effectively unchanged, the existing licensing calculations remain valid for BWFC reload fuel.

The position of the BWFC is that the existing SBLOCA calculations contained in the McGuire and Catawba FSARs are valid and bounding for the planned BWFC fuel reloads. The reactor coolant system, decay heat levels, and other system controlling parameters remain unchanged by the reload fuel. A significant safety margin exists between the calculated results and 10 CFR 50.46 limits. Adequate core cooling has already been demonstrated and does not need to be repeated. The present SBLOCA evaluation calculations remain valid for the McGuire and Catawba fuel reloads supplied by the Babcock & Wilcox Fuel Company. No additional SBLOCA evaluations are required.