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Dr. J. Nelson Grace, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Grace:

ADDITIONAL INFORMATION CONCERNING GAS ENTRAINMENT IN THE PRIMARY HEAT TRANSPORT SYSTEM (PHTS)

Enclosed are amended Preliminary Safety Analysis Report (PSAR) pages to Clinch River Breeder Reactor Plant PSAR Chapters 4 and 5 that provide additional information concerning the potential for gas entrainment in the PHTS. These pages are in response to questions raised by the Nuclear Regulatory Commission staff reviewer and will be included in the next amendment to the PSAR.

Questions regarding the enclosure may be addressed to Mr. D. Hornstra (FTS 626-6110) or Mr. D. Robinson (FTS 626-6098) of the Project Office Oak Ridge staff.

Sincerely,

Robert L. Woolley for

John R. Longenecker
Acting Director, Office of
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the Doppler feedback in CRBRP. Reducing the Reference 9 uncertainty for the already conservative neglect of global temperature-importance weighting results in an extrapolated Doppler uncertainty of less than ± 10% (1σ) for CRBRP.

Further confirmation of the accuracy of LMFBR Doppler constant predictions is provided by the small-sample measurements in zero power critical mockups. Small, heated-sample Doppler constant measurements have been performed in the Zero Power Plutonium Reactor (ZPPR) critical assemblies simulating the CRBRP core configuration. The analysis of the Doppler constant experiments in ZPPR-2, 3, and 5 are reported in References 10, 11, and 12, respectively. Although these small-sample measurements do not represent a direct experimental determination of the total core Doppler constant, the good agreement between calculated and measured values does provide substantial confidence that the U-238 resonance parameters in the core spectrum are accurately predicted. From the total of 52 small-sample Doppler measurements throughout the core under a variety of reactor conditions (flooded, voided, control rods inserted and withdrawn, etc.), the mean calculation-to-experiment ratio is 0.98 (slightly conservative underprediction of Doppler) with an uncertainty of ± 5.4% (1σ).

The CRBRP Doppler constant uncertainty, based principally on the aforementioned SEFOR evaluation, and supported by the small-sample measurements in ZPPR, is ± 10% (1σ). This value is intended for use in operational and design transient evaluations within the reactor design duty cycle. The minimum Doppler feedback in anticipated design transients and the extremely unlikely class of events are based on Doppler constants with -2σ (80% of nominal Doppler) and -3σ (70% of nominal Doppler) uncertainties, respectively.

4.3.2.3.2 Sodium Void Worth

(see Section 4.4.4.1)

The sodium void worth relates the change in neutron multiplication to the presence of voids in the sodium coolant. Small, distributed voids such as gases entrained in the coolant are adequately treated by use of the sodium density coefficient (Section 4.3.2.3.3). Large voids in a region of the reactor or complete voiding is an extremely unlikely situation. In the following discussion, the reactivity associated with this latter type of voiding is developed for use in accident analysis in Chapter 15.

Figure 4.3-28 is a flow chart showing the method for calculating the sodium voiding reactivity worth. Cross sections are processed both with sodium and without sodium to properly account for resonance self-shielding and the change in spectrum when sodium is removed.

51

The amount of entrained gas at equilibrium shall be limited to less than 1% volume to maintain reactivity changes below 5¢. (This limiting entrainment level is lower than the level that would impair reactor heat transfer to the extent that fuel lifetime would be shortened.)

The above discussion refers to demonstrating the ability of the rods to meet their design requirements when subjected to the various categories of design transients. Such topics as transient effects due to rod failures and continued operations with failed rods are presented in Section 15.4.

4.4.3.7 Potentially Damaging Temperature Effects During Transients

In general, a single anticipated event is not damaging to the reactor structures; it is the total sum of all the occurrences (in this event classification) over the particular lifetime that may cause the structure to approach its design limit. As mentioned in Section 4.4.2.9, the current cladding requirement for fuel pins is that considering all normal and anticipated events, the cumulative cladding damage must not preclude the capability to survive at least one of the worst unlikely events without loss of cladding integrity.

Maximum 3σ fuel and blanket hot rod temperatures during various limiting core design events of the plant duty cycle which are described in Appendix B have been analyzed. Detailed temperatures for various axial and radial positions along the rods have been evaluated in determining the core design adequacy as described in Section 4.2.1. Conservative assumptions used in calculating these temperatures are described in Section 15.1.4. These assumptions include: full power operation, hot rod analyzed in highest power and temperature fuel assembly of all core conditions, worst case Doppler coefficient with uncertainties included, a 200 millisecond delay between trip signal and the start of control rod insertion, 3σ hot channel factors, the single most reactive control rod assumed to be stuck in the withdrawn position for both sets of control rods, highest core pressure drop, and the most rapid flow coastdown of the primary pumps following pump trip. In addition, Section 15.1.4 shows that of the safety related events, both of the overpower and undercooling
58 type, the Safe Shutdown Earthquake (SSE) event (60¢ step reactivity insertion occurring under SSE conditions) results in the highest core temperatures. The conclusion of the Chapter 15.0 safety evaluations and the Section 4.2.1 design evaluations is that the design changes incurred in going from the homogeneous to the heterogeneous scheme are not expected to significantly change the design or safety capability of the core.

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4.4.3.8 Thermal Description of the Direct Heat Removal Service (DHRS)

The thermal description of the Direct Heat Removal Service will be found in Section 5.6.2.

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4.4.4 Testing and Verification

At the present state-of-the-art in reactor design, scale model
51 tests of reactor flow systems provide the most useful tool for studying reactor hydraulics. Pressure drop through complicated flow paths, thermal

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An evaluation has also been made to determine if expected levels of gas entrained in the sodium could impair heat transfer in the reactor. It was found that at the limiting value of one volume percent established for reactor neutronics effects (Section 4.3.2.3.2) there is no significant effect on heat transfer. The expected entrainment level (Section 4.4.4.1) is much lower than this limiting value.

inlet plenum as a function of the plenum height, inlet nozzle discharge angle, and the elevation of the nozzles relative to the Core Support Structure. The test served a scoping purpose for use in designing the 1/4-scale IPFM test previously described.

The gross flow patterns were predictable and produced no unusual or unexpected results.

Integral Reactor Flow Model, Phase I Testing - Outlet Plenum Feature Flow and Vibration Test

This test: 1) measured the velocity pattern in the outlet plenum and in the vicinity of the major outlet plenum structures; 2) determined the pressure drop characteristics of the outlet plenum and major outlet plenum structures; 3) determined the mixing characteristics and transport times in the outlet plenum and at locations of probable hot/cold interfaces; 4) evaluated flow induced vibration characteristics of selected outlet plenum structures; and 5) evaluated the gas entrainment characteristics of the suppressor plate.

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Integral Reactor Flow Model, Phase II Testing

The purpose of this test which is in progress, is to verify the final design for hydraulic and vibration performance using the integral reactor flow model. Those components used in the Phase I testing, whose design has changed to the extent that hydraulic and vibration performance is influenced, will be modified in a subsequent Phase II.

Outlet Plenum Flow Stratification Test

The Outlet Plenum Flow Stratification Test was performed in a model of 0.55 scale simulating a 120° sector of the CRBRP reactor vessel outlet plenum, containing a portion of the UIS, and an outlet main coolant pipe.

Some important conclusions and observations derived from the evaluation of the test data (Ref. 60) are:

1) The transient temperature response at the outlet nozzle is less severe than that predicted by the plant simulation model in the DEMØ code, thus demonstrating that this UIS design goal is satisfied.

2) The standard height UIS chimneys resulted in less severe transient temperature ramp rates at the outlet nozzle than shorter chimneys.

3) The nominal prototypic gap beneath the UIS skirt of 1.0 inch resulted in less severe outlet nozzle transient temperatures than either larger or smaller gaps.

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The gas entrainment tests were performed to determine the amount of cover gas that might be entrained at the free surface inside the CRBRP reactor vessel. Using conservative Weber Number modeling of the entrainment process, the average reactor vessel outlet void fraction when at normal liquid level and scaled to 115 percent of CRBRP average full flow is 1.25×10^{-4}. The maximum void fraction measured at any reactor vessel outlet nozzle for this same case and extrapolated to 115 percent flow was 3.0×10^{-4}.

In addition to determining primary entrainment phenomena within the reactor vessel, tests were run to determine the equilibrium PHTS void fraction that results for a given gas entrainment rate in the reactor vessel. These tests were based on the conservative assumption that all gas removal occurs in the outlet plenum of the reactor. This assumption results in the highest equilibrium void fraction in the PHTS.

Based on the entrainment rate that gave the void fraction of 1.25×10^{-4} and Weber Number modeling of the entrainment process, the equilibrium void fraction in the PHTS at 115 percent of rated flow is 2.0×10^{-3}. Gas removal by the IHX vent and the primary pump standpipe bubbler will reduce this level.

The test results, as well as a complete description of the subject testing, are presented in Reference 76.

73. F. C. Engel, R. A. Markley and B. Minushkin, "Buoyancy Effects on Sodium Coolant Temperature Profiles Measured in an Electrically Heated Mock-up of a 61-rod Breeder Reactor Blanket Assembly", ASME 78-WA/HT-25.

74. F. C. Engel, R. A. Markley and B. Minushkin, "Heat Transfer Test Data of a 61-rod Electrically Heated LMFBR Blanket Assembly Mockup and their use for Subchannel Code Calibration", in Fluid Flow and Heat Transfer Over Rod or Tube Bundles symposium of ASME 1979 Winter Annual Meeting, pp. 223-229, December 1979.

75. F. C. Engel, B. Minushkin, R. J. Atkins and R. A. Markley, "Characterization of Heat Transfer and Temperature Distributions in an Electrically Heated Model of an LMFBR Assembly, to be published in a special issue of Nuclear Engineering Design.

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76. D. R. Dickson, "Final Tests of Entrainment and Equilibrium Void Fraction in the Integral Reactor Flow Model", Volume II of HEDL-TME-80-59, November 1982, (Availability: US DOE Technical Information Center)

(11 ft/sec) at design flow from the reactor vessel. Gas entrainment within the pump is minimized by designing and testing, to ensure that all pump parts which need to be submerged during operation at pony motor speed or restart to pony motor speed are located below the minimum sodium level.

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The maximum oxygen content in the primary system sodium is specified to be ≤2 ppm at 800°F or above and ≤5 ppm below 800°F. This level of sodium impurity will not affect the pump operating characteristics.

The biological shielding for the PHTS sodium pumps as well as the pump assemblies is designed to withstand the loadings associated with the Safe Shutdown Earthquake (SSE) and the transient overpressures for extremely unlikely plant conditions. The analyses required to demonstrate this treated the PHTS pump as a Class I component in accordance with the rules of the ASME Code Section III and modifying RDT Standards.

The biological shielding for the PHTS sodium pumps is provided by (1) an annular shield tank which surrounds the pump shaft, (2) the pump shaft itself which is designed to provide an integral part of the shield requirements, (3) the pump support structure which is part of the operating floor, and (4) special precautions to preclude streaming along the instrumentation penetrations. The annular shield assembly is integral with the top closure flange of the pump pressure boundary containment vessel which is designed in accordance with the ASME Code for Class 1 nuclear components. The pump shaft supporting assembly and the annular shield structural assembly are supported on the pump tank flange which is mounted on a pump support ledge designed into the operating floor pump motor well. The design of this joint provides for the dual function of resisting static and dynamic loads and provides a seal at the boundary between the pump atmosphere and the RCB atmosphere that consists of a double metallic "o" ring seal with argon purge gas in the annulus between rings.

For the PHTS pump SSE seismic analysis, a 2% damping value was used. The SSE loadings were considered to occur in conjunction with a plant trip. Following the SSE, the Intermediate Heat Transport System, Steam Generator System, and Steam Generator Auxiliary Heat Removal System must provide for removal of stored and decay heat. The primary pump is designed to maintain pony motor flow without loss of structural integrity after the SSE. Computer programs, such as SAP IV and ANSYS, will be utilized to perform seismic analyses on the primary pumps. Descriptions of these computer programs can be found in Appendix A.

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The pump standpipe bubbler also provides a vent for gas de-entrained from the PHTS. The maximum gas flow that the standpipe can accommodate without flow instability corresponds to an entrained gas volume fraction of 3×10^{-4}. This volume fraction has been shown to have no impact on pump performance or heat transfer in the IHX. Outlet plenum tests which give expected gas entrainment levels are discussed in Section 4.4.4.1.

5.3.3.4 Valve Characterisitics

The essential characteristics and features for the check valve design to satisfy the system operating requirements are included in Table 5.3-13.

The CRBRP Cold Leg Check Valve (CLCV) is hydraulically similar to the FFTF 16" check valve. As noted in Reference 62, extensive tests have been performed on the FFTF valve to confirm its hydraulic performance. In operation at FFTF, no problems due to gas accumulation in the CLCV have been identified. Secondly, tests performed on the FFTF hot leg isolation valve verified the self-venting characteristics (that any accumulated gas will be swept out) of the main valve body of both valves. Since the valve bodies of both CRBRP and FFTF CLCV's and the FFTF hot leg isolation valve are geometrically equivalent, the results of these tests are applicable to the CRBRP CLCV. Air deliberately introduced into the valve was quickly entrained and removed by the turbulent fluid at pipe flow velocities greater than 7 fps, which correspond to a 2 fps (average) velocity in the valve. The experimenters noted that, "This is the approximate limiting value for air removal from pipelines as established by previous research" (Reference 63). Under full flow conditions the flow velocity in the CLCV inlet pipe is about 25 fps and about 6 fps (average) in the valve body. At 40 percent flow the velocity in the valve body is still above the 2 fps level, and at this reduced flow rate there is little likelihood for any gas entrainment at the principal gas entrainment site (the reactor vessel outlet plenum). Thus, no gas accumulation is expected in the main body of the CLCV.

Although no mechanism has been identified which would allow the de-entrainment of gas in the valve, an evaluation has been performed with the following conservative assumptions. The dome on the top of the check valve is initially full of gas at the conditions consistent with the thermal hydraulic design condition, the pumps are tripped, and the pressure in the valve has been minimized by assuming that the sodium level in the reactor is at the minimum safe level. The evaluation showed no break of siphon. Therefore, gas accumulation in the check valve would not be a problem for CRBRP.

References:

54. P. Soo, "Selection of Coolant-Boundary Materials for the Clinch River Breeder Reactor Plant", WARD-D-0010, August 1974.

55. Deleted

56. Deleted

57. P. Soo (Compiler), "Analysis of Structural Materials for LMFBR Coolant-Boundary Components - Materials Property Evaluations", WARD-3045T3-5, November 1972.

58. Deleted

59. R. A. Leasure, "Effect of Carbon and Nitrogen and Sodium Environment on the Mechanical Properties of Austenitic Stainless Steels", WARD-NA-94000-5, December 1980.

60. ES-LPD-82-007, CRBRP Engineering Study Report, "CRBRP Transition Joints", April 1982.

61. ES-LPD-82-008, CRBRP Engineering Study Report, "CRBRP Materials Data Base", May 1982.

62. J.W. Ball and J.P. Tullis, "Report of Hydraulic Tests, 16-inch Cold Leg Check Valve," HydroMachinery Laboratory Report No. 36, Colorado State University, June 1974

63. J.P. Tullis, "Flow Test of the 28-inch HLI Valve," Hydro Machinery Laboratory Report No. 33, Colorado State University, November 1973.