



Department of Energy
Washington, D.C. 20545

Docket No. 50-537
HQ:S:82:020

APR 21 1982



Mr. Paul S. Check, Director
CRBRP Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Check:

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION - CHEMICAL ENGINEERING

Reference: Letter, P. S. Check to J. R. Longenecker, "CRBRP Request For Additional Information," dated March 23, 1982

This letter formally responds to your request for additional information contained in the reference letter.

Enclosed are responses to questions CS 281.1, 2, 3, 4, 5, 7, 10, 11, and 13 that will also be incorporated into the PSAR in Amendment 68, scheduled for April 30. The remaining questions from the reference letter (CS 218.6, 8, 9, and 12) will be provided under separate cover by April 23.

Sincerely,

John R. Longenecker, Manager
Licensing & Environmental
Coordination
Office of Nuclear Energy

Enclosure

cc: Service List
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Question CS281.1 (4.2)

Degradation of CFR fuel cladding may be caused by (a) selective leaching of nickel and chromium due to sodium exposure, (b) formation of double oxides and intergranular attack by sodium corrosion, (c) loss of mechanical strength due to loss of carbon and nitrogen, or (d) fission product attack on the cladding internal surface. In your cladding strain and Cumulative Damage Function Analysis of the cladding integrity, you assumed uniform cladding wastage. Provide additional cladding performance analysis including the effect of localized attack and the loss of strength due to interstitial transfer and the formation of a ferrite layer on the cladding surface, to demonstrate that the cladding integrity will be maintained under normal operating and design basis accident conditions.

Response:

As described in Reference 58 of PSAR Section 4.2, the CDF procedure incorporates those phenomena which either degrade the strength of the cladding or which reduce its effective load bearing cross section. Specifically, the following component effects are considered as explicit functions of the prevailing local environmental conditions:

- (a) Sodium corrosion (i.e., the physical loss of cladding to flowing sodium).
- (b) The formation of the ferrite layer at the cladding/sodium interface (i.e., a substrate devoid of substitutional alloying elements, e.g., Cr and Ni).
- (c) Fission product attack at the inside surface of the cladding.
- (d) The change in the cladding's bulk content of interstitial elements (i.e., C and N).

In application, the change in the interstitial content is taken to affect the cladding's mechanical properties; the other phenomena serve to reduce the effective load bearing cross section of the cladding.

Since the above phenomena have already been factored into the cladding performance calculations, additional analyses are not required.

Question CS281.2 (4.2)

In the PSAR, you state that the design objective of the primary control rod system (PCRS) is to achieve a service life of minimum of 17,000 ft. travel and 732 scrams. To improve the resistance to wear and friction, carbide coating was applied on certain PCRS component surfaces. Describe the design criteria and limits established for maximum carbon transfer in the primary coolant system due to the presence of carbide coating. Include in your analysis the problem of carburization of the fuel cladding surface which leads to cladding embrittlement due to the use of chromium carbide on the coupling head of PCRS.

Response

The principal use of chromium carbide coatings in the CRBRP is on removable assembly load pads, as described in PSAR Section 4.2.1.2.2 (p. 4.2-48). This coating is used on fuel, blanket and control assembly ducts. Reference is also made to the possible use of chrome carbide in the secondary driveline/control assembly coupling in PSAR Section 4.2.3.1.7 (p. 4.2-254), but the potential surface area exposed to flowing sodium in this application would be small in comparison with the load pad surfaces. This coating is not used in primary driveline couplings.

There are no design criteria and no "limits" identified or established for carbon transfer in the primary coolant system that relate specifically to carbide coatings present on certain parts of the PCRS. The rationale for not considering carburization from this source as a significant issue is given below.

The concern over the possibility that carbide coatings would produce carburization of core component arises, not because of the chromium carbide itself, but largely because of free carbon that may be present in the binder material.

Free carbon is present in the raw material fed into the detonation gun. During the coating process, much of this is oxidized and lost to the atmosphere. Tests for 2000-hours in vacuum have demonstrated that at 1160°F, the Cr/C ratio in the coating is sufficiently high to favor the stabilization of Cr_7C_3 . Similar exposures in sodium, however, produced Cr_{23}C_6 as the predominant carbide presumably through some initial loss of the free carbon.

After each exposure several carbide phases were present; in the former case " $M_{23}C_6$ " accounted for only 10% of the carbide total, while in the latter it was about 70% of a $M_{23}C_6$, M_7C_3 , M_3C_2 mix.

Although definitive experiments have not been conducted to determine whether or not significant core component carburization is possible through carbon release from the chromium - carbide coated regions of the PCRS, the following statements may be made:

- o The surface area of the PCRS carbide coatings is a negligible fraction of the total surface area exposed to sodium in the primary system. Carbon release from these areas would not be discernible from the general carbon release within the core.
- o Carbon loss from the coatings is not likely to be at a level that would increase the carbon activity of the sodium to a level that would produce carburization on the fuel cladding. The coating contains a substantial amount of free chromium from the nichrome binder and continued breakdown of M_7C_3 carbides and other types is more likely to produce additional $M_{23}C_6$ within the 3-mil thick coating, than to result in significant carbon release to the sodium.

General Reference Q281.2-1:

G. A. Whitlow, R. L. Miller, S. L. Schrock, and P. C. S. Wu, "Sodium Compatibility Studies of Low Friction Carbide Coatings for Reactor Application", Corrosion V30, No. 12 p.p. 420-426 (1974).

Question CS281.3 (4.2)

Carbides formed by Nb and Ti in the alloy 718 are thermodynamically more stable than the chromium carbides formed in the type 316SS fuel cladding. Provide analysis that the presence of the Inconel 718 in the CRBR core region would not lead to decarburization and subsequent loss of strength of the fuel cladding.

Response:

It is true that from a thermodynamic standpoint Nb and Ti carbides will form in preference to Cr carbides; however, there is essentially no "free" Nb or Ti in Alloy 718. These elements are already tied up in a very stable form as NbC, Ni_3Nb or as δ' and δ'' the coherent precipitates ($\text{Ni}_3(\text{Al Ti Nb})$ and Ni_xNb , respectively) responsible for strengthening the alloy.

The fuel cladding exposed to hot spot temperatures (1250-1300°F) is expected to experience decarburization to surface levels of about 50 ppm. The rate of carbon loss from the cladding will be controlled by the breakdown of M_{23}C_6 in the 316SS which is virtually independent of the sodium carbon activity at the low activities expected in CRBR. The effect of adding another carbon sink, to the already large sink available (i.e., cold leg regions, cold trap, IHX, mass transfer deposits) would have a negligible effect on carbon loss from the cladding.

The supportive data on expected sodium carbon activities and decarburization rates can be found in the Nuclear Systems Materials Handbook.

In conclusion, Alloy 718 cannot be considered as a carbon sink in the CRBR primary heat transport system that would in any way influence decarburization of the fuel cladding.

Question CS 281.4(5.3)

On page 5.3-16 of the PSAR you state that sodium leak tests have shown corrosion rates of steam generator tubes by Na-water reaction product; e.g., NaOH, to be 0.12 mils/hr at 1050°F, 1000 vppm H₂O and 1.2 v/o O₂ or less, and this accelerated corrosion from the presence of water vapor and sodium is acceptable in that propagation of a leak from corrosion at this rate will not significantly affect plant capability to safe shutdown and maintain safe shutdown conditions. Provide the technical basis and analysis for the above statement, and demonstrate the validity of leak before break criteria in your analysis.

Response

- a) The steam generator tubes are not addressed in the section referenced. Information regarding the steam generator tubes is found in PSAR Section 5.5. Additional information regarding the effects of sodium-water reaction products on the steam generator tubes can be found in the response to Question 281.8.
- b) The quote from Section 5.3 on corrosion rates deals with the primary heat transport system. WARD-D-0185, "Integrity of Primary and Intermediate Heat Transport system Piping in Containment" provides the basis for the Project's conclusion that rapid catastrophic failure of the in-containment HTS piping need not be considered in the plant design basis. Section 2.2 provides the Piping Integrity Rationale; Project standards for Heat Transport System Piping are summarized in Section 3.1.5, and Section 5.5 and provide additional information dealing with piping corrosion due to leakage. The rationale for the leak before break concept is addressed throughout WARD-D-0185.

Question CS 281.5 (5.3, 9.3)

The heat transport system liquid metal (Na and NaK) chemistry is selected to minimize corrosion and to ensure the fuel cladding integrity and to prevent radiation leakage. In the PSAR, you state that periodic analysis of the liquid metal chemistry is performed to verify that the Na and NaK quality meets the proposed specification. The liquid metal purity is maintained by the use of cold traps. Describe the frequency and chemical and radiochemical analysis to be performed for liquid metal analysis and the criteria for cold trap replacement to ensure the liquid metal quality meets the proposed specifications.

Response

Sodium

PHTS and IHTS sodium samples will be taken by the multipurpose sampler (MPS) for laboratory chemical analyses. Chemical analysis will be performed monthly with the following exceptions:

1. During periods of anticipated changes (initial fill; refueling, maintenance activities) in impurity levels, the frequency will be increased.
2. When it has been established that an element impurity level is no longer changing after several years of plant operations, the frequency of analysis may be reduced.

Indication of oxygen and hydrogen concentrations by the plugging meter technique will be obtained routinely once a day, or if there is no apparent change over a long period of operation, at a decreased frequency.

The determination of oxygen, hydrogen, and carbon concentrations is by the equilibration method in which metal tabs or wires are exposed to flowing sodium in the MPS for a time sufficient to establish equilibrium with respect to the impurities. Subsequent measurement of the impurity concentration in the wire or tab is used to determine the impurity concentration in the sodium.

Uranium and plutonium concentrations in liquid metal are determined fluorometrically and by alpha assay, respectively, using a vacuum distillation residue. Tritium concentrations are measured using liquid scintillation counting techniques.

The concentrations of various elements (Ca, Co, Cr, Cu, Fe, Li, Mg, Mn, Mo, Ni, K, Cs, Rb, Si, B) are determined using standard techniques of atomic absorption or flame emission spectrophotometry. Several of the analysis are performed with a sodium distillation residue.

The current basis for developing CRBR procedures for chemical and radiochemical analyses of liquid metal is RDT Standard F 3-40T which contains typical analytical procedures.

Forced Circulation Sodium Cold Trap Replacement Criteria

The criteria for forced circulation cold trap replacement is an increase in cold trap Δp at design flow. This limiting Δp is currently estimated to be 22 psid. When this limiting Δp is reached, cold trapping may be continued at decreased flows provided impurity limits are not exceeded.

NaK

It is expected that no sampling will be necessary because the diffusion cold traps have been sized to clean up the system initially and to remove the oxygen and hydrogen introduced when each system is opened four times during the life of the plant. If operating experience indicates leakage or additional contamination, a decision would be made at that time to take a sample. Provisions will be available for obtaining NaK samples from the NaK storage vessels. If this sample indicates that the cold trap is not maintaining acceptable oxygen and hydrogen concentrations, a decision may be made to replace the diffusion cold trap.

Chemical analysis of the NaK samples would be performed in the laboratory with methods similar to those for sodium. There is no plan for radiochemical analysis of the NaK.

Response to NRC Question CS281.7Question

Provide a Failure Mode and Effects Analysis (FMEA) to demonstrate that the sodium dump subsystem of the CRBR steam generating system is designed such that no single failure of the isolation and dump equipment causes the loss of shutdown heat removal capability, including that a single failure does not cause two sodium dump valves in the same dump path to open.

Response

The IHTS has three independent loops, any one of which can transport all the residual and decay heat from the reactor to the steam generator system. There are two normally closed valves in series on each dump line. A single failure of one dump valve actuator would not cause the loss of an IHTS loop. Failure of both dump valve actuators, causing the valves to open, would be necessary in order to lose an IHTS loop. CRBRP is designed to accommodate failure of two sodium dump valves in series and retain the capability in the remaining loops to perform its shutdown heat removal function.

A failure modes and effects analysis performed on the IHTS has considered the following sodium dump valve failure modes:

- o Failure to open or close due to solenoid failure or loss of compressed air.
- o Opens falsely due to operator error or I&C failure.
- o Loss in integrity of the sodium boundary at the valve.

The sodium dump valve control for each loop is separate and independent. Within each loop the control of the upstream sodium dump valves is independent of an IHTS loop, interlock features are provided on the dump valve operation. IHTS dump valve controls are located on the Main Control Panel with the following features:

- o Push-button controls are located on the Main Control Panel by loop and properly designated.
- o There will be separate protective covers over each of the two controls for each loop to preclude inadvertent operation.
- o An interlock requires operation of downstream valves before upstream valves can be opened. (Interlock bypass capability exists to allow valve exercising for maintenance and testing.)

These features preclude the simultaneous draining of all three loops.

Question CS 281.10 (9.1)

Describe the chemistry and radiochemical limits, monitoring frequency, and criteria for cold trap replacement to ensure the sodium purity in the spent fuel storage pool. Provide the basis for establishing these limits.

Response

The limits for sodium purity in the "Spent Fuel Storage Pool" which is known as the Ex-Vessel Storage Tank (EVST) are 5 ppm oxygen and 0.4 ppm hydrogen. The basis for establishing these oxygen and hydrogen limits is corrosion, which is temperature dependent. The corrosion effects due to these impurity levels for the EVST (operating at 500°F) are comparable to those for 2 ppm O₂ and 0.2 ppm H₂ for the reactor sodium (operating at 1,000°F).

The radiochemical limits currently imposed for the EVST sodium are 10 ppb Pu and U. These limits are established to minimize potential maintenance problems resulting from contamination of components and to minimize contamination resulting from potential sodium spills.

EVST sodium samples will be taken by the multipurpose sampler (MPS) for laboratory chemical analyses. Chemical analysis will be performed monthly with the following exceptions:

1. During periods of anticipated changes (initial fill; refueling, maintenance activities) in impurity levels, the frequency will be increased.
2. When it has been established that an element impurity level is no longer changing after several years of plant operations, the frequency of analysis may be reduced.

Indication of oxygen and hydrogen concentrations by the plugging meter technique will be obtained routinely once a day, or if there is no apparent change over a long period of operation, at a decreased frequency.

The criteria for forced circulation cold trap replacement is an increase in cold trap Δp at design flow. This limiting Δp is currently estimated to be 22 psid. When this limiting Δp is reached, cold trapping may be continued at decreased flows provided impurity limits are not exceeded.

Question CS 281.11 (9.8)

Provide any analysis and any experimental results to demonstrate that the plugging meter, on-line impurity monitoring device, can provide quantitative results on sodium chemistry. Include correlation of these results with those obtained by the vanadium wire equilibration device (VWED) technique.

Response

Plugging temperature indicators (PTI) have been in use for several years as on-line monitoring devices for sodium systems at EBR II and FFTF. A PTI of the type planned for use in CRBR was developed and tested by HEDL and is currently in successful operation at the FFTF.

Table CS281.11-1 contains references to documents published on various experimental results and other studies. These reference documents provide the basis for utilization of the plugging meter method to provide sodium chemistry information.

The PTI is utilized to indicate the saturation (plugging) temperature for all the species (impurities) present in the sodium stream which precipitate out at that temperature. The plugging temperature cannot be correlated directly to the concentration of any specific impurity such as O_2 or H_2 . However, as long as the plugging temperature is below the maximum permissible, then the corresponding oxygen and hydrogen impurities in the sodium stream do not exceed the specified limits of 2.0 ppm and 0.2 ppm, respectively.

TABLE CS281.11-1

Reference Documents for Plugging Temperature
Monitoring of Sodium Systems

1. HEDL TME 73-41, Prototype Application Loop "PAL" 1 yr of Operations, Interim Report, J. J. McCown (3-1973).
2. HEDL TME 74-24, Prototype Application Loop "PAL" 2 yrs of Operations, J. J. McCown (3-1974).

Question CS281.13

The NRC position and clarifications on post accident sampling capability of light water reactors is presented in Item II.B.3 of NUREG-0737. Provide the design bases and criteria for a system that will provide post accident sampling and analysis capability for the CRBR which will be equivalent to the functional requirements for light water reactor plants.

Response

Appendix H of the PSAR provides CRBRP's evaluation of and resolution to the requirements delineated in NUREG-0718. NUREG-0718 defines requirements of NUREG-0737 applicable to Applications for Construction Permits. Item II.B.3 is addressed in the Appendix H.

Means to assess the degree of core damage in the event of an accident will be provided. However, the approach to be utilized has not yet been finalized.