UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 3 0 1980

Admiral H. G. Rickover, Director
Division of Naval Reactors
U. S. Department of Energy
Washington, D. C. 20510

Dear Admiral Rickover:

SUBJECT: CONTINUED OPERATION OF SHIPPINGPORT LWBR

In your letter of March 10, 1980, you described briefly your plans to continue operation of the Light Water Breeder Reactor (LWBR) core at the Shippingport Atomic Power Station beyond the 18,000 effective full power hours (EFPH) originally planned. You asked that any NRC comments on such operation be given to you by April 30, 1980.

In our Safety Evaluation Report and Supplement (NUREG-0083 and Supplement 1) of July and November 1976, respectively, we reported the results of our review of the proposed LWBR operation. At that time it was expected that the core would be operated for about three years, the time needed to attain the performance objective of at least 18,000 EFPH. The three-year scheduled operating life was a factor that was considered in some of our review areas.

As a result of your March 10 request, we have considered the proposed continuation of operation as it might affect our conclusions in those areas, which are concerned with (1) the proposed augmented inservice inspection program, (2) reactor vessel irradiation, (3) control rod drive mechanisms, (4) burnup effects on the fuel, (5) steam generator tube integrity, (6) post- LOCA hydrogen control, (7) consequences of postulated loss-of-coolant and fuel handling accidents, and (8) loss-of-coolant accident calculation models.

We have also considered the effects of extended operation on (9) reactivity coefficients. Our comments are given in the Enclosure.

As a result of the TMI-2 accident, we have issued to the commercial nuclear power industry a number of bulletins, orders and reports. These have provided guidance to the industry as to improvements in nuclear power plant design and operation that we consider to be important in improving the safety of commercial nuclear plants. We realize that some of the matters addressed in those documents may

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not be applicable to the proposed extension of LWBR operations. However, we suggest that you give careful consideration to them and effect those that you judge to be necessary for the continued maintenance of the health and safety of the public.

Sincerely,

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Comments on Proposed
Extension of LWBR
Operation

APR 3 0 1980

ENCLOSURE

COMMENTS ON PROPOSED EXTENSION OF LWBR OPERATION

(1) Augmented Inservice Inspection

As noted in our Safety Evaluation Report of July 1976, we considered your proposed augmented in-service inspection program to be an acceptable alternative to installation of additional piping restraints. Such a program, if performed in accordance with ASME Code Section XI, requires periodic inservice inspections to be performed within 40 month intervals. The Code also permits an extension of the interval up to one year, for a total of 52 calendar months.

You expect the 24,000 EFPH point to be reached within this time, thus not requiring an ASME Code inspection prior to reaching this burnup.

We understand that you are evaluating an additional inservice inspection for the plant to support core operation beyond 24,000 EFPH. This inservice inspection will include visual and hydrostatic tests, and volumetric inspection of selected pipe welds in order to verify that there has been no degradation of the welds. We suggest that you give appropriate consideration in your evaluation to the ASME Code requirements when establishing your additional inservice inspection.

(2) Reactor Vessel Irradiation

With respect to the effect of increased radiation dosage on the reactor vessel integrity, only a negligible effect is anticipated. The reactor vessel brittle fracture analysis presented in the SAR was based on 21,000 EFPH of operation for the LWBR core. The analysis was extended to 24,000 EFPH and shows that the reference transition temperature (RTT) will increase only about 3 F. For 30,000 EFPH an RTT increase of only 9 F is predicted. The pressure-temperature limits for heatup and cooldown will be revised, prior to exceeding 21,000 EFPH, to reflect the change in RTT from the current value of 420 F to 429 F. We conclude that the small increase in the reactor vessel RTT will not appreciably change the crack initiation and arrest depths calculated for postulated LWBR accident conditions.

(3) Control Drive Mechanisms

We had reviewed, in 1976, the life expectancy of the LWBR control drive mechanisms. These components, installed during the LWBR conversion, were generally designed for a 15 year fatigue life. They have now been in service approximately three years, and have sufficient remaining fatigue life to operate up to 30,000 EFPH and beyond.

(4) Burnup Effects on Fuel

The extension of core operation to higher burnups than originally expected exposes the fuel and cladding to potentially greater damage. Greater fission gas release contributes to higher internal rod pressure and greater cladding stress. The potential for collapse of cladding may be increased due to larger gaps resulting from pellet axial movement. Pellet-to-clad interaction becomes significant and so does fuel rod bowing. Greater fuel rod growth could produce

interference between the rod and the baseplate. The continuation of buildup of crud on the outside of the fuel rod cladding will reduce heat transfer capabilities.

From the additional information your staff provided to us on April 25, 1980, it is apparent that you have considered these matters in your evaluation of extended operation. The continued operation of on-line radiation monitoring and periodic sampling of reactor coolant provide reasonable assurance that, should fuel failures occur, they would be detected at an early stage.

(5) Steam Generator Tube Integrity

We understand that there have been no leaks in the U-tube steam generators since they were repaired about ten years ago nor in the straight-tube units that began service in 1977. All present steam generators are operating at about one-half their rated loads, and all-volatile water treatment is being used to maintain secondary water quality. You have further stated that sampling for fluorine in the boiler water can detect leaks as small as one gallon per hour.

Based upon the above, we suggest that steam generator tube inspection be performed if there are signs of excessive leakage or other degradation.

(6) Post-LOCA Hydrogen Control

In Section 6.2 of our July 1976 Safety Evaluation Report, we noted that DNR has designed a back-up post accident hydrogen purge system for the LWBR containment building. However, due to (a) the long time anticipated following an accident until containment purging might be necessary in the event of failure of the redundant safety grade recombiner system (i.e., 79 days minimum) and (b) the limited anticipated operation of the plant (three years), DNR would purchase, fabricate, install and check out the system after a loss-of-coolant accident occurred. We found this commitment to be acceptable.

The analysis on which the system design and our conclusions were based was performed in conformance to Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." Compliance with Branch Technical Position CSB 6-2 assures that the post accident hydrogen control system design meets the requirements of 10 CFR Part 50.44 and Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident", which are the Commission's current design requirements for post-accident combustible gas control. Accordingly, we find the DNR commitment regarding the back-up hydrogen purge system to be acceptable for the expected extended duration of plant operations.

The accident at TMI-2 resulted in a large hydrogen release, in the containment building, due to reaction of the fuel cladding with reactor coolant water. The hydrogen released greatly exceeded the Commission's current

design requirements. Although current combustible gas control system design requirements have not been changed, the subject of revision of those requirements will be included in proposed rulemaking proceedings for accidents that may result in molten or degraded reactor cores. We have recently written a paper to the Commission, "Proposed Interim Hydrogen Control Requirements for Small Containments" - SECY-80-107, February 22, 1980, in which we reported on studies of the effects of large hydrogen releases in different types of containment buildings. We concluded that large dry containments of the type like the Shippingport containment were the least affected and would probably survive a TMI-2 type accident in a manner similar to that which was experienced at TMI-2, i.e., a moderate pressure spike due to hydrogen combustion with no loss of containment integrity or damage to systems important to maintaining the plant in a safe shutdown condition.

(7) Radiological Consequences of Accidents

In Tables 15-1 and 15-2 of our July 1976 Safety Evaluation Report, we listed the assumptions we used in evaluating the radiological consequences of postulated loss-of-coolant and fuel handling accidents. An operating time of three years was listed. This is the same operating time we assume in evaluating commercial plants; the fission product inventory used in our calculation is the equilibrium value. An extension of operating time past three years would not increase the applicable fission product inventory and, therefore, would not increase the calculated radiation doses for these accidents.

(8) LOCA Calculation Models

In the November 1976 Supplement to our Safety Evaluation Report we noted that our review of the FLASH-6 loss-of-coolant accident model and its application to the LWBR was limited to an audit of compliance with 10 CFR Part 50, Appendix K. Extending LWBR operation beyond the three years originally planned does not alter the validity of that audit.

As a result of the accident at TMI-2, we asked commercial reactor vendors to evaluate their small break analysis methods against available small break data (semi-scale, LOFT, TLTA). We suggest that DNR do the same, on a continuing basis, and improve those portions of the model that may lead to under-prediction of cladding temperature or oxidation.

(9) Reactivity Coefficients

Concerning the effects of extended burnup on reactivity coefficients, you have noted that the Doppler coefficient becomes less negative with burnup, due mainly to the change in power shape in the core caused by motion of the seed assemblies. Since these assemblies will not have reached their upward motion limit by 18,000 EFPH, the coefficient should not have reached its least negative value. The effect of the continued withdrawal of the

seed assemblies will be offset by the reduction in power so that the Doppler coefficient should remain within the bounds of the nominal values given in the LWBR Safety Analysis Report.

The moderator temperature coefficient becomes increasingly negative with burnup. However, reduction in power causes an offsetting decrease in magnitude of this coefficient. The result should be only a small change in the value of the moderator coefficient, which, in any case, remains negative.

You have also included a 25 percent uncertainty factor in these coefficients for additional margin in the safety analysis.

In view of these considerations we conclude that the core reactivity coefficients will remain within the bounds of the safety analyses.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 6, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: EXTENDED OPERATION OF SHIPPINGPORT LIGHT WATER BREEDER REACTOR

Dear Dr. Ahearne:

The Division of Naval Reactors, Department of Energy, in its letter of March 10, 1980, discussed its plan to operate the Light Water Breeder Reactor (LWBR) core at Shippingport Atomic Power Station beyond the 18,000 effective full power hours (EFPH) originally planned, and requested NRC comments by April 30, 1980 regarding the extended operation.

During its 241st meeting, May 1-3, 1980, the Advisory Committee on Reactor Safeguards discussed this proposal with representatives of the Westinghouse Electric Corporation (Bettis), Duquesne Light Company, the Division of Naval Reactors of DOE, and the Nuclear Regulatory Commission Staff. The Committee also had the benefit of the documents listed.

The Committee concurs with the NRC Staff's letter to Admiral Rickover dated April 30, 1980, which recommended that consideration be given to the bulletins, orders, and requests issued to the commercial nuclear power industry as a result of the TMI-2 accident.

Subject to the above, the Committee believes it to be acceptable to operate the Shippingport Atomic Power Station Light Water Breeder Reactor core to 24,000 EFPH as proposed.

Sincerely yours,

Milton S. Plesset

Milton S. Plesset
Chairman

References:
1. Letter from H. R. Denton, NRC, to Adm. H. G. Rickover, DOE Naval Reactors, Subject: Continued Operation of Shippingport LWBR, dated April 30, 1980
2. Letter from H. G. Rickover, DOE Naval Reactors, to H. R. Denton, NRC (NR:D:H.G.Rickover Z#818) Subject: Light Water Breeder Reactor - Plans to Continue Operation of the Present Reactor Core
3. NBI Log No. 0203-80/0081L, "Information Report Concerning Extended Operation of the LWBR Core at Shippingport"

cc: Admiral H. G. Rickover

Attachment (c) to State of PA Letter