C O P Y

September 7, 1960

Mr. E. T. Morris, General Manager
Westinghouse Electric Corporation
P. O. Box 1075
Pittsburgh 30, Pennsylvania

Dear Mr. Morris:

By letter dated July 12, 1960, you were advised that the modified high pressure experimental loop of the Westinghouse Testing Reactor described in WTR-40 was not to be fueled or operated without further authorization in writing from the Commission.

We have reviewed the proposed modifications to the high pressure loop and have found that operation of the Westinghouse Testing Reactor with the proposed modified high pressure experimental loop would not present any substantial change in the hazards to the health and safety of the public from those presented by the previously authorized operation of the reactor, and will present no undue hazards to the health and safety of the public. A copy of our analysis of the safety of the operation of the WTR with the modified loop is attached hereto.

In view of the foregoing, approval is hereby granted to Westinghouse to fuel and operate the modified high pressure experimental loop in accordance with WTR-40, dated June 17, 1960.

Sincerely yours,

H. L. Price, Director
Division of Licensing
and Regulation

Attachment

A154

WESTINGHOUSE TESTING REACTOR
MODIFICATIONS TO A HIGH PRESSURE LOOP

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Introduction

By letter, serial WTR-SS-597 dated June 17, 1960 and report WTR-40 attached thereto, Westinghouse has proposed to modify certain mechanical and hydraulic characteristics of one of the WTR high pressure experimental loops. These modifications are:

1. Use of larger orifice in the main flow line to increase the primary loop coolant flow rate capability from 50 to 100 gpm.
2. Use of a top entry U-shaped in-pile loop with a zircaloy-4 test section in place of the previously authorized type 316 stainless steel bottom entry reentrant thimble.

Hazards Evaluation

The increased flow rate capability is a requirement of the particular experiment to be inserted in the proposed loop, and introduces no additional safety considerations from those previously evaluated. The hazards aspects of the new U shaped in-pile loop and zircaloy-4 test section involves the possibility that the in-pile section could fail and the consequences that would result from its failure.

The type 304 stainless steel portions of the proposed loop were designed in accordance with the ASME Boiler and Pressure Vessel Code, Section I and the ASA Code for Pressure Piping, Section B 31.1. The proposed zircaloy-4 test section was designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section I. Since zircaloy is not a reference code material, the allowable stress values were established on the basis of the ASME code for non-ferrous materials and Code Case No. 1205-3 for pressure vessels of small diameter with no weldments. The yield and tensile strengths used were the minimum for annealed strip, and no advantage was taken of the increased mechanical properties which would result from fabrication and irradiation. We conclude that the design of the new in-pile loop and test section is conservative and adequate for the proposed operating conditions insofar as initial strength is concerned.

The possible effects of fast neutron embrittlement and corrosion with attendant hydrogen pickup and embrittlement on the mechanical strength and integrity of the zircaloy-4 test section have also been analyzed. For zircaloy, fast neutron irradiation causes an increase in the tensile strength, an increase in the yield strength and a corresponding decrease in ductility. Unpublished Bettis data indicate that for fast neutron exposure levels up to 6×10^{21} nvt, adequate ductility is retained. During the proposed nine month period of testing in the WTR, the zircaloy test section will receive a fast neutron exposure of approximately 1.2×10^{20} nvt. We thus conclude that no serious impairment in the important mechanical properties and integrity of the zircaloy test section would be expected from the fast neutron irradiation anticipated.

Zircaloy is readily oxidized by water or oxygen but forms a protective oxide film which inhibits further reaction. The rate of further oxidation is temperature dependent and, for temperatures which would exist in the in-pile test section, is extremely small and unimportant. Experience indicates that the service life of zircaloy in reactors is limited by hydriding rather than oxidation. The amount of hydriding of the material, which concentrates at the cold surface, depends on the initial hydrogen content of the alloy and the rate at which hydrogen is absorbed by the material at the operating conditions. The material of which this test section is made has been analyzed, and found to contain from 6 to 13 parts per million of hydrogen. Chalk River work indicates that hydrogen pick-up in reactor environments is limited to the amount of hydrogen formed by the oxidation reaction. This is small, approximately 0.3 ppm per year, for the WTR conditions. Hanford data indicate that the strength of zircaloy is affected little if at all by hydrogen concentrations less than 100 ppm at temperatures between 100° and 400° C. We thus would conclude that gross corrosion and hydriding of the test section and failure therefrom would not be expected under WTR conditions.

The proposed design of the in-pile test section and enclosed fuel assembly contains a large number of crevices between the zircaloy section and the close fitting thermal baffles and between dissimilar materials at the mechanical tube joints. With such a design it is possible that some crevice or galvanic corrosion may occur in localized areas. Available experience indicates no gross sensitivity to crevice or galvanic corrosion of zircaloy alone, or zircaloy to 304 or 410 stainless steel couples. However, the available experience is not sufficient to be fully confident that some local crevice or galvanic corrosion might not occur in the proposed experiment and, in fact, one of the purposes of the experiment is to examine this point. We conclude that although the likelihood of the zircaloy test section failing in service may be somewhat greater than the likelihood of failure of the previously authorized type 316 stainless steel in-pile thimble, the design of the modified in-pile loop and test section is adequate from a safety standpoint.

With respect to the possible consequences of this modified in-pile loop thimble failing, both the mass and enthalpy of the coolant in the test loop, and therefore the total disruptive mechanical energy released in the event of a loop failure, are less than that of the previously authorized loops. The in-pile section of the modified loop is located in the reflector region of the reactor rather than in the core as was the case for the previously authorized thimbles, so that any interaction between a failure of this loop and the WTR core would be expected to be less than that previously considered. The maximum increase in reactivity which could result from the displacement of the fuel contained is 0.1% delta k/k, compared with the 1% increase in reactivity previously in the event of an experimental loop failure. The power level of the modified loop experiment is 200 Kw compared with the 500 Kw previously authorized, thus the fission product inventory of this experiment would be less than that previously considered. Accordingly we would expect that the consequences of a failure of the modified loop would be considerably less than the consequences of a loop failure previously evaluated.

Conclusions

Based upon the preceding analyses, we conclude that operation of the Westinghouse Testing Reactor with the proposed modified high pressure experimental loop does not present any substantial change in the hazards to the health and safety of the public from those presented by the previously authorized operation of the reactor, and will present no undue hazards to the health and safety of the public.