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Research Reactors Division

THE OAK RIDGE RESEARCH REACTOR - SAFETY ANALYSIS

Volume 2
Supplement 3

Compiled by

D. H. Cook
T. P. Hamrick

APPROVALS

Reactor Operations Section Head	19 June 87 Date
Research Reactors Division	22 June 87 Date
ORNL Office of Operational Safety	26 June 87 Date
Department of Energy, Oak Ridge Operations Office	2 July 87 Date

OAK RIDGE NATIONAL LABORATORY

OPERATED BY MARTIN MARIETTA ENERGY SYSTEMS, INC.

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June 29, 1987

Mr. Joseph A. Lenhard, Assistant Manager
Energy Research and Development
Department of Energy, Oak Ridge Operations
Post Office Box E
Oak Ridge, Tennessee 37831

Dear Mr. Lenhard:

Volume 2, Supplement 3 to the Safety Analysis Report for the Oak Ridge Research Reactor

Reference: Letter to J. A. Lenhard from F. R. Mynatt dated June 22, 1987; Subject: Information Copy of Volume 2, Supplement 3 to the Safety Analysis Report for the Oak Ridge Research Reactor

Enclosed is Volume 2, Supplement 3 to the Oak Ridge Research Reactor (ORR) Safety Analysis Report for your review and approval. This supplement, which addresses the life-limiting factors for the ORR facility, has been reviewed and approved by the Laboratory in accordance with Section 6 of the technical specifications and DOE Order 5480.6.

The enclosed Volume 2, Supplement 3 replaces the information copy sent to you on June 22, 1987. Please advise us if you need additional information on these subjects.

Sincerely,

Fred R Mynatt

Fred R. Mynatt
Associate Director
for Reactor Systems

FRM:jdb

Enclosure

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June 19, 1987

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
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for the
DEPARTMENT OF ENERGY

TABLE OF CONTENTS

1. INTRODUCTION

The Oak Ridge Research Reactor (ORR) was constructed in the mid-1950s. Since it is an older facility, the issue of life-limiting conditions or material deterioration resulting from prolonged exposure to the normal operating environment is an item that should be addressed in the safety analysis for the ORR. Life-limiting conditions were considered in the original design of ORR; but due to the limited data that were available at that time on material performance in research reactors, various studies [8,11] were completed during the first 10 years of operation at ORR to verify the applicable life-limiting parameters. Based on today's knowledge of life limiting conditions and the previous 30 years of operating experience at the ORR facility, the three specific areas of concern are addressed in this supplement: (1) embrittlement of the structures due to radiation damage, which is described in Section 2; (2) fatigue due to the effects of both thermal cycling and vibration, which is addressed in Section 3; and (3) the effects of corrosion on the integrity of the primary system, which is described in Section 4.

The purpose of this document is to provide a review of the applicable safety studies which have been performed, and to state the status of the ORR with regard to embrittlement, fatigue (due to thermal cycling and vibration), and corrosion.

2. EMBRITTLEMENT OF STRUCTURES DUE TO RADIATION DAMAGE

The following discussion, entitled "Radiation Embrittlement of ORR Components" was written in May of 1987 by K. Farrell of ORNL,[5] and is reproduced here.

> "The structural components of the core region of ORR, including the core box, or tank, are undoubtedly embrittled by radiation damage. Removable components are replaced before embrittlement is of concern. The permanent pool side plate, or "window", of the tank, which is made of 5052 aluminum alloy, has accumulated the highest neutron fluences of about $5 \times 10^{22} n/cm^2$ (>0.1eV) and about $1.5 \times 10^{23} n/cm^2$ (thermal). According to creep and tensile test data for 5052 alloy irradiated in HFIR[1,2] and for a similar alloy[3] irradiated in the High Flux Reactor at Petten, The Netherlands, it is anticipated that the residual tensile ductility in the pool side window at the water temperature of about 50°C is probably less than 2% total elongation; correspondingly, the yield strength of the alloy is expected to be 3 to 5 times stronger than before irradiation.
>
> This embrittlement of the tank window is not considered to be detrimental to the safe operation of the reactor nor to proper functioning of the window. Since aluminum alloys are not prone to

the sudden ductile-to-brittle fracture transition displayed by pressure vessel ferritic steels, abrupt failure of the tank window is not expected under normal operating conditions. The reactor tank is not so much a pressure boundary as a separation barrier between the flowing core coolant water and the relatively static pool water. The pressure differential across the walls of the tank is no more than about 40 psi. So the stresses on the window are small, much less than the irradiated yield strength. Furthermore, the consequences of a fracture in the window are estimated[4] to be minor; even complete removal of the window will not dangerously disrupt the core coolant flow.

Periodic visual examinations of the pool side window, the most recent in February 1987, have revealed no signs of corrosion or cracking. In view of these considerations there seem to be no good technical reasons to oppose continued safe operation of the window under the existing regular surveillance program."

The actual pressure differential across the walls of the tank is approximately 30 psi as described in Section 5.

3. FATIGUE OF STRUCTURES

3.1. Thermal Cycling

The issue of fatigue of ORR structures due to thermal cycling was a concern reviewed in 1982 by the RORC.[6] The RORC's recommendations and the response of the Reactor Operations Division are presented below.[7]

"Recommendation 7: That an evaluation be made of the combined radiation damage and thermal cycle effects on the north and south test facilities.

A comprehensive evaluation of the thermally induced cyclic strains in the large facility dished heads was published in 1962 by J. M. Corum and B. L. Greenstreet.[8] At that time it was recognized that these cyclic strains should be minimized and plans to increase the reactor power to 45 MW were abandoned. Also, reactor operating procedures were developed to minimize the thermal cycling of the system. The work of Corum and Greenstreet will be reviewed to determine the feasibility of updating the calculations with the inclusion of radiation-damage information."

Subsequently, the review was made by the RORC and as a result of that review, an additional study was performed in 1984 by G. T. Yahr.[9] A summary of that work is given below.

"Since the cyclic stress and strains induced by a reactor cycle were well established in the 1962 evaluation, our current effort was focused on a consideration of new fatigue data for the aluminum alloy of the dished heads and on the effects of irradiation. Briefly, it was concluded that irradiation has only a small effect on fatigue life in the range of interest. Furthermore, the new fatigue data, including data for weldments, substantiate the fatigue behavior assumed in the 1962 study. Thus, we still predict that cracking of the dished heads could occur after 20,000 cycles of 30 MW operation. Recognizing that the reactor has only experienced about 1400 shutdown cycles to date, we conclude that failure of the dished heads is highly unlikely."

3.2. Vibration

The vibration of the ORR structures during operation in 1987 is essentially the same as it has been during the reactor's lifetime. Flow-induced vibrations experienced during normal operation are quite small and are not considered a credible source for structural failure or degradation. a vibration surveillance program is conducted by the staff of ORR during their normal shift surveillance routines. Unusual noises or vibrations are investigated or monitored to determine and correct the condition.

In 1986, there was an occasion that caused such an investigation. The suspected vibration was analyzed by use of state-of-the-art electronic vibrational analysis equipment. The analysis showed that the equipment in question was operating in the "good" to "extremely smooth" range. The same results were obtained six months later in a follow-up investigation. Both analyses are documented and on file at the ORR.[10]

4. CORROSION OF THE PRIMARY SYSTEM

During the operation of the ORR from 1957 to 1961, a corrosion program was conducted by P. D. Neuman[11] which led him to the conclusion in 1961 that the possibility of any massive failure due to corrosion in the next 50 years of operation of the ORR would seem remote. An internal evaluation by J. C. Griess in June 1987 supported this previous conclusion by Neuman. An excerpt from the evaluation by Griess is presented in the following paragraphs:

"Corrosion has been observed on aluminum in contact with concrete where water has collected, and this problem was solved either by rerouting the aluminum lines or by preventing the accumulation of water in crevice regions between aluminum and concrete.

In 1983 leaks were found in the aluminum pipe lines to and from the main reactor heat exchangers. These leaks occurred as the result of pits that developed on the outside of the pipes. At each point of failure the metal was dented, suggesting that the pipe and its bitumastic coating had been damaged during installation of the lines. The rest of the surfaces were smooth with no visible signs of corrosion. These leaks were repaired by patching from inside the pipes.[12] Since the patches were installed, the leak rate has remained very low, indicating that the patches are effective and that new leaks have not developed since 1983.

Pits have also been found on the outside of the same aluminum lines near the pump house. These pits were not related to mechanical damage of the pipe and they did not penetrate the pipe wall. Pits were located in places where the coating on the pipe had become detached and groundwater could contact the aluminum directly. This section of pipe was left uncovered to prevent further pit growth.

Recent examination of the radiographs of select welds in the primary system indicates that defects exist in some of them but none of the radiographs suggests that corrosion has occurred in these regions.

Corrosion tests conducted in various locations in the ORR, including the core, the core cooling loop, the pool surrounding the reactor, and on the secondary side of the aluminum pool heat exchanger showed very low corrosion rates. While a few shallow pits were noted, it was concluded that pitting was not a major problem in these locations.

In the stainless steel part of the system a few unexplained small leaks in the primary heat exchanger tubes have occurred, and these tubes were plugged. Also, a crack that resulted in a small leak developed in the aluminum decay tank. The cause of the crack was not established. Since the vessel was repaired several years ago, no new leaks have developed.

The above observations plus consideration of the general corrosion resistance of aluminum in aqueous environments clearly indicate that a catastrophic corrosion failure of the ORR system is extremely unlikely, if not impossible. The aluminum alloys used in the ORR are not subject to stress corrosion cracking and the uniform corrosion rates are very low. Failures in aluminum alloys such as those used in the ORR are most likely to be by pitting, and even pitting is very unlikely in flowing water. It is probable, however, that additional pitting originating on the outside of buried pipes will be experienced in the future as the pipe coatings disintegrate with age. Any such pitting would result in low leak rates that would increase with time, as was previously observed; sudden loss of coolant would not occur."[13]

The defects reference in the report by Griess refers to porosity and lack of penetration which have existed since the original construction.

5. ANALYSIS OF LEAKS IN THE ORR PRIMARY SYSTEM

An analysis was made by T. P. Hamrick of the effects of a crack in the ORR primary piping and pool side window.[14]

"A micro computer software program, NOZZLE, has been developed by Scientific Micro of Rochester, New York, that computes flow rates through nozzles, venturis, and orifices. NOZZLE can be used to analyze flow through any of seventeen shapes of nozzles, venturis, and orifices. The program has built-in properties of thirty liquids that are commonly found in various industrial applications. For this analysis, water under moderate pressure flowing through a short exit tube from a reservoir was chosen. A mean water temperature of 125^{o}F was used for pressures ranging from 5 psig to 70 psig. A short tube size of 0.375 in. was chosen for the primary piping and 1.0 in. was chosen for the pool side window. The exit pressure of the leak in the piping is assumed to be 0 psig since the leak is to the atmosphere. In the case of the piping, the gauge pressure is also the differential pressure.

In the case of the pool side window, the normal internal operating pressure at the top of the window is between 36 and 40 psig and the internal pressure at the bottom of the window is between 16 and 17 psig. Since the window is in the reactor pool, it is subjected to an external pressure of 10 psig at the top of the window and 11 psig at the bottom. This means that the differential pressure on the window during normal operation is between 26 and 30 psi at the top of the window and approximately 6 psi at the bottom. For this reason, the analysis was made for a range of 6 to 30 psi differential pressure."

The results of this analysis are shown in reference 14, and lead to the conclusions discussed in the following section.

6. CONCLUSIONS

The review of the material presented in Sections 2 through 4 of this supplement leads to the conclusion that the effects of embrittlement, fatigue due to thermal cycling or vibration, and corrosion are not detrimental to the safe operation of the ORR. It is recognized that while embrittlement, fatigue or corrosion may limit the operating life of a component or system, their effects will have no catastrophic consequence on the safety of the reactor.

The analysis of leaks in the primary system (which could be caused by any of the three effects considered) is presented in reference 14. That analysis concludes:

1. Any leak in the primary system must be made up through the equalizer line from the pool system. The equalizer line is equipped with an orifice and the flow from the pool to the primary system is constantly monitored in the ORR control room. If the make-up flow rate reaches 30 gallons per minute, an alarm is sounded. At 60 gallons per minute, a reactor scram occurs and at 75 gallons per minute the primary pumps are automatically shut down.

2. An abrupt failure of the tank window is not expected under normal operating conditions since aluminum alloys are not prone to sudden ductile-to-brittle fracture transition displayed by pressure vessel ferritic steels. (The normal maximum differential pressure exerted on the window is approximately 30 psi so the reactor tank is not so much a pressure boundary as a separation barrier between the flowing core coolant water and the relatively static pool water).

3. It is expected that any leak will begin small and gradually increase over a period of time. Water made up to the primary system through the equalizer line from the pool must be made up to the pool through the make-up system. The make-up rate is calculated daily.

4. A leak area of somewhere between 0.03 sq. in. at 60 psig (the highest pressure in the system) and 0.085 sq in. at 10 psig (the lowest pressure in the primary piping) will cause a 6 gpm leak. This corresponds to a crack in the pool side window of between 0.038 sq in. and 0.086 sq in. depending on the location.

5. A study[4] has indicated that if the entire window were to suddenly drop out, the safety of the reactor would not be jeopardized.

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