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16 OTHER LICENSE CONSIDERATIONS

This Chapter discusses the prior use of components in the NBSR and documents that the reactor has only been used solely for non-medical research.

16.1 Prior Use Of Reactor Components

The NBSR initiated neutron research in 1969. Chapter 1.8 discusses the facility history and major modifications made to the facility. With the exceptions of the modifications described, all the systems, structures, and components installed and operated in the past will continue to be used in the NBSR. Design and operating limits, the potential for age-related degradation, inspections to assess aging, and a discussion on the suitability for continued use have been included for specific systems, structures, and components in the appropriate chapter herein (Chapter 3 Systems, Structures, and Components; Chapter 4 Reactor including fuel, control elements, and Chapter 6 Reactor Coolant System). Additional information on the continued acceptability of use of the reactor vessel and major core components is discussed in this Chapter.

16.1.1 Reactor Vessel and Core Components

Opportunities to inspect the NBSR reactor vessel are limited since the core must be offloaded to reduce interior radiation levels and allow better visibility of the vessel walls. Two types of inspections are possible: a visual inspection using binoculars when the refueling head is removed from the top of the vessel and an inspection using remote imaging equipment. In preparation for continuing facility operation, a visual inspection (using binoculars) was performed in February 2000. No flaws or unusual discolorations were observed. This type of inspection will be repeated only if the refueling head is removed from the vessel, not a frequent occurrence.

Another inspection using remote imaging equipment was performed in January 2002, which included a detailed inspection of the vessel interior. A visual record (VHS tape) of this inspection was recorded using a radiation hardened, 2.75" diameter, 10X zoom, closed circuit video camera with its own control, signal, and power cable. This inspection was performed with the reactor vessel fully defueled, but still filled with heavy water. The taped video was transferred to a DVD to provide a permanent record. The exterior of the vessel and the interior of the thermal shield were also examined during the same period, and the results recorded on the same DVD. The results of this inspection are summarized below. Figures 16.1 through 16.3 are examples of the images recorded and obtainable from the DVD. The inspection of the vessel interior will be repeated at an appropriate time, when the core is offloaded, estimated within the next six to eight years.

16.1.1.1 Reactor Vessel Interior Inspection

The specific areas remotely examined were all or part of the tip welds for beam ports, through tubes, the cold source, and pneumatic sample thimbles; vessel penetration welds for beam ports, through tubes, the cold source, pneumatic sample thimbles, the outer plenum, outlet pipes, the dump line, and the dry transfer pipe; and vessel fabrication welds. Sections of the upper and lower grid plates, the hold-up pan, the emergency cooling system distribution pan on the upper grid, accessible vertical thimbles, and structural members for different internal components were also inspected.

The inspection revealed no flaws that indicated a decrease in the integrity of the welds. The tip welds were in excellent condition. The other items examined also appeared to have no defects that would affect their design function.

16.1.1.2 Reactor Vessel Exterior Inspection

The vertical portion of the exterior of the vessel may only be seen via a one inch (2.54 cm) gap between the thermal shield and the reactor vessel. The bottom of the reactor vessel is in an area bounded by the vessel, the thermal shield, and the one inch gap. During a cryostat maintenance activity, access to the gap was gained via the biological shield penetration for the Cold Source cryostat. The camera was inserted through the gap to the area containing the bottom of the vessel. The examination of the reactor vessel wall, the dry transfer pipe, the outlet piping, and the outer plenum did not uncover any material condition that could lead to a failure. The thermal shield within the inspection area was also found in good condition.

16.1.1.3 Reactor Vessel Embrittlement

The NBSR vessel is fabricated from aluminum alloys 5052 and 6061, which have been extensively studied in radiation fields. As part of the Advanced Neutron Source Project at Oak Ridge National Laboratory (ORNL), heavily irradiated (4.2×10^{23} thermal, 2.0×10^{22} fast neutrons/cm^2) samples of the 6061-T6 alloy were obtained from a control rod drive follower tube used in the High Flux Beam Reactor (HFBR) at Brookhaven National Laboratory (BNL). These samples were cut into proper shapes for mechanical testing (e.g., tensile, Charpy impact, and metallographic examination) at ORNL. The results are presented in (Weeks, et al., 1993) and are summarized in Table 16.1. The authors were able to correlate their new results with all prior measurements by noting that the relevant parameters were total thermal neutron fluence and the ratio of thermal to fast fluence. Thus, the data can be assumed to be typical of results for the conditions specified above i.e. $>4 \times 10^{23}$ n-cm^{-2}-s^{-1} thermal neutron fluence, and a thermal to fast fluence ratio of 20.

The most heavily irradiated portions of the NBSR vessel are the tips of the beam tubes, and the total irradiation of these components has been calculated (Rowe and Williams, 2002) using the models described in Chapter 4. The salient result is that the most heavily irradiated positions will conservatively (no extended shutdowns are included in the calculations) have accumulated less than 2×10^{23} n-cm^{-2}-s^{-1} thermal neutron fluence by 2024. The calculations also showed that the ratio of thermal to fast fluxes ranged from 17 to 21, very analogous to those of the samples tested. Thus, the data above are a conservative estimate of the condition of the beam tube tips in the NBSR at the end-of-license. As can be seen from the irradiation data, the ductility, while reduced, retains approximately 70% of the original value, although the Charpy energy has dropped by over a factor of 6. Thus, the tips remain a ductile material, with reduced impact strength and toughness. The primary effects of the changes in properties will be reduced resistance to crack propagation under tensile stress, and reduced resistance to sudden pressure applications and impacts.

Within the NBSR, the beam tips are never under significant tensile stress. The beam tube penetrations are filled with CO_2 at 2 inches of water pressure (0.005 atmospheres). The CO_2 system is protected from over-pressure by two relief valves set at 0.5 psi (0.03 atmospheres)(and by a large gasholder). Even if these protective devices failed, there is a pressure regulator from the high pressure tank that reduces the pressure to 10 psi (0.68 atmospheres)(with a pressure relief valve set at 12.5 psi) (0.85 atmospheres), followed by another to reduce pressures to the 2 inches of water value. No experimental systems are inside the beam tips, which are closed at the thermal shield (except for the cold source, which has a vessel designed to be much stronger than the vessel). Therefore, tensile stress is not an issue. The surrounding D_2O does exert a compressive force (due to the static head as there is no coolant over-pressure), which causes a compressive stress, but this stress will not create cracks that can propagate quickly. The vessel is entirely closed, and therefore there can be no impacts against the beam tube tips while the core is loaded.

From these considerations, it is concluded that the noted embrittlement of the vessel creates no hazard to continued operation of the NBSR.

16.2 Medical Use of Non-Power Reactors

The NBSR has not been used for medical purposes, nor are there any plans for such use.

16.3 References

Weeks, J. R., Czajkowski C. J., and Farrell, K. (1993). *Effects of High Thermal Neutron Fluences on Type 6061 Aluminum,* in Effects of Radiation on Materials: 16^{th} International Symposium, ASTM STP 1175, A. S. Kumar, D. S. Gelles, R. K. Nanstad, and E. A. Little, Eds., American Society for Testing and Materials: Philadelphia.

Rowe, J.M. and Williams, R.E. to Weiss, Seymour (2002). Memorandum, April 4, 2002, *Reactor Vessel*.

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Table 16.1: NBSR Reactor Vessel Mechanical Test Results

	Irradiated Specimens	**Unirradiated Specimens**
Ultimate Tensile Strength (un-notched bar) MPa	682 ± 10	330 ± 4
Ultimate Tensile Strength (notched bar) MPa	200	316 ± 27
% elongation	7.2 ± 0.75	10.3 ± 0.5
Charpy Impact Energy Joules	0.34 ± 0.03	2.21 ± 0.26
Fracture Toughness K_{Ic} (derived) (MPa $m^{1/2}$)	8.7 ± 0.8	21.75

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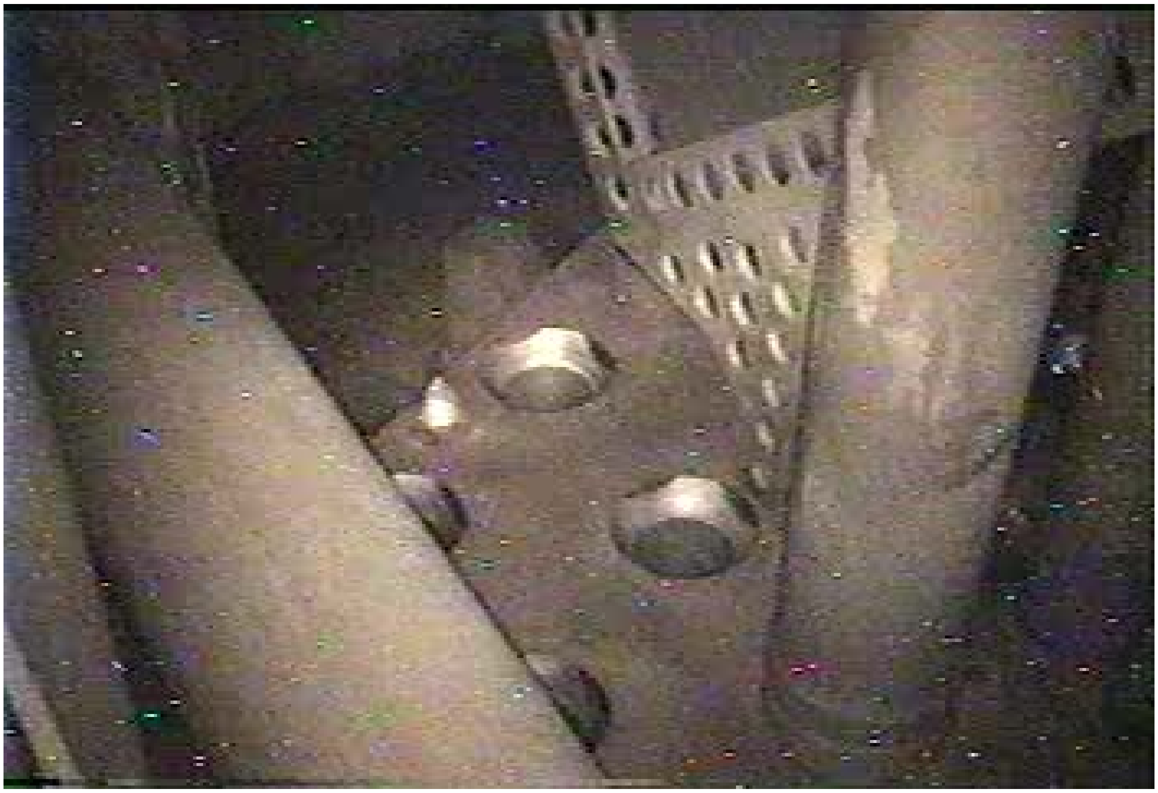

Figure 16.1: Lower Grid Plate

Figure 16.2: Beam tube Number 5 (BT-5) Weld

Figure 16.3: Cold Source Tip Weld