PENN STATE BREAZEALE REACTOR

Annual Operating Report, FY 96-97
PSBR Technical Specifications 6.6.1
License R-2, Docket No. 50-5

Reactor Utilization

The Penn State Breazeale Reactor (PSBR) is a TRIGA Mark III facility capable of 1 MW steady state operation, and 2000 MW peak power pulsing operation. Utilization of the reactor and its associated facilities falls into two major categories:

EDUCATION utilization is primarily in the form of laboratory classes conducted for graduate and undergraduate students and numerous high school science groups. These classes vary from neutron activation analysis of an unknown sample to the calibration of a reactor control rod. In addition, an average of 2000 visitors tour the PSBR facility each year.

RESEARCH/SERVICE accounts for a large portion of reactor time which involves Radionuclear Applications, Neutron Radiography, a myriad of research programs by faculty and graduate students throughout the University, and various applications by the industrial sector.

The PSBR facility operates on an 8 AM - 5 PM shift, five days a week, with an occasional 8 AM - 8 PM or 3 AM - 12 Midnight shift to accommodate laboratory courses or research projects.

Summary of Reactor Operating Experience - Tech Specs requirement 6.6.1.a.

Between July 1, 1996 and June 30, 1997, the PSBR was

critical for	440 hours	or 1.7 hrs/shift
subcritical for	348 hours	or 1.3 hrs/shift
used while shutdown for	381 hours	or 1.4 hrs/shift
not available	381 hours	or 1.4 hrs/shift
Total usage	1550 hours	or 5.8 hrs/shift

The reactor was pulsed a total of 76 times with the following reactivities:

< $2.00	17
$2.00 to $2.50	15
> $2.50	44
>= $3.00	0

The square wave mode of operation was used 35 times to power levels between 100 and 500 KW.

Total energy produced during this report period was 182 MWH with a consumption of 9 grams of U-235.

Unscheduled Shutdowns - Tech Specs requirement 6.6.1.b.

The 5 unplanned shutdowns during the July 1, 1996 to June 30, 1997 period are described below.

October 16, 1996 - During a reactor startup, the reactor operator tripped the reactor when an experimenter reported that a sample he had reported returned to the laboratory terminus (based on sound) during a previous experiment was not in the terminus when he went to remove it. The sequence of events was as follows. At 1132, the last in a series of 16 samples was reported returned to the terminus by the experimenter. At 1133, the reactor was taken to standby and at 1137 the reactor was moved to the other end of the reactor pool for startup for another experiment. At 1140, during the reactor startup, the reactor was tripped by the operator when the experimenter reported the rabbit sample run 8 minutes earlier was not in the terminus; this trip was as procedure since the sample could still have been in the reactor core. The system was surveyed and the sample was found several feet from the laboratory terminus. The experimenter was depending on sound to know that the sample had returned to the terminus and was leaving the sample in the terminus for several minutes for decay before removal for counting. The SOP-9, Pneumatic Transfer System Procedure, did not specifically call for visual inspection to verify that the sample was back. Dose calculations showed that a person standing under the area where the sample was located would have received 0.1 mRem in 30 minutes. Following the occurrence, a see through window was installed in the rabbit terminus to allow visual verification of sample return. The SOP-9 procedure was modified to require the visual verification of the sample's return.

October 30, 1996 - While the reactor was operating at 750 kW, the operator initiated a reactor trip when he received a Rabbit I Radiation Level High Alarm. This alarm disables the fan and opens a bypass valve to relieve system pressure through a charcoal filter and an absolute filter. At the time of the radiation alarm and reactor trip, the last in a series of 18 samples was still in the core. The SRO reset the alarm on the reactor bay alarm panel and at this point the fan came back on and the sample unexpectedly returned to the laboratory terminus. It is an SOP-9, Pneumatic Transfer System Operation, procedure requirement to not run the fan without Health Physics permission upon the receipt of a high radiation alarm to prevent the possibility of radiation being spread through the system due to a broken sample encapsulation (no contamination problem resulted in this case since the sample capsule was found intact). The RO had correctly tripped the reactor but erred in that he had not secured the rabbit system fan. SOP-9 was extensively re-written to provide better instructions for reactor operators and experimenters. The source of the radiation alarm was believed to be an Argon-41 build-up in the system.

December 3, 1996 - While the reactor was operating at 50 watts, the reactor operator inadvertently pressed a console calibrate button while putting a log book stamp away. The calibrate button being pressed caused a "do not operate" condition and a reactor trip. The reactor safety system is designed to operate this way since in calibration mode the actual detector signal is bypassed and an induced signal is inserted into the safety system. The root cause was operator error.

December 16, 1996 - During the fourteenth in a series of pulses for an experimenter, a normal peak power was recorded but Fuel Temp 1 and Fuel Temp 2 read only 33.7 degrees C and 32.5 degrees C respectively instead of an expected approximate 490 degrees C. This was reported to the NRC as a Tech Spec violation on December 17, 1997 since the fuel element temperature limiting safety system setting was inoperative during pre-pulse steady-state operation and during the subsequent pulse. An investigation revealed that when the experimenter exceeded the design specs of his data acquisition system, a ground loop caused the reactor safety system fuel element thermocouples to short, thus disabling them just prior to and during a reactor pulse. The reactor safety system did not have true isolation from the experimental setup. The reactor was not

operated from December 16, 1996 to January 24, 1997 while this event was investigated and completely understood. As a result, no reactor signals will go into or out of the safety system operations boundary (other than to the DCC-X monitoring computer which is a part of the reactor control system) unless adequate isolation is provided between the experimental setup and the reactor safety system. Further details of this event can be found in the 14-day letter dated December 20, 1996 and an addendum to that letter dated on January 23, 1997, both submitted to the NRC.

February 6, 1997 - While the reactor was operating at 750 kw against beam port #4, the reactor operator tripped the reactor as per procedure when he received a neutron beam laboratory beam gate open alarm. The gate opened when a reactor staff member pushed against the gate with a survey meter while checking radiation levels. The gate had been closed (alarm condition had cleared) but had not securely latched. Adjustments were made to the door alignment and the door was fitted with a deadbolt lock so a key must now be used to close the door, thus indicating a definitive closed position to the operator.

Major Maintenance With Safety Significance - Tech Specs requirement 6.6.1.c.

No major preventative or corrective maintenance operations with safety significance have been performed during this report period.

Major Changes Reportable Under 10 CFR 50.59 - Tech Specs requirement 6.6.1.d.

Facility Changes -

On February 13, 1997, the lead shield door on beam port #4 was modified for auto operation by a pneumatic cylinder. Previously the door was opened before the reactor was moved against the beam port and closed after the reactor was moved away from the beam port. With this modification the door is only open during the collection of experimental data, thus minimizing dose to experimenters, unnecessary activation of experimental equipment and degradation of the real-time neutron radiography camera and associated equipment.

During April of 1997, a new D_2O tank was installed to replace an existing D_2O tank. The south side of the pool was drained during installation. This new tank was designed to enhance the neutron beam quality. It was positioned tangentially to the core in an effort to improve the neutron to gamma ratio.

On June 10, 1997, the bridge west radiation monitor and bridge hi-range monitor were moved about six inches higher on the bridge to eliminate interference with west movement of the reactor bridge tower against a new Fast Neutron Irradiator. Measurements indicate a decrease of 0.5 mR/hr as measured by the bridge west monitor at 750 kw. This is not a significant change since a normal reading on this monitor at 750 kw is about 15 mR/hr and the alert and alarm set-points are 50 mR/hr and 200 mR/hr, respectively.

On June 18, 1997, the alert and alarm set-points on the neutron beam laboratory radiation monitor were permanently changed to 10 mR/hr for the alert and 20 mR/hr for the alarm. The set-points had historically been 3 mR/hr for the alert and 6 mR/hr for the alarm. These adjustments followed the installation of a new D_2O tank that has greatly enhanced the neutron beam in the neutron beam laboratory.

Procedures -

All procedures are reviewed as a minimum biennially, and on an as needed basis. Changes during the year were numerous and no attempt will be made to list them. A current copy of all facility procedures will be made available on request.

New Tests and Experiments -

D2O Tank

On January 6, 1997, a safety evaluation was done for a new D_2O tank . This new tank was designed to operate tangentially to the core to increase the neutron to gamma ratio. The new smaller tank places less D_2O between the core and the beam lab for an increased neutron beam intensity.

The reactivity effect of a leak of D_2O replacing the water in the core is negative. The tank was leak tested and care was taken during installation to minimize and monitor stress on the beam port penetration. Materials were selected to minimize any future radioactive waste disposal problems for the tank. A method of monitoring the level of D_2O in the tank was provided. The location of the core neutron detector away from the core to tank interface minimizes any effect on the detector calibration. The collimator tube shielding was designed to withstand an impact of 4 tons, greatly exceeding the lifting capacity of the 3 ton overhead crane in the reactor bay. Calculations showed the tank and support structure were not buoyant with the tank empty. Because of known interactions between D_2O and new aluminum, the tank was equipped with a pressure gauge and relief valve. No pressure build-ups were noted following installation.

The safety evaluation concluded that there are no reactor components affected by this D_2O tank. No reactor safety functions are affected and there are no potential effects on reactor safety functions from this tank. There is no credible mechanism to cause an equipment malfunction that would increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR. There is no credible possibility for an accident or malfunction of a different type than any evaluated previously in the SAR nor does the tank create a possibility for an accident or malfunction of a different type than any previously evaluated in the SAR. The use of the tank does not reduce the margin of safety as defined in any basis for any Technical Specification. No change in Technical Specifications or License were required.

Fast Neutron Irradiator (FNI)

On January 13, 1997 (with a final revision on June 9, 1997), a safety evaluation was done for the installation and use of the FNI. The FNI was designed to be used for the fast neutron irradiation of silicon wafers. The FNI accommodates larger wafers (8 inch) than a previous Fast Flux Tube (FFT) used safely since 1984 that could accommodate up to only 5 inch wafers. The FFT uses an annular design for both the dry irradiation tube and the surrounding lead, boron and cadmium shielding. The FNI uses an annular dry irradiation tube, but the lead and boron shielding are rectangular by design to provide a flat coupling face between the FNI and the reactor core face to minimize the water moderator effect. Provisions are made to allow for the expulsion of water from the enclosed aluminum cowling that surrounds the lead and boron shield to eliminate any water moderation within the shield. The tube is designed with a negative buoyancy under all conditions. Materials were selected to minimize eventual radioactive waste disposal problems.

The support structure was designed and constructed to support the weight of the shielding material and dry tube, to distribute the weight over a sufficient area of the floor, and to restrict its horizontal movement. Flooding of the irradiation tube or the shield cowling will not cause a significant reactivity effect; any effect there would be would be masked by the effect of the boron in the shield and this was verified by experiment after FNI installation. The reactivity effect of the boron shield was judged to be well within any limits on experiments as defined in the Technical Specifications and was measured and found acceptable as part of the FNI certification process following installation. An analysis of the effect of the restricting of the core flow area on the side of the core against the FNI was analyzed as not being significant; this was verified by

fuel temperature measurements after installation of the FNI. Argon-41 production is judged to be acceptable but will be monitored. Administrative controls to assure the shield plug is in when the tube is used have proven adequate with the existing FFT.

Radioactive Effluents Released - Tech Specs requirement 6.6.1.e.

Liquid

There were no liquid effluent releases under the reactor license for the report period. Liquid from the regeneration of the reactor demineralizer is evaporated and the distillate recycled for pool water makeup. The evaporator concentrate is dried and the solid salt residue is disposed of in the same way as other solid radioactive waste at the University. Presently, the demineralizer beds are not normally regenerated but are replaced when depleted. The depleted beds are solidified for shipment to licensed disposal sites.

Liquid radioactive waste from the radioisotope laboratories at the PSBR is under the University byproduct materials license and is transferred to the Health Physics Office for disposal with the waste from other campus laboratories. Liquid waste disposal techniques include storage for decay, release to the sanitary sewer as per 10 CFR 20, and solidification for shipment to licensed disposal sites.

Gaseous

Gaseous effluent Ar-41 is released from dissolved air in the reactor pool water, air in dry irradiation tubes, and air leakage to and from the carbon-dioxide purged pneumatic sample transfer system. The amount of Ar-41 released from the reactor pool is very dependent upon the operating power level and the length of time at power. The release per MWH is highest for extended high power runs and lowest for intermittent low power runs. The concentration of Ar-41 in the reactor bay and the bay exhaust was measured by the Health Physics staff during the summer of 1986. Measurements were made for conditions of low and high power runs simulating typical operating cycles. Based on these measurements, an annual release of between 138 mCi and 418 mCi of Ar-41 is calculated for July 1, 1996 to June 30, 1997, resulting in an average concentration at ground level outside the reactor building that is 0.3 % to 0.8 % of the effluent concentration limit in Appendix B to 10 CFR 20.1001 - 20.2402. The concentration at ground level is estimated using only dilution by a 1 m/s wind into the lee of the 200 m^2 cross section of the reactor bay.

During the report period, several irradiation tubes were used at high enough power levels and for long enough runs to produce significant amounts of Ar-41. The calculated annual production was 99 mCi. Since this production occurred in a stagnant volume of air confined by close fitting shield plugs, most of the Ar-41 decayed in place before being released to the reactor bay. The reported releases from dissolved air in the reactor pool are based on measurements made, in part, when a dry irradiation tube was in use at high power levels; the Ar-41 releases from the tubes are part of rather than in addition to the release figures quoted in the previous paragraph. The use of the pneumatic transfer system was minimal during this period and any Ar-41 release would be insignificant since the system operates with CO-2 as the fill gas.

Tritium release from the reactor pool is another gaseous release. The evaporation rate of the reactor pool was checked by measuring the loss of water from a flat plastic dish floating in the pool. The dish had a surface area of 0.38 ft^2 and showed a loss of 139.7 grams of water over a 71.9 hour period giving a loss rate of 5.11 g ft^{-2} hr^{-1}. Based on a pool area of about 395 ft^2 the annual evaporation rate would be 4680 gallons. This is of course dependent upon relative humidity, temperature of air and water, air movement, etc. For a pool ^3H concentration of 63,190 pCi/l (the average for July 1, 1996 to June 30, 1997) the tritium activity released from the

ventilation system would be 1119 μCi. A dilution factor of 2×10^8 ml s^{-1} was used to calculate the unrestricted area concentration. This is from 200 m^2 (cross-section of the building) times 1 m s^{-1} (wind velocity). These are the values used in the safety analysis in the reactor license. A sample of air conditioner condensate showed no detectable ^3H. Thus, there is probably very little ^3H recycled into the pool by way of the air conditioner condensate and all evaporation can be assumed to be released.

^3H released	1119 μC
Average concentration, unrestricted area	1.8×10^{-13} μCi/ml
Permissible concentration, unrestricted area	1×10^{-7} μCi/ml
Percentage of permissible concentration	1.8×10^{-4} %
Calculated effective dose, unrestricted area	9×10^{-5} mRem

Environmental Surveys - Tech Specs requirement 6.6.1.f.

The only environmental surveys performed were the routine TLD gamma-ray dose measurements at the facility fence line and at control points in residential areas several miles away. This reporting year's measurements (in millirems) tabulated below represent the July 1, 1996 to June 30, 1997 period. A comparison of the North, West, East, and South fence line measurements with the control measurements at Houserville (1 mile away) show the differences to be slightly higher but similar to those in the past

	3rd Qtr '96	4th Qtr '96	1st Qtr '97	2nd Qtr '97	Total
Fence North	27.8	29.7	27.5	28.8	113.8
Fence West	17.7	19.9	17.6	30.2	85.4
Fence East	19.0	21.1	21.2	26.9	88.2
Fence South	19.7	21.9	22.7	23.7	88.0
Control-Houserville	15.8	16.4	17.4	20.7	70.3

Personnel Exposures - Tech Specs requirement 6.6.1.g.

No reactor personnel or visitors received an effective dose equivalent in excess of 10% of the permissible limits under 10 CFR 20.