IDAHO STATE UNIVERSITY

College of Engineering
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June 29, 1998

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Transmittal of Annual Facility Operating Report for 1997.

Dear Sir/Madam:

Enclosed please find a copy of the Annual Operating Report for the Idaho State University AGN-201M Reactor, License No. R-110, Docket No. 50-284, for calendar year 1997. Submission of this report fulfills the requirements of AGN Technical Specification 6.9.1. A copy of this report has also been sent to the Region IV Administrator, as required by the aforementioned technical specification.

If you have any questions concerning the report, please contact me at (208) 236-3351.

Sincerely,

John S. Bennion
Reactor Administrator

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cc: Mr. Marvin M. Mendonca, Project Manager
Non-Power Reactors and Decommissioning Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

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Draft Minutes of the Idaho State University
Reactor Safety Committee (RSC)

Date: December 29, 1997
Start Time: 1:40 P.M.
End Time: 3:10 P.M.
Location: Lillibridge Engineering Laboratory Rm 218

Members Present:
Mr. Frank Just, Chairman
Dr. Jay Kunze, Dean of Engineering
Dr. John Bennion, Reactor Administrator and Acting Reactor Supervisor
Dr. Tom Gesell, Radiation Safety Officer
Dr. Frank Harmon
Mr. Mike Vaughn
Mr. Terry Smith

Members Excused:
Mr. Rick Cummings

Others Present:
Mr. Todd Gansauge

1. The meeting was called to order at 1:40 P.M.

2. Mr. Todd Gansauge was introduced as the future Reactor Supervisor pending passing the NRC SRO exam.

3. The minutes of the October 29, 1996 Reactor Safety Committee meeting were reviewed and approved.

4. Mr. Mike Vaughn asked the status of missing comments about the console upgrade regarding issues raised by Mr. Jim Larson. Dr. Bennion committed to forwarding clarifying information to the committee.

5. Dr. John Bennion reviewed the reportable occurrence of June 29, 1997. Failure of a primary fission product barrier caused by a cladding failure on Safety Rod No. 2. The details of the event, and the steps taken to recover were discussed.
   a. Mr. Mike Vaughn asked if metallurgical analysis had been preformed on the cladding failure. Dr. Bennion indicated that he and Mr. Kermit Bunde had taken some photomicrographs of the fracture surface. Mr. Vaughn indicated that he would speak with the metallurgists in his section at the INEEL and ask them if they would be willing to take a look at the fracture. The possibility of decontaminating the end cap and sending it to INEEL for analysis was discussed. Mr. Frank Just asked Dr. Bennion to follow up on this matter and make it an agenda item for the next Reactor Safety Committee meeting.

6. The proposed modifications of the dashpot mounting brackets were discussed.
   a. Mr. Mike Vaughn raised a question about the bearing surface between the proposed dashpot modification and the dashpot shaft. He suggested this bearing surface be made out of brass. Dr. Bennion commented that the staff had discussed the possibility of fabrication of this surface from polyethylene.
   b. Mr. Terry Smith suggested that the dashpot mounting modification be validated on the test stand before being installed in the reactor. He raised the issue of dashpot malfunction by jamming in the fully extended position and asked how the resulting position of the elements would effect the shutdown margin. The staff committed to look into this and include it in the 50.59 analysis.
   c. Dr. Jay Kunze asked about a formal 50.59 analysis for this modification. Dr. Bennion commented that the attached handout was the first steps in such an analysis and the analysis was underway.
   d. Mr. Mike Vaughn asked if the manufacture of the dashpot had been contacted about this problem with dashpot mounting. Mr. Todd Gansauge responded stating that discussions had been held with the manufacturer and that the manufacturer indicated that these dashpots were not designed to be bottomed or operated without external shaft support.

7. Dr. Harmon moved to allow restart of the reactor. Dr. Gesell amended the motion to include removal of the dashpot between runs. After discussion reactor restart was approved unanimously with the following conditions:
   a. The replacement dashpot used during control rod calibration following the reportable occurrence be replaced with one of the unused dashpots until mounting plate modifications can be completed.
   b. The SR-2 dashpot will be removed from the reactor between runs to minimize the time spent with a static load on the graphite piston until modifications to the mounting hardware can be completed.

8. Mr. Terry Smith volunteered to do the two annual audits required of the RSC. He will contact the reactor staff near the end of January and make arrangements to preform these audits.

9. Dr. John Bennion reviewed the results of the latest NRC inspection. He stated that all previous items have been closed. Two non-citatable violations were listed in the inspection report that had been self-identified by the staff.

10. Mr. Mike Vaughn brought up the issue of a document control system and storing duplicate records in a fireproof cabinet off site. This issue was discussed and the staff indicated that a lot of effort has gone into organizing and maintaining the reactor records and that the staff is not aware of any requirements for a document control system or off site storage.

11. Dr. John Bennion updated information on the current status of the reactor console upgrade. Mr. Mike Vaughn mentioned the need for detailed procedures on console changeover.

12. Dr. John Bennion briefed the committee on the status of the relicensing effort for the reactor. Commenting that he had committed to responding to the NRC's Request for Additional Information as soon as possible.

13. Dr. Tom Gesell gave the Radiation Safety Officer report. He reminded the committee that the Technical Safety Office operates under a written agreement for services to the reactor facility. The TSO provided special services, verifying no measurable I-131 uptake, as requested by the reactor staff. He mentioned the fact that Mr. Kermit Bunde underwent a full body assay preformed at the INEEL at Mr. Bunde's request. No evidence of uptake was found. All external doses to facility personnel have been within ALARA limits, based on his reviews of dosimetry records.

14. Mr. Frank Just stated that Mr. Rick Cummings has asked to step down from the Reactor Safety Committee due to lack of time to fulfill this position. Mr. Ken Waitham and Mr. Bob Boston were suggested for consideration for RSC appointments. The staff committed to look at the RSC charter and verify staffing requirements for the RSC.

15. Dr. John Bennion reviewed the emergency drill held in January of 1996. He stated that he needs to finish the final report on this drill and distribute it to the committee.

16. The meeting was adjourned at 3:10 P.M.

Submitted by: Todd C. Ganzauge

Approved:_____ _____
Reactor Safety Committee Chair Date

**Idaho State University**
**AGN-201M Reactor Facility**
**License R-110, Docket No. 50-284**
**Annual Operating Report for 1997**

## 1. Narrative Summary

### A. Changes in Facility Design, Performance Characteristics, and Operating Procedures:

No changes in facility design or performance characteristics relating to reactor safety occurred during the reporting period. An application to amend the facility operating license was submitted to the Nuclear Regulatory Commission to increase the possession limit for uranium-235 from 700 to 995 grams in order to allow the facility to receive spare control elements from Oregon State University (OSU). The spare control elements, complete with fuel material, were from an AGN reactor that was operated at OSU and subsequently decommissioned during the period 1978-1980. The license amendment was issued on August 18, 1997.

### B. Results of Major Surveillance Tests and Inspections:

(1) Channel tests performed on all safety channels and scram interlocks were found to be satisfactory and within specifications.

(2) Power and period calibrations were performed with satisfactory results.

(3) The shield water tank was inspected and no leaks or excessive corrosion were observed.

(4) The seismic displacement interlock was tested satisfactorily.

(5) (a) The control rod drive mechanisms were inspected and tested with satisfactory results.

(b) Ejection times were measured for all scrammable rods and found to be less than 130 milliseconds.

(c) The reactivity worths of all safety and control rods were measured, as well as the time required to drive each rod to its fully inserted position. Reactivity insertion rates were determined to be less than 0.039% $\Delta k/k\ s^{-1}$ (\$0.052 $s^{-1}$) for all rods.

(d) The shutdown margin was determined to be greater than 1.64% $\Delta k/k$ (\$2.22) with both the most reactive scrammable rod and the fine control rod fully inserted.

(e) All surveillances were within the appropriate Technical Specification requirements.

2. Operating History and Energy Output.

The reactor was operated at power levels up to approximately 4.8 watts for a total 77.8 hours thereby generating 1.98 watt-days (47.6 watt-hours) of thermal energy during this reporting period. A summary of monthly operations for 1997 is given in Table I.

Table I. Summary of Monthly Reactor Operations
(1 January 1997 through 31 December 1997)

| Month | Hours | Energy (W-hr) |
|---|---|---|
| January | 4.4 | 0.13 |
| February | 14.3 | 15.64 |
| March | 16.9 | 0.34 |
| April | 8.8 | 1.98 |
| May | 10.6 | 4.10 |
| June | 14.6 | 25.09 |
| July | 0.0 | 0.00 |
| August | 0.0 | 0.00 |
| September | 0.0 | 0.00 |
| October | 8.2 | 0.33 |
| November | 0.0 | 0.00 |
| December | 0.0 | 0.00 |
| Total | 77.8 hr | 47.61 W-hr |

3. A. Unscheduled Shutdowns and Corrective Actions Taken.

3/29/97 The Coarse Control Rod (CCR) drive switch became inoperable during FCR calibration. The reactor was scrammed to allow cleaning of the switch. The switch was thoroughly cleaned and returned to service.

6/25/97 During restart following a planned low level trip on Nuclear Instrument Channel No. 1, Nuclear Instrument Channel No. 3 tripped high due to improper

range switching by the operator-in-training. Attempts to restart resulted in Safety Rod No. 2 (SR-2) unlatching from the electromagnet as soon as fully inserted in the core. Investigation revealed a failed dashpot, a component of the SR-2 drive assembly. The run was then terminated and further operations were suspended pending replacement of the dashpot. Later inspection revealed that the aluminum capsule which comprises the cladding of SR-2 had failed. This discovery resulted in the immediate notification of the Nuclear Regulatory Commission as a reportable event, i.e., breach of a primary fission-product barrier (see attached Reportable Occurrence Report). Further reactor operations were suspended until the aluminum capsule could be replaced and the event could be reviewed by the Reactor Safety Committee. In accordance with the recovery plan for the reportable event, the reactor was operated at low power following repair of the SR-2 and replacement of the dashpot during October in order to perform required surveillances necessary to demonstrate that operation of the reactor was within technical specifications. The event was reviewed by the Reactor Safety Committee (RSC) on December 29, 1997 (see draft copy of minutes attached to this report). At that time, the RSC granted permission to resume normal operation of the reactor with the provision that the SR-2 dashpot be removed from the drive assembly between reactor operations. This action will continue until the dashpot mounting assembly can be modified such that the static load of the control element is transferred from the dashpot piston to the mounting assembly. The operations staff is in the process of designing the necessary modifications to the dashpot mounting assembly and preparing the required review to ensure no unreviewed safety questions will result from the proposed modifications. The RSC will then review the proposed modification and staff review of unreviewed safety questions before granting approval to implement the redesigned dashpot mounting assemblies.

B. Inadvertent Scrams and Action Taken.

5/28/97: During ascension to the planned power level, a high trip occurred on Nuclear Instrument Channel No. 1 due to improper switching of by the operator-in-training. The reactor was restarted. After restart, there were 3 spurious high level trips on Nuclear Instrument Channel No. 2 attributed to power fluctuations in the building. The reactor was restarted following verification and readjustment of the Channel No. 2 zero and amplifier balance set points.

6/10/97 While irradiating a sample at 4.5 watts, noise in Nuclear Instrument Channel No. 2 caused a high level trip. The reactor was restarted and taken a power level of 4.2 watts. Noise in Channel No. 2 resulted in another high level trip several minutes later. The reactor was restarted and taken to a power level of 4.0 watts and irradiation continued.

6/19/97 After maintaining a steady state power of 4.7 watts for 52 minutes, Nuclear Instrument Channel No. 1 tripped high. The reactor was restarted and taken to the same power level. The problem did not reoccur.

6/25/97 During restart following a planned low level trip on Nuclear Instrument Channel No. 1, Nuclear Instrument Channel No. 3 tripped high due to improper range switching by the operator-in-training. Attempts to restart resulted in Safety Rod No. 2 (SR-2) unlatching from the electromagnet as soon as fully inserted in the core. Investigation revealed a failed dashpot, a component of the SR-2 drive assembly. The run was then terminated. Later inspection revealed that the aluminum capsule which comprises the cladding of SR-2 had failed, as reported in item 3.A. above.

10/16/97 During determination of the Fine Control Rod (FCR) reactivity worth, fluctuations in magnet current caused a reactor trip. The reactor was restarted.

4. Safety-Related Corrective Maintenance

3/11/97 Two small squares of neoprene rubber were attached to the contact surface of the dashpot for Safety Rod No. 1 (SR-1) when it was observed that SR-1 was not making contact with the electromagnet preventing startup.

3/22/97 The alignment of SR-1 was adjusted after to make SR-1 release upon shorting the electromagnet during rod worth calibrations by the rod drop method.

3/29/97 The Coarse Control Rod (CCR) drive switch was thoroughly cleaned using an electronic solvent and returned to service.

5/31/97 The Nuclear Safety Channel No. 3 strip chart recorder was found to operate intermittently. The chart recorder contacts were cleaned and the chart recorder was returned to service.

7/23/97 The failed SR-2 was dismantled, inspected, and inventoried. Components were placed in sealed plastic bags and stored in an approved location.

9/15/97 Disassembled the Oregon State University CCR. Contents were sealed in plastic bags.

9/16/97 Reassembled the ISU SR-2, using the OSU CCR capsule and extension shaft.

9/22/97 to 10/15/97
Installed the repaired SR-2 and drive assembly in the reactor, and installed

replacement dashpot. Performed required Technical Specification surveillances. All Technical Specification requirements were satisfied.

5. Modifications.

A. Changes in Facility Design.

There were no changes to the facility design during 1997 to the extent that changed the description of the facility in the application for license and amendments thereto.

B. Changes to Procedures.

None.

C. Experiments.

No new or untried experiments or tests were performed during 1997.

D. Reactor Safety Committee.

As of the end of the reporting period, membership of the Reactor Safety Committee (RSC) consisted of the following individuals:

Frank H. Just - Chair
Jay F. Kunze - Dean, College of Engineering
John S. Bennion - Reactor Administrator and Acting Reactor Supervisor
Thomas F. Gesell - Radiation Safety Officer
J. Frank Harmon
Terry W. Smith
Michael E. Vaughan

6. Summary of Changes Reportable under 10 CFR 50.59.

None.

7. Radioactive Effluents.

A. Liquid Waste - Total Activity Released: None.

B. Gaseous Waste - Total Estimated Activity Released: 1.04 $\mu$Ci.

The AGN-201 Reactor was operated for 77.8 hours at power levels up to approximately 4.8 watts. At this power level argon-41 production is negligible and substantially below the effluent concentration limit given in 10 CFR 20 Appendix B, Table 2. The total activity of Ar-41 released to the environment was conservatively estimated at 1.04 $\mu$Ci. This activity corresponds to the total activity of all gaseous radioactive effluent from the facility. A monthly summary of gaseous releases is given in Table II.

Table II. Summary of Monthly Gaseous Radioactive Effluent Releases
(1 January 1997 through 31 December 1997)

| Month | Ar-41 ($\mu$Ci) |
|---|---|
| January | 0.003 |
| February | 0.341 |
| March | 0.007 |
| April | 0.043 |
| May | 0.089 |
| June | 0.548 |
| July | 0.000 |
| August | 0.000 |
| September | 0.000 |
| October | 0.000 |
| November | 0.000 |
| December | 0.007 |
| Total activity: | 1.038$\mu$Ci |

C. Solid Waste - Total Activity: None.

8. Environmental radiation surveys, performed at the facility boundary while the reactor was operating at 94% of full licensed power (4.7 watts), measured a maximum combined neutron and gamma dose equivalent rate of less than 1.6 mrem $hr^{-1}$ at the outside walls of the building proximal to the reactor.

9. Radiation Exposures.

Personnel radiation exposures are reviewed quarterly by the Radiation Safety Officer. Annual reports of ionizing radiation doses are provided by the Radiation Safety Officer to all monitored personnel as required under the provisions of 10 CFR 19.

Personnel with duties in the reactor laboratory on either a regular or occasional basis have been issued radiation dosimeters by the Idaho State University Technical Safety Office. These dosimeters are changed out on a quarterly basis. The duty category and monitoring period of personnel are summarized in Table III:

Table III. Personnel Monitored for Exposure to Ionizing Radiation

| Name | Monitoring Period | Duty Category |
|---|---|---|
| Kazi Ahmed | 1/1/97 - 12/31/97 | Regular |
| Rick Baker | 1/1/97 - 12/31/97 | Occasional |
| John S. Bennion | 1/1/97 - 12/31/97 | Regular |
| Robert D. Boston | 1/1/97 - 12/31/97 | Terminated 8/15/97 |
| Kermit A. Bunde | 1/1/97 - 12/31/97 | Regular |
| Todd C. Gansauge | 5/1/97 - 12/31/97 | Regular |
| Dirk Howlett | 1/1/97 - 12/31/97 | Occasional |
| Raed Jaber | 1/1/97 - 12/31/97 | Occasional |
| Michael F. Jolley | 1/1/97 - 12/31/97 | Occasional |
| Jay F. Kunze | 1/1/97 - 12/31/97 | Regular |
| Sad Jarall | 1/1/97 - 12/31/97 | Occasional |
| James K. Sample | 1/1/97 - 12/31/97 | Occasional |
| Alan G. Stephens | 1/1/97 - 12/31/97 | Occasional |
| William Taylor | 1/1/97 - 12/31/97 | Occasional |
| Miles Whiting | 1/1/97 - 12/31/97 | Occasional |

Dose Equivalent summary for Reporting Period:

Measured Doses

1/1/97 - 12/31/97 Whole-body dose equivalents: ≤ 10 mrem for most personnel.
Maximum measured quarterly dose = 50 mrem shallow dose from a ring badge.
Minimum detectable dose equivalent per quarterly badge ≐ 10 mrem.

None of the 122 visitors to the facility during 1997 received a measurable dose. Therefore, the average and maximum doses are all within NRC guidelines. A summary of whole-body exposures for facility personnel is presented in Table IV.

Table IV. Summary of Whole-Body Exposures
(1 January 1997 through 31 December 1997)

| Estimated whole-body exposure range (rem): | Number of individuals in each range: |
|---|---|
| No Measurable Dose | 7 |
| Less than 0.10 | 8 |
| 0.10 to 0.25 | 0 |
| 0.25 to 0.50 | 0 |
| 0.50 to 0.75 | 0 |
| 0.75 to 1.00 | 0 |
| 1.00 to 2.00 | 0 |
| 2.00 to 3.00 | 0 |
| 3.00 to 4.00 | 0 |
| 4.00 to 5.00 | 0 |
| Greater than 5 rem | 0 |
| Total number of individuals reported: | 15 |

Report submitted by:

Todd C. Gansuage, Reactor Supervisor (April 1998 to present)
John S. Bennion, Reactor Administrator and Acting Reactor Supervisor (to April 1998)