DEFENSE NUCLEAR FACILITIES SAFETY BOARD

September 16, 1994

MEMORANDUM FOR: G. W. Cunningham, Technical Director

COPIES: Board Members

FROM: Daniel G. Ogg, Program Manager, INEL

SUBJECT: Tritium Testing and Safety Analyses, Idaho National Engineering Laboratory, Advanced Test Reactor, Report of Site Visit, August 30-September 1, 1994

1. **Purpose:** This memorandum documents the results of the DNFSB staff visit to the Idaho National Engineering Laboratory (INEL). The trip focused on a new tritium production test being conducted at the Advanced Test Reactor (ATR) for the Office of Reconfiguration (DP-25). Additionally, the staff reviewed the design basis of the reactor, excluding the in-pile loop experiments. The review team included DNFSB staff members Daniel Ogg, Joseph Roarty, and Sol Pearlstein.

2. **Summary:** The tritium production feasibility test at the ATR presents little challenge to the safety of the core of the reactor. The test targets or "shadow slugs" are located around the periphery of the core, and the amount of tritium to be produced will measure only a few grams. The integrity of the shadow slugs, fabricated at the Savannah River Site (SRS), has been poor, with a failure rate of 80 percent. Measures to prevent a defective shadow slug from reaching the ATR are essential to preclude a tritium release during the test.

   The DNFSB staff review of the design basis of the ATR disclosed several characteristics that indicate the safety margin in the ATR is less than that of the SRS K Reactor (after restart) or of commercial reactors. The following observations were noted:

   a. Irradiation induced growth of beryllium (Be) results in cracking and bowing of the ATR reflector such that a complete replacement of core internals is required every 6-8 years. Although this condition could constitute a potential safety issue due to the release of Be fragments and distortion of the reflector, considerable experience has been gained from the operation of the core since 1968. The Staff acknowledges that this problem has long been identified and evaluated and believes that continued careful monitoring of the reflector is appropriate to mitigate challenges to the safety of the core.

   b. The thermal design of the ATR can accommodate a primary coolant pipe break equivalent to 3 inches in diameter. This break size is less than that in the design basis for the SRS K Reactor. The K Reactor analysis assumed a Double Ended Guillotine Break of a primary system pipe.

c. The ability to inspect ATR primary coolant piping is limited as some piping is inaccessible. Intergranular Stress Corrosion Cracking (IGSCC) was noted at K Reactor during a comprehensive inspection of the reactor vessel and piping, and sections of piping were replaced. ATR operators conduct periodic inspections for leaks and no leaks due to IGSCC have been detected throughout the life of the ATR.

d. The ATR primary coolant nozzles are located below the reactor core; a break in this piping would drain the core. Further, the location of heat exchangers is below the elevation of the core, which nullifies natural convection cooling of the core following a loss of pumping power. These conditions are not permitted in commercial pressurized water reactors, but did exist in K Reactor. An additional emergency coolant injection loop and a seismically qualified gadolinium-nitrate poison injection system were added as safety enhancements prior to the restart of K Reactor. ATR has provided a fire main emergency cooling system.

e. The maximum power density in the ATR is about twice that of commercial reactors or K Reactor. During a reactivity addition accident, the maximum local heat flux exceeds $3x10^6$ Btu/hr-ft^2 in ATR. The DNFSB staff believes that investigation of a rate response system and lowered trip settings is appropriate to mitigate this type of accident.

3. **Background:** The ATR is operated by EG&G for the DOE to test reactor materials and to produce radioisotopes and has been in operation since 1968. The EG&G operations staff at ATR completed the third Core Internals Change-out (CIC) in July 1994, and commenced a new test cycle on August 22, 1994. While materials testing for the Naval Nuclear Propulsion Program continues at the ATR, the current test cycle also includes a new test, sponsored by the DOE Office of Reconfiguration (DP-25), called the Tritium Validation and Feasibility Study. The purpose of this test is to validate predicted tritium production, demonstrate target survivability, and predict optimum tritium production in the ATR.

4. **Discussion:**

a. <u>Tritium Demonstration Test</u>: Lithium targets, originally fabricated for the K Reactor at SRS, are being irradiated in the ATR. The objective is to demonstrate that modest but sufficient amounts of tritium can be produced in the ATR and that such production could extend the date, by approximately two years, when DOE must decide its long-term method for producing tritium. The targets will be placed in ATR for two cycles, 40 days each. At the end of that time the irradiated targets will be returned to SRS for extraction of tritium. The integrity of the lithium targets is important to avoid release of tritium in the ATR system. At SRS, an 80 percent failure rate was observed for the test specimens examined. Only targets showing no defects were sent to the ATR for irradiation. If this program is to continue, strict quality assurance procedures must be used to select or manufacture targets for irradiation in the ATR.

b. ATR Design Review: The ATR contains a number of similarities to the Savannah River Site K Reactor that allow an approximate comparison to be made between the thermal design margin of each reactor. The cores are both designed with curved aluminum plates, and for down flow of the coolant, and operate at relatively low temperature and pressure. The local power density in ATR is significantly higher than that of the K Reactor with heat fluxes about two times higher.

The K Reactor has, as a design basis accident, a Double Ended Guillotine Break of a large diameter primary coolant pipe, while the ATR design is limited to an equivalent 3-inch diameter primary coolant break.

Another approximate comparison of the thermal design margin in ATR and K Reactor is provided by examining the heat flux reached during a design basis reactivity addition accident. In ATR, the steady state power of 250 MW increases to 460 MW, and a local heat flux of 1.8×10^6 Btu/hr-ft^2 increases to a value of over 3×10^6 Btu/hr-ft^2. In K Reactor, the peak heat flux is about one-half of these values. A reevaluation of the ATR thermal design, based on modern methodologies (including single fault assumptions as required by DOE Order 5480.30, *Nuclear Reactor Safety Design Criteria*), is appropriate to identify ways (e.g. lower scram trip, rate protection) to mitigate overpower accidents.

c. ATR Aging Effects: The ATR has operated for about 25 years and is potentially vulnerable to IGSCC of the 304L primary coolant piping. Such a condition was noted in Savannah River Site reactors, and a section of the primary coolant piping was replaced in K Reactor. Reactor vessel repair was also required in C Reactor.

ATR conducts periodic in-service inspections and a walk-down following each start-up to confirm the absence of leaks. None has been found to date; however, it is noted that some primary coolant piping is inaccessible and cannot be directly inspected.

The ATR organization does not include a materials/metallurgy specialist. Such an individual is available on an on-call basis. In view of the safety dependence on materials integrity, additional support in this area may be warranted.

The DNFSB staff believes that the ATR organization can be aided significantly by a stronger tie with the Savannah River Site organization. Although ATR has some awareness of SRS activities, contacts have been limited. The K Reactor Safety Analysis Report (SAR) completed in 1992 is particularly relevant as are several issues (corrosion, water-hammer, corrosion-erosion, etc.) which have been investigated by personnel at the Defense Waste Processing Facility, F-Canyon, and the Savannah River Technology Center.

d. ATR Beryllium Reflector Cracking: Irradiation induced growth of beryllium (Be) results in cracking and distortion of the ATR reflector, such that a complete replacement of core internals is required every 6-8 years. This condition constitutes a potential safety issue as the release of Be fragments and/or the distortion of the reflector could block a coolant channel or interfere with movement of a safety rod. This problem was identified early in the life of the ATR and, upon evaluation, led to the current practice of periodic replacement of the reflector. The DNFSB staff believes that continued monitoring of the reflector is appropriate to prevent challenges to the safety of the core.

In view of this situation, it is appropriate to recall an experience which occurred at the Rochester Gas and Electric Plant (GINNA) where a loose part led to the rupture of a steam generator tube and loss of primary coolant. It is to be noted that many commercial nuclear power plant operators responded by installing a Loose Parts Monitoring System. This equipment was also installed at the K Reactor.

e. Reactivity Analysis: The principal reactivity analysis tool used at ATR is PDQ, a 4-energy group, 2- and 3-dimensional fine mesh diffusion theory code. This code is widely used in the design of thermal reactors and is also used for burnup calculations. However, the use of PDQ rectilinear coordinates to describe the cylindrical shape of cells and reflectors found in ATR can introduce jagged boundaries.

Data sets of temperature dependent 4-group cross sections were obtained from a variety of sources and, therefore, do not constitute an easily documentable reference set of data, but the ATR group is working towards deriving data from a standard reference data set. It is also not clear that 4 energy groups are adequate to describe the Be hardened spectra which differ from those found in conventional light water reactors. PDQ calculations yielded k_{eff}=0.985 for measured critical loadings. This tendency to underestimate criticality is non-conservative. The analysis group was urged to work toward improving its understanding of physics and methods to reduce the discrepancy between calculated and measured absolute criticalities to perhaps within 0.5 percent or at least understand the biases that exist and where they should be applied. Recently, the Monte Carlo code, MCNP, with continuous energy and combinatorial geometry treatments has been implemented and can be used to explore these factors.

The calculation of incremental reactivity changes, e.g., temperature defect, burnup, sample and control worths, is quite accurate. Calculated and measured flux distributions agreed to within a few percent generally and within 8 percent at boundaries. Furthermore, reliance on calculations to establish safety margins is alleviated by use of the ATR critical facility (ATRC). This full scale look-a-like facility is used to verify design changes, target perturbations, and fuel loadings that combine new and spent fuel. The ATRC is considered a vital factor in the ATR program. At a cost of $300K per year, it is a small investment within the $44M ATR program and is cost effective in obtaining safety information and savings in ATR time.

f. Core Internals Change-out (CIC) and Equipment Upgrades: The CICs at 6-8 year intervals reflect the application of good ALARA principles. The last change-out resulted in an total worker dose of 25 person-rem compared to 60 person-rem for the previous change-out. It has been noted that during a CIC there is no appreciable decay in the radiation background from the permanent part of the ATR. This suggests that some radioactive isotopes are formed with half-lives comparable with or longer than the few months time required to complete the change-out. If the half-lives are comparable with the 6-8 year interval between change-outs, the background radiation could continue to increase making change-outs more difficult with increasing ATR age. The longest lived gamma activity from the stainless steel components is expected to be from ^{60}Co ($t_{1/2}$=5.3 years). It might be prudent to analyze whether there are known impurities or minor constituents that, if activated, might increase the radiation background so as to limit the useful life of the core. No radiation background measurements from previous change-outs are available for comparison.

It was also noted that during the life of the beryllium reflectors, about 100 grams of tritium is expected to be formed through the ^{9}Be (n,t) reaction. The feasibility of extracting tritium from the reflector was not discussed.

There is a program to systematically replace aging components. The yearly capital improvement budget is about $500K. Within the last few years, the reactor control system and many electrical components have been replaced. An upgrade to the battery room ventilation system is also planned. Currently, the emergency batteries are charged one at a time because the ventilation system is judged by EG&G to be inadequate for the removal of hydrogen generated during a two-battery charge.

5. **Future Staff Reviews:** Future activity relative to the production of tritium will be closely followed by the DNFSB staff. Other reviews of ATR will continue on a periodic basis at a frequency of approximately three per year.