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**RICHLAND FIVE-YEAR
02 R&D PROGRAM**

NUCLEAR SAFETY

**RICHLAND OPERATIONS OFFICE
ATLANTIC RICHFIELD HANFORD COMPANY
BATTELLE-NORTHWEST
DOUGLAS UNITED NUCLEAR, INC.**

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June 30, 1968

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Verified By Julie Mally
10-19-94

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R I C H L A N D F I V E - Y E A R
O 2 R & D P R O G R A M

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Introduction

The nuclear safety record achieved at the Richland facilities is outstanding.

The production facilities have been in operation now for nearly a quarter of a century. During this time there have been no injuries to personnel attributed to exposure to ionizing radiation; there have been no releases of radioactivity from these facilities that have created loss or damage to private or public property outside the project boundaries.

The achievement of this record reflects the importance ascribed throughout the history of the Project to nuclear safety in the design, the operation, and the modification of the nuclear facilities and the development of new and improved processes. The location and the climate of the Richland site are themselves important plant safety factors, and were factors leading to the selection of the relatively isolated Hanford site during the Manhattan Project.

The approach to nuclear safety at the Richland site evolved logically as follows:

1. Great emphasis was placed during the design and construction phases on providing high levels of reliability, adequacy and redundancy in systems whose proper operation would prevent the occurrence of a nuclear safety incident.
2. Operational and administrative controls were emphasized to assure that the facilities would be operated within known-safe system capabilities.

Despite the extreme efforts on accident avoidance, some residual possibility for a major accident remained. Site isolation and inherent features of the systems have been the main factors depended upon to reduce the consequences of this residual risk to tolerable bounds, although accident mitigation has received increased emphasis in recent years. The continued operation of Richland facilities has been deemed prudent; in part, because of the national security needs for the nuclear products. The wisdom of this judgement to-date has been vindicated by the excellent safety experience.

While the safety status of the Richland facilities has in the past been deemed adequate, all aspects of nuclear technology have progressed and evolved including standards of nuclear safety. Hence, the national nuclear safety environment within which the Richland facilities are operated is grossly different now than was the case ten years ago; a commensurate change can be expected in the future. Further, the urgency of national security aspects of Hanford operation seems less acute now than in the past and may become less so in the future. Finally, the isolation of the Hanford site has been reduced by the release of land, and the concentration of people and valuable property in near proximity may increase in the future.

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Thus, even discounting the obvious obligation for a government-owned facility to show leadership in an area of such basic public responsibility as nuclear safety, the nuclear safety status which seemed quite adequate in the 1940's in the future could well be considered clearly unacceptable.

The above should not be inferred to mean that the Richland Operation has to-date been unresponsive to the trends for more stringent levels of safety. The safety of the C and K reactors has been continually upgraded; provision of fission product confinement and improved safety instrumentation are two obvious examples. Further, the N Reactor was designed with sophisticated engineered safeguard systems in its original design and these safeguards will be further improved by the Effluent Control Program.

However, improvements in nuclear safety, implemented and proposed, will largely use up the existing store of technology. Any further improvements of note and any improved definition of the basic safety of plants as they now exist will require a better understanding of accident mechanisms, system behavior during accidents, and the physics, chemistry and metallurgy of fissile material and fission products in the "accident environment".

During FY 1968 the nuclear safety R&D efforts continued along the lines and at about the schedule predicted at the beginning of the year. Hence, the safety problems to be studied have been identified and the program seems well-defined.

While some program modifications and changes in emphasis are to be expected, the five-year outline shown here is considered to be a reasonable representation of the safety work of highest priority to be studied.

The Nuclear Safety Program consists of seven, concurrent subprograms. These are:

- 8RLa - Fuel Temperature Transients Under Accident Conditions
- 8RLb - Chemical and Metallurgical Reactions and Fission Product Release from Overheated Fuel
- 8RLc - Control of Fission Gases
- 8RLd - Meteorological Studies
- 8RLe - Ground Fixation of Radioactive Material in Liquid Wastes
- 8RLf - Particle Formation and Release from Overheated Nuclear Material
- 8RLg - Seismic Studies

The Scope and Objective and Incentives of the overall program will be discussed and then details provided on each of the seven subprograms.

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Scope and Objectives

The scope and objectives of the Nuclear Safety Program are:

1. Obtain a better understanding of the course of reactor loss of cooling accidents including temperature transients, chemical and metallurgical reactions in the overheated core, and the extent, rate and nature of fission product release from overheated fuel. This will permit a much more quantitative assessment of the basic risk of facility operation replacing current assumptions which are quite possibly pessimistic.
2. Determine methods for controlling fission gases released following reactor accidents. This may point to practical ways for reducing the most acute consequences of reactor accidents.
3. Determine the efficiency and reliability for fission product removal from liquid wastes by fixation on soils. This will ensure that liquid wastes are disposed in a manner that prevents contamination of the Columbia River during accidents or from chronic release.
4. Determine the characteristics of particle and aerosol formation from overheated nuclear materials such as plutonium and its compounds. This will provide a better understanding of shipping hazards.
5. Obtain a better understanding of the reaction of structures at the Richland site to seismic shocks. Determine response of instruments and equipment components to seismic activity.
6. Obtain quantitative information on meteorological parameters at the reactor sites along the Columbia River. Provide an appropriate diffusion model for evaluating an atmospheric release of fission products following a postulated accident involving a Hanford production reactor.

Incentives

The main incentives for conducting the Nuclear Safety Program are:

1. By substituting facts for conservative conjectures, show that the continued operation of the Hanford facilities is fully consistent with the imposition of increasingly stringent safety requirements. In cases where improvements in safeguards may be required, assure that the necessary capital expenditure is correctly directed to obtain maximum benefits.

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2. Obtain optimum use of Hanford facilities to further programs deemed within the public interest without introducing undue hazard to persons operating the facilities and residing near the plant boundaries, to public and private property outside project boundaries, or to the plant capital investment.
3. Develop technology and techniques applicable to other government sites and to the nuclear industry as a whole.

This Nuclear Safety Program is, of course, complementary to the safety research and development sponsored by the Division of Reactor Development and Technology. Specific features of the Hanford reactors and site are studied in light of progress and developments in the O4 program R&D. Examples of the special Richland site conditions requiring R&D study are: Zircaloy-clad metallic uranium fuel, aluminum clad metallic uranium fuel, large masses of uranium in the reactors, unique piping, fuel, and moderator geometry, unique emergency cooling systems, fission product confinement systems, soil disposal of wastes, large distances from the reactors to populated areas, unique terrain near the reactors, and unique site geological and seismic characteristics.

STATISTICAL SUMMARY SCHEDULE

	<u>FISCAL YEAR</u>				
<u>Dollars (IN THOUSANDS)</u>	<u>1969</u>	<u>1970</u>	<u>1971</u>	<u>1972</u>	<u>1973</u>
DUN - C & K Reactors	200	250	250	250	250
DUN - N Reactor	400	450	470	450	470
ARHCO	<u>70</u>	<u>70</u>	<u>100</u>	<u>100</u>	<u>100</u>
Total	670	770	820	800	820
<u>Man Years</u>					
DUN - C & K Reactors	2.8	3.8			
DUN - N Reactor	10.8	10.9			
ARHCO	<u>0.2</u>	<u>0.2</u>			
Total	13.8	14.9			
Equipment	283.5	47			

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PROGRAM DETAILS

Subprogram 8RLa

Fuel Temperature Transients Under Accident Conditions

Scope and Objectives of Subprogram 8RLa

The program will involve experimental and analytical evaluation of the heat transfer and coolant fluid flow as a function of location in the reactor and time after initiation of the accident. These studies will be related to the various fuel geometries used for production loads. The techniques developed will benefit all Richland production reactors as well as all other reactors using process tubes. Descriptions will be developed for both the dual purpose N Reactor and the low pressure C and K Reactors.

Results of this program will permit accurate calculation of fuel temperature transients as a function of location in the reactor following loss of adequate cooling. The calculated transients in turn will be used to determine the fraction of fuel in the reactor which would release fission products as a function of time after initiation of an accident.

Incentives for Subprogram 8RLa

An event affecting the capability to remove heat from the reactor fuel could result in a major fission product release. This type accident is considered the principal hazard of operating the Richland production reactors.

The resulting fission product release is a function of the fuel time-temperature history. Thus, ability to predict accurately the fuel temperature transients following a loss-of-coolant accident is an essential factor in assessing the real risk of reactor operation. The information developed by this part of the program will provide a basis for such accurate predictions.

In the absence of firm data, temperature transients are calculated using best available estimates of key parameters. It is very desirable to remove all possible uncertainties from calculations of such importance.

Limits are now imposed on the power density of N Reactor fuels. These limits are based on delaying fuel failure caused by cooling loss so that engineered safeguards - emergency cooling and confinement - could function properly. Results of studies carried out under the Fuel Temperature Transients Subprogram will improve the validity of these limits permitting maximum, but safe, power-density limits. This in turn may permit reactor operation at increased power levels.

Progress During Report Period

Experiments were completed to determine the pressure and temperature transients following a complete stoppage of coolant flow to a K Reactor process tube. The

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data will be used to normalize a computer program which is being developed to calculate the coolant boilout transients and void coefficients in process tubes resulting from a sudden reduction in coolant flow. Preliminary progress in this program is reported in BNWL-CC-1495.

The ability to analyze post-accident fuel temperature transients for the C and K Reactors was improved. A computer program was developed to calculate the reactor coolant flow, pressure, and temperature transients following sudden reductions in coolant flow which could result from complete breaks of major inlet piping. A method was developed using these data to estimate the post-accident fuel temperature transients. Analytical studies of the crossheader failure accident were improved. A computer program was written which provides more precise estimates of the coolant flow distribution in the tubes fed by the stricken header, thereby refining estimates of the post-accident fuel temperature transients.

Analysis supporting design of the N Reactor loss-of-coolant heat transfer experiment has been completed. The experimental apparatus required to determine critical heat transfer parameters following loss of cooling has been designed, fabricated, and installed. Experiments are in progress. A computer program for analysis of N Reactor core temperature history following loss of coolant (NLOC) has been prepared. The program uses current technology to compute core thermal transients for fuels of the tube-and-tube or target-and-tube type. Analysis of core thermal transients resulting from loss of coolant to a Mark IV core has been completed.

Formal documentation of the fuel temperature transient calculational model for the N Reactor was completed, including a study of the sensitivity of the results to various input parameters.

Evaluation of Effort on Subprogram 8RLa

Five milestones to be accomplished in FY 1968 were identified in the program document prepared last year, RL 3-8. Three were accomplished:

1. N Reactor Mark IV temperature transients were calculated.
2. Coolant boilout transients for the C and K Reactors were updated.
3. Temperature transients in the C and K Reactors were predicted.

One was cancelled:

1. The Mark V fuel (advanced coproduct design) program for N Reactor was cancelled, therefore the Mark V fuel temperature transients were not calculated.

One milestone was missed:

1. Measurements of radiative heat transfer from zircaloy-clad fuel have not been made as planned but the preliminary work is well underway.
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In summary, the very important work included in Subprogram 8RLa is proceeding well and on schedule in nearly all respects.

Budget Period Plans Subprogram 8RLa

FY 1969

Advanced design fuel elements for use in the C and K Reactors with enlarged process channels will be tested and analysis initiated to determine post-accident fuel temperature transients. The results of the experimental program to study the effects of sudden loss of coolant to a process tube will be documented. Calculation of coolant pressure transients in the process channels following sudden loss of coolant due to pipe ruptures will be completed.

Experiments measuring the net heat transfer from fuel to process tube using the N Reactor geometry and materials will be completed. While the test results will not attempt to separate radiative, conductive, and convective heat transfer rates, the overall measurement will provide a measure of the validity of the analytical calculations which does examine the various heat transfer mechanisms in detail. If the agreement is reasonably good, more sophisticated very expensive and time consuming experiments will not be required. If agreement between experiment and analysis is not suitably close, the experimental program will be expanded. If the need is identified, test apparatus will be prepared for experiments to determine individual emissivities of zirconium surfaces undergoing reaction with steam following an accidental loss of coolant. These experiments will provide basic data required to evaluate core temperature history of advanced N Reactor and C and K Reactor cores following loss of cooling.

FY 1970

The analytical programs will be modified to include the experimental heat transfer measurements to the degree possible. Also, results from metal-water reaction measurements (an energy input term) will be factored into the calculations as will the release of f.p.'s (reduction in energy generation) as these results are defined in Subprogram 8RLb.

Experiments establishing surface emissivities will be completed if this program is engaged in. Data from these tests will be incorporated into the NLOC program providing general capability of computing core temperature histories of zirconium clad fuels.

Analytical programs will continue to be improved by utilizing experimental test results and by utilizing the results of the metal-water reaction studies. In this latter regard, the actual measured rates of Zircaloy-2 and uranium-metal reactions rates will be factored into the calculations as well as any conclusions on the possibility that the reaction may be limited by tube plugging or insufficient rate of delivery of water (steam) to the core.

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FY 1971-1973

Advanced design N Reactor fuel elements will be tested and analyzed to determine temperature transients under loss of fuel coolant conditions.

The computer models will be refined to include more accurate heat addition rates from chemical reactions and the effect of melting on the rate of heat transfer from the fuel to the graphite. A model of the heat transfer to adjacent cooled channels will be added to the programs. The fuel temperature transients for different conditions of partial and complete loss-of-coolant will be completely characterized. This information is expected to allow a more realistic evaluation of the consequences of coolant loss accidents in both the C, K and N Reactors.

The feasibility and need for developing a computer code including both physics and thermal hydraulic parameters will be examined. Such a code would permit a more sophisticated investigation of nuclear excursions and might lead to a basis for relaxation of limits now deemed necessary to satisfy reactor speed-of-control requirements. This work may have applicability to both reactor types.

Milestones

1. Experimentally measure heat transfer from Zircaloy-clad fuel to the reactor core at temperatures up to 1100°C.	October, 1968
2. Refine analysis of coolant pressure transients in C and K Reactors following sudden loss of coolant.	June, 1969
3. Perform tests on advanced fuel element designs for the C and K Reactors to determine post-accident fuel temperature transients.	June, 1969
4. Perform analysis of advanced design fuel elements for the C and K Reactors to determine post-accident fission product releases.	June, 1970
5. Experimentally measure individual Zircaloy-clad fuel surface emissivities.	June, 1970

Reference Reports Issued

BNWL-CC-1495, "Digital Program for the Prediction and Analysis of One-Dimensional Two Phase Flow Transients Following Sudden Loss of Coolant or Loss of Pressure," D. C. England, K. C. Garlick, R. W. Moulton, and D. S. Rowe, December, 1967 (Unclassified).

RL-GEN-1371, "Postulated Loss of Coolant to Coproduct Fuel in N Reactor - Results for Mark II," P. Riggle, June 30, 1967 (Confidential).

RL-GEN-1550, "Fuel Heating Following Loss of Cooling," N. R. Miller, March 29, 1967 (Confidential).

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Equipment Funding

FY 1969

Switchgear, control, and other equipment for
emissivity tests.

\$30,000

FY 1970

Equipment for emissivity tests.

\$10,000

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Subprogram 8RLb
Chemical and Metallurgical Reactions and Fission Product
Releases from Overheated Fuel

Scope and Objectives of Subprogram 8RLb

This subprogram is designed to develop, primarily by experimental methods, a characterization of the rate and extent of metal-water and metal-metal reactions and the quantities and rates of release of biologically significant radioisotopes from overheated fuel. Full-size metallic fuel and target materials will be used.

The subprogram will characterize isotope removal by plate-out on hot or cold surfaces, rain-out by fog sprays and retention on filters or collectors. The results will permit more realistic evaluation of the consequences of failing or melting a given quantity of fuel or target material in a reactor core.

The results of this subprogram interact with Subprogram 8RLa. Information obtained from fuel temperature transient studies show the rate of temperature increase and the terminal fuel temperatures which should be studied. Conversely, metal-metal and metal-water reactions constitute a heat input source which is considered in the calculation of the temperature transients.

Also, information on fission product release presents inputs to Subprogram 8RLc and 8RLd, studies of the control of fission gases and studies of fixation of radioactive materials by soil.

Incentive for Subprogram 8RLb

In the absence of information, arbitrary assumptions are made on the rate, the extent, and the nature of fission products released from overheated fuel. In some cases these assumptions are probably in error by 1-2 orders of magnitude on the pessimistic side. Demonstration that this error does exist would have an extremely favorable impact on the assessment of the level of risk associated with the operation of the reactors. In turn, this could prove to be pivotal in assessing the comparability of the Hanford reactors to the safety standards for commercial plants and to decisions on the utilization of land near reactor boundaries.

Metal-water and metal-metal reaction problems are vexing to the entire nuclear industry. Insight gained on this subject will greatly reduce the uncertainty in the assessment of the risk of Hanford reactor operation and could apply to important degrees to other reactor types.

Progress During the Reporting Period

Six tests using full-size unirradiated fuel elements from a K Reactor were conducted to determine the effect of aluminum and zircaloy process tubes on the uranium-aluminum reactions. In previous tests, the aluminum tubes

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contributed to the exothermic reaction initiated at about 1080°C. Tests with the zircaloy tubes showed no reaction between the tubes and other metals. The change in diameter of the reacted fuel elements observed in some of the tests indicated that it would be likely at least one element in a channel heated to 1000°C or higher during an accident would plug the channel.

Four full-size N Reactor irradiated fuel failure tests have been performed in FY 1968 for evaluation of fuel failure temperatures (below the uranium melting point), identification of mechanisms and extent of failure, and measurement of fission product release fractions. Results from these tests, performed with Mark II alloy (U-Fe-Al) fuel at exposures slightly above the average goal exposure, showed more release of uranium foam from the cladding than had been observed in the earlier six tests with Mark I alloy fuel at lower exposures. As a result of the foaming, noble gas releases were higher than measured earlier, but still only about 50 percent of the conservative value used in current safety analyses for failed but unmolten fuel. On the plus side, the foaming could appear to limit the access of steam to the fuel column and thus reduce the magnitude of the metal-water reaction. Fuel failure temperatures were found to be in the 1860°F to 1920°F range, compared to a 1900°F failure temperature used in previous safety analyses.

The metal-water reaction experimental program has developed significantly with the experimental determination of zirconium-steam reaction rates for full sized unirradiated N Reactor fuels. These reaction rate laws show a reduced reaction rate compared to the Baker-Just (of Argonne) laws that have been used in current N Reactor safety analyses. Successful use of a mass flowmeter for continuous measurement of hydrogen evolution during these tests has greatly enhanced the overall metal-water reaction study program. This represents a major step towards continuous determination of the metal-water reaction rate with failed and molten fuels where sudden changes in physical geometry may result in widely varying reaction rates.

All major modifications of the 324-D hot cell for use by this program in testing molten fuels are complete. Various small jobs are now being completed, and final checkout in preparation for tests will begin soon. Delivery of support equipment for the induction heater has delayed initial testing to early July, 1968. Tests in this facility will provide a better description of the effects of metal-water reactions, and the character and extent of fission product release from molten fuel.

Eight-inch length full-diameter fuel elements of the Mark II coproduct geometry are being irradiated in the reactor at the present time, and fuel of the Mark IV geometry has been fabricated for later irradiation in the reactor. Irradiation and subsequent testing of new fuels will ensure that the metal-water reaction and fission product release results will apply to fuel currently in use at N Reactor.

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Evaluation of Effort on Subprogram 8RLb

While delay in project initiation and delays in procurement of some items of equipment prevented attainment of beneficial use of the 324-D facility in FY 1968, it appears the facility will be fully operable in early FY 1969 and will constitute a powerful tool in studying the behavior of overheated fuel.

Failure testing of the four N Reactor Mark II fuel elements together with six tests conducted previously with Mark I fuel provides a good basis for establishing the temperature at which rapid failure of irradiated fuel can be anticipated. This temperature in turn constitutes a firm basis for establishing limits or specific fuel powers to assure fuel failures would not be induced during transient following accidents requiring reactor cooling with the emergency cooling system.

Data from the ten tests with the N Reactor Zircaloy-clad fuel together with data from an additional six tests with C and K Reactor aluminum-clad fuel product definitive proof that previously assumed fission product release fractions from failed fuel are erroneous. In all cases the assumed fission product release fraction, 100 percent of the noble gases, 50 percent of the volatile solids and halogens and 1 percent of the non-volatile solids have been demonstrated to be high.

The metal-water reaction studies which show that the actual Zircaloy fuel clad surfaces react with steam at a lower rate than predicted by tests with laboratory-sized coupons (Baker and Just work at ANL) are considered quite important. The data provide a further basis for evaluating results of tests with irradiated Zircaloy-clad fuel.

Observation of the behavior of irradiated fuel heated to failure shows that on failure uranium is extruded from the clad with a copious increase in volume. It appears possible that this effect may be significant enough to block the flow passage through the fuel column, prevent steam from reaching surfaces and hence reduce the potential consequences of the metal-water reaction. This effect will be carefully studied in future testing.

The metal-water reaction test program in total lagged behind expectations. Work which was scheduled during the year but not accomplished included:

- Measurement of the reaction rate of irradiated fuel cladding and steam.
- Measurement of the reaction rate of unmolten irradiated uranium extruded from failed fuel and steam.

This lag resulted from difficulties in perfecting the technique for monitoring hydrogen generation rates in turn used to measure the rate of the reaction:

Where "M" is metallic zirconium or uranium.

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A technique involving measurement of the gas mass flow rate from the test section and the hydrogen content of the gas (using a gas thermal conductivity method) has been developed and produced good results in the second series of metal-water tests noted before. However, the gas mass flowmeter seems not to be a sturdy instrument and the work is still plagued with instrumentation difficulties.

During FY 1969, the metal-water reaction studies will be emphasized with the aim of regaining the original schedule.

The initial experiments to determine the heat of formation of uranium-aluminum compounds were unsuccessful. Further efforts on experimental techniques and apparatus would be required before this could be resumed.

The fission product release tests on the aluminum-clad fuel were delayed pending completion of the 324-D facility. Tests with unirradiated fuel to explore further the uranium-aluminum reactions were completed as planned.

Plans and Expected Results Subprogram 8RLb

FY 1969

The new 324-D hot cell facility will be utilized for experiments with full-sized Zircaloy-clad and aluminum-clad metallic uranium fuels. These elements will be heated to and held at various temperatures including those above the uranium melting point. The 292-T facility will still be used for additional tests with non-molten fuel with an insignificant effect on costs of the program. The metal-water and metal-metal reactions will be measured for both the total heat of reaction and the rate of reaction for both type fuels. Zircaloy-water reaction rates will show if irradiation effects change reaction rates (swelling induced by fission product gases could damage protective oxide layers). The first full scale measurements of unmolten uranium-water reactions following fuel cladding failure will be obtained, followed by measurement of the molten uranium-water reactions.

The relationships between fission gas blanketing, oxide thickness, and physical changes on the metal-water reactions will be determined. The effects of irradiation exposure on the fuel failure characteristics and metal-water reaction effects will be tested.

Coincident with the measurements of the metal-metal and metal-water reactions, a comprehensive study of fission product releases will be accomplished for both failed fuel (cladding breached) and molten fuel of geometries currently in use at the C, K and N Reactors. The fuel elements will be inductively heated at rates expected in reactor loss of coolant incidents to various goal temperatures. The time that fuel elements are held at goal temperatures will be sufficient to ensure that fission product release is complete or that the rate of release has reached a constant value.

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Particular emphasis will be placed on determination of the forms of iodine and other fission products in the atmosphere leaving the fuel piece, to provide an improved basis for future evaluation and testing of removal techniques such as plate-out, washing by sprays, and filtration.

FY 1970

Experiments designed to determine the rate of heat generation for metal-metal reactions under simulated accident conditions will be performed. These experiments will be aimed at establishing the basis for more accurate prediction of fuel temperature transients.

Experiments to determine the fission product release rates above the uranium melting point will be continued, the deposition of fission products on reactor piping and other surfaces will be measured, and the effect of temperature and atmosphere will be correlated with fission product character and formation. The performance of sprays and filters in removing true fission product aerosols will be studied. The experimental arrangement will essentially be a mockup of a section of a reactor fuel channel with the clad uranium element heated to goal temperature at the inlet end. The process tube and exit hardware will represent the remaining pathway to a pipe break location. Fission products escaping this point will pass through filters and charcoal traps, representative of the C, K and N Reactor confinement systems. A measurement of the penetration of fission products to each section of this path will be made. These tests, using actual fission products, will be fully integrated with the tests of Program 8RLc using simulated fission products.

Testing of metal-water reaction will continue to add statistical significance and add necessary information on new type fuels.

FY 1971-1973

Failure of N Reactor fuel caused by delayed injection of cooling will be studied. The fission product release rate subsequent to failure will be determined. Attention will be given to the plate-out of fission products on N Reactor core components under conditions simulating accident conditions as nearly as feasible. The physical nature and chemical composition of released fission products will be identified where pertinent and temperature effects on chemical form will be noted. Tests will be continued on N Reactor fuel to provide enough data to be of statistical significance. Concurrent experiments will be planned to test systems designed to mitigate the consequences of radioactive noble gas release.

Experiments designed to simulate the conditions in the C and K Reactors tube channels with either zirconium or aluminum tubes will be run if tests with N Reactor geometry indicate a need. The effects of reactor graphite on the release of fission products from ruptured or melted tubes will be experimentally evaluated.

Work will be extended to cover new fuel types being developed for reactor use. For example, use of oralloy fuel consisting of an aluminum-uranium or zirconium-uranium alloy would require considerable extension of the subprogram.

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Milestones

1. Measure the Zircaloy-water reaction rate using irradiated fuel elements heated to failure in the 292-T facility. Obtain gross estimate of rate when irradiated fuel fails and irradiated uranium is extruded into a steam atmosphere. July, 1968
2. Initiate metal-water and fission product release tests where irradiated N Reactor fuel is heated to temperatures including those above uranium melting. July, 1968
3. Confirm preliminary fission product release rates from aluminum-clad fuel below the uranium melting temperature. December, 1968
4. Determine rate of fission product release from molten aluminum-clad and molten Zircaloy-clad fuel. December, 1969
5. Determine the rate heat is generated in uranium-aluminum compound formation at various temperatures. July, 1970
6. Complete the definition of the rate, nature, and extent of fission product release caused by a loss-of-cooling accident by integrating the results of the fission product release studies, metal-water reaction studies, and the fuel temperature transient studies. July, 1972

Reference Reports Issued

DUN-3955, "Fission Product Release Rate from Aluminum Clad Uranium Fuel," F. C. Traxler, May 1, 1968 (Secret).

RL-GEN-1535, "Alternate Formations of the Rate Laws for the High Temperature Metal-Water Reactions Involving Uranium and Zircaloy-2," T. W. Evans, March 23, 1967 (Unclassified).

RL-GEN-1541, "Metal-Water Reactions During a Loss-Of-Coolant Accident in N Reactor," T. W. Evans, March 23, 1967 (Unclassified).

Equipment Needed

FY 1969

\$91,000

FY 1970

\$31,000

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Subprogram 8RLc
Control of Fission Gases

Scope and Objectives of Subprogram 8RLc

Radioactive noble gases and iodine released from fuel during and following an accident would be transported by various pathways from the reactor. The transport agents would include water, steam and air. The factors affecting the quantity of fission gases transported through the various paths and the amount finally released to the atmosphere must be known if valid assessments of the accidents or systems to reduce the amounts released are to be made. The program will include:

- o Improved definition of the absorption of noble gases and halogens into turbulent streams.
- o Better understanding of the kinetics of release of fission gases from water pools.
- o Determination of the leakage of noble gases from water entering a porous media such as the earth below a discharge drain field.
- o Experimental and actual measurements to determine the extent that organic iodides are formed in a confinement system.
- o Experimental measurements on the effect of continuous use on the effectiveness of charcoal filters.
- o Analysis and demonstration of the effectiveness of fog spray in removing elemental and organic compounds of iodine.

Incentives for Subprogram 8RLc

The existing safeguards at the Hanford reactors would serve to efficiently prevent release of particulate fission products from the confiner following an accident. It is also probable that these safeguards, fog sprays and filters, would also efficiently remove halogens. As matters now stand, however, it is expected that the noble gas release would be essentially complete.

Since fission product release from the confiner is the principal hazard of reactor accidents, it is important that the magnitude of the noble gas and halogen release be known accurately and be reduced to the extent practical.

Favorable results from the study or improved safeguards provided as a result of the study could have the effect of assuring that the reactors meet the site criteria, 10 CFR 100, with large margins to spare.

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Progress During Report Period

An analysis was performed to predict the behavior of noble gases released from failed fuels in the C and K Reactors. For the accident conditions assumed it was determined that a large fraction, 90 percent or more, of the noble gases escaping from the failed fuel might remain dissolved in the water of a covered quench tank, and not escape to the air in the tank. Subsequent disposal of this water to a long, gravel-filled trench was studied analytically. It was determined that if the noble gases are released in the trench, they would be expelled into the soil by pressure created by the rising water and would initially occupy voids below the soil surface. An assumption was made that an impermeable sheet such as vinyl would be used to seal the crib ten feet beyond the edge of the excavation. Conservative assumptions were made that the gas disengaged vertically from the water into the space over the water in the covered crib or trench and that the slowly rising water displaced the air and gases into the soil pores through the sloping side of the crib. The noble gas which is moved out into the soil would diffuse slowly through the pores of the soil and over a period of time would be released from the soil along the crib. Further study is underway to evaluate analytically the diffusion from soil to air, and the resulting downwind concentrations. The study pointed up the need for experimental data to define the movement of noble gases in soil.

Apparatus was assembled for evaluating the efficiency of candidate charcoals for reactor filter banks. The efficiency of aged charcoal for methyl iodine removal under conditions of moderately high humidity is of particular concern.

The oxidant detector which is being tested as a potential measurement instrument for in-place charcoal testing has not proven as sensitive as expected. Radioactive tracers and activation analysis are being tested in the laboratory as alternates for in-place efficiency measurements.

Evaluation of Effort on Subprogram 8RLc

Theoretical relationships for the absorption and retention of fission gases in reactor effluent waterstreams and storage basins were established. Analysis using these relationships indicated an emergency disposal system could be designed to limit the release rate of noble gases sufficiently to enable the confinement system to handle larger releases. An experimental program was also established.

Progress on development of methods for in-place testing of filter charcoals for iodine retention has been slower than expected.

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Plans and Expected Results Subprogram 8RLc

FY 1969

Two areas will be studied to better define the consequences of reactor accidents. First, the release of noble gas isotopes from water pools will be better defined. This study will continue the theoretical estimates of noble gas released from water systems and add experimental measurements as well. The experimental measurements will be made using nominal size vessels, and will include the effects of accompanying non-condensable gas, entry rate, sparge rate, and mixing. It is anticipated that the data will better define release of noble gases from water systems.

Secondly, the use of charcoals for reactor air cleaning systems will be studied. These studies will continue the current work to better define the behavior of promising charcoals for iodine and methyl iodide removal. Assistance will be given to develop appropriate unit and in-place charcoal filter tests for iodine removal and other performance requirements. It is anticipated that this research could lead to adequate methods for determining charcoal efficiency in day to day operation and for qualifying new filters.

FY 1970

In FY 1970 studies will be undertaken to increase the understanding of the behavior of noble gases and iodine following a reactor accident. More refined analyses of release from fuel and improved definition of escape pathways will require re-evaluation of accident consequences. It is anticipated that studies will be made of the conversion of iodine to methyl iodide in reactor atmospheres if it is shown that charcoal effectiveness for methyl iodide is low enough to give added importance to methyl iodide release in a reactor accident. Assistance will be given in perfecting and placing charcoal efficiency measurements in routine use.

The feasibility of coupling iodine and noble gas release studies to fuel failure research to be performed in 324 Building under Subprogram 8RLb will be investigated. Designs for accomplishing this will be developed should feasibility be shown.

FY 1971-1973

The research undertaken in these years will be responsive to the particular problems arising from continued safety analysis of the production reactors. It is anticipated that some additional noble gas release studies will be required to demonstrate the retention of noble gas in the entire waste stream discharge system installed for handling reactor loss-of-coolant waste water. The feasibility of performing in-place tracer studies with ⁸⁵Kr will be determined. Definition of the displacement factor for noble gases in water discharge to shallow trenches or cribs will be made, either through in-place testing, or through modeling studies. These studies are anticipated to firm up earlier estimates based on smaller scale studies and would provide the added assurance that assumptions made earlier were justified, or otherwise point to need for further controls.

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Milestones

1. Evaluate charcoals with additives after aging in confinement system service to determine their effectiveness in removing elemental iodine and methyl iodide.	July, 1969
2. Perform confirmatory experiments for the theoretical fission gases release rates simulating the range of conditions which might be encountered in an emergency liquid waste disposal system.	December, 1969
3. Establish the release rates of fission gases through typical reactor site porous media.	July, 1970
4. Perform preliminary evaluation of methods of retarding fission gases from water pools.	July, 1971
5. Determine extent that methyl iodide formation might be expected in the actual confinement systems.	July, 1971

Reference Reports Issued

None.

Equipment Funding

FY 1969

Tanks and flow measurement equipment	\$9,000

FY 1970

Miscellaneous equipment	\$6,000

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Subprogram 8RLd
Meteorological Studies

Scope and Objectives of Subprogram 8RLd

The meteorological program is designed to investigate the atmospheric transport, diffusion, and deposition of postulated fission product releases from the Hanford production reactor sites. The objectives of this program are:

1. To define and measure the meteorological parameters which would describe the path, speed, and dilution of radioactive materials released from a production reactor following a postulated accident.
2. To experimentally determine the dilution rates of tracers released at production reactor sites.
3. To provide a valid model which describes the atmospheric transport, diffusion, and deposition on the Wahluke Slope of fission products released from a production reactor following a postulated accident.

The scope of the program includes the measurement and evaluation of meteorological parameters in the vicinity of the Hanford production reactor sites and over the Wahluke Slope. Meteorological parameters will be measured which are expected to have an effect on the transport, diffusion, and deposition of a plume during its passage over the slope. An automatic data acquisition system will be installed at the N Reactor site to collect meteorological data. The data will be collected on magnetic tape. Portable meteorological sensor systems will be placed on the Wahluke Slope and in the vicinity of the reactor. These data will be collected on punch paper tape.

The evaluation of meteorological data will include calculations of joint probabilities of wind direction, speed, and variability as a function of atmospheric stability as measured at the N Reactor Meteorology Tower. This evaluation will also provide correlations of meteorological data between the N Reactor tower, portable meteorological measurement systems located on the Wahluke Slope, and the Hanford Meteorology Tower located on the 200 Area plateau.

A comparison of observed trajectories with those predicted from meteorological data will be used to evaluate the parameters which control the plume transport. Observed trajectories will be obtained from smoke releases and tracking of constant level balloons. These trajectory studies will enhance the understanding of the dynamics of the atmosphere over the Wahluke Slope. The trajectories will also be used to aid in establishing an air sampling grid for that part of the study which is designed to determine the diffusion and deposition of a plume.

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A series of experiments involving tracer release and sampling will be conducted to simulate the diffusion and deposition characteristics of postulated fission products released from the production reactors. The tracer sampling will be conducted at distances that will provide data that will characterize the atmospheric diffusion over Wahluke Slope. A diffusion model, selected for its applicability to the geography surrounding N Reactor, will be used to compute time-integrated air concentration values for comparison with observed values.

The model will be evaluated and its modification or replacement initiated as deemed necessary. The N Reactor site is chosen as the prime source point for field tests because of the availability of pertinent meteorological data from the tower facility.

The objectives will be accomplished, within the known techniques of sampling, using zinc sulfide as the primary tracer. As tracer and air sampler techniques improve, they may be designed into the program.

Incentives for Subprogram 8RLd

Safety analysis reports for the N Reactor have routinely ^{used} assumed the elevated release of fission products from the 200-foot stack following an accident as a description of the associated radiological hazards. Review of these safety analyses, by regulatory bodies, has resulted in a suggestion that the release should be taken at ground level (a more severe case) because of the Wahluke Slope rise to an elevation higher than the stacks. Data, to be collected during this program, holds promise of resolving the question and the uncertainty in offsite dose calculations. The present diffusion model, developed from Hanford experimental data over relatively flat terrain using point source meteorological information, will be compared with meteorological and diffusion data to be obtained in the Wahluke Slope region. Tracer studies are a most important aspect of the meteorological program as it will provide experimental diffusion data over the Wahluke Slope, responding to questions posed by regulatory and safety review bodies.

It has been the practice to extrapolate the 622-R Hanford Meteorology Tower data to the 100 Areas on the Columbia River for calculations in the diffusion model, and to indicate the speed and direction of movement of the material released. It is known that the maximum ground exposure varies as a function of meteorological conditions. The variation of meteorological parameters between the 622-R Hanford Meteorology Tower and the 100 Areas as well as the variation of parameters between the reactor areas and Wahluke Slope has indicated a need for measurements of the meteorological conditions along the Columbia River and over Wahluke Slope.

Progress During This Report Period

The automatic meteorological data acquisition system (N Met System) has been purchased and is now being assembled by the vendor at his factory. Wind sensors have already been received at the N site. Installation and check out of this system is expected to be completed early in FY 1969. The 300-foot

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tower (N Met Tower) has already been erected and utilities have been installed. Software is being specified at this time for the writing of the computer programs needed to process the data acquired by the system.

Some small scale smoke releases from the N exhaust stack were accomplished with marginal results. Because of poor visibility due to dilution with normal stack air flow and an unreliable smoke generator, these tests were temporarily discontinued.

Four portable, wind sensors, with strip chart recorders, have been installed on the Wahluke Slope and at a river location a few miles upstream of the reactor site to help determine surface wind flows in the area. Two bivane wind sensors and two temperature sensors have been temporarily installed at the 200-foot and ground levels on the N Met Tower to record meteorological data at this site until the permanent N Met System is installed and operational. Data reduction from these sensors has been partially completed and preliminary reports prepared.

Trial runs have been completed using constant-level balloons and preparations are complete for a concentrated effort to determine trajectories of air parcels by this method in and around the reactor areas. Two theodolites will be used to determine more accurately balloon position and speed.

Smoke releases from the ground and from the N Met Tower will use military smoke bombs as a source. Photographs will be taken of these plumes in order to qualitatively determine turbulence characteristics during various atmospheric conditions.

Evaluation of Progress on Subprogram 8RLd

The meteorological program continues to develop at a good pace. In FY 1967, system specifications for the N Met System were prepared. During FY 1968, bids were evaluated, orders were placed and some components received. Acquisition of local meteorological data using temporary sensors represents a significant milestone in acquiring the desired quantitative understanding.

Plans and Expected Results Subprogram 8RLd

FY 1969

Meteorological data will be acquired continuously by the N Met System and will be used in generating statistical climatological summary reports.

Balloon releases will continue to ascertain winds up to 5000 feet and low level wind trajectories. These tests will consist of double-theodolite tracking of constant level (0-1000 feet) and pilot balloons (0-5000 feet). Smoke will be released and photographed, both from the N Met Tower and from ground sites around the reactors and on the Wahluke Slope.

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Preparations will be made for diffusion experiments to test the application of the Hanford model to the N Reactor site. These experiments will consist of releases of tracer material and subsequent sampling of the air downwind from the release using both surface and airborne methods.

Two wind stations will be added to the devices now on the Slope to more fully define wind flows there. These will be automatic stations powered by thermo-electric generators with measurements recorded on punched paper tape.

Initial correlations of data between the BNW meteorological tower, the N meteorological tower, and portable stations on the Wahluke Slope will be made to provide a preliminary insight into the overall wind patterns and aid in the analysis of meteorological parameters.

Normalized exposures (E/Q) will be computed using the bivariate normal diffusion model for those meteorological conditions recorded at the 100N Met Tower, and from established climatology at the 622-R Hanford meteorology tower. The E/Q values will be used to indicate those meteorological conditions which create the greatest exposure over Wahluke Slope.

FY 1970

Meteorological data collection, reduction and summarization will continue. An initial climatic summary will be made, and joint probability statements concerning the occurrence of certain meteorological conditions and their duration will be developed based on two years of data.

The continuing trajectory study will emphasize conditions believed most likely to produce the worst case situations on Wahluke Slope. The observed smoke and balloon trajectories will confirm the meteorological data for better understanding of the dynamics of the atmosphere, if adequate meteorological instrumentation is available on the Wahluke Slope.

Portable sampling procedures for measuring the dilution of a tracer released to the atmosphere will be checked. Present procedures typical of fixed grid systems will provide a basis for evaluation, and alterations will be made to these procedures to provide portable sampling techniques. Other techniques known to be portable will be compared and evaluated. The sampling technique will then be checked on the Wahluke Slope to assure that no unforeseen problems have arisen.

FY 1971-1973

The dilution of tracers will be measured downwind of release in a series of experiments aimed at evaluating the dilution model for fission products released to the atmosphere. The experiments will originate primarily from the N Reactor and data extrapolated and verified as necessary for other reactor sites. Model modification will be initiated if data evaluation

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indicates that it is necessary. These tests will include both surface and real-time airborne measurements of time integrated and instantaneous concentrations of tracer released during both stable and unstable conditions and in each season of the year.

The results from the meteorological studies will be evaluated and a final report issued describing the results of the relationships observed on the dispersion and path of fission products released from reactor sites.

The joint probabilities of adverse dispersion conditions and unfavorable wind directions will be evaluated and factored into safety analysis calculations.

Milestones

1. Issue initial reports on meteorological parameters at the N Reactor site, including possible effect on fission product release consequences.	June, 1969
2. Issue major report on meteorological parameters at the N Reactor site, including correlations with the BNW tower data and data from portable stations on the Wahluke Slope.	November, 1970
3. Complete diffusion testing at the reactor sites.	June, 1972
4. Prepare final report on all phases of the program.	June, 1973

Reference Reports Issued

RL-GEN-1665, "AREA: A Computer Code for Calculating Isopleth Areas,"
D. A. Baker, R. L. Engel, August 4, 1967.

Equipment Costs

FY 1969

\$30,000

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Subprogram 8RLe
Ground Fixation of Radioactive Materials in Liquid Wastes

Scope and Objectives of Subprogram 8RLe

The objective of this program is to evaluate the adequacy of ground disposal of contaminated liquid wastes generated by normal processes or accidents.

This program will characterize the depth and flow of ground water (as a function of river level, and season of the year) in the vicinity of N Reactor, initially, and at other reactor and chemical processing sites later.

The capacity of the soil for permanent retention of radionuclides will be determined from laboratory evaluation of core samples taken from test wells and from tracer studies in the field which will disclose the nature and rate of transport of radionuclides in the soil and ground water at selected sites.

This effort will be coordinated with the Columbia River Studies (Mission 10) to reduce the effluent activity entering the river from the C and K Reactors. Information from that program on ground water flow patterns and flow times near the C and K Reactors will be valuable in applying the results of this program to other locations.

Incentives for Subprogram 8RLe

Ground disposal of radioactive wastes generated during normal operation or during non-standard conditions is very attractive if it can be shown that hold-up would be adequate to assure that no excessive contamination of ground or river water would eventually result. Further, information obtained from these studies will serve to guide the design requirements and scope of facilities required to clean up solutions prior to ground or river release.

Progress During Report Period

Actual irradiated fuel rupture debris was passed through laboratory soil columns typical of those at 100N to determine the soils retention of strontium and cesium as a function of water volume passed through the soil. Results verified those reported earlier from simulated waste water tests in which the decontamination factors were less than desired until a significant quantity of water (five-column volumes) had passed through the soil. Simultaneous with efforts to identify the cause of this undesirable effect, laboratory tests were performed to develop possible corrective actions. The most promising treatments to date include either pre-leaching the soil with water or use of a combination of a polyelectrolyte (such as Magnifloc) and alum. Such a treatment of waste water to the new disposal basin being installed at 100N by the Effluent Control Project could probably be done at reasonable cost. However, laboratory results must be further evaluated by field tests before incorporation into an accident-waste disposal system can be done with confidence.

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Chemical waste disposal criteria were established for use in evaluating specific waste disposal problems. The operating characteristics of the 1301N crib are currently being examined by review of the considerably larger body of sample data and reactor operating history than was available when ground water travel times and decontamination factors were previously evaluated. Plans are being made to conduct a dye test to provide additional information on travel time distributions.

Radiochemical analyses of test well soil samples from sites near the 100N crib has been partially completed and will provide valuable information on radionuclide movement through the soil.

Evaluation of Effort on Subprogram 8RLc

The successful utilization of soil disposal of radioactive wastes is dependent upon several radioisotope holdup or delay processes. Ion-exchange and filtration are important for retention of several key elements, including strontium and cesium, while others, such as iodine, are not retained to a significant degree and reduction in activity is brought about mainly by decay during water travel through the soil to the river.

One objective of this program has been to understand the basic processes and develop methods for maximizing the retention of strontium and cesium by the filtration and ion-exchange processes. A FY 1968 milestone was to develop an understanding of the early break-through phenomenon for strontium and cesium. The fact that pre-leaching was successful indicates that the phenomena is a reflection of a "soil process" rather than a characteristic of the form of the radioactive particles. However, a complete understanding was not achieved, and studies are continuing. When it became apparent that a full understanding of the break-through process would not be readily attained, development of treatment methods to reduce the break-through was given greater emphasis, and several highly promising methods have been developed in the laboratory. This part of the effort is now ahead of the schedule outlined one year ago, although final choice of a treatment process awaits both actual field tests and a better understanding of the mechanism to lend confidence in the long term applicability of treatment techniques.

The development of chemical waste disposal criteria for the existing 100N crib was completed on schedule. Results from this effort have frequent applications as various chemicals used at the reactor are separately evaluated for crib disposal to ensure that radionuclides now held in the soil below the 100N crib will not be disturbed.

Evaluation of cesium and strontium migration through the soil by analysis of test well samples has not been completed as scheduled due to delays in the analytical laboratory.

There was no effort scheduled in FY 1968 for evaluation of ground water flow paths from the existing 100N crib and from the new disposal facility to be constructed. Studies in this area will be engaged in FY 1969 to provide

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more definite answers on the water travel directions and travel times to permit a better description of the Iodine-131 concentrations expected at the river bank following a postulated accident.

Plans and Expected Results Subprogram 8RLe

FY 1969

Work will be continued to evaluate the basic cause of the early break-through of strontium and cesium in the soils. An understanding of the cause of this phenomenon is necessary to ensure that treatment methods being simultaneously developed will be of long term rather than temporary value. Further evaluation of treatment methods will continue, and planning and initiation of field testing will be done.

Initial estimates will be made of liquid movement rates from the emergency disposal facility to the Columbia River using the best available transient flow analysis technique and hydrological data.

Sample analyses of soil drillings obtained near the 100N crib will resume to determine radionuclide migration rates.

FY 1970

The field testing program will be completed in which simulated wastes will be introduced to small-scale field disposal sites and the migration of radionuclides and particulates from the sites measured to supplement laboratory results from previous studies. The success of particular treatment methods will be finally evaluated. Initial field data will be gathered for inputs to a more sophisticated hydrological system model necessary to better predict the movement of radioactive wastes from the present crib and from the new disposal basin.

FY 1971-1973

The hydrological system model will be verified by extensive field testing at each reactor area. A transport model will be developed that will allow diffusion, dispersion, radioactive decay, and other interactions to be evaluated. The model will be supported by field data on permeability and ground water potential.

Milestones

1. Evaluate data on cesium and strontium migration obtained from test wells near the 100N crib and relate to laboratory studies. June, 1968
2. Complete field evaluation of means for reducing cesium and strontium break-through. June, 1969

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3. Interim flow model of emergency disposal facility.

June, 1969

4. Final flow models and evaluation of long range criteria for waste handling completed.

June, 1972

Reference Reports Issued

None.

Equipment and Construction Costs

FY 1969

For test wells.

\$25,000

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Subprogram 8RLf
Particles and Aerosols from Burning Nuclear Materials

Scope and Objectives of Subprogram 8RLf

The purpose of this program is to determine the technological parameters which influence the accidental airborne release of plutonium and the ultimate consequences which would ensue. Factors which govern the nature of the released material and the fraction airborne and transported to points where exposure might result are to be determined. Better understanding of the release mechanisms, the fraction released, the nature of the aerosols, and any other parameters affecting ultimate disposition should assist materially in arriving at more realistic hazards evaluations and help to establish preventive and recovery measures. Accidents involving plutonium and its compounds in which these materials may be subjected to high temperatures are of particular concern in this facet of nuclear safety research. The research, however, is not necessarily restricted to incidents involving overheating of plutonium and its compounds, but will embrace as appropriate the movement and deposition of other aerosols in a variety of situations. The release rates of fission product aerosols as defined in previous sections of this Mission should provide data that will be a corollary to the present plutonium work.

Incentives for Subprogram 8RLf

By substituting facts for pessimistic estimates of maximum severity, this program will permit the elimination of excessive conservatism in hazards evaluation of both Hanford nuclear facilities and shipment of nuclear materials. By combining the actual release rates of nuclear materials subjected to high temperatures and the probability of high temperature accidents a more realistic hazards evaluation of postulated accidents can be obtained.

Progress During Report Period

The fractional release of plutonium from several compounds of plutonium was measured as a function of temperature and air velocity. The experimental arrangement for this study was chosen to simulate in a general way the approach of air in convective flow up to the sample and vertically above the heated material. Plutonium oxide, partially oxidized oxalate, oxalate, and fluoride were subjected to temperatures of 400, 700, and 1000°C. Air flow in the chimney above the compound was maintained at 10, 50 and 100 cm/sec in separate experiments. The material produced during the heating of partially oxidized plutonium oxalate was most readily airborne at the lowest rate of heating and air flow. At all temperatures and air flows, measurable releases occurred; the rates ranged from 0.057 to as much as 0.82 wt. %/hour. The fractional release from "fresh" plutonium oxalate was a marked function of the air velocity used. Rates ranged from 0.007 to 0.9 wt. %/hour. The amount airborne from a mass of finely divided oxide ranged from 5×10^{-6} to 2.5×10^{-2} wt. %/hour. Plutonium fluoride, although containing the

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largest fraction of fine particles in the original material in most cases, released a smaller fraction, 0.007 to 0.05 wt. %/hour than did partially oxidized and fresh plutonium oxalate, presumably due to changes during heating. Although a wide range of release rates were observed, the data are valuable in bracketing the anticipated airborne release fraction. Size distribution data were obtained in these studies.

A series of experiments was performed using plutonium nitrate as the starting material. The fractional release from the slow evaporation to near dryness of the concentrated solution, during which a current of air was passed across the shallow pool, was first measured. Unless visible boiling occurred (which was deliberately avoided) the airborne release was very low. The release to the air stream ranged from $< 2 \times 10^{-9}$ wt. % to 8×10^{-4} wt. % of solution. The "dried" residue nitrate was subjected to heating at 400 to 1000°C in the apparatus used for the compound release studies. Results were quite erratic due to differing amounts of moisture in the plutonium nitrate. Fractional release was around 0.12% under the worst conditions.

Another series of experiments was performed to gain insight into the degree to which plutonium nitrate aerosols would be generated from deliberately boiled solutions of plutonium nitrate. Four boiling regimes were characterized by physical description and boil-off rates. The total released plutonium nitrate was collected and measured. Release rates ranged from 3×10^{-4} to 0.35 wt. %/hour and were reasonably well correlated with boil-off rates.

Apparatus was readied for measuring aerosol release from burning material containing plutonium compounds. Several experiments were completed during FY 1968.

Evaluation of Effort on Subprogram 8RLf

During the reporting period much has been learned concerning the airborne release of plutonium and its compounds during overheating with air currents directed across the various specimens. As might be anticipated, the observed release rates are more indicative than definitive, or absolute, and a rather wide range of release rates are observed. These data are of worth in understanding the formation of aerosols from plutonium and its compounds. The anticipated work should yield information of further value in assessing airborne releases in non-standard incidents.

Plans and Expected Results Subprogram 8RLf

FY 1969

An objective of the work in FY 1969 will be to further define the nature and fraction of plutonium aerosols released from overheated materials containing plutonium and its compounds. Emphasis will be placed on aerosols released during the course of an accident involving transportation of plutonium. The aerosol released when porous media (paper, cloth, wood, etc.) containing plutonium nitrate are contacted with fire and flame will be investigated in these studies.

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A continuing effort will be made to determine the areas of plutonium handling and processing which are most likely to result in overheating incidents, using accidents to date and the operating experience at this and other sites as guides. The definition of the potential for plutonium releases in working areas with or without accompanying fires should help determine the nature of useful larger scale experiments which may be performed. Larger scale experiments will be designed when it is determined that a simulant can be used and measured in samples to an adequate sensitivity, and that cleanup time between experiments can be practical. These experiments would be designed to determine the release fraction to a room-size atmosphere when plutonium is released accidentally through failure of containment. The airborne concentration as a function of time with and without the benefit of air cleanup designed into the system would be measured.

FY 1970

It is anticipated that accidental releases of plutonium and its compounds would be simulated in room-scale studies. The objective of these experiments is to gain information in the persistence of airborne aerosol concentrations in selected off-standard incidents, to determine the rate of removal from an atmosphere with various air moving and filtration concepts. The impact of such inadvertent releases of plutonium aerosols on the operation from a standpoint of personnel exposure, atmospheric releases, and recovery will be investigated. Procedures and facility modifications will be sought to mitigate the consequences of accidental releases.

FY 1971-1973

The research would be directed toward further investigation of aerosol generation and behavior following accidents involving release of plutonium and its compounds. Studies and experiments will be made of leak rates of plutonium aerosols through openings of various dimensions. Variables of opening size and shape, pressures, time, and particle size would be considered. This study would place in better perspective the consequences of various degrees of containment failure and may disclose corrective measures in particular situations. The facility, prepared earlier, would be used in mocking up accident situations with added emphasis on fires. Possibility of conducting full scale or near full scale transportation accident simulation would be investigated and initial experiments undertaken.

Milestones

1. Reach decision on the feasibility of scale-up tests and design of a facility based on the burning and high-temperature work now in progress. October, 1968
2. Complete review of accident statistics to identify accident situations likely to involve radioisotopes. October, 1968

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3. Reach decision on the magnitude of tests to be made based on the initial experiments in the new test facility. August, 1969
4. Determine need for further studies using plutonium. Large field tests may be indicated using a simulant. December, 1970
5. Provide statistically valid data on the behavior of radioisotopes under environmental conditions likely to be encountered in probable accident situations. December, 1971

Reference Reports Issued

BNWL-786, "Plutonium Release Studies III: Release from Heated Plutonium Bearing Powders," J. Mishima, C. A. Radasch, L. C. Schwendiman, April, 1968.

Previous Reports

HW-83668, "A Review of Research on Plutonium Releases During Overheating and Fires," J. Mishima, August, 1964.

BNWL-205, "Plutonium Release Studies I: Release from the Ignited Metal," J. Mishima, December, 1965.

BNWL-357, "Plutonium Release Studies II: Release from Ignited, Bulk Metallic Pieces," J. Mishima, November, 1966.

Equipment Needed

FY 1969

For laboratory

\$7,500

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Subprogram 8RLg
Seismic Studies

Scope and Objectives of Subprogram 8RLg

This program will characterize the Richland site response to major seismic activities measured and observed at distant offsite locations and minor seismic activities measured in the immediate vicinity of the Richland site. The measurements will be used to evaluate the coupling effect between the bedrock and the overlying unconsolidated sediments and between the unconsolidated sediments and building structures at the ground surface. The information generated by this program will be related to U. S. Coast and Geodetic Survey Stations in the Blue Mountains and at Newport, Washington, and to other seismic studies being conducted by the University of Washington near the plant site. These data will be used, together with the results of consultant findings, to identify the potential maximum earthquake intensity for the Hanford plant site as a basis for evaluating existing structures and equipment and for developing design criteria for new structures.

Incentives for Subprogram 8RLg

Severe seismic shocks have been identified as being among the more probable causative events leading to a serious nuclear safety incident. The occurrence of such a shock has the potential of damaging nuclear control systems, plant structures - including shielding, coolant systems, electrical power supplies and waste storage tanks. Identification of the possible magnitude and location of potential seismic events and the capability of Hanford structures and systems to withstand the resulting seismic excitation imparted to them are important factors in assuring the safety of a nuclear facility.

The geologic structure of the Richland site consists of basalt bedrock to be over 10,000 feet thick which is folded in a series of anticlinal ridges and synclinal basins. These downwarped basins are filled with up to 800 feet of unconsolidated and semi-consolidated sediments. Seismic data of the Hanford region tends to indicate that the basin-fill sediments have a decoupling effect between the seismic shock waves transmitted through the bedrock and the building structures at the ground surface. The amount of damping caused by these sedimentary beds may be great enough to render surface structures immune from large lateral or vertical displacements occurring in the underlying bedrock. The incentives are large to confirm the degree to which seismic waves are attenuated by sedimentary beds. If it can be shown that ground shocks are effectively attenuated, it would provide some assurances that future operation will be accident-free. Also, such a conclusion would be a factor in making Hanford an attractive site for additional nuclear facilities.

Progress During Report Period

Preliminary measurements of reactor components and building response to background vibrations have indicated some low frequency resonance sensitivities

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in the reactor building structures. A permanent seismograph station was established on Gable Mountain where basalt rock extends to the surface. Extensive modifications to the amplifiers and recorders were made to increase their sensitivity and stability.

Specifications were prepared for a weak motion seismic detection system that can be used to determine seismic response of the soil adjacent to a building being monitored. Evaluation and analysis of historical data on benchmark changes in the Columbia Basin were started.

Evaluation of Effort on Subprogram 8RLg

Measurement of seismic motion at the Gable Mountain seismic station and a reactor site has been delayed by instrument instability problems and the apparent lack of seismic disturbances sufficient for measurement while the instruments were in place. The specifications for an additional weak motion instrument package were completed.

Plans and Expected Results Subprogram 8RLg

FY 1969

Instruments will be set to monitor the motion of bedrock, a reactor building foundation and a key structural part of the building during minor seismic disturbances. Computer programs for analysis of accelerograms will be procured and analyses of accelerograms will be initiated. Comparison of building measurements with measurements on the basalt will be made to determine coupling factors. An additional weak motion seismic detection monitor will be purchased and installed on the surface of the ground near the reactor building being monitored. High sensitivity velocity measurements will also be made with this monitor. These data will be used to help identify the potential maximum earthquake intensity which might be expected in the process areas.

A strong motion seismograph will be purchased and installed at the Gable Mountain Seismograph Station.

Compilation of Geologic Survey data on benchmark changes in the Columbia Basin will be completed.

A scouting program will be initiated to test component operability while being externally excited at amplitudes and frequencies corresponding to those which might be produced by severe seismic events.

FY 1970

Continue to operate the Gable Mountain Seismic Station strong and weak motion seismic recording system and the monitors on other plant structures. Complete assembly of seismic data analysis computer programs and begin theoretical analysis of soil coupling factors for comparison to measurements.

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Component testing will be continued at a level warranted by the usefulness of the preliminary scouting tests.

FY 1971-1973

Continue to operate the seismic recording systems and analyze significant disturbance recordings. Periodic reports summarizing the data with pertinent conclusions will be published.

Milestones

1. Evaluate results of preliminary scouting tests of component operability being excited to simulate earthquake effects.	December, 1968
2. Installation of new weak motion seismic detection monitor at a reactor.	July, 1969
3. Reporting of benchmark changes from Geologic Survey data compilation.	December, 1969
4. Complete theoretical analysis of soil coupling factors for comparison with measured seismic response.	July, 1970
5. Report seismic measurement results at a reactor location.	July, 1970
6. Report seismic measurement results at a separations building.	July, 1971
7. Report seismic measurements and data analysis.	July, 1973

Reference Reports Issued

None.

Equipment Needed

FY 1969

Weak motion seismograph	\$15,000
Shaker table	\$73,500
Strong motion seismograph	\$ 2,500

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U.S. ATOMIC ENERGY COMMISSION
Research & Development Statistical Summary

02 Program

1. Mission No. & Title Nuclear Safety Mission 8 (C, K) 2. Budget Activity No. _____
Mission Subprogram Title Summary

3. Contractor & a. DUN AT(45-1)-1857 4. Person in Charge: J. W. Riches
Contract No. b. _____ Investigator: _____
c. _____ Investigator: _____

FIVE YEAR FORECAST (In Thousands of Dollars)

Current Budget Years

	<u>FY 1969</u>	<u>FY 1970</u>	<u>FY 1971</u>	<u>FY 1972</u>	<u>FY 1973</u>
5. Manyears					
a) Scientific & Technical	2.4	3.0			
b) Other	0.4	0.8			
Total	<u>2.8</u>	<u>3.8</u>			
6. Costs (\$000's)					
a) Direct Labor (Incl. Cost of Serv)	77	109			
b) Materials, Tech Serv, Sub Contracts	100	110	100	100	100
c) Sponsored with PNL	177	219			
Total Direct	23	31			
d) Indirect Expenses	<u>200</u>	<u>250</u>	<u>250</u>	<u>250</u>	<u>250</u>
Total Operating Costs	20	15	15	15	15
e) Equipment*(New)					
f) Irradiation Unit Costs					

7. Comments:

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*Includes prorated portion of equipment requirements for PNL.

Prepared By: _____
Date: _____

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Research & Development Statistical Summary

02 Program

1. Mission No. & Title Nuclear Safety Mission 8 (N) 2. Budget Activity No. _____
Mission Subprogram Title _____

3. Contractor & a. DUN AT(45-1)-1857 4. Person in Charge: J. W. Riches
Contract No. b. _____ Investigator: _____
c. _____ Investigator: _____

FIVE YEAR FORECAST (In Thousands of Dollars)

Current Budget Years

	<u>FY 1969</u>	<u>FY 1970</u>	<u>FY 1971</u>	<u>FY 1972</u>	<u>FY 1973</u>
5. Manyears					
a) Scientific & Technical	7.3	7.3			
b) Other	3.5	3.6			
Total	<u>10.8</u>	<u>10.9</u>			
6. Costs (\$000's)					
a) Direct Labor (Incl. Cost of Serv)	160	201.9			
b) Materials, Tech Serv, Sub Contracts	18	18.0			
c) Sponsored with BNW	170	170.0			
Total Direct	348	389.9			
d) Indirect Expenses	52	60.1			
Total Operating Costs	<u>400</u>	<u>450</u>	<u>470</u>	<u>450</u>	<u>470</u>
e) Equipment (New)	241.3	24.5			
f) Irradiation Unit Costs					

7. Comments:

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Prepared By: _____
Date: _____

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Research & Development Statistical Summary

02 Program

1. Mission No. & Title Nuclear Safety Mission 8
Mission Subprogram Title _____

2. Budget Activity No. _____

3. Contractor & a. ARMCO AT(45-1)-2130
Contract No. b. _____
c. _____

4. Person in Charge: R. E. Tomlinson
Investigator: _____
Investigator: _____

FIVE YEAR FORECAST (In Thousands of Dollars)

Current Budget Years

	<u>FY 1969</u>	<u>FY 1970</u>	<u>FY 1971</u>	<u>FY 1972</u>	<u>FY 1973</u>
5. Manyears					
a) Scientific & Technical	0.2	0.2			
b) Other	0.0	0.0			
Total	<u>0.2</u>	<u>0.2</u>			
6. Costs (\$000's)					
a) Direct Labor (Incl. Cost of Serv)	2.3	2.3			
&b) Materials, Tech Serv, Sub Contracts	61.7	61.7			
c) Sponsored with PNL	64	64			
Total Direct	<u>6</u>	<u>6</u>			
d) Indirect Expenses	70	70	100	100	100
Total Operating Costs	<u>70</u>	<u>70</u>	<u>100</u>	<u>100</u>	<u>100</u>
e) Equipment*(New)	22.2	7.5	50	20	20
f) Irradiation Unit Costs					
7. Comments:					

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*Includes prorated portion of equipment requirements for PNL.

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Date: _____

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SCHEDULE CHART FOR NUCLEAR SAFETY PROGRAM

Schedule Chart for Nuclear Safety Program

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NUCLEAR SAFETY PROGRAM INTERRELATIONS

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Nuclear Safety Program Interrelations

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70. N. R. Miller	81. R. H. Shoemaker
	82. J. R. Spink

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