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DATE
July 28, 1967

TITLE

DRAFT OF
MISSION No. 8
NUCLEAR SAFETY PROGRAM

MISSION 8
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35 pages.

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Introduction

The nuclear safety reactor 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. Indeed, the concern for nuclear safety was a decisive factor 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. The three original production reactors, B, D, and F were provided with horizontal rods for nuclear control, vertical rods for emergency scram and a liquid poison system for "last-ditch" nuclear control. Analogous provisions were made for reactor cooling.
2. Operational and administrative controls were emphasized to assure that the facilities would be operated within known-safe system capabilities. Operating personnel have been competent and qualified. Independent technical staffs have been maintained to assist in assuring safe operation, to audit the operations and to investigate the safety implication of existing processes, and proposed system modifications and processes.

In short, great effort has been directed toward preventing the occurrence of a nuclear safety accident. It has always been realized that 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. Thus, while the exact potential for a major accident has not been, and cannot be, quantitatively described and while the potential consequences of such an accident could, and can, be

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described only imperfectly, 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. The inception and maturation of the commercial nuclear industry has had large impact on nuclear safety considerations:

- o Isolated sites such as the Richland site are few in number and generally not suitably located for commercial plants.
- o The maintenance of large, safety-oriented, technical staffs at the commercial facilities is not a logical outgrowth of the static nature of the commercial plant operation. There would be little, if any, economic benefit to the facility owner. Further, in such an environment it would be difficult if not impossible to recruit and maintain a staff of technical people with the diversity of backgrounds and with the technical competence possible at Richland where the changing processes and programs have provided a continued stimulating and challenging atmosphere.
- o The magnitude of the commercial industry in itself is a potent factor. The number and size of nuclear plants now on order and the projected growth of the industry indicates that soon a large proportion of the nation's population will have a major-sized nuclear facility as a not-distant neighbor. An accident, highly improbable for a single facility, can become uncomfortably probable in the overall with the operation of, say, a hundred nuclear facilities.

Thus, there is little wonder that along with emphasis on accident prevention there evolved additional concern on establishing a better understanding of accident consequences and on minimizing these consequences.

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 can only increase in the future.

Thus, even discounting the implied obligation for a government-owned facility to show leadership in an area of such basic public responsibility as nuclear

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safety, the nuclear safety status which seemed quite adequate in the 1940's now seems marginal and 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 single-purpose 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".

Modest programs have been underway in these areas for the past several years. However, the establishment of the Nuclear Safety Mission with increased financial support served to integrate and emphasize the importance of these efforts. During FY 1967, the program gained momentum, objectives were better crystalized, and the incentives and possible benefits can now be better outlined. This document, then, revises and extends the Nuclear Safety Program as now envisioned for the coming five-year period. Among things certain, this program is bound to be modified in the future. As results of the Hanford Program are obtained and evaluated and as information derived from safety studies conducted at other sites is obtained and integrated, the emphasis will be changed to obtain maximum cost effectiveness.

However, the basic scope, the incentives, and the general direction now perceived are considered valid and may be expected to shape the program during the projected period.

The Nuclear Safety Program now 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

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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 seismology of the Richland site and the reaction of structures to seismic shock.
6. Obtain quantitative information on meteorological parameters at the reactor sites along the Columbia River. This information will provide a better, probably less pessimistic basis for evaluating offsite consequences of a fission product release from one of the production reactors.

Incentives

The incentives for conducting the Nuclear Safety Program are listed below. The first two items are considered to provide full justification for the program effort. The remaining two items represent valuable spin-off benefits that will automatically result.

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 found 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.
4. Continue programs which will provide a challenging, stimulating environment to attract, develop, and hold competent technical personnel expert in the science and philosophy of nuclear safety.

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PROGRAM DETAILS

Subprogram 8RLa
Fuel Temperature Transients Under Accident Conditions

Scope and Objectives of Subprogram 8RLa

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.

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 reactor (N Reactor) and the production-only reactors.

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 rate and extent of fission product release are functions of fuel temperature. 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 arbitrary and probably pessimistic estimates of some key parameters. This may have led to overstatement of the risk of plant operation which would be corrected by the data to be developed.

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

Analytical studies of post-accident fuel temperature transient were developed and improved. For the N Reactor geometries (Mark I or tube-in-tube fuels and

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Mark II or target-in-tube fuels) this included acquiring the capability to calculate self consistent reactor and fuel operating parameters (such as spatial distribution of uranium temperatures, process tube temperatures and graphite moderator temperatures) to determine initial conditions at the start of temperature transients. Ability was developed to account more satisfactorily for heat generated during loss-of-cooling accidents due to the exothermic reaction between Zircaloy and water (or steam). Capability to calculate spatial fractions of fuel reaching specific temperatures following loss-of-cooling accidents was achieved. This was applied to the Mark I and Mark II fuel designs to establish the fractions of fuel reaching failure temperatures following a sudden and complete loss-of-cooling accident.

Parametric studies of the Mark II fuel provide an improved basis for the design of experiments which will focus on the most critical parameters that control post-accident heat transfer rates. Specifically, the emissivity of the fuel and tube surface have first-order effects on heat transfer rates. Since data or emissivities at the specific conditions of interest are incomplete, measurement of emissivities will receive first priority.

The Accidental Loss of Coolant (ALOC) computer program was modified. This program was developed to model the single-purpose production reactor conditions affecting fuel temperature transients after loss of adequate coolant. The modification made the program capable of making computations for varied fuel geometries and materials as well as target materials.

Evaluation of Effort on Subprogram 8RLa

On balance, progress during FY 1967 approximated that predicted a year ago. Progress on analytical studies of fuel temperature transients exceeded that expected. Experimental work, expected to begin during the fiscal year, was delayed because funds were not available early enough and because priority was given to the analytical studies. However, the better insight gained from the analytical work has undoubtedly improved the experimental design.

Budget Period Plans Subprogram 8RLa

FY 1968

Further analytical work in support of the N Reactor will involve calculations of the Mark IV (improved tube-in-tube) and Mark V (improved target-in-tube) fuel. It is believed that any basic improvement in the analytical technique must await development of experimental data. Hence, emphasis will be placed on the determination of the emissivity of the fuel, target and process tubes under conditions of loss of adequate cooling.

Results of studies of metal-water and metal-metal reactions derived from Subprogram 8RLb will be integrated as appropriate with the fuel temperature calculations for both the dual-purpose and single-purpose production reactor configurations.

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Characterization of the coolant boilout transient for the production reactors will be updated based on new laboratory experiments with simulated fuel geometries. The boilout experiments are expected to show the effect of changed fuel geometry on the initial temperature transient.

FY 1969

Steam-water flow transients upon restoration of coolant will be analyzed. The analysis will be applied to special N Reactor fuel elements and the fuel temperature history after loss-of-cooling will be calculated.

The computer model for the single-purpose production reactors will be expanded to predict spatial reactor temperature transients. This work will be based on the computer programs developed for the N Reactor and thermal hydraulic experiments conducted in support of other R&D Missions for the single-purpose reactors. Experiments will be conducted to determine the flow rates for partial flow loss in hot channels and the time to flow recovery. A further refinement of the fuel temperature profiles vs. time is expected. These results would be used with the results of the fission product release studies to characterize the fission product release rate from the reactor.

FY 1970-1972

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 production and dual-purpose 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. Calculate temperature transients for Mark IV, N Reactor, fuel.
2. Calculate temperature transients for Mark V, N Reactor, fuel.

August, 1967

October, 1967

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3. Update coolant boilout transients for single-purpose production reactors. March, 1968
4. Experimentally measure radiative heat transfer from Zircaloy-clad fuel to the reactor core at temperatures up to 1100°C. June, 1968
5. Predict temperature transients in single-purpose production reactors. June, 1968
6. Refine analytical techniques for determining temperature transients to include chemical and metallurgical reactions and heat transfer and fluid flow test results for both the single-purpose and the dual-purpose production reactors. January, 1970

Reference Reports Issued

DUN-200, "Status Report for Confinement Studies - Douglas United Nuclear Operated Reactors," N. F. Fifer and J. R. Spink, December 30, 1965 (Secret).

Equipment Needed - Subprogram 8RLa

FY 1968

System components for emissivity tests. \$10,000

FY 1969

System components and temperatures and coolant flow measurement devices. \$20,000

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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 plate-out on hot or cold surfaces, rain-out by fog sprays and retention on filters or collectors of isotopes will be characterized. The results will permit more realistic evaluation of the consequences of rupturing 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 represent inputs to 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. These assumptions, in some cases, are likely in error, perhaps 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 dual purpose and the production 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 Report Period

Six tests using full size irradiated fuel elements typical of the single-purpose production reactors were conducted simulating the predicted temperature transient following a complete loss-of-coolant supply. The release of fission

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products during the test was monitored by taking the off-gas through special filters and traps. The tests indicated:

1. The rate of fission product release below the melting point of uranium is much lower than in previous experiments using miniature, unclad fuel.
2. There is a uranium-aluminum reaction below the uranium melting point which forms a case around the uranium and adds heat to the fuel.
3. The release of krypton progresses at a significantly lower rate than the release of xenon.
4. A secondary reaction of uranium and aluminum is initiated at about 1080°C which rapidly adds heat to aluminum-clad fuels.
5. The iodine release rate was not appreciably increased when some uranium melting occurred. The rate appeared to be related to the oxidation of the uranium.
6. Uranium oxidation in a steam atmosphere was restricted by the coating of the uranium-aluminum compound.

Results 1 and 3 serve to confirm similar results obtained in FY 1966 from tests with full-sized N Reactor fuel pieces.

Test N Reactor fuels using Mark II alloy and of nominal 8" length have been fabricated, charged, irradiated, and six of these elements discharged as part of the continuous program to supply fuel elements for fission product release and metal-water reaction testing. The discharged elements are presently being cooled in the 105N storage basin in preparation for recommencing tests in August, 1967.

The potential reaction during accidents between reactor core components and coolants, particularly Zirconium and water and Uranium and water have been a matter of increasing concern. Thus, a project proposal for a fission product release laboratory has been completed and approved and design of the facility is underway. This laboratory, scheduled for completion in December, 1967, will provide capability for performing tests using molten irradiated fuel. Present facilities are not capable of handling molten fuel. The behavior of fully irradiated N Reactor fuels will be observed. Upon heating of the fuels on a temperature transient expected in various postulated accidents, the extent and rate of fission product release and the extent and rate of hydrogen evolution will be measured.

Metal-water reaction tests using unirradiated fuel elements are presently underway in the 292-T facility. These tests provide at least a tentative and preliminary verification of the metal-water reaction calculational model presently being used in N Reactor safety analysis reports.

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An oxidant detector was procured as a candidate iodine monitor for use in routine monitoring of the charcoal halogen collectors in the reactor confinement systems. Preliminary testing indicates it is highly sensitive to iodine. The detector has been revised and coupled to recorders suitable for reactor service.

Evaluation of Effort - Subprogram 8RLb

Progress on this extremely important area of study during the past year is considered good. Information on the metal-metal reaction between aluminum and uranium for aluminum clad fuel will remove much of the uncertainty in the consequences of loss-of-cooling accidents at the single-purpose production reactors. The additional information developed on fission product release from fuel has exciting implications on current risk assessments. Preliminary results of metal-water reaction studies seem to show that the proposed experimental techniques are indeed workable increasing confidence in the ultimate ability to quantitatively describe the processes involved.

The ultimate success of the entire program was sharply improved by the success of efforts to obtain funding to outfit the 324-D facility to permit melting full-sized irradiated elements.

The one area of lag involves the resumption of failure testing of N Reactor fuel in the 292-T facility. The availability of test fuels was delayed by the work stoppage beginning in July, 1966. A further delay resulted from the higher priority given to the initiation of metal-water tests using unirradiated fuel. While this delay is disappointing it does not indicate any serious, long-range perturbation to the program.

Plans and Expected Results

FY 1968

Metal-water reaction studies using unirradiated, Zircaloy-clad fuel will be completed using the 292-T facility. These studies will provide a first check on the validity of current models now in use and will provide a base for evaluating data obtained with irradiated fuel.

Failure testing of irradiated Zircaloy-clad fuel will be resumed in the 292-T facility. Zircaloy-water reaction rates will show if irradiation effects change reaction rates (swelling induced by fission product gases could damage protective oxide layers). Preliminary results will be obtained on the reaction of unmolten irradiated uranium extruded from failed fuel with water. Failure temperatures will be established for the Mark II fuel alloy (U-Fe-Al). Data on fission product releases from failed unmolten fuel will be used to confirm results from previous similar tests.

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Experiments to determine the rate of fission product release from aluminum-clad fuel for revised heating transients will be conducted. Emphasis will be placed on establishing the total heat reaction of metal-metal and metal-water reactions as well as the rate of these reactions. Confirmation of the lower fission product release rates from aluminum-clad fuels indicated by the first tests will be obtained. Experiments with non-irradiated fuel and fuel samples will be used to characterize the conditions affecting the secondary metal-metal reaction observed at higher temperatures. Examination will be made of test pieces to identify the metal compounds formed in the ranges above and below the uranium melting point.

Emphasis will be placed on the outfitting of the 324-D facility to permit studies where full-sized, Zirconium-clad irradiated elements are heated to and held at various temperatures including those above the uranium melting point. Metal-water reaction studies will be carried out in these tests permitting better approximation of effects to be expected in a reactor loss-of-cooling accident. Surfaces of cladding components will be studied to identify any significant changes which would lead to changes in the surface emissivity.

Research to be undertaken in the 324-D facility will also include a systematic study of the release of fission products from fully irradiated, full-size uranium fuel elements. The fuel elements will be inductively heated at rates expected in reactor loss-of-coolant incidents to various goal temperatures, including the melting point. The time at which fuel elements are held at goal temperature will be sufficient to ensure that fission product release is complete or that the rate of release has reached a constant value. In these experiments particular attention will be given to the rate at which biologically significant radionuclides are released. The forms of iodine and cesium in the atmosphere leaving the fuel piece will be investigated. Data to date suggest that the form of iodine released is dependent upon the temperature and the time at temperature during release. The release form also appears to be dependent on the presence of uranium and other fission products. Little is known of the effects of temperature and cladding on cesium release.

The program to develop means for routinely monitoring the performance of the charcoal halogen collectors will be continued. The oxidant detector will be tested in the reactor air system to observe operation for the possible effects of ozone and other oxidants in the reactor building environs. The system will be calibrated and tested with elemental iodine in conjunction with laboratory charcoal evaluations.

FY 1969

Fission product release laboratory work will emphasize interrelation between fission gas blanketing and oxide thickness and metal-water reactions. The simultaneous fission product release rates will be measured. The effects of exposure on rupture temperature and the release of volatile fission products will be measured. The possible effect of fuel exposure on the metal-water reaction rate will be measured.

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Studies will determine if the rate of heating significantly affects the degree of rupture and the subsequent rate of fission product release and hydrogen evolution extent and rate.

Location of the fuel failures will be studied to determine if a pattern exists since the rupture location affects the magnitude of the fission product release. Fuel elements will be sectioned after testing to determine the amount of material affected during a rupture.

Emphasis will be placed at the characterization of fission products released as to physical form and chemical composition. Other data expected to be generated are the influence of cladding, the influence of atmosphere, degree of oxidation, and the importance of radiation level on release. The latter is known to be important to 10-20 nvt. Irradiation level effect on fraction and rate of release at higher levels is yet to be shown.

Other experimental work will develop information in two areas in which there are currently only meager data which can be confidently applied in reactor accident analysis. (1) The deposition and transport of fission products and fuel channels and related piping in the pathway from the overheated fuel elements to the reactor confinement space, and (2) the efficiency of currently favored confinement features and new concepts for retaining fission products actually released from fuels overheated to the point of melting.

The experimental arrangement will essentially be a mock-up of a section of an N Reactor fuel channel at the inlet end of which the clad uranium specimen will be heated to a goal temperature. The downstream section will be a reasonable facsimile of a process tube containing electrically heated rods simulating the fuel pieces near the rear face of the reactor. Further downstream will be exit hardware representing the remaining pathway to a break point. Fission products escaping this point will pass through filters and charcoal traps representative of the N Reactor confinement system. A measurement of the penetration of fission products to each section of this path will be made.

While efforts will emphasize Zirconium-2 clad fuel, fission product release rate experiments with aluminum-clad fuel will be extended to include evaluation at temperatures above the uranium melting point. Tests of low-heat-generation, aluminum-clad target elements containing fission products may be run.

FY 1970-1972

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

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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 single-purpose reactor tube channels with either Zirconium or aluminum tubes will be run. The effects of reactor graphite on the release of fission products from ruptured or melted tubes would be experimentally evaluated.

Milestones

1. Complete metal-water reaction studies using unirradiated fuel.	October, 1967
2. Determine failure temperatures and fission product release fractions for Zircaloy-clad, British alloy (U-Fe-Al) fuel.	December, 1967
3. Initiate metal-water and fission product release tests where irradiated fuel elements are heated to temperatures including those above uranium melting.	December, 1967
4. Determine rate of fission product release from aluminum-clad fuel using revised heating transients.	April, 1968
5. Define fractional release rate of fission products and particle characteristics including fission products released from molten fuel.	July, 1969
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 studies, and the fuel temperature transients studies.	July, 1971

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Subprogram 8RLc
Control of Fission Gases

Scope and Objectives of Subprogram 8RLc

Studies will include the release of fission gases (noble gases and halogens and their compounds) to air from water and methods for preventing or slowing the release of these gases from the confinement system.

Facets of the program will include:

- o Improved definition of the adsorption of noble gases and halogens into turbulent streams.
- o Better understanding of the kinetics of release of fission gases from water pools.
- o Further determination of the effectiveness of the fog spray system in removing halogens and other fission products from the confiner atmosphere, including the removal of organic halogen compounds.
- o A study of the effectiveness of oil films or floating layers of a diffusion barrier in preventing or retarding the release of fission gases from water pools.
- o Further work on the foam encapsulation of noble gases.

Incentives for Subprogram 8RLc

The existing safeguards at the production and dual-purpose 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 noble gas release would be essentially quantitative.

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, 10CFR100, with large margins to spare.

Progress During Report Period

A small start on this subprogram had been scheduled for FY 1967. However, the start was not made in order to provide fundign for other Mission 8 programs.

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Plans and Expected Results

FY 1968

Laboratory apparatus will be designed and employed to measure the fraction of noble gases and iodines removed when a mixture of steam, noble gas, and/or iodine, and water is discharged into a pipe or tank full of water. These experiments would give insight into the behavior of a small quantity of a low solubility gas and the relatively high soluble iodines under simulated reactor accident conditions. Other measurements to determine the initial rate of release of these isotopes from a pool or column of water subsequent to the event would be undertaken. Various degrees of mechanical, air sparge or thermal gradient mixing would be applied and the release rates measured.

A parallel study will be directed toward establishing the theoretical basis for predicting iodine and methyl iodine removal with fine water droplets. Input data relating to drop size and water flow rate in existing reactor confinement systems will be determined and a transport and removal model developed. Available data will be used, but experimental work to better characterize the fog sprays may be needed.

FY 1969

Experiments will be continued on the adsorption and retention of noble gases and iodines in water with the emphasis shifted toward the longer term release rates. The influence of air movement across the surface of a water pool containing known amounts of noble gas and/or iodine would be measured. The effectiveness of water sprays for removing iodine from air and steam atmospheres would also be experimentally studied.

FY 1970-1972

After obtaining some understanding of the degree to which the low-solubility gases and soluble iodines were released under a variety of conditions, methods of suppressing or slowing the rate of release will be investigated. Potential use of oils in films or floating layers of a diffusion barrier would be investigated to determine their effectiveness. Pilot scale apparatus would be used to confirm laboratory findings. Research in these years is expected to permit selection and design of the best system for retention of both noble gases and iodines for the effluent paths of concern. The research effort would be expected to provide a basis for optimizing the final design.

Work will be done in foam encapsulation of noble gases with emphasis on studies of high stability foams.

Milestones

1. Determine the adsorption and retention of fission gases in reactor effluent water streams and storage basins.

July, 1969

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Reference Reports Issued

None.

Equipment Funding

None.

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Subprogram 8RLd
Meteorological Studies

Scope and Objectives of Subprogram 8RLd

This program is aimed at defining the meteorological parameters which would control the path, speed of movement and dilution of a radioactive release from the production reactors. The program will involve the measurement and correlation of the pertinent meteorological parameters, i.e., wind speed, direction, and variability and atmospheric stability at the Columbia River reactor sites and applying these results to calculate expected dose rates in the plant environs for given fission product releases.

The studies will be made utilizing a meteorological facility which is to be installed at the N Reactor site. Also, this program contemplates a series of experiments involving release of harmless gases or aerosols at N Reactor and other reactors in order to characterize the path, size and dilution of the cloud as a function of weather conditions. Such studies will, as a minimum, permit more realistic preparations for evacuation of an area in the event of an accident.

Incentives for Subprogram 8RLd

The offsite consequences of an accident at a production reactor would be largely controlled by the path of the radioactive cloud, its speed of movement, and its dispersion. Indeed, with the isolation enjoyed at the plants, the actual maximum dose rate at plant boundaries could vary greatly depending on the meteorological variables.

Most of the data obtained at Hanford on atmospheric dispersion has been obtained on the flat plateau where the meteorology station is located. This site is ideal for such studies, but unfortunately the terrain is significantly different than that at the reactor site. This difference makes it impossible to describe the joint probabilities for the various important parameters at these sites using the present meteorological station. Further, the existence of the Wahluke Slope cross-river from the reactors has raised doubts in some persons on the basic applicability of the dispersion model. Data are urgently needed to resolve these questions.

If current assessments are confirmed quantitatively, the Wahluke Slope problem (a problem first raised about 10 years ago) will be resolved. Further, there is fair likelihood that the current estimates will themselves be proved excessively conservative and that the risk of reactor operation found to have been significantly overstated.

Progress During This Report Period

A meteorological system for the N Reactor area was specified and is being procured for installation in early FY 1968. This system includes a remote station

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consisting of four sets of sensors to measure wind speed, horizontal direction and air temperature including temperature differential. These would be placed at four levels on a tower at the N site. The data would be telemetered by FM radio back to a master station located at the 1100-N Building where it would be formatted and recorded on magnetic tape to be correlated by computer later.

Two efforts are being carried on to estimate possible terrain effects on dispersion which might be caused by the Wahluke slopes. Smoke release tests from the N Reactor exhaust stack are underway.

A theoretical approach to the problem is also underway. A mathematical model of the air currents over the slope is being devised. A model in which density is a function of only the vertical coordinate has been proposed and is being developed as a first step in the solution.

Evaluation of Progress on Subprogram 8RLd

Technical and engineering made during the year led to the development of specifications for the meteorology system. Procurement in FY 1968 will permit the start of data acquisition expected to produce a much better understanding of the local meteorological conditions in the years immediately ahead.

Plans and Expected Results

FY 1968

Installation of the N Reactor meteorological system will be completed, and data will be collected and analyzed. Air temperatures and wind velocities will be recorded, analyzed, and ordered according to stability conditions. In order to calculate input parameters for dispersion calculations related to estimating potential environmental consequences during N Reactor accidents, joint probabilities of adverse dispersion conditions and unfavorable wind directions will be determined. These data will be factored into the hazards calculations.

Small scale smoke tests will be continued to provide a qualitative and illustrative description of the performance of a plume released from the N Reactor stack during stable atmospheric conditions.

Onsite effects of accidents will be more accurately estimated.

Coincident with these studies is a comprehensive program being undertaken by Battelle. The principal objectives of this program are:

1. To evaluate expected trajectories of material from the 100-N stack using data from the N Reactor Meteorological system (to be installed), the existing Battelle telemetry network data, and portable wind

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stations. These trajectories will be determined for each season and for day and night-time conditions as well as for a variety of wind speeds expected. Probability statements for the arrival of a plume at the Wahluke Slope under the above conditions will be derived.

2. To evaluate expected exposures for each season in both day and night situations using data from the N Reactor Meteorological system along with results of trajectory studies and the Hanford diffusion model (Fuquay-Simpson model) for elevated continuous release.
3. To conduct smoke and balloon releases to determine air current trajectories by visual tracking over the Wahluke Slope. These experiments will be compared with calculated trajectories.
4. To summarize N Reactor Meteorological system data for diffusion climatology.

FY 1969

Studies and experiments conducted in FY 1968 will be continued during FY 1969.

Dispersion tests will be conducted with sampling on Wahluke Slope to check the validity of the existing Hanford diffusion model as applied to the N Reactor site. These tests are to include both surface and real time airborne measurements of exposure and concentration for releases during both stable and unstable conditions and for each season.

The Hanford model will be modified for the N Reactor site if necessary based on the reevaluation of diffusion and meteorological data.

During the entire program, the N Reactor meteorological system will be continuously sensing and storing climatological data for the continued updating of joint probabilities of the adverse dispersion conditions.

Hazards calculations will be appropriately revised based on the revised models.

FY 1970-1972

The importance of terrain effects will be established particularly with respect to onsite effects.

Milestones

1. Begin acquisition of data on meteorological parameters. April, 1969

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2. Determine year around joint probabilities for the various meteorological parameters at the reactor site and incorporate in fission product release calculations.
3. Define the effect of terrain on the dispersion and path of gaseous releases from reactor sites.

June, 1969

January, 1970

Reference Reports Issued

None.

Probable Equipment Costs

None.

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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 potential of the soil to adequately treat radionuclides found in radioactive solutions 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 production-only reactors. Information from that program on ground water flow patterns and flow times near the low pressure 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 Current Report Period

Laboratory soil column investigations with high activity cesium and strontium solutions were carried out to evaluate the extent of radionuclide movement from a ground disposal facility for waste generated in the event of a reactor primary loop failure incident. The experimental results show the cesium and strontium breakthrough values to be appreciably higher than previously extrapolated predictions for N Area soils. An initial breakthrough of $10^{-1}\%$ ($C/Co^* = 0.1\%$) occurred with both Cesium 137 and Strontium 85. At less than 0.1 column volumes of solution throughput, activity leakage fluctuated as additional solution permeated through the column; ranging from background values

*Co = Concentration in effluent to column.

C = Concentration in effluent from column.

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to momentary values as great as 1% breakthrough. After approximately 5 column values throughput, activity leakage stopped or was reduced to values below limits of detection (at present C/Co less than $10^{-3}\%$). From the breakthrough obtained in the initial phase of this study it was evident that some phenomenon other than ion exchange was responsible for the effluent activity. The only mechanism thought possible for this pattern of activity transportation was the migration of activity-carrying colloids or particulates.

Test wells were dug near the 100N crib to obtain information on the migration of cesium and strontium.

Evaluation of Effort

The discovery of the initial breakthrough phenomenon adds an important new dimension to the soil study program. While not a favorable development, this discovery permits the focusing of increased attention on this problem to obtain a better understanding of the processes and a feasible remedy.

Plans and Expected Results

FY 1968

Since greater than expected concentrations of cesium and strontium were found in the laboratory studies for some accident cases, research will be continued on the characterization of the colloids migrating through soil columns. Suspensions containing rupture debris are available for characterizing the particulates in solution and for additional soil column studies. The objectives of these studies would be: (1) to determine more specifically the chemical and physical forms of strontium and cesium which are not retained in the column, (2) to determine the physical chemical and mineral characteristics of particulates in fuel element rupture solutions, and (3) relate the nature of these particulates to those found in column effluent solutions.

Laboratory methods and facilities to be used in this study include electron microscopy, X-ray diffraction and emission, high speed centrifugation, microfiltration and other more routine procedures such as soil equilibrium, column studies and radio-analysis.

Soil sampling near the 1301-N crib will be continued to determine the distribution of radioisotopes on the soil. The distribution of Sr^{90} , Cs^{137} , and other isotopes of radiological significance will be determined. The soil sampling wells will provide points at which the piezometric head as a function of depth may be measured for input data modeling programs associated with flow system evaluation and waste movement analysis. The field data needed for input to a model and verification of a model will be determined.

Information gained from tests on the ground disposal of the effluent cooling water from the production reactors (single-pass cooling) will be integrated into other results.

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FY 1969

Based on results found in previous studies, appropriate treatment procedures will be further investigated for reduction of initial activity breakthrough to acceptable levels. Knowing the physical and chemical characteristics of the migrating colloids, treatments to eliminate this transport could be accomplished by several possible alternatives, depending on the source of the particulates. Chemical treatment such as pH adjustment and/or addition of coagulants such as those used in water treatment operations would promote coagulation of colloids in solution and enhance their adsorption within the soil interstices. Use of these chemicals or other soil stabilizing agents could also be a means of reducing the movement of fine clay colloids originating in the soil itself. Other treatments such as preleaching the disposal facility with reactor waste effluent will also be considered.

Proposed Study - FY 1970 and Beyond

Work will continue on the investigation of treatment methods needed to create a more favorable soil-waste water relationship. Emphasis will be placed on a field testing program to implement laboratory results from previous studies. Small-scale field disposal sites will be set up and operated in a manner analogous to an emergency disposal facility, using simulated wastes and employing detection procedures for determination of radionuclide and particulate migration in field sites and to measure the success of particular treatment methods in reducing this occurrence.

The hydrological system model will be verified by extensive field testing. A transport model will be developed which will allow diffusion, dispersion, radioactive decay, and other interactions which can be described, to be evaluated. This model would allow concentration distributions in the field to be determined.

Structures will be installed to collect the necessary potential and permeability data needed for input and verification of the model. The model which will describe the system in 3 dimensions, giving velocities, travel times, and flow paths, should be operational.

Milestones

1. Identify cause for the early breakthrough of cesium and strontium.	April, 1968
2. Evaluate data on cesium and strontium migration obtained from test wells drilled near the 100N crib and relate to laboratory studies.	June, 1968
3. Determine means for reducing the breakthrough of cesium and strontium.	June, 1969
4. Define long-range criteria for radioactive liquid waste handling.	June, 1971

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Proposed Obligation for Related Construction Projects

\$25,000 in FY 1969 for construction of hydrological structures (wells) necessary for monitoring and obtaining hydrological input data to the simulation model.

Reference Reports Issued

BNWL-CC-995, "Soil Radionuclide Adsorption and Particulate Filtration in an N Area Soil," B. L. Carlile and B. F. Hajek, January 18, 1967.

BNWL-CC-1032, "Field Evaluation of Ground Disposal of Reactor Effluent - 1301-N Crib," J. R. Eliason, February 8, 1967.

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Subprogram 8RLf
Particles and Aerosols from Burning Nuclear Materials

Scope and Objectives for 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 ensuing. 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 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 of 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 those used in shipping. By combining the actual release rates of nuclear materials subjected to high temperatures and the probability of high temperature accidents (from actual experience) a more realistic hazards evaluation of postulated accidents can be obtained.

Progress During Report Period

Research is currently under way to characterize the nature of plutonium releases from plutonium and its compounds when they are subjected to a range of temperatures. Metallic plutonium pieces ranging in weight from 455 to 1700 grams were ignited and allowed to oxidize completely in air with a velocity of about 500 cm/sec. Airborne release rates of 0.032 to 0.0045 weight percent per hour were observed for the bare metal. Covering the ignited metal with magnesium oxide sand during oxidation reduced the release rate by a factor of 10 to 100. The median mass diameter of particles airborne during release from the bare metal was found to be 4.2 microns with no particles greater than 8 microns. Many experiments were performed to determine the influence of temperature on the enhancement of airborne release of particles from plutonium oxide-oxalate. Currently, these experiments are still in progress with the data for the oxide-oxalate material than with the same air flow at ambient temperatures. Compounds of plutonium oxalate, fluoride, and plutonium nitrate will be examined similarly during the remaining months of FY 1967.

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Evaluation of Effort

Progress towards understanding of release mechanisms and particulate characterization, during plutonium fire situations, has been made. Continued efforts should contribute greatly to realistic estimates of hazard associated with fire situations and lead to more realistic (less conservative) estimates of consequences.

Plans and Expected Results

FY 1968

During this period experiments will be designed and carried out to help establish the fraction of plutonium which may become airborne when plutonium and its compounds are heated with quantities of combustible materials which might be present. Experiments will be conducted to measure plutonium aerosol release during hypothetical shipping accidents which would involve release of liquids to a soil which is subsequently exposed to high temperatures. Feasibility of performing realistic room-size studies to establish plutonium transport and deposition patterns following a serious accident will be determined. The feasibility study would include definition of a simulant material and determining the most likely accident situations to mock-up. Rapid recovery from the room type experiments would have to be assured. Facility designed would follow assurance of feasibility.

An important part of the work will be compiling data on actual accidents, accident potential and consequences from all sources available. Aerosol physics information of relevance to the problem would be factored into the studies. The better definition of research needs in this area of plutonium aerosol release which would result should help materially in determining the most realistic approach to accident evaluation and point out the most important areas for study.

FY 1969

Should the FY 1968 studies prove that a practical experimental design could be developed for plutonium accident mock-ups, a facility would be designed, made available, and experiments performed which would realistically simulate certain accident situations. The spatial distribution and nature of plutonium releases in these simulated accident situations would be measured. Aerosol persistence and other factors would be investigated. Interpretation of the observations are anticipated to yield information for safer plutonium processing plant designs, air control features, and help establish the consequences of various accidents. Possible application of clean-room techniques for minimizing contamination would be investigated.

Plant and capital equipment construction funds in the sums of approximately \$40,000 are estimated for the test facility. The actual sum will depend upon the availability of enclosed space. The estimate assumes that a sheet metal building and floor slab will have to be erected first to house the test facility.

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FY 1970-1972

Further studies of plutonium releases using the facility prepared for this purpose will be performed. Study of the air reentrainment of settled (simulated) plutonium oxide particles from fire situations would be carried out and the aerosol characteristics and behavior clarified. Studies relating to recovery methods using the experimental techniques developed in the earlier studies would be performed. Evaluation of various cleaning methods or recovery procedures to remove or immobilize particulate contamination in recovering from a plutonium accident would be explored. Data obtained in the overall study will be factored into recommendations for improved plutonium processing facility construction, arrangement, and operation. Continued review of fire or explosive caused release of plutonium in these facilities and in plutonium shipping accidents would be made concurrently with these studies. Data from actual fire or other accidental releases would be evaluated and the implications to the research direction taken into account. A continuing study of the applications to the research direction taken into account. A continuing study of the application of the findings of the research to safety reviews will be made. At an early point in these years a detailed evaluation of the high efficiency filter performance in various accidental release situations will be made. The penetration of these filters by plutonium aerosols in overheating situations and involving extraordinary operating conditions will be measured.

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. May, 1968
2. Complete review of accident statistics to identify accident situations likely to involve radioisotopes in transit. July, 1968
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 on 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

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Reference Reports Issued

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

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.

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Subprogram 8RLg
Seismic Studies

Scope and Objectives for Subprogram 8RLg

This program will characterize the Richland site response to minor seismic activities in relation to measurements and observations at offsite locations, and evaluate the response at the reactor and chemical processing buildings. The measurements will evaluate the ground coupling to bed rock and response of the building structures and critical equipment to seismic excitation frequencies. 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 on the potential earthquake intensity for the Hanford plant as a basis for determining the strengthening requirements of existing critical structures and equipment and for developing design criteria for new structures.

Incentives of 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 reactor nuclear control system, plant structures - including shielding, coolant systems and electrical power supplies. Evaluation of the probability for seismic events of given magnitudes and the capability of structures and system to withstand these events are important factors in assuring the safety of a nuclear facility.

The geology of the Hanford site provides the basis for an argument that the region is not subject to destructive seismic events. Plant structures are built on a thick sediment on an underlying basaltic structure. Hence, plant structures may well be decoupled from seismic shocks and immune from large lateral or vertical displacements. If this argument can be sustained, better assurance would result 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

The first of three transducer-recorder sets was delivered and used to measure building and equipment response to background vibration at one low pressure reactor. Provisions for permanently installing the seismic monitor station were made and telephone lines coupling this station to other potential sites for the two portable transducer-recorder sets were installed. A mechanical oscillator was obtained for use in artificially inducing vibration in the structures and equipment of interest. These measurements will be used as the basis for seismic response measurements excited by low level natural seismic activity.

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Evaluation of Effort

Procurement of the equipment described above is an important step in the execution of the program.

Plans and Expected Results

FY 1968

The permanent seismic monitoring station installation will be completed and seismic history recording initiated. One of the two portable transducer units will be located at various reactor buildings to determine coupling factors between the buildings and bedrock formations. Explosive charges may be used for initial readings. Mechanically induced shaking tests would be continued and recordings analyzed using a spectral density analyzer to determine the frequency response of various structures and equipment.

FY 1969

The seismic coupling measurements would be expanded to cover the separations area structures and the response of separations facilities to seismic excitation would be evaluated. The response data taken at the reactors would be analyzed for potential strong motion earthquake excitation not specifically shown in tests at low excitation amplitudes.

FY 1970-1972

The seismic coupling measurements would be completed. Vibration response characteristics measurements of reactor and separations facility components will be completed in FY 1971. Development of mathematical correlations will be completed by the end of FY 1972. The recording of seismic activity at the Hanford plant would be continued through this period.

Milestones

1. Measure seismic vibrative response of Richland facilities.	June, 1968
2. Complete seismic coupling measurements and structural vibration characteristics.	June, 1971
3. Develop mathematical correlations.	June, 1972

Reference Reports Issued

BNWL-CC-409, "Research Program to Determine the Oscillator Response Characteristics of Critical Hanford Structures and Related Geological Factors," R. T. Jaske and R. E. Brown, January, 1966.

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Equipment Needed

FY 1968 - None

FY 1969 - None

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