NUREG/BR-0112

RESEARCH NEWS

OFFICE OF NUCLEAR REGULATORY RESEARCH
VOLUME 4, NUMBER 1

U.S. NUCLEAR REGULATORY COMMISSION
MAY 1991

U.S. Working Group 3 Participates in Reannealing of Soviet Novovoronezh-3 Reactor Vessel

Charles Z. Serpan, DE/MEB

JCCCNRS Working Group 3 Agreement

During the October 1990 meeting of the Joint Coordinating Committee for Civilian Nuclear Reactor Safety (JCCCNRS) between the USNRC and the Kurchatov Institute in Moscow, USSR, it was agreed that U.S. Working Group 3 (WG-3) would send a team of experts to witness the reannealing of the Novovoronezh-3 (NV-3) VVER-440 reactor vessel. The purpose of this trip would be to follow in detail and to evaluate the in situ annealing of the NV-3 reactor vessel and, as a result of this evaluation, provide an assessment of the applicability to U.S. plants of the many different procedures and activities undertaken and necessary for such an annealing in light of current U.S. vessel and plant designs, industry practices, and regulatory requirements.

Reactor Vessel Embrittlement

Reactor pressure vessels are fabricated from low-alloy ferritic steels, which are subject to embrittlement caused by exposure to high-energy neutrons emanating from the reactor core. The rate of embrittlement is significantly affected by the exact chemical composition of the steel, the highest rates being caused by the presence of copper and nickel in U.S. reactors and phosphorus in Soviet reactors. Radiation embrittlement becomes apparent as an increase in the nil ductility transition temperature (NDT) of the steel. Because the NDT temperature marks the important boundary between ductility and brittleness in a reactor vessel, it is essential that the reactor operate above the NDT temperature at all times to ensure ductile behavior. Normal operation of reactors, however, causes the NDT temperature to increase, possibly to the point where brittle rather than ductile behavior of the vessel could be predicted under pressurized thermal shock or low-temperature overpressure accident conditions. A number of U.S. plants have vessels that will closely approach or possibly even exceed NDT limits during their original 40-year license period; other plants contemplating license renewal will almost certainly have to face the prospect that their vessel will reach or exceed the NDT limit.

Reactor Vessel Annealing

Embrittlement of reactor vessels can, however, be reversed or eliminated by "annealing." This is accomplished by raising the vessel temperature as much as 300 F above the operating temperature for a time period of about one week. The Soviets find that such conditions will result in virtually complete recovery of the initial properties. Although no U.S. reactor vessel has yet been found to be so embrittled as to require annealing, nine Soviet VVER-440 reactor vessels have already been found to be sufficiently embrittled to require annealing as a prerequisite for continued safe operation. The NV-3 vessel was the first to be annealed by the Soviets although the temperature was lower than that presently used for reasons of conservatism; it has now been annealed again but at a higher temperature to provide better assurance that the desired properties were recovered and also for the Soviets to check the rate of reembrittlement following the first anneal.

Soviet Reactor Annealing Experience

As of March 1991, the Soviets have conducted ten annealings of reactor vessels of the VVER-440 Model 230 type. The Model 230 vessels are the most important ones to be annealed because the phosphorous and copper contents are higher in these vessels than in the later Model 213 reactors. It is noted in addition, however, that the newer Model 213 plants have certain safety improvements, including ECCS systems. Both models nevertheless have small-diameter vessels with a very narrow water gap; this coupled with the high P and Cu content is the primary cause for early high embrittlement in Soviet reactors. The following VVER-440 reactor vessels have been annealed:

1. Novovoronezh-3 (1987 and 1991),
2. Armenia-1 (1988),
3. Nord-1, 2, 3 (in what was East Germany) (1988, 1990),
4. Kozloduy-1, 3 (Bulgaria) (1989),
5. Kola-1, 2 (1989).

U.S. Preparations for Reannealing Observation

A team of four U.S. personnel from NRC and a contractor, MPR Associates of Washington, D.C., traveled to Moscow in November 1990 to gather information about the activities related to Soviet vessel annealing and their sequence. The MPR personnel were especially important to this information-gathering visit because they have been intimately involved with many different kinds of activities conducted inside containment of a number of reactors during shutdown and maintenance operations. They are also very familiar with U.S. regulations concerning permissible operations inside containment and were able to quickly focus on the kinds of activities that were conducted for annealing and how they were carried out to ensure safety and minimum radiation exposure.

Soviet Annealing Activities

For the NV-3 reannealing, the Soviets first took "template" samples of vessel material just under 1/4 inch thick by about 1 x 3 inches, from which they have cut sub-size Charpy specimens to establish as-irradiated material properties. (After annealing, additional "template" specimens were taken to establish the as-annealed condition.) For all previous annealings, a manned shielded "cabin" equipped with a hardness tester was lowered into the dry vessel to establish the as-irradiated material condition in a different way. (Such hardness measurements are also taken after annealing to determine the effectiveness of the annealing.) Finally, the heater assembly was lowered into the vessel. The assembly had three rows of 18 heaters each and covered almost six vertical feet of vessel surface. The heaters raised the vessel temperature to 460 C (about 850 F) where it remained for 100 hours. Cooldown of the vessel required only a few days. If done on a round-the-clock basis, the entire annealing cycle, including before and after "template sampling," heatup, hold time, and cooldown, can be accomplished in about 20 days.

U.S. WG-3 Team Schedule and Activities

The actual reannealing of NV-3 took place during February and March 1991. The U.S. sent two teams of personnel to observe the sequence of operations conducted by the Soviets for this reannealing. The first team observed the "template" sampling operations, the checkout and insertion of the annealing rig, and the start of the annealing heatup and temperature stabilization. The first team returned to the U.S. after about three weeks at the site. The reactor was annealed for about four days, and the cooldown required almost that much additional time. The second team arrived at the site just as the reactor reached the end of the cooldown period; they observed post-anneal "template" sampling operations and the start of the packaging of the annealing rig for storage. Grinding operations were postponed to a later time. Personnel on the two teams included representatives from the Offices of Research (RES) and of Reactor Regulation (NRR), Region I, MPR Associates (who will be responsible for preparing the description of Soviet annealing in terms of application in the U.S.), U.S. utilities having PWRs that could require annealing in the future, the Oak Ridge National Laboratory (for information relative to metallurgical aspects of annealing), and the Electric Power Research Institute.

Summary and Objectives

The objective of this activity was to observe in detail and to evaluate the in situ annealing of the NV-3 reactor vessel and, as a result of this evaluation, to provide an assessment of the applicability to U.S. plants of the many different procedures and activities undertaken and necessary for such an annealing in light of current U.S. vessel and plant designs, industry practices, and U.S. regulatory requirements. In addition to observing the operations specific to annealing, the WG-3 team gained much information on other operations related to the annealing of reactor vessels in the Soviet Union, including movement of the annealing rig into the containment, removal of internals, placement of the seal between vessel and refueling canal, decontamination, water chemistry, and control of radiation levels in the plant.

As result of these visits to the Soviet Union to witness the reannealing of the NV-3 reactor vessel, NRC has gained important information on the actual conduct of reactor vessel annealing for application to U.S. plants that might need to perform annealing in the future. The teams identified a number of differences in the application of the Soviet technology to U.S. plants. However, because of the

close interaction with the Soviet personnel and the on-site observations afforded by these visits, the differences were clearly defined. We now believe that positive consideration of anne... in the U.S. can proceed on a much more inf... basis and with greater assurance of ... From a somewhat different viewpoint, ... provided the Soviets with an opportunity ...nstrate their capability for annealing to a ... U.S. utility personnel. The Soviets ... desirous of marketing this technology in the U.S. as an excellent way to earn "hard" currency, so the visits were as important to the Soviets for economic reasons as they were to the NRC for technical reasons.

PRA Code Developed for Use on Personal Computers

Dale M. Rasmuson, DSR/PRAB

In 1985 the NRC started to develop an easy-to-use computer code to perform accident frequency analysis. At about that time, the increased power and capabilities of the personal computer (PC) made it possible for the PC to perform some of the data preparation tasks for the primarily main-frame program. However, the rapid increase in PC power and the new capabilities introduced made it possible to do most PRA tasks and calculations on the PC. As a result, the Integrated Reliability and Risk Analysis System (IRRAS), an integrated PRA analysis tool, gives the user the ability to create and analyze fault trees and event trees using a personal computer. This program provides functions for fault tree and event tree construction and analysis.

The fault tree functions range from graphical fault tree construction to fault tree cut set generation and quantification. The event tree functions include graphical event tree construction, linking of fault trees, defining accident sequences, generating accident sequence cut sets, and quantifying them. IRRAS also contains capabilities for performing uncertainty analyses, importance analyses, and sensitivity analyses.

IRRAS is being used in many NRC projects. Some of these are the seismic margins research program, resolution of generic safety issues, and the development of risk-based inspection strategies. It is currently being used in the low-power/shutdown PRA's being done for RES by Sandia National Laboratories and Brookhaven National Laboratory and is being evaluated for use in event assessment by the Offices of Nuclear Reactor Regulation and Analysis and Evaluation of Operational Data. It is also used by such Government Agencies as the National Aeronautics and Space Administration and the Departments of Energy and Defense for their reliability and risk assessment work, e.g., Galileo mission and the space shuttle.

IRRAS version 1.0 was completed in February 1987. Over 400 copies have been distributed to national laboratories, utilities, Government Agencies, and other organizations. Version 2.0, which includes the capabilities for accident sequence analysis, was distributed in the spring of 1990 and is documented in NUREG/CR-5111. IRRAS 2.5 has just been completed and is being placed in the National Energy Software Center. A graphical event tree editor and an improved accident sequence cut set algorithm were added in this version, which is documented in NUREG/CR-5300.

New 10 CFR Part 20 Published

Alan K. Roecklein, DRA/RPHEB

On December 13, 1990, the Commission approved the issuance of the revised 10 CFR Part 20, "Standards for Protection Against Radiation," and the rule was published in the *Federal Register* in December. This comprehensive revision of the basic radiation protection standards applies to all activities licensed by the USNRC. Agreement States will be adopting similar regulations.

The revised 10 CFR Part 20 is based upon the 1977 recommendations of the International Commission on Radiological Protection and is generally consistent with the 1987 recommendations of the National Council on Radiation Protection and Measurements. The revised Part 20 contains significant changes from past practice and procedures for estimating, measuring, combining, recording, and reporting doses. These changes are associated with the introduction of new concepts and methods of assessing doses.

The revised Part 20 differs from the previous Part 20 in many ways, the most important of which are:

1. The use of the "effective dose equivalent" (risk weighting of organ doses) concept instead of the "critical organ approach" represents a major departure from past practice.
2. The concept of controlling the sum of internal and external doses will require new procedures and records.

3. Reliance on annual instead of quarterly dose limits and elimination of a cumulative lifetime limit will necessitate tighter dose control by the licensee and modification of operating procedures.
4. New limits on doses to members of the public and the embryo/fetus will also require new procedures and records for their implementation.
5. Allowance for modification of dose estimates and airborne concentrations for such parameters as actual particle size, solubility, and retention will require guidance on acceptable technical approaches.
6. Revised intake and concentration limits will require modification of procedures and dose calculation manuals.
7. New recordkeeping and reporting requirements (Form 4 and Form 5) may call for additional instruction.

A major staff effort is under way to deve[illegible] new and revised guidance to assist licens[illegible] their conversion to the new standard. [illegible]latory guides are being prepared on estima[illegible]dding internal and external doses, calcula[illegible]se to the embryo/fetus, high and very high [illegible] areas at nuclear power reactors, the content of radiation protection programs, controlling external doses from airborne radionuclides, recordkeeping and reporting radiation exposure data, and estimating doses from bioassay measurements.

Essential Service Water System Failures at Multi-Unit Sites

Demetrios Basdekas, DSIR/RPSIB

The essential service water (ESW) system is required to provide cooling in nuclear power plans during normal operation and accident conditions. Typical equipment supported by the ESW system are the component cooling water heat exchangers (and therefore the reactor coolant pump seals), containment spray heat exchangers, high-pressure injection pump oil coolers, emergency diesel generators, and air conditioning and ventilation systems. Failure of the ESW function could lead to severe consequences.

A generic issue, GI-130, "Essential Service Water System Failures At Multi-Unit Sites," was identified in 1986 as a result of a probabilistic risk assessment of the ESW system performed for Byron Unit 1. Multi-unit sites like Byron have two ESW pumps per unit (one per train) with one of the two pumps being shared between the units via a crosstie. However, ESW system support from Byron Unit 2 via the crosstie between the two units was not available while Unit 2 was under construction. The insights derived from that study indicated that the core damage frequency due to the unavailability of a two-train (one pump per train) ESW system could present a significant risk to the public health and safety, particularly if one ESW pump from the adjacent unit via an ESW system crosstie is not available.

Fourteen units at seven sites having the basic Byron ESW configuration were evaluated as part of the resolution of this issue. Other design configurations of ESW systems, including those of single-unit sites, will be evaluated under GI-153, "Loss of Essential Service Water in LWRs."

It should be noted that the success criteria for the ESW systems in providing adequate cooling capability during normal, accident, and postaccident conditions are design specific, depending on the plant configuration, the capacities of the ESW pumps, and equipment dependencies on ESW cooling. Although the success criteria may be as varied as the ESW systems, this evaluation assumed a generic set of success criteria as a representative model for purposes of quantifying the events leading to possible core damage accidents. These generic criteria are discussed below and apply only to multi-unit sites having two ESW pumps per unit with a crosstie between them.

During normal operation, one ESW pump per unit provides adequate cooling to the required systems and components. The second ESW pump per unit is assumed to be normally in a standby mode. Because of load shedding (isolation of nonessential equipment), one ESW pump per unit is assumed capable of handling accident and cooldown heat loads. With one plant in power operation and the second plant in the shutdown or refueling mode of operation, the criteria assume that one ESW pump can provide adequate cooling to shut down the operating plant through the crosstie connection should the need arise.

A review of operational experience showed that a number of different components in the ESW system may fail to perform their intended function in a variety of ways. However, some important failure modes for the ESW system are associated with failures of certain components. For example, failures of the traveling screens or other common-cause problems at the intake structure can lead to the partial or complete loss of the water supply. The

ESW pumps and their electric power supply are other important contributors to the partial or total loss of the ESW system.

The comprehensive evaluation of the operating experience showed that, excluding system fouling (sediment, biofouling, corrosion, erosion), there were a total of 12 events involving a possible complete loss of the ESW function in 667 reactor-years. System fouling data were noted but were excluded from the current analysis because of the earlier resolution of Generic Issue 51, "Improving the Reliability of Open Cycle Service Water Systems."

A detailed examination of the loss-of-ESW events indicates that a number of them occurred during shutdown. Some of these events may not have been a complete loss of ESW in terms of total stoppage of ESW flow, even though the ESW system may have been declared inoperable. The differences in the ESW system between power operation and shutdown are primarily the actual heat load and the equipment affected by the loss of ESW. In addition, the different administrative requirements imposed by the technical specifications make these two operational modes more distinct.

The evaluation of ESW system failures at multi-unit sites included a determination of the initiating frequency of loss of ESW, the core damage frequency (CDF) due to ESW failure, and the dose consequence.

In order to estimate the core damage vulnerability caused by the failure of the ESW system, a full-scale PRA model that included initiating event frequency categories, event tree and fault tree analysis, and support system dependencies was developed. The PRA model was then appropriately modified to reflect various plant operating configurations to analyze the consequences of the loss of ESW function in each operating state. In addition, the probability of a loss-of-coolant accident (LOCA) caused by failure of a reactor coolant pump seal was established based on a recent pump seal failure model.

The initiating event frequency representing the loss of ESW for multi-unit site operations was derived from operational experience for single-unit PWRs. This initiating event frequency, as modified, is assumed to be valid for multi-unit PWR sites and is not specifically limited to single PWR units. As the system configuration for various operating states may be different, the initiating event frequency for each state was determined separately. An approximation method involving the combination of the experience data with an analytical technique was used. A multi-unit ESW system fault tree similar to the existing model for Byron Unit 1 was developed. This modified model represents the unavailability of the second unit to supply ESW to the first unit in the event of complete loss of ESW in the first unit.

To calculate the initiating event frequency for loss of ESW, the total of 667 reactor-years was divided into 487 reactor-years at power and 180 reactor-years at shutdown. The initiating event frequencies were calculated to be 1.1E-03 per reactor-year at power, 3.2E-02 per reactor-year at shutdown (with one pump running and one at standby), and 2.9E-01 per reactor-year at shutdown (with one pump running and the other in maintenance).

The evaluation showed that the small LOCA due to failure of the reactor coolant pump seal is the dominant accident sequence.

Operating experience data consisting mostly of LER submittals showed that the duration of the ESW system failure has ranged from 1 hour to a few days before recovery. Approximately 70% of the ESW failures were recovered within one hour. About 20 to 26% of all events involved more problematical hardware or other failures and were recovered within 5 hours. The remaining events involved the most serious hardware problems, and recovery took a relatively long time. It is estimated that, by the end of 24 hours, only about 1% of the events were not recovered.

For each of the operating states, a conditional core damage probability was calculated by renormalizing the original base case with the respective configuration-dependent initiating frequency and weighting the state-dependent initiating event frequency. The dominant sequence is the reactor coolant pump seal LOCA with a CDF of 8.8E-05 per reactor-year, which is about 60% of the total CDF due to loss of ESW of 1.5E-04 per reactor-year.

The total CDF (1.5E-04 per reactor-year) is judged to be substantial compared to the Commission's subsidiary safety goal of 1.0E-4 per reactor-year.

Potential alternatives for improvements that could lower this core damage frequency were selected by considering the dominant failure modes of the ESW system and the dominant accident sequences contributing to the relatively high CDF.

The following potential improvements were analyzed:

1. An additional crosstie to reduce the possibility of a malfunction of the cross-connection between units.
2. An electric power cross-connection to increase the redundancy of the electric power supplies to ESW pumps.
3. A separate intake structure or bay with an additional swing ESW pump to increase the redundancy of the ultimate heat sink or source of cooling and increase the availability of the ESW pumps.
4. Changes in Technical Specification requirements and emergency procedures.
5. Installation of an independent reactor coolant pump seal cooling system.
6. A combination of the reactor coolant pump seal cooling system and changes in Technical Specifications and Procedures.

Costs, reductions in CDF, and collective dose averted have been calculated for each alternative. The results are being evaluated, and a proposed resolution of GI-130 is being coordinated with other related generic issues.

Volcanology Research Program Plan Under Development

Linda A. Kovach, DE/WMB

Volcanic activity has occurred in the vicinity of Yucca Mountain during the Tertiary and Quaternary. Evaluations of a high-level waste repository at Yucca Mountain will require an estimation of volcanic hazards in the Yucca Mountain area over the next 10,000 years. Current scientific methods have not demonstrated, with any certainty, the ability to predict future volcanic activity. This is due in part to a lack of understanding of the controlling mechanisms of magma transfer and storage and even the rates of magma production. The external manifestations of magmatic activity in terms of dramatic eruptions are abundant, but it is much less clear how this phenomenon is quantitatively related to deeper crustal processes.

The Department of Energy (DOE) and the State of Nevada are investigating specific aspects of the potential hazard of volcanic activity at the Yucca Mountain site. The Waste Management Branch of the NRC Office of Nuclear Regulatory Research is also developing a modest research program to provide an independent technical basis for the review and assessment of the claims of DOE and Nevada based on their investigations. The proposed program is intended to critically review two potential mechanisms for volcanism in the Basin and Range, the tectonic region containing Yucca Mountain, and to investigate the extent to which it is possible to assign a probability for future volcanic activity in the region surrounding Yucca Mountain. Three studies will be performed in an attempt to address the above issues. The first study will analyze the existing volcanic and tectonic data of the Great Basin to identify spatial and temporal relationships between volcanism and tectonism.

The second study will encompass investigations performed at several different analog sites throughout the Southwest. Because of the size of volcanic systems, e.g., with roots in the upper mantle and a subareal extent of cubic kilometers, it is virtually impossible to understand the dynamics of a volcanic system through the study of one volcanic field unless extensive drilling is performed at the site. The alternative approach being used in the NRC work involves investigations of at least two sites that exhibit varying degrees of erosion and hence exposure of the internal plumbing system. The first investigation will examine the dynamics of a volcanic vent and the potential for polycyclic volcanism through investigation of an analogous but deeply eroded Tertiary basaltic volcano in the Sierra Nevada, California, thereby providing vital information regarding the internal plumbing system. The second investigation will attempt to identify the various eruptive scenarios through consideration of the following issues: structural control on volcanism within a volcanic field, coupled magmatic-hydrologic processes, temporal and spatial distribution of volcanic events, and the character of the crust and upper mantle.

The third study will be designed to integrate the results of the other work into the NRC's performance assessment efforts by modeling the mantle systems that control the tectonic behavior of the Basin and Range. This will provide the driving force that defines the boundary conditions for the regional models evaluated in the first study. These models will then be used to estimate the probability and probable type of volcanic events that may occur over the projected life of a repository.

RES plans to begin this work in FY1991 with the placement of the first project at the Center for Nuclear Waste Regulatory Analyses. A proposal has been made to the Office of Research by the Johns Hopkins University (JHU) to invite the author to come to JHU as a part-time Associate Research Scientist in the area of volcanology. If this proposal is approved, she will begin work on the volcanic vent investigations as part of her duties at JHU. The studies of an extended volcanic field are designed to follow the volcanic vent studies if funds are available, and the final integration study is designed as a follow-on to the first study of available information on the Basin and Range.

Division of Engineering to Support Review of Advanced Reactors

John A. O'Brien, DE/SSEB

With the aim of assisting NRR in its technical review of evolutionary, passive, and other advanced reactors, the RES Division of Engineering has developed a 5-year research plan that covers structural, seismic, mechanical, electrical, and materials engineering. The plan has been coordinated with NRR, other RES divisions, and the Nuclear Safety Research Review Committee. Some of the important issues treated in the plan are:

1. Assessment of unique construction techniques and unusual structural configurations,
2. Reliability of components in passive systems,
3. Definition of the OBE for advanced reactors,
4. Qualification of advanced instrumentation and control systems,
5. Design criteria for interfacing systems LOCAs,
6. Deficiencies in ASME fatigue requirements applied to advanced reactors with a 60-year life,
7. Extent of application of leak-before-break behavior in advanced reactors,
8. Tornado design requirements for advanced reactors,
9. Pressure vessel materials for advanced reactors,
10. Containment performance goal for severe accidents.

Besides these topics, special sections of the plan deal with reactor types very different from those presently in service in the United States. These include the liquid-metal-cooled PRISM design of General Electric, the PIUS pool-type design being developed by Combustion Engineering, the heavy water CANDU-3 being proposed by Atomic Energy of Canada Limited, and the modular high temperature gas reactor of General Atomics. The plan recognizes that future licensing practices focus on design certification of standardized units and addresses questions resulting from the staff review of the EPRI advanced light water reactor requirements document.

National Seismographic Network Dedicated

DE/SSEB

On April 3, 1991, the National Seismographic Network (NSN) was officially dedicated at the National Earthquake Information Center of the U.S. Geological Survey (USGS) in Golden, Colorado. The NSN was established as a cooperative effort between RES and the USGS. The USGS has long been active in studying the seismicity of the United States, especially the Western U.S., whereas the NRC has placed more emphasis on the seismicity of the Eastern U.S. The NRC, for over a decade and a half, has sponsored regional seismographic networks covering areas of the Central and Eastern U.S. An interagency agreement signed in 1986 established the NSN, which will take over the function of the regional networks in the Eastern and Central regions after 1992.

The NRC was instrumental in identifying the need for establishing the NSN program and using more advanced earthquake monitoring technology than had been used previously. The NRC and USGS jointly directed the implementation of the program. The NRC is primarily responsible for the portion of the network east of the Rocky Mountains, whereas the USGS emphasizes implementation of the western portion. The NRC is therefore a major partner in this pioneering effort that combines the most advanced seismograph stations with data transmission via satellite to a central recording facility.

The dedication ceremony was attended by representatives from the Department of the Interior, the USGS, the NRC, and other interested parties, including the news media and seismologists from all parts of the U.S. Participants were addressed by, among others, Frank Bracken, Deputy Secretary of the Interior; Dallas Peck, Director of the USGS; and Eric Beckjord, Director, Office of Nuclear Regulatory Research. Dallas Peck and others strongly emphasized that this undertaking has been marked by excellent cooperation between the USGS and the NRC, an achievement that does not always accompany agreements between Federal agencies.

Feasibility of Using Cellular and Molecular Research to Reduce Uncertainties in Health Effects from Low-Level Radiation Studied

DRA/RPHEB

RES has sponsored a study to examine the feasibility of reducing the uncertainties in the estimation of risk due to protracted low doses of ionizing radiation through studies at the cellular and molecular levels. The idea that such studies would be timely was derived from the discovery of oncogenes, the development in the past 15 years of the techniques of recombinant DNA molecular biology, and the evident and significant progress that has been made in the characterization of certain human cancers in genetic terms.

The study was performed by Science Applications International Corporation with participation of eminent scientists from the Department of Radiology and Radiation Biology, Colorado State University, under the leadership of Professor Mortimer M. Elkind.

In addressing the question of feasibility, the study group reviewed the cellular, molecular, and mammalian radiation data that are available; the way in which altered oncogene properties could be involved in the loss of growth control that culminates in tumorigeneses; and the progress that had been made in the genetic characterizations of several human and animal neoplasms. On the basis of this analysis, the study group concluded that, at the present time, it is feasible to mount a program of radiation research directed at the mechanisms of radiation-induced cancer with special reference to risk of neoplasia due to protracted low doses of sparsely ionizing radiation.

A report on this study, NUREG/CR-5635, has been distributed to individual scientists and Federal agencies working in this field. The staff intends to explore with them the possibility of developing a coordinated program of research designed to improve the understanding of the effects of low-level radiation.

RESEARCH NEWS is published by the USNRC Office of Nuclear Regulatory Research, Edward L. Hill, Editor.

Comments, suggestions, and articles for future issues should be directed to the Editor, RESEARCH NEWS, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

120555139531 1 1A01AR19L19R
US NRC-OADM
DIV FOIA & PUBLICATIONS SVCS
TPS-PDR-NUREG
P-223
WASHINGTON DC 20555