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Reviewing Advanced Light Water Reactor Designs

T. E. Murley

The NRR staff has entered a period of intense activity in reviewing the designs of the next generation of light water reactors. The results of our reviews will set the standard for the safety design of nuclear plants to be built and operated in the United States well into the middle of the next century.

Part 52 requires that the NRC make final safety judgments on a design at the time of certification, before a plant has been ordered and constructed. The rule further requires that the NRC make final safety judgments on site-related matters before issuing a combined operating license (COL). These requirements place a greater level of discipline upon the staff reviewers than was the case under the two-step licensing process for Part 50, because the staff cannot defer making a judgment on an issue in the expectation of making a final decision when the plant is under construction.

A further need for more discipline in the review process under Part 52 is that the staff must specify the inspections, tests and acceptance criteria (ITAAC) that are to be used to verify that the conditions of the COL are met during construction. This is complicated in some areas in which practical limitations have prevented the vendor from completing the design. In these areas, the staff will have to make final safety judgments by specifying design acceptance criteria (DAC) to be verified by the staff after the COL is issued. The concept of design acceptance criteria is new to the staff, and it will require innovative thinking, as well as discipline, to make it work.

In spite of these challenges, I believe the staff reviews will be more thorough than those conducted in the past and will result in safer designs. We have a great deal more experience now, because we have the benefit of some 1600 reactor-years of operating experience in the U.S. Many problems in the early designs have been revealed through operating experience and have been dealt with through backfit regulations (e.g., TMI, ATWS and Station Blackout rules). A great deal of high quality research over the years has added to our understanding of reactor safety as well.

Because we are using what I call both a top-down review and a bottom-up review approach, the result is expected to be an integrated, balanced design review. The top-down review comes from our requirement that the designer feed back the insights from his probabilistic risk assessment (PRA) into the design process. This has already led to a number of design changes to improve the plant's ability to cope with core melt accidents, for example.

The bottom-up review deals with specific issues that have arisen with the current plant designs. Some of these specific safety issues are:

- Fire protection will be assured through better separation of safety systems.
- Intersystem LOCAs can be prevented by design.
- The risk from accidents during shutdown operational modes can be reduced by design.
- Defense in depth can be enhanced by considering core melt accident mitigation features in the design.
- A reliability assurance program can integrate requirements for maintenance, surveillance, in-service inspection and in-service testing with the technical specifications to help assure that the initial reliability of equipment (assumed in the PRA during design) will be maintained throughout the operating life of the plant.

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When we say that we expect the Advanced Light Water Reactors will be safer than current plants, there are three aspects to that assertion. First, the PRAs will show the predicted core damage frequency to be lower for advanced plants because of less frequent challenges and a better ability of the plant to cope with the challenges. Second, the improved mitigation features included in the design will improve the containment performance under severe core damage accident conditions. Third, the designs are expected to reduce the number and complexity of actions required of the operators, thereby reducing the chances of human errors and increasing our confidence in the predictions of better prevention and mitigation of accidents.

BWR Hydrogen Water Chemistry and Zinc Injection Passivation in the United States

Frank J. Witt, DET

Hydrogen water chemistry (HWC) is a countermeasure to mitigate intergranular stress corrosion cracking (IGSCC) in reactor recirculation piping and reactor internal components in boiling water reactors (BWRs). Many U.S. utilities are adopting HWC for their BWRs, with 8 units operating permanently with HWC, 12 units having completed preimplementation test programs, and 9 other units installing permanent equipment. The water environment contributes in two primary ways to IGSCC: the oxidizing power of the water indicated by the electrochemical potential (ECP) of stainless steel and the concentrations of ionic impurities, particularly sulfate and chloride. To suppress IGSCC, reactor coolant conductivity must be maintained below 0.3 microsiemens per centimeter, and sufficient hydrogen must be added to the feedwater to reduce the ECP below -0.23V (standard hydrogen electrode). Excess hydrogen in a high radiation field results in the catalytic recombination of hydrogen with radiolytic decomposition products (O_2 and H_2O_2 which cause IGSCC) to reform water.

The feedwater hydrogen injections required to establish HWC ECP conditions vary between plants and between regions within a plant. At the Hope Creek Generating Station and the Nine Mile Point Nuclear Station, the concentration of oxygen in the reactor coolant is reduced to 1 ppb by maintaining a feedwater hydrogen concentration of 0.6 ppm and 1.9 ppm, respectively. The amount of feedwater hydrogen and resulting O_2 in the recirculation water depend primarily on the individual design characteristics of the reactor and can be correlated with the downcomer dose rate and the mass flows in the core regions. A General Electric-Harwell BWR water radiolysis model predicts the concentration of radiolysis products in various regions of the BWR primary system. Comparing O_2 levels in the reactor plant recirculation system with this model indicates reasonable agreement (References 2 and 3). The radiolysis model shows the different regions of the BWR respond differently to hydrogen additions and may require increased concentrations of hydrogen in the reactor coolant to attain the -0.23V ECP needed to mitigate IGSCC in the reactor's internal structures. Utilities have measured the ECP in the reactor core to determine the quantity of hydrogen that should be added to suppress IGSCC.

At the Edwin I. Hatch Nuclear Plant Unit 1 in the U.S. and Oskarshamn Unit in Sweden, copper ions in the coolant prevented the HWC from yielding the results predicted by the BWR water radiolysis model. These copper ions reduce the effectiveness of the hydrogen recombination with radiolytic decomposition products. These plants have Admiralty brass condensers coupled with filter/demineralizer condensate cleanup systems which typically release into the coolant copper concentrations in the 15-30 ppb range and zinc at concentrations in the 2-10 ppb range. At Hatch Unit 1, the licensee has replaced the condenser Admirality brass tubes with titanium tubes. This action reduced the copper concentration in the feedwater, which significantly decreased the hydrogen concentration in the feedwater which reduced recirulation coolant ECP to less than -0.23V. Before the condenser tubes were replaced, sufficient hydrogen could not be added to mitigate IGSCC (-0.23V ECP) because of excessive N-16 radiation levels measured by the main steam line radiation monitor (MSLRM).

When hydrogen is added to the feedwater to yield the proper HWC, the N-16 steam activity increases, resulting in higher MSLRM levels. Because hydrogen dissolved in the reactor coolant reduces the oxidizing potential, the N-16 is converted from a nitrate to a more volatile ammonium ion, increasing the amount of steam N-16 activity. To address this problem, the licensees for some BWR plants have increased the MSLRM set points above the values specified in the technical specifications to allow HWC.

The Electric Power Research Institute (EPRI) and experienced industry personnel have developed generic guidelines for designing, installing, and implementing permanent HWC at BWR plant sites. These guidelines discuss recommended safety features and address possible increased N-16 dose rates and failure modes of systems that store and handle hydrogen on the plant safety systems. To prevent damage to safety related structures, the licensees provide separation distances of hydrogen storage facilities (liquid and gaseous H_2) to protect against a storage facility explosion. Separation distances are also provided to prevent flammable gas mixtures from forming at safety-related air intakes if a pipe breaks at a storage facility. The NRC has approved these guidelines for the licensees to use in safely implementing HWC.

To manage the increased dose rate from the HWC-enhanced production of N-16, the licensees can perform an initial and continuous radiation survey program and change the procedures for plant shielding and maintenance. N-16 has a half-life of 7.1 seconds and thus causes the radiation level to increase only upstream of the main condenser. These changes to programmatic procedures, in addition to the normal procedures for plant radiation protection, are sufficient to ensure that, while the licensee is adding hydrogen, the plant will continue to meet the requirements of Part 20 of Title 10 of the Code of Federal Regulations (10 CFR Part 20) and the recommendations of Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be As Low As Reasonably Achievable."

In addition to reporting increased amounts of N-16 in the main steam line, licensees have recently reported increases in shutdown dose rates in some U.S. HWC plants. The increased shutdown dose rates primarily result from Co-60 that is deposited in the corrosion film on piping and component, and in hot spots in crud traps. This increase varies among plants where an increase in soluble Co-60 and the spiking of insoluble crud (Co-60, 1.17 and 1.33 Mev; Mn-54, 0.84 Mev; Zn-65, 1.11 Mev) have been observed. These dose rates probably increase because HWC reduces the corrosion potential. This reduction resulted in a restructuring of the oxide corrosion deposits on surfaces inside and outside the core. Such a restructuring would result in the release of particulate and soluble Co-60. The General Electric Company (GE) believes that the effect may be temporary but may last for more than one cycle. To correct this problem, the licensee for the Edwin I. Hatch Nuclear Plant, Unit 1, used additional temporary lead shielding during the last refueling outage and is considering chemically decontaminating the recirculation piping to maintain an as-low-as-reasonably achievable (ALARA) program. Utilities that operate Swedish HWC plants have not observed increases in the shutdown dose rate because these plants have minimized the cobalt content in materials that contact the reactor coolant. To minimize the accumulation of radioactive materials that could cause a shutdown condition, the licensees for operating plants should minimize the cobalt in replacement materials, and the designers of advanced reactors should eliminate cobalt where possible.

The licensees can manage the increased occurrence of shutdown dose rates attributed to HWC by performing normal radiation protection procedures including using temporary shielding and chemically decontaminating recirculation piping. Although HWC may increase person-rem, such an increase would be relatively minor compared to the amount of person-rem exposure from the inspections and repairs needed to maintain overlay welds and to replace the recirculation piping associated with pipe cracking in BWR plants operating with normal water chemistry. The BWR Owners' Group has submitted a topical report to the NRC justifying inservice inspection (ISI) credit for plants operating with HWC. If the NRC approves this initiative, it would allow licensees operating HWC plants to reduce the frequency of ISI inspections, with an accompanying reduction in person-rem. Therefore, ISI credit would offer additional incentive to implement HWC.

The control of the build-up of radiation in reactor systems has been of concern in BWR plants. GE reviewed operating plant correlations and found that plants having 5 to 15 ppb of soluble zinc in the reactor water (as a result of the unique plant design parameters discussed previously for Hatch 1 and Oskarshamn-1) had lower piping dose rates than plants that had only trace amounts of zinc. Laboratory tests confirmed that using soluble zinc causes much less Co-60 to be deposited in pipes having both normal and hydrogen water chemistries. These studies led GE to develop the General Electric Zinc Injection Passivation (GEZIP) process in which small quantities of zinc are injected to the final feedwater. Currently, eight U.S. plants have implemented zinc injection and four other plants will start zinc injection in a few months. Dose rate measurements made at several of these plants indicate significantly lower levels of Co-60 than are found at plants not using zinc. As a side effect of injecting zinc, the activated isotope Zn-65 (half-life 244 days) has become a significant contributor to the dose rates from reactor water and piping. To enhance the benefits of injecting zinc, GE is developing methods to produce the zinc oxide that is depleted in the precursor Zn-64. The reference water chemistry for the advanced boiling water reactor is HWC and GEZIP. The NRC continues to monitor the implementation of HWC and GEZIP and their effectiveness in mitigating IGSCC and reducing the buildup of shutdown radiation levels.

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Industry Reports For License Renewal

P. T. Kuo, PDLR

On December 13, 1991, the Commission published the 10 CFR Part 54 rule on license renewal. Currently, the Commission issues licenses to operate nuclear power plants for a fixed period of time not to exceed 40 years. However, the Commission may renew these licenses to extend the period of operation up to 20 years under the regulatory requirements set forth in 10 CFR Part 54.

The regulatory philosophy and approach for the 10 CFR Part 54 rule are based on two key principles. The first principle is that the regulatory process is adequate to ensure that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety for operation. The second principle is that each plant's current licensing basis (CLB) must be maintained during the renewal term. To maintain the CLB during the period of extended operation, the licensee may need to implement a program to manage the age-related degradation of systems, structures, and components (SSCs) that are important to license renewal.

To assist the NRC in renewing licenses, the nuclear industry is developing a series of technical reports (industry reports (IRs)) on generic resolution of, and programs to manage, the age-related degradation of a variety of structures and components (SCs) important to license renewal. The IRs are coordinated by the Nuclear Management and Resources Council (NUMARC) and sponsored by the U.S. Department of Energy (DOE) and the Electric Power Research Institute (EPRI). In the IRs, the industry evaluates the effects of age-related degradation on specific SCs, describes the bases for the manner in which existing regulatory programs address the degradation concerns, and recommends specific corrective actions for specific SCs not presently addressed by effective age-related management programs.

By October 1990, NUMARC had submitted 11 IRs for the staff to review. The industry has written these IRs on subjects affecting both boiling water reactors (BWRs) and pressurized water reactors (PWRs). The staff expects to complete its review and issue safety evaluation reports (SERs) on the IRs as follows:

Industry Report	SER Date
Cable Inside Containment	1/93
Class 1 Structures	8/92
BWR Containment	8/92
BWR Reactor Coolant System	11/92
BWR Reactor Vessel	11/92
BWR Reactor Vessel Internals	11/92
PWR Containment	5/92
PWR Reactor Coolant System	8/93
PWR Reactor Vessel	9/92
PWR Reactor Vessel Internals	3/93
Screening Methodology	10/92

The staff found that the technical justifications in these IRs generally did not support conclusions on managing age-related degradation. These justifications were not sufficient because the industry prepared the IRs (1) before the proposed 10 CFR Part 54 was published, and (2) were based on an unsubstantiated assumption that existing programs were, by definition, effective in managing aging. Nevertheless, these IRs contain valuable information concerning the effects of age-related degradation on SCs. The staff is using this information in developing the review and acceptance criteria that will be incorporated into the regulatory guide and the standard review plan on license renewal (SRP-LR).

To minimize unnecessary revisions of the IRs and to focus resources on the technical issues in each IR, NUMARC and the staff agreed to initiate the following review process in revising the IRs:

1. NUMARC submits the IR.
2. The staff reviews the IR and provides written comments.
3. NUMARC reviews the staff's comments and submits a written response to each comment.
4. The staff reviews NUMARC's responses and determines if they are adequate.
5. The parties hold a public meeting to discuss any comments which require additional review and response and to discuss any issues regarding the IR under review. (This step may be repeated if necessary).
6. NUMARC submits the revised IR.

This cooperative process has enabled the staff and NUMARC to resolve many technical issues identified in the IRs. The remaining steps in the IR review process will be to review the revised IRs after they are submitted by NUMARC, resolve any technical issues that remain, and prepare an SER for each of the IRs. The draft SERs will be reviewed by the Committee to Review Generic Requirements (CRGR) and the Advisory Committee on Reactor Safeguards (ACRS). The staff will publish the draft SERs for public comment before issuing the final SERs.

The IRs have been reviewed by the staff from the Office of Nuclear Reactor Regulation and the Office of Nuclear Regulatory Research and by expert consultants from several national laboratories. The staff has provided an average of 80 technical comments on each IR. NUMARC has provided responses to staff's comments and met with the staff to discuss the responses on all IRs, except the IR on PWR vessel internals. The staff and NUMARC have reached an agreement in principle (AIP) on approximately 85 percent of the comments on the IRs by reviewing of NUMARC's responses and discussing these issues during public meetings. An AIP is a general agreement between NUMARC and the staff on the manner in which to resolve a particular comment and indicates that NUMARC will revise the IR to be consistent with the agreement. The following list provides the current disposition of comments provided to NUMARC:

Industry Report	Number of Comments	Number of Remaining Comments
Cable Inside Containment	114	18
Class I Structures	99	5
BWR Containment	128	8
BWR Reactor Coolant System	50	5
BWR Reactor Vessel	67	13
BWR Reactor Vessel Internals	51	3
PWR Containment	95	14
PWR Reactor Coolant System	45	9
PWR Reactor Vessel	49	25
PWR Reactor Vessel Internals	65	No Meeting Yet
Screening Methodology	Reevaluating per 10 CFR Part 54	

Although the numbers by themselves do not indicate the extent of acceptability of the IRs, they do indicate the intensive effort expended by NUMARC and the staff in the IR review process.

The staff is reviewing the revised IRs that NUMARC recently submitted on PWR containment, BWR containment, and Class I structures.

The IR review process enables NUMARC and staff to focus their efforts on resolving open issues in a systematic and technically sound manner. By employing this review process, the staff and NUMARC have identified several technical issues that must be resolved. These issues include the evaluation of fatigue in metal components, the environmental qualification (EQ) of electrical equipment, and the thermal embrittlement of cast stainless steel components. These are complex issues for which the design requirements for the currently licensed plants vary considerably. By interacting together, the industry and the staff have more clearly defined the aspects of these issues that pertain to age-related degradation. This interaction assisted the staff in developing draft branch technical positions (BTPs) for fatigue and EQ to be considered for incorporation into the SRP-LR.

While the staff has not completed its review of these IRs, it found that they contain valuable information on the effects of age-related degradation on SCs. Once approved by the staff, the IRs will provide the generic technical basis for evaluating the effects that age-related degradation could have during the license renewal term. The IRs provide for a single review and approval with subsequent referencing, rather than repetitive reviews of the same subject. By referencing the IRs, license renewal applicant will enable itself and the staff to focus their resources on resolving for specific plants the license renewal technical issues which are idenified in the IRs as requiring enhanced plant-specific aging management and which are outside the scope of the IRs.

Indian Point Unit 2 Steam Generator Girth Weld Cracking

Herbert J. Kaplan and Alfred Lohmeier
Region I

Inservice inspection of the upper shell to transition cone girth welds during the 1989 spring outage of Indian Point Unit 2, revealed circumferential cracks in the weld- and heat-affected zone material of four steam generators. When girth weld cracking was discovered in these steam generators during the 1987 outage, the licensee removed the defects by grinding and did not repair any of the welds by rewelding.

The licensee removed metallurgical samples during the 1989 outage and determined that a corrosion-fatigue mechanism had caused the cracking. The corrosion resulted from cold feedwater injection in the transition cone region during startup, plant operations, hot standby, and other plant transient modes of operation.

This time, the licensee used an automatic tungsten inert gas (TIG) process to remove and repair weld the defects. All girth welds were subjected to a 4-hour, 1125° F local stress relief treatment.

Following magnetic particle inspection, the licensee found more indications in previously ground-out cavities, in weld repair areas, and within the original weld metal. The licensee ground out and retested these defects and returned the plant to service.

In addition to the weld repairs and heat treatment, the licensee removed the steam generator downcomer resistance plates and installed a timer to delay feedwater bypass valve closure to lessen the thermal shock effect during cold feedwater injection.

During the midcycle inspection after the 1989 startup, the licensee discovered more indications in the weld metal of the four steam generator girth welds. The licensee again used the TIG process, for repair welding, but without the 1125°F post-weld heat treatment. Instead, the licensee applied a 500°F post-weld treatment to produce a tempered heat-affected zone.

The licensee, assisted by several consultants and research organizations, concluded that the principle cause of crack initiation and progression was related to the dissolved oxygen content in the secondary water, which acted on susceptible material in a high-tensile stress field. The licensee then machined a 360-degree, 6-inch wide by 3/4-inch deep groove around the girth welds of the steam generators and filled the grooves with a low sulfur weld metal, followed by an 1125°F heat treatment. The licensee also will maintain low oxygen levels by using a nitrogen blanket in the condensate storage tank and will reduce the tensile stress by providing a smooth radius at the shell/cone junction. The licensee's actions should provide a basis for continued safe operation of the steam generators.

Reviews of Recent Epidemiological Studies of Radiation Risks

Frank J. Congel and Charles A. Willis

Introduction

The NRC is responsible for protection of the public and workers from the ill effects of exposure to ionizing radiation. To meet this responsibility effectively, the NRC needs to understand the magnitude of the risks associated with radiation exposure. Thus, when new studies are reported that purport to cast new light in this issue, the staff examines them carefully. The staff recently reviewed five new epidemiological studies of radiation risks and the results of those reviews are summarized here.

Background

Everyone is exposed to radiation at all times. This has always been true, although no one knew about it until 1895 when x-rays were discovered. Radiation injuries were reported only a few months after radiation was discovered. Since that time, radiation and its biological effects have been the subject of intense world-wide scientific investigation. The important effects were soon identified. Even the possibility of genetic damage was reported in 1911. The fundamentals of radiation protection also were identified within a few years of the discovery of radiation. The first person known to be killed by man-made radiation was Thomas Edison's assistant, Clarence Dally. By the time of Mr. Dally's death in 1904, Edison reported that proper precautionary measures had been developed and that "I would continue the work myself but my wife won't let me."

Radiation protection measures were not always applied and, as a result, hundreds of people died of radiation-induced cancer and others suffered radiation injuries. Early injuries initiated public controversy before 1900. Radiation injuries caused by the use of x-rays to investigate wounds during World War I contributed to the controversy. Despite the controversy, radiation was misused. Misuse is exemplified by Radium tonics being sold through the mail and fluoroscopes being available in most shoe stores. The public controversy, legal actions, and voluntary control measures combined to eliminate most of the gross misuses of radiation by the end of World War II.

The development of the atomic bomb provided new impetus and funding for radiobiological research. In 1956, the National Academy of Sciences declared that radiation was the best understood environmental hazard. Research has continued and, today, radiation risks are very well understood. However, we have not yet determined the magnitude, if any, of risks from exposure to low levels of radiation (such as less than about 10 rem per year).

If the mechanisms of radiation injury were known, the question of risk from low-level exposure could be answered with laboratory investigations. Since the mechanisms are not known, epidemiological studies are conducted to try to reduce the degree of uncertainty.

Current risk estimates for low-level exposure are based primarily on the results of epidemiological studies of the survivors of the nuclear weapon detonations at Hiroshima and Nagasaki. These results are supported by studies of other highly irradiated groups such as the radium dial painters, patients irradiated as a treatment for ankylosing spondylitis, and women irradiated as a treatment for cervical cancer. Where doses are high (above about 50 rem) cancer rates are increased. For example, the cancer rates for the survivors of Hiroshima and Nagasaki apparently were increased about 5 percent.

Researchers have conducted numerous studies of groups receiving lower doses, but the results are inconclusive. Generally, they have found no increase in cancer rates, even in Guarapari, Brazil, where 12,000 people receive doses of about 0.64 rem per year, which is about 5 times the average background dose; in Kerala, India (0.38 rem per year); or in Yanjiang County, China (0.3 to 0.4 rem per year).

No radiation-induced genetic effects have been observed in any human population.

Epidemiological Studies Reviewed

The five studies reviewed were conducted by the National Cancer Institute (NCI) [1], the Three Mile Island Public Health Fund (TMIPH) [2], the Massachusetts Department of Public Health (MDPH) [3], Steve Wing, et al. [4, 5]; and Sternglass and Gould [6]. The populations investigated differed in most respects between each of the studies and the investigators reached markedly differing conclusions. The extremes were the NCI and the Sternglass-Gould studies. The NCI study reported no detectable ill effect in the populations around any nuclear power plant or Department of Energy (DOE) facility in the U.S. However, Sternglass and Gould contend that effluents from the Trojan nuclear plant are killing thousands of people annually in Oregon.

We reviewed these studies and concluded that none of them convincingly showed any discernible effect of low-level radiation or provided any reason to believe that the NRC should revise its effluent control practi-

The Sternglass-Gould Report

E. Sternglass has long had the reputation of being one of the most uninhibited of the antinuclear activists, and J. M. Gould is rapidly developing a similar reputation. In conducting their work, which was funded by a political group trying to shut down the Trojan plant, they concluded that radioactive effluents are killing over 8,000 people each year in Oregon. Their basis for this conclusion is that the death rate in Oregon declined in the 2 years preceding the startup of Trojan. If this "trend" had continued, the death rate in Oregon would be far below its current value. Sternglass and Gould contend that the effluents from Trojan caused the difference between the actual and the projected death rates. However, if this "trend" had continued, the present death rate in Oregon would be far below the national average and, in another 75 years, the life expectancy in Oregon would reach 1,000 years. Clearly, the logic and the conclusion lack substance.

The contention by Sternglass and Gould is made even more dubious by the Trojan's outstanding effluent control record. Radioactive effluents have been so limited that the total calculated population dose from all releases through 1986 was only 1.0 person-rem (NUREG/CR-2850, Vol. 8). By comparison, the dose from natural background radiation to the population in the vicinity of Trojan exceeds 1 person-rem every 4 minutes.

The Massachusetts Department of Public Health Study

The MDPH study also is seriously flawed. This was a case-control study of leukemia (other than chronic lymphatic leukemia (CLL), which other investigators show to be non-radiogenic) in people over 12 years of age in the 22 communities within 25 miles of the Pilgrim Nuclear Power Station. The MDPH identified "cases," people for whom leukemia had been diagnosed, from medical records and selected matching "controls". The researchers estimated relative doses and assumed the extent to which the cases had doses higher than the control group to be a measure of the impact of radiation. The MDPH researchers found one time period in which the estimated doses were higher for the cases than for the controls. The MDPH concluded that radiation had quadrupled the leukemia rate in that period for the more highly exposed group.

In our review, we found the MDPH conclusion untenable for several reasons. First, the short duration of the increased incidence of leukemia is inconsistent with the increase being radiogenic; that is, the elevated incidence disappeared just when it would have been approaching a maximum if it had been caused by radiation. Second, the distribution of doses from effluents assumed by the MDPH is totally inconsistent with the actual calculated doses from effluents; the doses from natural background and other radiation are ignored. Third, the method used for determining the location of the people is highly inaccurate: questioning a surviving friend or relative by telephone to determine where the person lived and worked many years ago. Fourth, the leukemia incidence of the low dose group was well below the average for the state. Fifth, the presumed consequences are totally inconsistent with the doses, based on generally accepted methods.

The total calculated population dose from Pilgrim effluents was only 260 person-rem. The National Academy of Sciences (BEIR-V) estimates that this dose could cause a total of from 0 to 0.05 cancer deaths. Furthermore, the doses from effluents were only a small fraction of doses from natural background radiation. Thus, even if very high values are assumed for the radiation risk factors, the effluent doses could not have caused a discernible increase in leukemia.

Oak Ridge Workers Study

Steve Wing proclaimed on national television that he had shown that radiation risks are 10 times greater than the National Academy of Sciences' estimates. Before the proclamation, there was little interest in this work because the sponsor, DOE, said no ill effects were being found. DOE epidemiologists expressed surprise at both the conclusion and the public announcement.

Our review was complicated by the omissions from the publications: they did not contain either the data or a full description of the methodology. We were told that neither is available.

The population studied consisted of the white males hired to work at Oak Ridge between 1943 and 1972. This population excluded people who worked there for less than a month, those who worked at other nuclear facilities, and those for whom the dose or [illegible]ographic data were incomplete. The researchers drew their conclusions from the following reported observations: (1) the lifetime doses were quite low, with the average being only 1.7 rem and with only 321 people (3.8 percent) getting more than 1 rem; (2) the average death rate of these people was significantly less than expected (based on the data for white male Americans); (3) the cancer death rate also was less than expected; (4) the leukemia death rate was 63 percent higher than expected; and (5) with the arbitrary selection of a "lag time" (latent period), the death rate could be correlated with dose.

In our review, we noted the following. The observed effects could not be related to radiation dose because the measured doses were only a small fraction of the total doses and because exposure to other carcinogens was not taken into account. Second, the higher-than-expected leukemia rate is a common variation in situations where radiation exposure cannot be the cause. Third, resorting to an anomalous "lag time" tends to invalidate any correlation found between dose and effect. Fourth, the short time between completion of the study and publication precluded meaningful peer review; one reviewer was not given time for even a cursory review. Fifth, the report contains obvious errors in the few instances in which numbers can be checked. Finally, the publication contains an antinuclear discourse that indicates a nonscientific agenda. Therefore, we concluded that the Oak Ridge study did not constitute a basis for changing regulatory practice.

The TMI Public Health Fund Study

In this study researchers investigated the population living within 10 miles of the Three Mile Island Nuclear Station (TMI) for childhood cancers, leukemia, lymphoma, lung cancer, and all cancers. The researchers estimated the radiation doses from the TMI accident, from the normal effluents and from natural background radiation. The researchers found exposure rates to vary from 50 to 90 millirem per year from external sources of natural background radiation. The maximum exposure from the TMI accident was less than 100 millirem and the exposure from normal effluents was much less than that value. The authors acknowledged being unable to find any effect, while admitting their disappointment. However, they should have been expected to find no effects of exposure to plant effluents because effluent exposures were less than variations in natural background.

The National Cancer Institute Study

This study was a heroic effort, covering 51 nuclear power plant sites and 10 DOE facilities for a 35-year period. The researchers divided cancers into 15 categories and analyzed the populations in 5-year age groups. The researchers based the study primarily on mortality data but also included incidence data where they were available. They calculated both standard mortality and relative risk ratios and produced over 23,000 statistical tests for evaluation.

One especially important feature of the NCI study was the inclusion of calculations for the years before the plants went into operation. This clearly showed the variation in the data that could not be the result of plant effluents. For example, of the six standard mortality ratios for childhood leukemia that were significantly greater than one, four occurred before startup. Similarly, in those instances in which relative risks were significantly different from one, only four were greater than 1 while 14 were less than 1. Thus, even though the study included old DOE facilities in which releases were relatively high during World War II, the NCI found "no suggestion that nuclear facilities may be linked causally with deaths from leukemia or other cancers."

In reviewing this document, we note that the results are what would be expected from current knowledge of releases and radiation risks.

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Electrical Distribution System Functional Inspections Program Review And Lessons Learned

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Previous NRC inspection teams had identified generic deficiencies related to plant electrical systems which had the potential for compromising plant safety margins during postulated accidents. As a result, the Special Inspections Branch (RSIB), in conjunction with the Electrical Systems Branch and the regions, developed guidance and techniques for inspecting plant electrical distribution systems and conducted five pilot electrical distribution system functional inspections (EDSFIs) during 1989-1990 to validate these techniques. Based on the findings of the pilot inspections, RSIB developed Temporary Instruction 2515/107 and conducted a training course for regional inspectors. As of January 1, 1992, EDSFIs have been completed at 31 plants. The regions are scheduled to complete EDSFIs at all operating plants by early 1993.

EDSFIs evaluate the capability of the electrical system to perform its intended functions during all plant operating and accident conditions, and the effectiveness of the licensee's engineering and technical support for the system. Inspection teams generally consist of two electrical design engineers, one mechanical design engineer, two electrical field inspectors, and the team leader. The teams perform the inspections on site and, when necessary for the design portion of the inspection, at the licensee's corporate engineering office. The inspection teams use a vertical slice sampling technique to evaluate one or more electrical load flow paths between power sources and loads. The inspectors evaluate attributes such as postulated worst case electrical loads during design basis events, the capacity and operation of power sources, the electrical protection and coordination of protective devices, and the interfaces to supporting mechanical systems. The inspectors also assess the performance of the licensee's organizations regarding engineering involvement in operations, root cause analysis and corrective actions, and self-assessment.

The results of the EDSFI program have confirmed the existence of significant deficiencies, some of which required licensees to promptly determine operability and to take corrective action. The inspection teams identified deficiencies in the design basis that affected the capability of electrical equipment to perform safety functions. Undervoltage relay settings were not adequate to provide sufficient voltage to accident mitigating loads during degraded grid conditions. Plant load studies did not consider conservative "worst case" loads and voltage drops for degraded voltage conditions. Unanalyzed relay drift increased the time required to transfer power between sources. Medium voltage circuit breakers had unconservative short circuit ratings. Neutral resistors were not large enough to handle the current and dissipate the heat.

The largest number of findings identified during the EDFIs involved emergency diesel generators and their mechanical interfaces. The inspection teams identified deficiencies in the capacity of diesel generators to carry transient loads; the multiple start capability; the setpoints for fuel oil tank levels; the quantity, quality and transfer of fuel oil; and the air start accumulator pressure. In addition, the diesel load sequencers had undersized relay contacts and unanalyzed timer drift and transient loads. The inspection teams determined that these deficiencies could have compromised the startup and operation of the diesels and could have affected the supply of emergency power to accident-mitigating loads.

The inspection teams identified certain weaknesses in the capability and performance of licensee engineering and technical support (E&TS) groups. Licensees had not reviewed electrical design calculations performed by the architect engineer for accuracy and completeness and had not adequately performed self-assessments of the design basis of the electrical system. Design basis documentation for the electrical system, such as calculations covering all credible failures and modes of operation, was missing or incomplete, particularly at older plants. Test procedures did not include correct acceptance criteria and some critical electrical equipment was not being tested.

Some inspection teams identified strengths with regard to E&TS groups. Personnel in the field understood numerous technical issues regarding electrical equipment and the safety significance of these issues. Engineering control of plant modifications appeared well established. Root cause analysis of failures of electrical equipment appeared thorough.

By conducting the EDSFIs, the staff has helped licensees improve the functional capability of the electrical distribution systems. For example, EDSFI findings have focused the licensees' engineering and technical support groups on (1) the necessity for controlling loads relative to the capacity of offsite and onsite power sources and (2) the fault protection of electrical equipment. In addition to correcting specific deficiencies, licensees have increased their attention toward evaluating and improving the design basis of the electrical distribution system and also toward related engineering and technical support programs. Several licensees have conducted their own EDSFIs to assess the electrical system design basis and its implementation. Various licensees have identified and corrected problems before the NRC inspected their facilities. Some licensees have increased the scope of their design reconstitution programs to include problem areas, such as instrument setpoints, identified by EDSFIs. EDSFIs have also made licensees aware of the importance of retaining and updating design basis documentation developed by the architect engineer, vendor, and licensee to demonstrate that the electrical system operates properly and to justify modifications to its components. The results of the EDSFIs, including a number of positive findings, have provided increased assurance that the electrical equipment in nuclear plants will perform its safety functions. RSIB is developing a database of EDSFI findings for further evaluation and for inclusion in NRC information notices as appropriate.