

November 29, 2001

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

Ann Marshall Young, Chair

Charles N. Kelber, Administrative Judge

Lester S. Rubenstein, Administrative Judge

In the Matter of

DUKE ENERGY CORPORATION

McGuire Nuclear Station, Units 1 and 2; Docket Nos. 50-369 & 50-370

Catawba Nuclear Station, Units 1 and 2 Docket Nos. 50-413 & 50-4

Blue Ridge Environmental Defense League submittal of contentions in the matter of the renewal of licenses for Duke Energy Corporation (DUKE) McGuire Nuclear Stations 1 and 2 [McGUIRE] and Catawba Nuclear Stations 1 and 2 [CATAWBA].

And

Support for Motion to Suspend Proceeding Filed by NIRS 11/29/01

1. In accordance with 10 CFR 2.1203 (a) and 10 CFR 2.1203 (e), BREDL, hereby submits its formal written contentions to be considered for a hearing by the ASLBP regarding the renewal of licenses for Duke Energy Corporation (DUKE) McGuire Nuclear Stations 1 and 2 [McGUIRE] and Catawba Nuclear Stations 1 and 2 [CATAWBA].
2. BREDL hereby refers to the Atomic Safety and Licensing Board's October 16, 2001 for details (ASLBP No. 02-794-01-LR, with amended order regarding timing of submittals dated November 15, 2001) regarding history of the proceeding, form of contentions, location of filing date and other matters.
3. In accordance with 10C.F.R. § 2.714(b)(2), the following contentions are hereby filed in this original submittal:

Contention Number	Title
One	Radiological impacts of routine operations and accidents
Two	Human Reliability, Workforce Aging and Critical Skills Retention
Three	Steam Generator Aging Management Program
Four	Aging Management of Ice Condensers
Five	Assessment of Reactor Vessel Integrity
Six	
Seven	
Eight	
Nine	
Ten	

4. Support for Motion to Suspend Proceeding Filed by NIRS 11/29/01.

BREDL hereby supports the motion by NIRS to suspend proceedings until the Final Safety

Analysis Reports (FSARs) are available. During the preparation of these contentions

frequently encountered licensee references to the FSARs as validation of various aging management programs. There was no way for BREDL to review these claims.

A. Contention Number and Title

Contention One: Radiological impacts of routine operations and accidents

B. Contention

Offsite radiological impacts must analyzed as a Category 2 issue in Environmental Report.

C. Specific statement of the issue of law or fact to be raised or controverted

10CFR51, Subt. A, Appendix B identifies radiological exposures to the public during refurbishment, radiation exposure to the public from routine operations during the renewal term, collective offsite radiological impacts, and radiological doses during decommissioning as generic Category I NEPA issues for license renewal of nuclear power plants. The licensee applied the GEIS findings in its Environmental Report.

BREDL's contention is that analyses focused exclusively on the risks of cancer from ionizing radiation, and neglected to address information regarding birth defects (congenital anomalies), infant mortality, infant cancer incidence, and neurological effects.

D. Brief explanation of the basis or bases of the contention

This contention is based on the emergence of new information since the Commission rulemaking and the GEIS regarding health impacts from ionizing radiation:

1. A health study by Dr. Joseph Mangano focusing on the effects from operational closure of the Rancho Seco nuclear power plant near Sacramento California found "significant decreases in mortality (all causes and from congenital anomalies) and cancer incidence...for fetuses, infants, and small children" following operational closure.

A subsequent study by co-authored by Dr. Mangano has been accepted for publication by the *Archives of Environmental Health* will be published in the spring, and then would be available for public dissemination and review in this proceeding. This study examined health impacts at eight nuclear power plants "at least 70 miles from other reactors" where operations have ceased. The study found that:

Strontium-90 levels in local milk declined sharply after closing, as did deaths among infants living downwind and within 40 miles of each plant. These reductions occurred in the first two years after closing, were sustained for at least six years, and were especially pronounced for birth defects. Trends in infant deaths in proximate areas not downwind, and 40-80 miles downwind, from closed plants are not different than national patterns. In proximate downwind areas with available data, cancer incidence in children under age five fell significantly after shutdown. Changes in health after nuclear reactor closings may help in better understanding the relationship between low-dose radiation exposure and disease.

2. In a recently published health study by KGA Associates in the Chernobyl Nuclear Power Plant area near Kiev, Ukraine and sponsored by the U.S. Department of Defense, the authors concluded:

“Taken collectively, the results of the data analysis are rather frightening. Initial dosages were from 1 rad to 183 rads. Our research suggests neurocognitive and physical decrements in performance 12 years AFTER a nuclear accident.”

In addition, the introduction to the study contains information about hot particles not previously released outside of Ukraine regarding “radioactive particles from the brown forest.” The occurrence of hot particles following a major radionuclide release must be addressed in the Severe Accident Management Alternatives Analysis. (SAMAs).

3. This new information indicates that the licensee’s analyses of radiological health impacts are deficient and should be addressed as a Category 2 issue:

a. The licensee identified radiation impacts from decommissioning alternatives as a Category I NEPA issue covered within the GEIS, and concluded in its analysis of Decommissioning Impacts in the Catawba ER that the “impacts of decommissioning would not be significantly different if decommissioning occurs after 40 years or after 60 years of operation. Duke has reviewed the environmental impacts of decommissioning of Catawba. These impacts are expected to be comparable to those environmental impacts described in the GEIS for impacts to: land use, water, air quality, ecological resources, human health, social and economic structure, waste management, aesthetics, and cultural resources.”

However, this did not take into account the availability of published data on the positive impacts of operational closure at Rancho Seco or the neurotoxic impacts of acute and chronic radiation exposure.

b. The licensee also identified Radiation exposures to public (license renewal term) and offsite radiological impacts (collective effects) as applicable to both Catawba and McGuire but encompassed in the GEIS as Category I NEPA issues. However, this did not take into account the availability of published data on the positive impacts of operational closure at Rancho Seco.

E. Statement of all appropriate facts and expert opinion to support contention

1. Health effects of ionizing radiation on infants and fetuses.

a. The published article by Dr. Joseph Mangano is submitted as Exhibit 1.

b. The following information submitted by the Radiation and Public Health Project for the Peach Bottom NPP relicensing proceeding is excerpted as follows:

COMMENT ON ENVIRONMENTAL ISSUES

I. INTRODUCTION

The Radiation and Public Health Project (RPHP) is an independent, non-profit research and educational organization. The focus of RPHP's work is to assess the health effects of exposures to radioactive chemicals released into the environment by nuclear weapons tests and nuclear reactor operations. Founded in 1985, RPHP maintains a staff of professionals from the fields of radiation physics, toxicology, epidemiology, and statistics. Its members have published numerous medical journal articles and books on the radiation health issue (see Appendix).

RPHP has documented substantial evidence linking environmental radioactivity with increased cancer risk. Perhaps the strongest evidence is the correlation of levels of radioactive Strontium-90 in baby teeth with risk of childhood cancer in Long Island. The following comment outlines RPHP findings and considers implications for the environmental impact of extending the operating license of the Peach Bottom 2 and 3 reactors.

II. NUCLEAR REACTOR EMISSIONS AND HEALTH RISK

More Reactors Produce More Power. Currently, 103 nuclear power reactors (at 64 sites) are operating in the U.S., producing about 20% of the nation's electricity. (1) About two-thirds of Americans live within 100 miles of at least one nuclear reactor. Operating utilities have permanently closed a total of 22 reactors. In addition, 128 reactors that were proposed by utilities to federal regulators were later canceled before commencing operations. (2)

Startup of new reactors and increased use of existing ones have caused the generation of electricity from reactors to nearly triple (248 million to 727 million gigawatt hours) from 1980 to 1999. (1) Present trends suggest that use of nuclear power reactors may proliferate in the future. The U.S. Nuclear Regulatory Commission (NRC) has received applications to extend the licenses of 43 reactors from the current life span of 40 years to 60 years. In addition, the Nuclear Energy Institute announced a goal of starting 50 new nuclear reactors at its annual meeting in May 2001.

Government Assessment of Health Risks is Deficient. Because radioactivity can damage human health, an accurate assessment of risk to the public is warranted. **However, current regulatory policies do not include any such risk assessment** The NRC has approved the first five applications for reactor license extension, with no consideration of disease rates, including cancer, in persons living closest to reactors.

III. NEED FOR MORE INFORMATION ON HEALTH

Reactor Operations Release Cancer-Causing Chemicals. Nuclear reactors employ fission of uranium atoms to generate electricity. The fission process creates 100 to 200 radioactive chemicals not found in nature, which may damage the immune, genetic, and hormonal systems. These products include strontium, plutonium, iodine, and other carcinogenic isotopes. The only other source of these man-made chemicals is nuclear weapons explosions. Most fission products generated by reactors are contained as radioactive waste, but a fraction is emitted into air and water.

The NRC requires utilities that operate nuclear power plants to report levels of radioactive emissions into the environment each year, along with levels of radioactivity in local air, food, soil, and water. If levels fall below government-defined "permissible limits," the NRC presumes that the public has not been harmed.

Health Studies Are Lacking There has been a dearth of scientific, peer-reviewed studies evaluating disease rates near U.S. nuclear power plants since the first reactor began operations in 1957. Only one national study has been done. In 1990, at the insistence of Senator Edward M. Kennedy, the National Cancer Institute published data on cancer near nuclear plants. While the study concluded that there was no connection

between radioactive emissions and cancer deaths, rates near many reactors rose after reactor startup. (3) Since 1990, no federal agency, including the Environmental Protection Agency and Nuclear Regulatory Commission, has undertaken any studies of disease rates near nuclear plants.

In-Body Measurements Are Lacking The lack of health studies near American nuclear reactors is complemented by a lack of measurements of in-body levels of radioactivity for persons living near nuclear reactors. Government-supported programs to measure Strontium-90 in St. Louis baby teeth (4) and in New York City and San Francisco bones (5) were terminated in 1970 and 1982, respectively. Both measured the effects of bomb test fallout rather than nuclear power reactor emissions.

IV. SR-90 IN BABY TEETH AND CANCER RISK

RPHP Tooth Fairy Project. RPHP is addressing the shortage of information on radiation's health effects by documenting radioactivity levels in the human body and comparing them with cancer and other health patterns.

RPHP researchers are conducting **the first-ever study that measures radioactivity in the bodies of persons living near nuclear power reactors.** In 1996, RPHP launched the Tooth Fairy Project, which uses the same methodology of calculating levels of Strontium-90 (Sr-90) in baby teeth employed in St. Louis during the 1950s and 1960s. The chemical enters baby teeth through the mother's diet during pregnancy and through the mother's bones.

Sr-90 is just a marker for the 100-200 radioactive chemicals that are released in nuclear reactor operations, but it is a critical one. Like calcium, Sr-90 attaches to the bone and teeth when it enters the body, where it remains for many years due to its slow rate of decay (half-life of 28.7 years). It kills and impairs bone cells, and penetrates the bone marrow, in which the white blood cells critical to immune function are formed, making it a risk factor for all cancers. Of all man-made radioactive chemicals, Sr-90 was the one that caused the greatest health concern during the atmospheric bomb test years in the 1950s and 1960s. In 1956, Presidential candidate Adlai Stevenson remarked that Sr-90 was "the most dreadful poison in the world." (6)

To date, RPHP has collected over 3000 baby teeth, mostly from areas near reactors in California, Connecticut, Florida, New Jersey, New York, and Pennsylvania. Strontium-90 concentrations have been measured in nearly half (1463) of these teeth by Radiation Environmental Management Systems Inc., an independent laboratory in Waterloo, Canada.

The average current concentration of Sr-90 is similar to that in St. Louis in 1956, in the midst of the period of atmospheric nuclear weapons testing. Results of the Tooth Fairy Project have been published in three peer-reviewed medical journals. (7-9)

RISK FROM LOW-DOSE RADIOACTIVE NUCLIDES

The often held notion that reactions to chemicals and ionizing radiation follow a linear dose-response curve is not supported by fact. While a reaction may be proportional at high doses that impair or kill, a straight line dose-response is not borne out at low-dose exposures, (14) nor when an insult occurs at the critical periods of fetal development, and during cell division and repair. (15)

Internal exposures to toxic chemicals and radio nuclides below the level that kills a cell is critical: such sub-lethal exposures that alter cellular function or structure and are not repaired become expressed as cancer or functional alteration. The DES daughters and sons are prime examples. Diethylstilbestrol (DES) was administered to pregnant women in the misguided idea that it would protect against fetal loss during pregnancy. Children and now grandchildren were born with anatomic and functional genital abnormalities and developed genital cancers when they reached adulthood. (16) Cells undergoing replication are hundreds of times more susceptible to radiation and magnetic effects. (17) (18)

Internal radiation may involve exposure to nuclides such as plutonium-239 and strontium-90, which stay within a body essentially for life because of long half-lives. It also involves exposure to nuclides with a short half-life such as barium-140, cobalt-57, chromium-51, cesium-134, iodine-131, and others, which release significant amounts of radiation over a period of hours to days.

Many nuclides undergo sequential decay, an ideal condition for sub-lethal damage to promote the induction of genomic instability. (19) Thus, internal decay of such isotopes as plutonium-239 and carbon-14 deliver a biological effect of infinite duration and the potential to induce genetically transmitted defects. (20) In addition, very low levels of radiation exposure demonstrate an enhanced, supra-linear effect due to the release of free radicals, resulting in functional and physiological effects, separate from genetic or mutational alteration. (21) (22)

RADIOACTIVE STRONTIUM-90 (SR-90) IN BABY TEETH

Sr-90 is a reliably measured surrogate to determine radiological fallout because of its stability in the body and a long half-life of 28.7 years. With a half-life of 28 years, Sr-90 is persistent in the environment and in the bodies of humans. The uptake of radioactive Sr-90 follows that of calcium and becomes deposited in bones and teeth. The newborn's calcium and Sr-90 are derived from the mother's dietary intake and from her bone stores during pregnancy. (23) But Sr-90 was understood before the first atomic bomb was detonated when it was proposed by Enrico Fermi to use the bone-seeking isotope to poison the food supply of Germany during World War II. (24)

Measurements of Sr-90 deposited in human bones and teeth began after the onset of above-ground nuclear bomb tests in Nevada and were carried out by various governments, including the U.S. (25) (26) (27) An independent, comprehensive study by the Committee for Nuclear Information measured Sr-90 levels in about 300,000 baby teeth collected from children in St. Louis. (23) (28) Comparing 1949-50 births with those in 1964, Sr-90 levels increased in concentration from 0.20 to 11.03 picocuries per gram of calcium. The risk to health from this contamination and concern for the health of children worldwide led to a ban on above ground nuclear testing by the U.S. and U.S.S.R., a treaty signed by President Kennedy and Premier Khrushchev.

More recent testing followed Chernobyl releases, when the Otto Hug Institute in Germany documented a ten-fold increase in Sr-90 levels in baby teeth for children born in 1987, compared with those born in 1983-85. (29) These elevated levels are comparable to those documented in the St. Louis children at the height of above-ground nuclear bomb testing. In 1990, for unknown reasons, the U.S. Environmental Protection Agency program of reporting monthly levels of barium-140, cesium-137, and iodine-131 in pasteurized milk in 60 U.S. cities was discontinued after 33 years. (30)

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2. The Impacts of Ionizing Radiation on Human Performance.

The DoD sponsored report by KGA Associates is hereby submitted in its entirety

as Exhibit 2. Following is the abstract:

"In an effort to assess the effects of exposure to ionizing radiation on neuropsychological and physical abilities, a longitudinal study in and near Chernobyl, Ukraine was conducted.

In this report are findings from 1995 to 1998. Participants were volunteers who resided in Ukraine during and since the Chernobyl Nuclear Power Plant accident. A translated subset of the Automated Neuropsychological Assessment Metrics battery and the Gamache Physical Abilities Battery were administered to a control and three experimental groups. Controls were healthy volunteers who resided well outside of the exposed area. Eliminators were decontamination and reconstruction workers with known levels of exposure. Forestry and Agricultural workers resided and worked in contaminated areas. Analyses of 1995 - 1998 4-year averaged results indicated the Eliminators were significantly impaired on all measures of neurocognitive and physical performance as compared to controls. Forestry and Agricultural workers were impaired on subsets of the neurocognitive and physical batteries. Significant correlations between levels of radiation dosage and 4-year averaged physical and cognitive performance were observed on 21 of 24 tasks for the combined exposure groups. The results appear to reflect the existence of clinically meaningful neurotoxic effects of both acute and chronic exposure to radio nuclides."

These issues should be evaluated in terms of chronic exposure to low levels of ionizing radiation and acute exposures to high levels of radiation resulting from a catastrophic accident.

F. Summary of Contention One

Sufficient information has been presented to show a genuine dispute on a material issue of law or fact, including references to specific portions of the application that the petitioner disputes and the supporting reasons for each dispute, and/or identification of each asserted failure of the

application to contain information on a relevant matter as required by law, as well as the supporting reasons for the petitioners belief that the application fails to contain relevant information required by law.

A. Contention Number and Title:

Contention Two:

Human Reliability, Workforce Aging and Critical Skills Retention

B. Contention

The license renewal application fails to provide a Human Reliability Assessment (HRA) that analyzes the impacts of workforce aging, critical skills retention and availability, the impacts of advanced technology on human reliability, and the ability of the future workforce to adequately implement aging programs, prevent severe accidents and economic accidents, and to mitigate the effects of accidents.

C. Specific statement of the issue of law or fact to be raised or controverted

10CFR54.4 and 10CFR54.21 require evaluation of safety-related systems within the Integrated Plant Assessment (IPA).

10CFR51.53(c) requires each “applicant for renewal of a license to operate a nuclear power plant under part 54 of this chapter shall submit with its application the number of copies specified in §§51.55 of a separate document entitled "Applicant's Environmental Report – Operating License Renewal Stage...(2) The report must contain a description of the proposed action, including the applicant's plans to modify the facility or its administrative control procedures as described in accordance with §§54.21 of this chapter.”

BREDL disputes the absence of a Human Reliability Assessment in the presence of administrative controls to ensure safety in a high consequence facility.

D. Brief explanation of the basis or bases of the contention

1. Integrated safety management includes human resources as a safety system that should not be separated within an integrated safety analysis. The skills and knowledge necessary for safe operation of a nuclear power plant are as essential, if not more so, than the engineered components and structures within plants. This will remain a fact as long as “administrative safety controls” are in effect to prevent accidents from happening and to mitigate the impacts of accidents.

2. Critical skills in the nuclear power and weapons workforce are in high demand and will continue to remain in high demand. Present trends suggest supply does not meet demand. The nuclear industry is presently characterized by an aging workforce with insufficient recruitment of

replacement personnel. Although efforts are underway to try to reverse this eroding of critical skill availability, the existing trend is towards a less-qualified and less-experienced workforce.

This trend is aggravated by increasing concerns regarding continued enrollment and attendance of brilliant foreign nationals at American institutions of high learning, particularly nuclear physics and engineering programs, could further erode workforce capabilities and critical skills availability. An HRA would identify how this trend could impact safety and the ability to mitigate severe accidents.

3. Human error is the direct or contributing and/or root cause of most nuclear accidents, and, vice-versa, human intervention is necessary to prevent severe accidents or mitigate the impacts. Every safety-related system and non-safety related system are dependent upon human capabilities to successfully insure safe operation of the plant; and the management of component and structural aging is equally dependent upon human resources. BREDL argues that workforce capabilities and critical skills availability are the primary limiting factor in managing Catawba and McGuire Nuclear Power Plants. Identifying, listing, and describing essential aging management programs are meaningless without the presence of a thorough HRA

4. Therefore an HRA must be conducted as part of the aging management safety analyses for the following reasons:

a. The scope of the proceeding, according to 10CFR51.53(c)(3)(ii)(L), includes "consideration of alternatives to mitigate severe accidents." In its attempt to comply with this rule the licensee conducted "Severe Accident Mitigation Alternatives Analysis (SAMAs)" (Attachment H of the Environmental Review). In both the Catawba and McGuire SAMAs human reliability is frequently cited as an integral part of the severe accident mitigation plans:

i. Ongoing initiatives at Catawba include a "Maintenance Rule Program" that is "an administrative program to ensure that structures, systems, and components important to safety are available and capable of reliably performing their intended safety function." (Page 6 of Catawba SAMA).

ii. The Catawba Severe Accident Management Guideline (SAMG) Program "includes diagnostic tools and severe accident management guideline documents for developing strategies during an event to arrest core damage progression and mitigate fission product releases in the event of a severe accident." The fact that this program is entirely dependent upon human reliability is illustrated by the statement that "this SAMG program achieves an incremental risk reduction capability *without reliance on additional hardware and resources.*" (emphasis added).

(Page 6, Catawba SAMA).

iii. Table 2.1 of the Catawba SAMA identifies "procedure changes," "PRA Based Simulator Training," "Improve Plant Personnel's Awareness of SSS importance," "Administrative controls on SSS Unavailability," and "Procedure Enhancements" as risk reduction measures already implemented at Catawba. Five of the nine alternatives implemented are almost entirely dependent upon human reliability.

b. The scope of the proceeding, according to 10CFR54.4, includes safety-related nuclear power plant systems that ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate off-site radiation exposures that exceed regulatory limits.

All of these systems are dependent upon the ability of operators (10CFR55) to perform as expected and/or the reliability of personnel to properly test and monitor components and structures. For example:

i. The Ice Condenser system is an engineered safety feature but aging management is conducted entirely through visual inspections of ice baskets and ice condenser engineering inspectionsbasket

c. The scope of the proceeding, according to 10CFR54.4(a)(2), includes non-safety related systems “whose failure could prevent satisfactory accomplishment” of these functions and capabilities. Prevention of failure of these functions and capabilities is directly and indirectly dependent upon human reliability in the monitoring and testing of components.

d. The scope includes includes systems “relied on in safety analysis or plant evaluations to “perform a function that demonstrates compliance with the Commission’s regulations for fire protection, environmental qualification,etc. Examples here include fire brigades and environmental technicians who must follow rigorous quality assurance programs. .

e. 10CFR 54.21(a)(1) requires an integrated plant assessment (IPA) for all systems defined as being within the scope of license renewal. The IPA requires identification and listing of dozens of components and structures to be subjected to an aging management review and that are not subject to replacement based on a qualified life or specified time period.

f. 10CFR54.21(a)(3) requires that the “effects of aging will be adequately *managed* so that the intended function(s) will be maintained consistent with the CLB [Current Licensing Basis] for the period of the extended operation.”

Nearly every component and structure identified as subject to aging management depends upon the reliability of humans to adequately test, monitor, and make professional judgements.

E. Statement of all appropriate facts and expert opinion to support contention

The need for a human reliability assessment to determine if Catawba and McGuire NPPs can safely operate an additional 20 years is supported by an abundance of expert documentation supporting the premise that human error is prevalent as a causal factor in accidents and as a factor exacerbating the impacts of accidents. The erosion of human performance and reliability capabilities in the nuclear industry makes this need even greater.

1. Human error as a primary causal factor in nuclear reactor accidents has been recognized for decades. For example, avoiding human error prevailed throughout the forty-two conclusions and recommendations on nuclear power accidents cited by Thompson et al (1964). Experts on safety in high-consequence industrial systems at Sandia National Laboratory have stated:

“Analysis of major industrial accidents such as Three Mile Island, Chernobyl, and Bhopal have revealed that these incidents were not attributable to a single event or direct cause, but were the result of multiple factors that combined to create a condition ripe for an accident. In each case, human error was a critical factor contributing to the accident. Consequently, many authors have emphasized the need for greater appreciation of systemic factors and in particular, human activities.”

BREDL hereby submits Exhibit 3, Chapters 7 and 9 as an expert analysis of these facts.

2. The prevalence of human error as a causal factor in accidents and as a factor exacerbating the impacts of accidents is well documented:

a. One result of NRC efforts in the 1990's to "address limitations identified in current HRA [Human Reliability Assessment] approaches" was the development by Sandia National Laboratory for a new HRA called "A Technique for Human Events Analysis," called ATHEANA. Bley et al (1999) provided background on the need for ATHEANA:

"The record of significant incidents in nuclear power plant operations shows a substantially different picture of human performance than that represented by human failure events modeled in PRAs. The latter typically represent failures to perform required procedure steps. In contrast, human performance problems identified in real operational events often involve operators performing actions that are not required for accident response and, in fact, worsen the plant's condition (i.e., errors of commission). Further, accounts of the role of operators in serious accidents, such as those that occurred at Chernobyl 4 [2,3] and Three Mile Island 2 (TMI-2) [4], frequently leave the impression that the operator's actions were illogical and incredible. Consequently, the lessons learned from such events often are perceived as being very plant-specific or event-specific."

"However, there is increasing evidence that there may be a persistent and generic human performance problem that was revealed by TMI-2 (and Chernobyl) but not 'fixed: errors of

commission involving the intentional operator bypass of engineered safety features (ESF)."

b. Brookhaven National Laboratory is another recipient of NRC funding for analysis of the human factor. According to one report regarding control room modernization:

Changes in automation can have a major effect on the operator's role, defined as the integration of the responsibilities that the operator performs in fulfilling the mission of plant systems and functions. Since automation has been predominately technology driven, changes in automation often fail to result in a coherent role for operators."

"Automated systems have generally been designed with inadequate communication facilities which make them less observable and may impair the operator's ability to track their progress and understand their actions. In one case this problem led to operators defeating or otherwise circumventing a properly automated system because they believed it was malfunctioning."

c. Sandia also conducted an assessments that determined there is a higher likelihood of human error during NPP low power and shutdown mode:

"Human error with a reduced water inventory is a key contributor to core damage frequency (CDF)" during LPSD, and a limitation in assessing probabilistic risk in these situations is "human reliability during transitions and other shutdown activities."

3. The ongoing erosion of America's nuclear workforce capabilities and availability of critical skills is equally well documented. Three examples are provided to support this contention:

a. On February 28, 2001 NRC Chairman Richard Meserve described the "Human Capital" situation to the Vice President of the United States as follows:

"The NRC's ability to fulfill its mission is critically dependent upon the expertise of its staff

and contractors. As with many Federal agencies, it is becoming increasingly difficult for NRC to hire personnel with the knowledge, skills, and abilities to conduct the safety reviews, licensing, and oversight actions that are essential to our safety mission. In some important offices, nearly 25

percent of the staff are eligible to retire today. Moreover, the number of individuals with the technical skills critical to the achievement of our safety mission is rapidly declining in our Nation and our educational system is not replacing them.”

Chairman Meserve went on to offer potential remedies such as modifying conflict-of-interest provisions regarding Department of Energy contractors, increasing NRC staff salaries, hiring consultants from the pool of NRC retirees at full pay, and funding University training programs:

“A recent blue-ribbon engineering panel reporting to the Department of Energy has identified a significant decline in the number of nuclear-related academic programs. Moreover, many universities are contemplating the shut-down of research reactors, limiting the opportunities for students and researchers. Congress could help to reverse this trend by funding academic fellowships to attract engineering students, by sustaining important research facilities, and by enabling the NRC to establish a training program to address shortages of individuals with critical safety skills.”

b. The Department of Energy’s (DOE) Argonne National Laboratory (ANL) is a leading scientific R&D resource of the nuclear power industry. Operated by the University of Chicago, it serves as the lead laboratory, in collaboration with Idaho National Engineering and Environmental Laboratory, for nuclear reactor technology for DOE’s Office of Nuclear Energy, Science, and Technology.

Argonne National Laboratory believes that nuclear energy “must contribute increasingly to the world’s energy supply if major environmental goals are to be met.” The caveat behind this perception is that there is “the need for a major U.S. initiative in nuclear technology R&D.” According to the most recent ANL Institutional Plan, “the U.S. nuclear technology infrastructure, which once led the world, has been eroded seriously and could be lost entirely if present trends continue.”

Argonne’s nuclear technology program goals and objectives include “maintaining a set of technical capabilities in nuclear science and technology—including both expertise and infrastructure—sufficiently broad and deep to address a full range of national needs...maintain a complete core competency in nuclear technologies so that a nuclear option remains available to the United States for the long term...conduct educational and training activities for U.S. and international participants, to improve knowledge of nuclear technology worldwide and to ensure a high level of capability in the staffs of safety oversight and regulatory agencies.”

c. **The Chiles Commission** was established to review the nuclear weapons workforce and determine needs and priorities. The Commission concluded in its 1999 report that, “large numbers of workers are reaching retirement and a new generation of workers must be hired and trained in order to preserve essential skills.”

3. The licensee’s operational history indicates that most incidents, occurrences, and accidents have human error as a direct or contributing cause. Three events out of hundreds at Catawba and McGuire are cited as examples:

a. Significance: TBD Feb 16, 2001 Identified By: NRC Item Type: AV Apparent Violation **Failure to Promptly Identify and Correct the Unit 1 Residual Heat Removal System Water Hammer Condition** An apparent violation of 10 CFR 50, Appendix B, Criterion XVI was identified for the failure to identify a root cause and establish effective corrective actions to prevent repetitive water hammer events in the Unit 1 residual heat removal (ND) system which have caused the repeated failure of snubbers on supports 1-R-ND-0226 and 1-R-ND-0596. (Section 40A2.b.(2).2) Inspection Report# : [2001003\(pdf\)](#)

b. Significance: G Mar 30 2001 Identified By: NRC Item Type: FIN Finding **Failed to Demonstrate Performance of the Station Drinking Water System as Backup Cooling Water to the Unit 1 and 2 A Train Charging Pumps** The licensee failed to demonstrate that the performance or condition of the station drinking water system, a risk important system that provides backup cooling water to the Unit 1 and 2 A train charging pump motors and bearing oil coolers, was being effectively controlled through the performance of appropriate preventive maintenance (including surveillance activities). This resulted in a failure to recognize and correct a degraded system pressure condition, until it was identified by the inspectors. The degraded pressure condition was determined to be of very low safety significance because an analysis performed by the licensee demonstrated that the backup function to cool the charging pumps and motors would have been provided at the degraded pressure (Section 1R12.2). Inspection Report# : [2000006\(pdf\)](#)

c. Significance: G Jun 24, 2000 Identified By: Licensee Item Type: NCV NonCited Violation **Failure to Provide Adequate Procedures for Performing Maintenance on Safety-Related Sump Pump Level Switches** Residual heat removal and containment spray pump room sump level alarm function was lost for several months up to February 2000 due to inadequate maintenance procedures associated with sump level switch calibrations. This issue was characterized as a non-cited violation of Technical Specification 5.4.1 and was determined to be of very low safety significance due to the availability of other emergency core cooling system leak detection methods (Section 4OA3.2). Inspection Report# : [2000003\(pdf\)](#)

4. Recently published research points to a potential link between chronic exposure to radiation and a reduction in neurocognitive abilities. See Exhibit 2.

F. Summary.

BREDL has presented, as required by 10CFR2.714 "sufficient information to show a genuine dispute on a material issue of law or fact." BREDL has not cited "references to specific portions of the application that the petitioner disputes" because the reason for the dispute is the "failure of the application to contain information on a relevant matter as required by law." The "supporting reasons for the petitioner's belief that the application fails to contain relevant information required by law" is more than adequately defined and presented to be accepted as a contention in a hearing.

A. Contention Number and Title:

Contention Number Three: Steam Generator Aging Management Program

B. Contention

The aging management program for steam generators and associated components such as steam generator tubes is insufficient and incomplete, and does not assure safe operations that prevent design basis and severe catastrophic accidents. In addition, the DBA frequency for steam generator tube rupture is grossly underestimated

C. Specific statement of the issue of law or fact to be raised or controverted

The licensee's aging management program for Steam Generators is incomplete (10CFR54.13) and does not assure safe operation (10CFR54.21) . Licensee program to ensure the prevention of steam generator degradation is insufficient both in practice and in the renewal application.

Futhermore, the licensee's estimates of a Design Basis Accident involving steam generator tube rupture is

D. Brief explanation of the basis or bases of the contention;

1. Steam generators are large components which convert water into steam from the heat produced in nuclear reactor cores; and fall within the category of Reactor Vessel, Internals, and Reactor Coolant System in NUREG 1800 (Chapter 3.1) and NUREG 1801 (Part 4). Each steam generator consists of numerous sub-components.
2. The licensee identified (Table 3.1.1, Pages 3.1-21 to 3.1-24) twenty-two (sub)component types in its aging management review results and wrote that each component functions to "maintain mechanical pressure boundary integrity" (Page 3.1.26). Loss of integrity could lead to accidents that result in unacceptable radiation exposure to the off-site public, economic losses due to shutdown, and loss of electrical supply to the region.
3. There are four steam generators in each licensee reactor, and their sub-components are subject to aging management analysis in accordance with 10CFR 54.4, 10CFR54.21(a)(1), and 10CFR54.21(a)(3). The aging management analysis must adhere to 10CFR**54.13**(a) by providing complete and accurate information.
4. Each steam generator consists of thousands of steam generator tubes considered highly vulnerable to corrosion and deformation. These tubes must be closely monitored and problems corrected to avoid a rupture of one or more tubes. Two of the fifteen known steam generator tube rupture occurrences in U.S. NPPs occurred at McGuire 1.

The licensee's overall program for managing aging of steam generator tubes is encompassed within three aging management programs:

- a. The scope of the *Steam Generator Surveillance Program* includes "all steam generator tubes (including plugs and sleeves) in each steam generator and internal support structures." Because the description of this program in Appendix B, Part B.3.31-3 is simplistic, overly brief, and contains numerous discrepancies and omissions (see Part E), compliance with 10CFR54.13 and subsequently 10CFR54.21(a) is being disputed by BREDL.
 - b. The *Alloy 600 Aging Management Review* is a proposed program to rank susceptibility to primary water stress corrosion cracking, ensure that nickel-based alloy locations are adequately inspected by the *Inservice Inspection Plan* or other programs. However, the licensee states that the review will be complete by the end of the initial 40-year license period and as such does not provide the assurance required by 10CFR54.21.
 - c. The *Chemistry Control Program* is for managing "loss of material and/or cracking of components exposed to borated water, closed cooling water, fuel oil, and treated oil environments" and is described as a mitigation program. The licensee failed to identify past problems with chemistry control prevalent throughout the industry and the efforts required to prevent recurrence. .
5. The licensee has in practice sought and obtained "relief" from meeting regulatory requirements and industry standards for pre-service inspection of numerous steam generator subcomponents (See E). This practice resulted in the failure to develop a baseline for monitoring aging of these parts. The licensee failed to identify these issues, a violation of 10CFR54.17.

E. Statement of all appropriate facts and expert opinion to support contention

1. Background on Steam Generators and Steam Generator Tubes

a. Steam generators are large components which convert water into steam from the heat produced in nuclear reactor cores.

According to the NRC:

“These devices can measure up to 70 feet in height and weigh as much as 800 tons. Inside the steam generators, hot radioactive water is pumped through thousands of feet of tubing—each steam generator can contain anywhere from 3,000 to 16,000 tubes, each about three-quarters of an inch in diameter—under high pressure to prevent it from boiling. That water flowing through the inside of the tubes then heats non-radioactive water on the outside of the tubes. This produces steam that turns the blades of turbines to make electricity. The steam is subsequently condensed into water and returned to the steam generator to be heated once again.”

These tubes have an important safety role because they constitute one of the primary barriers between the radioactive and non-radioactive sides of the plant. For this reason, the integrity of the tubing is essential in minimizing the leakage of water from the two ‘sides’ of the plant. There is the potential that if a tube bursts while a plant is operating, radioactivity from the primary coolant system—the system that pumps water through the reactor core—could escape directly to the atmosphere in the form of steam. However, such a rupture has not occurred since March 14, 1993, when a tube burst at Palo Verde 2 in Arizona.”

Subsequent to publishing this information a tube rupture at Indian Point 2 caused that reactor to shut down and initiated an extensive investigation. *No mention of this event or any other steam generator tube rupture exists in the application.*

b. “Steam generator tubes have proved to be especially susceptible to corrosion,” and the primary problem today in PWRs is “stress corrosion cracking of the tube.” Stress corrosion cracking is difficult to predict and detect and “there is a need for better methods to both detect and to size” cracks produced by this mechanism..” as well as a need for better technology “to predict whether cracks will grow to unacceptable dimensions during future cycles of plant operations.”

The licensee made no mention of these difficulties and associated uncertainties.

c. **Stress corrosion** cracking is the “principal degradation model leading to tube plugging in the U.S. and worldwide.”

d. In 1995 the NRC wrote that:

“Both the NRC and the industry have identified the reliable detection and sizing of circumferential cracks in steam generator tubes as a technical issue of concern.”

Steam generator tube ruptures represent a “failure of one of the principal fission product boundaries and present a pathway for primary system activity release to the environment...”and

“Inspection practices should furnish assurance that steam generator tube degradation will be reliably detected” so that the potential for rupture is maintained at an acceptably low level.

e. The NRC wrote in its 1996 version of the GALL that steam generator tubes are susceptible to additional aging mechanisms such as attrition, wear, ...

f. Dr. Joram Hopenfeld, a recently retired NRC staffer, began writing "Differing Professional Opinions" regarding steam generator tubes in December of 1991, and subsequently issued DPOs (and addendums) in March 1992, September 1992, September 1998, April 2000, and April 2001. The series of documents available on ADAMS that are introduced as evidence are referenced as Attachment 1-1. It is useful to provide excerpts from the most recent DPO:

" It is now almost 10 years since I originally raised several serious safety issues concerning the NRC practice of permitting excessively degraded steam generators tubes to remain in service during plant operations. This practice while benefitting the nuclear industry, has had a serious negative potential impact on public safety. After many and continuing attempts by NRC management to ignore these DPO issues, they remain unresolved. As demonstrated by the Indian Point 2 (IP2) accident, excessively degraded tubes continue to threaten public safety.

"During the past ten years, the NRC has expended inordinate resources on my DPO safety issues and has publically claimed that they have been properly addressed. The new ACRS

findings, NUREG-1750, clearly indicate that the staff contentions were flawed and misleading,

and that the allocated resources have been wasted."

"The ACRS had concluded last November that the staff position on the issues raised by the DPO is indefensible. Accordingly, the Executive Director for Operations, EDO, was requested to resolve these issues and report the outcome to the ACRS. Instead, the EDO merely instructed the divisions of RES and NRR to draft a new action plan and closed the DPO. Closing the DPO without specifying how it will be resolved is a clear violation of Management Directive (MD) 10.159(C). The EDO's latest action compounds previous violations of MD 10.159, making a sham of the entire process of encouraging employees to raise safety concerns. The NTEU union filed a grievance on my behalf to keep the DPO open until it is resolved.

The transcripts from the ACRS hearings and the following quotations from NUREG-1750 clearly demonstrate the poor state of knowledge at the NRC regarding steam generator safety issues.

1. ' the **staff has not** adopted a **technically defensible** position on the choice of iodine spiking factor to be used on the analysis of design for compliance with requirements of 10 CFR Part 100 or General Design Criteria 19.'

2. 'The **staff need to develop a defensible analysis** of the uncertainties in its

risk **assessment**, including uncertainties in its assessments of human error probabilities" (during design basis accidents.)

3. 'The **staff has not** developed **persuasive arguments** to show that steam generator tubes will **remain intact** under the conditions of **risk-important accidents** which the reactor coolant remain pressurized.'

4. 'The Ad-Hoc Subcommittee found that the **staff did not have a technically**

defensible understanding of these processes to assess adequately the potential for **progression of damage of steam generator tubes.**'

5. 'The Ad-Hoc Subcommittee did not feel that the staff has **developed an adequate understanding** of how movements of the tube support plate during an event could damage the tubes.'

6. 'The Subcommittee did not attempt to reach conclusions concerning occasions when staff granted exemptions to these criteria (1 & 2 V) except to note that these exemptions **should have been accompanied by more complete risk analysis.**'

7. 'The databases for 7/8" tubes **need to be greatly improved to be useful.**'

8. 'This issue (tube shearing during depressurization), at the **current level of understanding cannot be used** to judge the adequacy of the alternative repair

criteria described in **GL-95-05.**'

9. 'the issue of the possible evolution of severe accidents to involve **gross failure of steam generator tubes** and bypass of the containment **is not yet resolved.**'

Steam generators were originally sold to the utilities with the understanding that they would operate acceptably within design parameters for the lifetime of the plant. Because of inadequate and improper material selection, this expectation has never been fulfilled and some steam generators have been replaced after only a few years of service. U.S. plants alone have experienced 11 steam generator tube failure accidents, which can be traced to poor design and lack of meaningful NRC oversight. Additional, and possibly catastrophic, steam generator tube failure accidents can be expected in the future since many nuclear power plants will be relicensed for another 20 years.

The nuclear industry, however, has done essentially nothing to seriously address the safety issue. Licensees have demonstrated that their main goal is to continue using severely degraded steam generators as long as they want to do so. The NRC has been unwilling to insist that safety take priority over economics.

The NRC practices regarding steam generators contributed significantly to the recent IP2 accident. Fortunately this accident did not have significant safety consequences, it was, however, a serious precursor to the type of accidents which are described by the DPO. The NRC takes the unacceptable position that if the DPO accidents have not occurred they will not occur in the future."

g. A few weeks after Dr. Hopenfeld's last DPO, David Lochbaum of the Union of Concerned Scientists testified before the Clean Air, Wetlands, Private Property, and Nuclear Safety Subcommittee of the United States Senate Committee on Environment and Public Works. In regard to Steam Generator Tube ruptures, Lochbaum stated:

"The NRC's Advisory Committee on Reactor Safeguards (ACRS) issued a report in February 2001. The ACRS substantiated many of Dr. Hopenfeld's concerns. For example, the ACRS concluded:

'The techniques [used to look for cracked steam generator tubes] are not nearly so reliable for determining the depth of a crack, and in particular, whether a crack penetrates through 40% of the tube wall thickness.' [NRC's regulations do not allow a nuclear plant to start up with any steam generator tube cracked more than 40 percent of its wall thickness, but the methods used to inspect the tubes for cracks cannot reliably determine the depth of cracks.]

'The NRC staff acknowledged that there would be some possibility that cracks of objectionable depth might be overlooked and left in the steam generator for an additional operating cycle.' [Exactly what actually happened at Indian Point 2 to cause last year's accident.]

'Both the [NRC] staff and the author of the DPO [Dr. Hopenfeld] agree that the alternative repair criteria' [used by the NRC staff to allow nuclear plants to continue operating with steam generator tubes known to be cracked] 'increase the probability of larger primary-to-secondary flows during the MSLB [main steam line break] and SGTR [steam generator tube rupture] accidents.'

'The [ACRS] also finds that this contention of the DPO [namely, that an accident at a nuclear plant with cracked steam generator tubes could cause those tubes to completely break] has merit and deserves investigation.'

'This seems to be a plausible contention [that an accident at a nuclear plant with cracked steam generator tubes could widen the cracks and result in larger leakage], and the staff has not produced analyses or test results to refute it.'

'The [ACRS] concluded that the issue of the possible evolution of severe

accident to involve gross failure of steam generator tubes and bypass of the containment is not yet resolved ... [and] that the issue needs consideration regardless of the criteria adopted for the repair and replacement of steam generator tubes.'

'Data available to the [ACRS] suggest that the constant probability of detection [of cracked steam generator tubes] adopted by the NRC staff is nonconservative for flaws producing voltage signals less than about 0.7 volts.' [In other words, the NRC staff assumes that methods used to find cracked tubes are much better than the data shows them to be.]

'The [ACRS] was unable to identify defensible technical bases for the [NRC] staff decisions to not consider the correlation of the iodine spiking factor with initial iodine concentration [when evaluating the potential offsite radiation dose consequences from accidents involving cracked steam generator tubes].'

'The [ACRS] found that the [NRC] staff did not have a technically defensible understanding of these processes to assess adequately the potential for procession of damage to steam generator tubes.' [In other words, the NRC staff has no sound basis for arguing that one broken tube will not cascade and cause the failures of other tubes.]

'The [NRC] staff has not developed persuasive arguments to show that steam generator tubes will remain intact under conditions of risk-important accidents in which the reactor coolant system remains pressurized. The current analyses dealing with loop seals in the coolant system are not yet adequate risk assessments.'

'In developing assessments of risk concerning these design basis accidents, the [NRC] staff must consider the probabilities of multiple tube ruptures until

adequate technical arguments have been developed to show damage progression is improbable.’[In other words, the risk studies to date, which only consider failure of a single tube, may understate the true risk and therefore should not be relied upon.]

The concerns raised by Dr. Hopenfeld are extremely important safety issues. As the ACRS stated:

‘Steam generators constitute more than 50% of the surface area of the primary pressure boundary in a pressurized water reactor.’

‘Unlike other parts of the reactor pressure boundary, the barrier to fission product release provided by the steam generator tubes is not reinforced by the reactor containment as an additional barrier.’

‘Leakage of primary coolant through openings in the steam generator tubes could deplete the inventory of water available for the long-term cooling of the core in the event of an accident.’”

2. The licensee’s steam generator aging management program actually involves five major aging management programs:

- Chemistry Control Program
- Inservice Inspection Plan
- Alloy 600 Aging Management Review
- Fluid Leak Management Program
- Steam generator surveillance program

Deficiencies exist in at least three of the program descriptions in the application as they pertain to steam generators, and these deficiencies are primarily errors of omission:

a. The application states that “the purpose of the *Steam Generator Surveillance Program* is to provide comprehensive examinations of the steam generator tubes to ensure that degradation is identified and corrective actions are taken prior to exceeding allowable limits. The *Steam Generator Surveillance Program* is a condition monitoring program that is credited with managing loss of material and cracking of Alloy 600 and 690 steam generator tubes...The scope of the *Steam Generator Surveillance Program* includes all steam generator tubes (including plugs and sleeves) in each steam generator and internal support structures.” The program is described as “equivalent,” not equal, to the program described in NUREG 1723.

Generic issues that were not identified within Table 3.1.1 of the Technical Review, Appendix B, Part B3.31 “Steam Generator Surveillance Program,” or the USFAR include:

- i. an aging management program applicable to either the existing steam generator or the replacement steam generator in Catawba 2;
 - ii. aging of steam generator tube materials due to “deformation due to corrosion at tube support plate intersections,” which was identified by the NRC in the SRP.
 - iii. the various cracking initiation mechanisms in steam generator tubes, i.e. stress corrosion cracking within the broader category of “cracking;”
 - iv. Further evaluation of Alloy 600 steam generator tubes, repair sleeves and plugs; steam generator shell assembly, and other steam generator components as recommended by the NRC in Table 3.1-1 of the SRP.
- b. The *Alloy 600 Aging Management Review* is a proposed program to rank susceptibility to primary water stress corrosion cracking, ensure that nickel-based alloy locations are adequately inspected by the *Inservice Inspection Plan* or other programs. However, the licensee states that the review will be complete “by” the end of the initial 40-year license period and as such does not provide the assurance required by 10CFR54.21 to identify its aging management program within the license application. A “review” is only a part of a “program.”
- c. The *Chemistry Control Program* is for managing “loss of material and/or cracking of components exposed to borated water, closed cooling water, fuel oil, and treated oil environments” and is described as a mitigation program. The licensee failed to identify past problems with chemistry control prevalent throughout the industry and the efforts required to prevent recurrence.

3. Deficiencies in the licensee’s operating experience warrant further scrutiny of the steam generator aging program:

The licensee provides a minimal background on operating experience related to steam generator issues, only citing the year of steam generator replacement and some observations on aging of tubes. Incidents not included in the discussions of operational history include:

- a. Two of the fifteen known steam generator tube rupture occurrences in U.S. NPPs occurred at McGuire 1.
- b. In June 1997 McGuire 2 was shut down “because of an increasing primary-to-secondary leak.”
- c. Steam generators were replaced after less than 20 years of operation in 3 of the 4 reactors, yet no reason was provided for this major refurbishment. The abbreviated life span of the first steam generators indicates an inability to implement a strong and durable aging management program.
- d. When the licensee replaced the Catawba 1, McGuire 1, and its steam generators it failed to conduct pre-service examinations on numerous subcomponents until after it installed the new generators. According to the NRC, “the preservice examinations were not performed during manufacturing or prior to installation of the SGs. Instead, the licensee performed onsite preservice examination of the SGs after installation.”

As a result the preservice examinations could not achieve the 100% examination volume required by industry standards. On June 4, 2000 the licensee requested relief from preservice inspection requirements of steam generators that were then 2-4 years old. Nearly a year passed before the

NRC acquiesced and granted the relief based upon the rationale that following codes was not economically feasible.

F. Summary.

BREDL has shown sufficient information to show a genuine dispute on a material issue of law or fact, including references to specific portions of the application that the petitioner disputes and the supporting reasons for each dispute, and/or identification of each asserted failure of the application to contain information on a relevant matter as required by law, as well as the supporting reasons for the petitioners belief that the application fails to contain relevant information required by law.

Contention Number and Title:

Contention Four: Aging Management of Ice Condensers

B. Contention:

The aging management programs associated with the Catawba and McGuire Ice Condenser systems are insufficient to assure safe operations and prevent design-basis and severe accidents.

C. Specific statement of the issue of law or fact to be raised or controverted

10CFR51.53(c)(3)(ii)(L) requires “consideration of alternatives to mitigate severe accidents,” which the licensee submitted as part of its Environmental Reports (ER).

10CFR51.53(c)(3) requires the ER to “contain a consideration of alternatives for reducing adverse impacts, as required by §§51.45(c), for all Category 2 license renewal issues in Appendix B to subpart A of this part.”

Aging management and time-limited aging management programs of numerous Ice Condenser systems and components are required to comply with 10CFR 54.4, 10CFR54.21(a)(1), and 10CFR54.21(a)(3) in order to insure safe operations and prevent design basis and severe accidents.

Brief explanation of the basis or bases of the contention;

1. Catawba and McGuire NPPs constitute four of the ten existing Pressurized Water Reactors with ice condenser containment systems. These ice-condenser containment systems are the most vulnerable among all U.S. NPPs to loss of containment accidents.
2. The licensee's aging management programs for ice condenser systems and components does not comply with 10CFR54.21(a) because it is incomplete and inaccurate (10CFR54.17) and fails to provide reasonable assurance that aging management will allow these systems to function as designed when necessary and prevent a catastrophic release of fission products to our environment.

3. The licensee's SAMA analysis is incomplete because it fails to incorporate new and extensive information regarding ice condenser vulnerabilities. In its "analysis of potential containment-related SAMAs," the licensee failed to even identify potentially dominant failure modes for a severe accident.

4. The licensee's operational experience shows a history of deficiencies and the application was incomplete and inaccurate about the extent and depth of deficiencies in the operational record.

D. Statement of all appropriate facts and expert opinion to support contention

1. The experts joke about ice condenser containment:

"I just wonder if ICE condensers had some peculiarity about them that I didn't know about other than vulnerable containment.

(Laughter)

Mr. Kress: You were reading my mind.

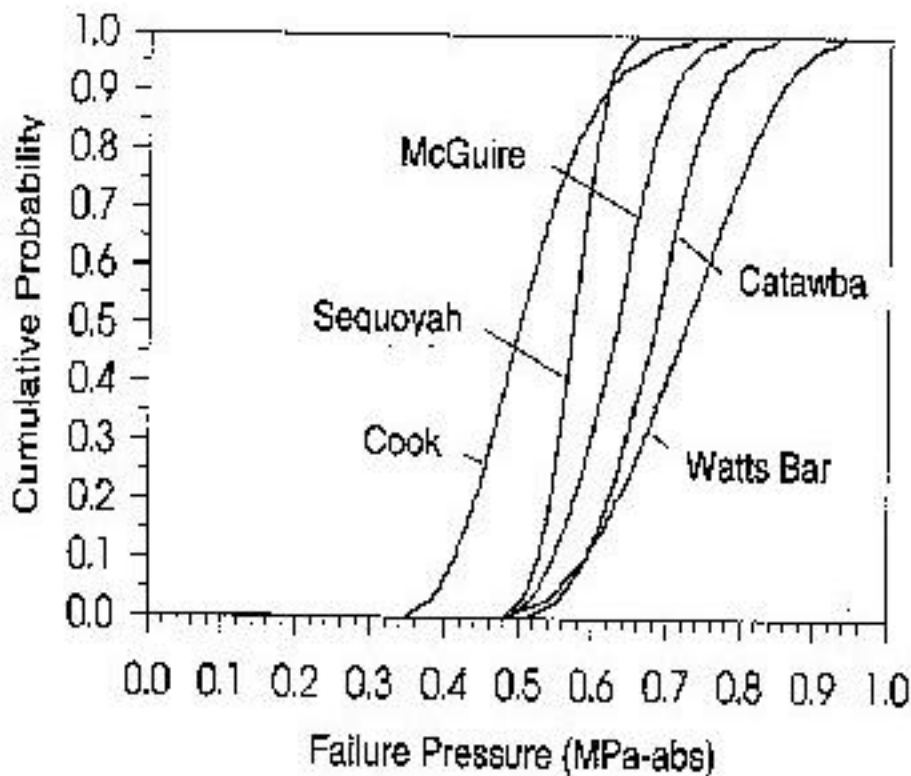
Mr. Powers: I saw you grinning over there."

Official Transcript of dialogue between Advisory Committee on Reactor Safeguards' members Mr. Dana M. Powers and Thomas S. Kress at the ACRS February 2, 2001 Meeting, in reference to the proposed use of Plutonium/MOX fuel in Catawba and McGuire NPPs.

2. Assessment of the DCH Issue for Plants with Ice Condenser Containments (NUREG/CR-6427. SAND99-2253) is a voluminous NRC sponsored study by Sandia National Laboratory published in April 2000, and therefore contains information that is considered new and relevant. While this report is far too in depth and voluminous to cite at length, two excerpts are provided:

We note that the ice condenser plants are substantially less robust than other Westinghouse plants with large dry or subatmospheric containments. Table 6.1 shows that the mean of the containment failure pressure for all ice condenser plants is 62.8 psig at a failure frequency of 10%. The comparable value for all Westinghouse plants with large dry or subatmospheric containments is 113.1 psig. Ice condenser containments can afford to be less robust because of their reliance on ice beds as a pressure suppression feature for design basis accidents.

a. From: Chapter 6. Quantification of Containment Fragility. Page 102.



b. Figure 6.1 showing Fragility Curves for all Westinghouse Ice Condenser NPPs:

A plant-specific evaluation of the CET showed that all plants, except McGuire, had an early failure probability (given core damage) within the range of 0.35% to 5.8% for full power internal events. These integral estimates of early containment failure are qualitatively consistent with published IPE results for these plants. The early containment failure probability, as computed here, was 13.9% for McGuire. This higher containment failure probability for McGuire is dominated by the relatively high SBO frequency and the relatively weak containment for McGuire. The IPE assessments of early containment failure at McGuire (2%) are significantly lower than our assessments; however, we have not investigated the reasons for the difference.

c. From Chapter 8.0 Summary and Recommendation, Page 124.

d. The licensee failed to even reference this landmark report in its Severe Accident Mitigation Alternative Analyses (Section 8, References), and searches of the 626-page application and Appendix B of the application for "Pilch" and "NUREG/CR-6427" yielded no information.

e. BREDL requests that this document be introduced as an Exhibit by the Panel as a central point of dispute in this proceeding. However, the size of the document (348 pages, 24.5 MB in ADAMS TIFF File format) makes it prohibitive for BREDL to copy and distribute this. Instead it is being placed on a CD-ROM along with other references as part of BREDL's submission.

2. The issue of ice condensers raised by Pilch et al was best summarized in a November 2000 report by Dr. Edwin S. Lyman of the Nuclear Control Institute, excerpted in part here:

“Vulnerabilities of Ice Condenser Containments

Nuclear power plants in the U.S. are required to have robust reactor containment buildings. The main purpose of these structures is to prevent the release of large quantities of radioactive materials in the event of a reactor core meltdown. In the aftermath of the 1986 Chernobyl accident in the former Soviet Union, the nuclear industry maintained that such a severe accident could never happen in the U.S. because U.S. reactors, unlike the Chernobyl reactor, were equipped with containments.

However, not all containments offer equal protection. Most pressurized-water reactors (PWRs) in the U.S. have "large dry" containments, which are typically massive concrete structures with walls several feet thick. Catawba and McGuire, on the other hand, are among a handful of PWRs worldwide with "ice condenser" containments. These are typically thin steel shells that have only half the volume and failure pressure of large dry containments. To compensate for the reduced strength of their containment buildings, ice condenser plants are equipped with "ice beds." These consist of baskets filled with blocks of ice that are supposed to cool and condense steam flowing past them during a core-melt accident, reducing the threat that the containment will become overpressurized and rupture from the rapid generation of steam.

However, even if the ice condensers do work as they are supposed to (which in itself is a questionable proposition), containment failure can still occur as a result of the combustion of hydrogen gas, which would be generated in large quantities during severe accidents when the metal cladding on fuel rods reacts with coolant water. During the Three Mile Island 2 (TMI-2) accident in 1979, a large amount of hydrogen was released to the containment and burned, although the pressure increase did not lead to rupture of TMI-2's large dry containment. The ice condensers not only cannot reduce the risk of hydrogen combustion but also can actually increase it, because they divide the containment volume into small compartments where hydrogen gas can more readily reach explosive concentrations.

The seriousness of this issue is clear from the following data on the strength of containment buildings. The pressure that can be generated in the containment from hydrogen combustion can typically reach a value of about 110 pounds per square inch

(psi). The average failure pressure of U.S. large dry containments is around 113 psi, whereas for ice condenser containments it is around 63 psi. Therefore, hydrogen burns can easily overpressurize and rupture ice condenser containments.

For this reason, after the TMI-2 accident, NRC required that ice condenser plants install hydrogen igniters, which are operator-initiated, AC-powered devices that are designed to burn hydrogen at a controlled rate before it reaches an explosive concentration.

However, the risk of hydrogen explosions in ice condensers has not been eliminated entirely by this requirement, since the hydrogen igniter systems now in use require AC power to operate. Therefore, in the event of a simultaneous loss of both off-site

and on-site AC power supplies, known as a station blackout (SBO), hydrogen control is lost.

Earlier this year, the Nuclear Regulatory Commission (NRC) released a report that analyzed the risk of containment failure during severe accidents at reactors with "ice condenser" containments. The report, entitled Assessment of the DCH

[Direct Containment Heating] Issue for Plants with Ice Condenser Containments, NUREG/CR-6427, finds that "no ice condenser plant is inherently robust to all credible hydrogen combustion events in a SBO accident." It also concludes that "ice condenser

plants are at least two orders of magnitude [one hundred times] more vulnerable to early containment failure than other U.S. PWRs" as a result of hydrogen explosions during core melt accidents. This study, which was performed by Sandia National Laboratories (SNL) in Albuquerque, calculated that for accidents in which the hydrogen igniters were not available, such as SBOs, the probability that the containment would rupture as a result of hydrogen combustion is 34% for Catawba and 58% for McGuire. Using the same methodology, previous NRC studies found that the risk of containment failure at large dry containments is less than 0.1%.

SNL found that certain SBO accidents --- namely, those in which the reactor coolant system remains at high pressure at the time that the reactor vessel is breached by molten fuel --- the probability of early containment failure as a result of detonation of pre-existing hydrogen is nearly 100% for both Catawba and McGuire. This means that if one of these sequences were to occur, there would be little difference between the ice condenser plants and nuclear plants without containments like Chernobyl.

NRC and the nuclear industry continue to argue that accidents as severe as an SBO are so unlikely that the weakness of ice condensers is not a high-priority concern. However, an SBO actually occurred at the Vogtle plant in Georgia in 1990, during which the plant lost all off- and on-site power supplies for 35 minutes. Other plants have come quite close to an SBO. For instance, in 1996 Catawba lost off-site power for more than a day with one of the two emergency diesel generators unavailable.

That means it was only one generator away from an SBO. NRC estimates that at that time, there was a 0.2% chance that the core of the reactor would have been damaged. In light of the SNL study, it is now known that this corresponded to a nearly one in a thousand chance of a Chernobyl-type accident.

According to Duke Energy's own data, provided to the NRC in its Individual Plant Examination (IPE) submittals (probabilistic risk assessments done by licensees, without peer review), McGuire has a relatively high probability of experiencing an SBO. Factoring in this probability, NRC obtained a containment failure probability given core damage of 13.9% for McGuire. This result is nearly seven times greater than the value of 2.4% reported by Duke in the McGuire IPE.

Although this value exceeds NRC's guideline that containment failure probability should not exceed 10%, NRC argues that it is "consistent with a general objective" of 10%. However, this result does not take into account "external events" such as earthquakes or tornadoes. A tornado caused a loss of off-site power at the Davis-Besse plant in 1998, and one of the diesel generators became inoperable afterward. Such events are associated with a much higher SBO risk than internal transients.

Therefore, the fraction of core damage scenarios that are also associated with SBOs would be much higher if external events were included.

For example, according to Duke Energy's own IPE data, the probability of an earthquake causing an SBO at Catawba is over ten chances per million per year. According to a recent NRC proposal, any accident sequence that had a probability of more

than one chance per million per year would have to have an early containment failure risk of less than 10%. Catawba, with an early containment risk of 34%, would be in violation of this guideline based on the seismic risk alone.

Station blackout can also occur as a result of sabotage, which hasn't been taken into account in the analysis. For instance, during a recent NRC force-on-force exercise at the Oconee plant, also owned by Duke Energy, mock attackers were able to cut off-site power (this is always assumed to be the case, because the power lines are not protected), defeat the security force and cause core damage. However, the probability of a sabotage-induced SBO cannot be quantified. Therefore, the best line of defense in this case is to ensure that the containment will not fail.

The SNL report concludes that "all [ice condenser] plants, especially McGuire, would benefit from reducing the station blackout frequency or some means of hydrogen control that is effective in station blackouts," noting that the latter course would reduce early containment failure probabilities "by more than an order of magnitude in all plants and especially McGuire."

However, according to the report, 'previous cost/benefit studies generally do not justify the expense in providing hydrogen control in SBO because ... the SBO probability is a small fraction of the core damage frequency ...'. This assumption has now been called into question.

NRC is in the process of reviewing its regulations on combustible gas control. NRC staff have recently proposed a requirement that ice condensers provide a means for controlling hydrogen in station blackouts unless it can be shown that the probability of a station blackout is acceptably low.

Meanwhile, Duke Energy has learned of the ice condenser report and is already raising doubts about its validity. Duke met with NRC staff on September 28 and vigorously opposed the idea that the installation of new equipment for controlling hydrogen gas accumulation in SBOs might be necessary."

In his response to the NCI report NRC Chair Richard Meserve wrote that:

Even though the vulnerability of ICC plants was judged [by the Sandia report] to be higher for particular severe accident sequences, the overall safety of the plants remains adequate considering the probabilities of these events in the context of the Commission's safety goals. The key finding of the report was that early containment failure in ICC plants is dominated by hydrogen combustion which largely depends on plant-specific probabilities for station blackout. As you stated, ICC plants have igniter systems for hydrogen control and these systems are not operable during station blackout events. The NRC staff shares your thoughts regarding the need to evaluate the functionality of hydrogen igniters during station blackout at ICC plants through the generic safety issue program. The NRC staff informed the Commission of our intention to perform such an evaluation consistent with the policy discussion on backfit considerations in SECY -00-0198, dated September 14, 2000.

3. Deficiencies in the licensee's operating experience warrant further scrutiny of the ice condenser system aging program(s).

a. As reported in BREDL's October 25, 2001 Petition to Dismiss:

" On October 8, 1999 the NRC granted the licensee an exemption to 10CFR.54.17.c., thus allowing the licensee to submit a license renewal application earlier than the 20 years before the expiration of the operating license currently in effect.¹

Part of the basis for the exemption, which was requested on June 22, 1999, was the licensee's assertion of a 'regular and systematic exchanges of information on plant-specific operating experience among all three Duke nuclear stations.' [One of the] two instances... where this statement [was found to be] in error is as follows:

“b. In 1998, the NRC’s Allegation Review Board found that ‘problems with D.C. Cook Ice Condenser Containment such as configuration and testing, and Ice Basket Bay Doors and Components were known but not reported by D.C. Cook, Watts Bar, McGuire, and Westinghouse.’” Although the ARB classified the concern as ‘low’ significance, it also illustrated a failure to exchange “information on plant-specific operating experience among all three Duke nuclear stations” in order to correct safety problems; and also implies by omission that McGuire personnel did not share this information with Catawba personnel.”

F. Summary

BREDL has presented sufficient information to show a genuine dispute on a material issue of law or fact, including references to specific portions of the application that the petitioner disputes and the supporting reasons for each dispute, and/or identification of each asserted failure of the application to contain information on a relevant matter as required by law, as well as the supporting reasons for the petitioner.s belief that the application fails to contain relevant information required by law. See 10

A. Contention Number and Title:

5. Assessment of Reactor Vessel Integrity

B. Contention

The assessment of reactor vessel integrity with regard to embrittlement and metal fatigue is insufficient and incomplete.

C. Specific statement of the issue of law or fact to be raised or controverted

10CFR54.29

10CFR54.21

D. Brief explanation of the basis or bases of the contention

Under 10 CFR 54.29 Standards for issuance of a renewed license, a condition for a renewed license includes the provision that:

“(a) Actions have been identified and have been or will be taken with respect to the matters identified in Paragraphs (a)(1) and (a)(2) of this section, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the [Current Licensing Basis], and that any changes made to the plant's CLB in order to comply with this paragraph are in accord with the Act and the Commission's regulations. These matters are: (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been

identified to require review under §§54.21(a)(1); and (2) time-limited aging analyses that have been identified to require review under §§54.21(c).”

Neutron bombardment resulting from the fission reaction degrades the metal parts of the reactor and the metal becomes brittle. Reactor embrittlement increases with age. An embrittled reactor may look unchanged, but it will not perform as well under extreme conditions. In the event of a drop in the level of reactor coolant, the heated water is replaced by cold water from outside the reactor. This cold water can cause the embrittled metal part to fail and a minor reactor failure becomes a major one. Embrittlement of reactor parts is a well-known phenomenon and has caused premature closing of commercial power reactors.

Assessment of reactor vessel integrity must account for all forms of vessel weakness caused by normal operations. The operator fails to include important factors in their assessment including prolonged cycles of heating and cooling and stress fatigue in critical reactor parts not revealed by current methods.

The impacts of aging on key mechanical and electrical parts as well as all other aging issues are required to be analyzed under this process by NRC rules and guidance, and reflected in the applicant’s Attachment B: Aging Management Programs and Activities.

E. Statement of all appropriate facts and expert opinion to support contention

1. Coupon Test Fails to Account for Stress Fatigue

Coupons are pieces of containment vessel metal which are installed in a new reactor assist in the monitoring of tensile strength losses. These coupons are limited in number and are insufficient to determine embrittlement effects during the 20-year license extension period. As outlined below, alternative methods of assessing reactor vessel embrittlement based on extrapolations of past performance will not provide adequate assurances of vessel integrity and protection of health and safety. Moreover, the coupon test itself fails to address an additional cause of metal component failure: stress fatigue caused by repeated cycles of heating and cooling.

Jesse Riley served as a spokesman for the intervenor Carolina Environmental Study Group during Nuclear Regulatory Commission construction and operating license proceedings for the original licensing of Duke Power’s McGuire and Catawba nuclear stations. Recent correspondence from Mr. Riley to NRC and local government officials details fundamental problems in the current and future assessment of reactor vessel integrity, embrittlement, and metal fatigue. Mr. Riley states:

“The reactor is currently limited to 200 refueling, i.e., cycles of heating and cooling. It is subjected to the stress of internal pressure and to stresses due to the thermal gradients from inside to outside making for a difference in thermal expansion. Fatigue is the term used to characterize the losses in tensile properties due to repeated cycles of stress. Tensile properties are also caused by irradiation from the reactor fuel.”¹

“This basis is a test performed on small pieces of metal from which the reactor was made, called coupons, which were placed in it at the beginning of operation. (It was obvious to the designers of these vessels and to the NRC that the failure of a reactor vessel would be catastrophic.) The coupon test is designed to show the extent to which the physical characteristics of the reactor vessel have deteriorated as a result of exposure to radiation. The problem is: unlike the

reactor vessel, the coupons have not been exposed to another weakening factor, stress fatigue.”²

*“The coupon test provides no information as to the effect of the fatigue on the reactor vessel which cycles between high load and no load. To the best of my knowledge this matter has not been examined in a licensing proceeding. It was not considered in the licensing of the McGuire and Catawba plants.”*² [emphasis in the original]

“The reactor stud bolts are exposed to greater stress than the reactor vessel. Are they replaced at refuelings? Are they the same material as the vessel? On what evidence are the tensile properties of the stud bolts based?”¹

1 Letter to Nuclear Regulatory Commission Jim Wilson from Jesse Riley, October 23, 2001

2 Letter to Mecklenburg County Commission from Jesse Riley, October 16, 2001

The licensee cannot assure the continued safe operation of the McGuire and Catawba plants for an additional twenty year period. Under normal operations the pressure inside reactor vessels is very high. A new vessel may perform well because the thickness of the vessel wall can withstand these tensile stresses. However, after 20 or 40 years of operation, repeated heating and cooling may cause a loss in strength, caused by stress fatigue, which would cause a reactor vessel to fail at pressures which would have been withstood by a new reactor vessel. The lack of real-world assessment of stress fatigue alone should prohibit the extension of McGuire and Catawba operating licenses.

2. Applicant’s program vs. NRC Regulations:

Duke Energy has not identified actions that have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of structures and components or time-limited aging analyses that have been identified under §54.29. Therefore, we contend that there is no reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the Current Licensing Basis.

3. Licensee’s Operating Experience:

The following non-cited violation was recorded in March 2001 during McGuire Unit 1 shutdown. Although cited by the NRC as having low immediate safety significance, the incident is instructive in that it reveals that near-normal operations over a period of time may put unusual stresses on critical reactor components. In this case the drop below the minimum temperature for criticality created unusual stresses as a result of the difference in thermal expansion caused by thermal gradients between the inside and the outside of the reactor vessel.

Initiating Events

Significance: G Mar 17, 2001

Identified By: Licensee

Item Type: NCV NonCited Violation

Inadequate Corrective Actions for Recurring Problems with Shutdown Operations Involving Loss of Letdown and/or Inadvertent Reactor Coolant System Cooldown

Transients

Inadequate corrective actions (10CFR50, Appendix B, Criterion XVI) for recurring problems with shutdown operations involving loss of letdown and/or inadvertent reactor coolant (NC) system cooldown transients. During a Unit 1 shutdown from Mode 2 to Mode 3 on March 9, 2001, NC system temperature went below minimum temperature for criticality due to overfeed of steam generators. This event occurred because of ineffective corrective actions to address procedural deficiencies and/or equipment problems complicating plant cooldown. This is captured in the licensee's corrective action program under PIP M-01-0986. This finding was determined to have very low safety significance and is being treated as a Non Cited Violation (Section 4OA7).
Inspection Report# : [2000007\(pdf\)](#)