MINNESOTA DEPARTMENT OF PUBLIC SERVICE

Volume IV

Direct Testimony and Exhibit of
Gregory C. Minor
MHB Technical Associates
ISFSI Health and Safety Risks

Before the
Minnesota Public Utilities Commission

Northern States Power Company
Docket No. E002/CN-91-19

September 30, 1991

DIRECT TESTIMONY AND EXHIBIT OF

GREGORY C. MINOR

MINNESOTA DEPARTMENT OF PUBLIC SERVICE

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BEFORE THE

MINNESOTA PUBLIC UTILITIES COMMISSION

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NORTHERN STATES POWER COMPANY

DOCKET NO. E002/CN-91-19

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NORTHERN STATES POWER COMPANY In the Matter of an application for a Certificate of Need for Construction of an Independent Spent Fuel Storage Installation	))))))))	Docket No. E-002/CN-91-19

Testimony of

GREGORY C. MINOR
MHB TECHNICAL ASSOCIATES

On Behalf of the

MINNESOTA DEPARTMENT OF PUBLIC SERVICE

COST IMPACT QUANTIFICATION OF NORMAL OPERATIONAL AND ACCIDENTAL RELEASES FROM THE PRAIRIE ISLAND INDEPENDENT SPENT FUEL STORAGE INSTALLATION

SEPTEMBER 1991

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BEFORE THE MINNESOTA PUBLIC UTILITIES COMMISSION
IN THE MATTER OF NORTHERN STATES POWER COMPANY
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TESTIMONY OF
GREGORY C. MINOR

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I. INTRODUCTION AND SCOPE

Q: Please state your name and affiliation.

A: My name is Gregory C. Minor, and I am a principal consultant and Vice President of MHB Technical Associates located at 1723 Hamilton Avenue, Suite K, San Jose, California 95125.

Q: On whose behalf do you appear in presenting this testimony?

A: I appear on behalf of the Minnesota Department of Public Service.

Q: Mr. Minor, what is the purpose of your testimony in this proceeding?

A: My testimony quantifies the risks arising from radiation exposures arising from accidental releases of radioactivity into the environment, as well as normal operational radiation exposures, resulting from the operation of the proposed Independent Spent Fuel Storage Installation ("ISFSI") at the Prairie Island Nuclear Power Station site. These risk values, expressed in units of person-rem per year, are then converted to dollar costs using a range of industry accepted cost conversion factors. Thus, my testimony is essentially a cost evaluation of what may be referred to as "environmental externalities" associated with the operation of the ISFSI.

Q: Please briefly describe your background and experience.

A: I have over 30 years experience in the nuclear industry including work at nuclear plant sites, design work related to nuclear plant systems, and consulting work related to nuclear plant cost and safety issues. A complete copy of my statement of qualifications and experience is provided as Attachment 1 hereto.

Q: Would you please describe the analytical approach which you used in the preparation of your testimony?

A: I first reviewed the Northern States Power Company ("the Company") 10 CFR Part 72 license application filing with the U.S. Nuclear Regulatory Commission in order to familiarize myself with NSP's proposal. The principal documents which I reviewed in this connection were as follows:

(a) Prairie Island Independent Spent Fuel Storage Installation Environmental Report, Docket No. 72-10, Rev. 0, August 1990.

(b) Prairie Island Independent Spent Fuel Storage Installation Technical Specifications and Safety Analysis Report, Docket No. 72-10, Rev. 0, August 1990; and Rev. 1, April 1991.

I then reviewed the Minnesota Environmental Quality Board ("MEQB") Final Environmental Impact Statement, Prairie Island Independent Spent Fuel Storage Installation (April 12, 1991), including the public comments on the Draft EIS. In addition, I reviewed NSP's April 29, 1991, application for a certificate of need before the Commission, as well as NSP's June 1991 supplemental filing before the Commission.

Then, in order to better evaluate NSP's submittals, I reviewed previous U.S. Nuclear Regulatory Commission ("NRC") environmental assessments of ISFSI applications at other sites and the Sandia National Laboratories report upon which the NRC bases its estimates of the dose consequences of accident and normal operational releases from ISFSIs.

Finally, I reviewed various NSP responses to discovery requests concerning the ISFSI certificate of need application.

Based on my review of this documentation, and my familiarity with the general practices of risk assessment in the nuclear industry, I performed the following steps: (a) identification of a set of bounding accident conditions for the ISFSI; (b) estimation of the likelihood or probability[1] of these accident conditions; (c) estimation of the magnitude of the resulting radiological releases; and (d) estimation of the dose consequences caused by these releases. In performing these steps, I derived conservative "*worst-case*" accidents in order to bound the consequences which might occur as a result of accidents involving the ISFSI.[2] These matters are described in more detail below.

Together with a review of NSP's estimates of normal ISFSI operational release dose consequence estimates, I estimated the annual dose consequences (in units of person-rem/year) arising from both accidents and normal operation of the ISFSI. I then performed a cost quantification of these dose consequences using accepted industry methodology (i.e., assigning a cost per person-rem of exposure, ranging from $1,000 to $10,000 per person-rem, as explained further below).

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1 Probability or likelihood can be expressed mathematically as the number of times an outcome may be expected in a number of samples or in a unit of time (such as years). This may be written in various forms. For instance, the following values all represent the same likelihood: (a) one chance in one thousand; (b) 1 in 1,000; (c) 1×10^{-3}; and (d) 1E-3.

2 In the context of this sentence, by "*conservative worst-case*" I mean that I have defined accidents which, while generally descriptive of what could occur, tend to overstate the resulting radiological release because they are more severe that what would most likely occur. A good illustration of this is the airplane crash accident which is described later in my testimony. My testimony assumes that all of the casks are in place at the ISFSI and that all are damaged as a result of the plane crash and subsequent fire, resulting in a very large radiological release. Moreover, the consequence calculations for this accident are based on 10-year old fuel; in reality, much of the fuel would be older than this and would, therefore, contain less radioactivity as a result of radioactive decay. This sort of analysis attempts to place bounds on risk which should not be exceeded.

II. ANALYSIS

II.A Background

Q: Would you please describe how spent reactor fuel is produced in the context of the Prairie Island facility?

A: Northern States Power Company (NSP) is the owner and operator of the Prairie Island Nuclear Power Plant, a two-unit pressurized water nuclear powered electric generating station located 28 miles southeast of Minneapolis, Minnesota. The station consists of a pair of Westinghouse pressurized water reactors with licensed thermal power levels of 1650 MWt each and associated nuclear safety and balance-of-plant structures, systems, and components.

As licensed by the NRC, the Prairie Island reactors have reactor cores consisting of 121 fuel assemblies each. The fuel assemblies consist of fuel rods containing low-enriched uranium dioxide fuel clad in a zirconium alloy. As the reactors are operated, part of the Uranium-235 component of the fuel is consumed (by fissioning). In addition, part of the Plutonium-239 created by transmutation (absorption of a neutron by Uranium-238, followed by decay to Plutonium-239) is also consumed.

Eventually, the power of the reactor will decline as the "*burnup*" of the fuel increases. Periodically, the reactors must be shut down in order to replace some of these fuel assemblies with unirradiated fuel in order to maintain full power operation. Spent fuel is discharged to a spent fuel pool for interim storage under water. Approximately 48 assemblies per unit are discharged for each 16-month cycle (or an average of about 70 assemblies per year). The design of the Prairie Island facility currently incorporates provisions for storage of 1586 spent fuel assemblies in high density racks.[3]

NSP proposes to supplement the current storage capabilities by removing the oldest spent fuel (cooled 10 years or more) from the spent fuel pool and placing it into metal dry cask storage containers, and placing the containers on concrete pads. Such a spent fuel storage facility is referred to in NRC jargon as an Independent Spent Fuel Storage Installation, or by the acronym ISFSI (pronounced "*iz-fizzy*").

Q: Please identify the composition of the Prairie Island spent fuel which will be stored in the proposed ISFSI.

A: During reactor operation, nuclear fuel undergoes fission. A portion of the original Uranium-235 and some of the Plutonium-239 (created by transmutation resulting from absorption by Uranium-238 of a neutron) is fissioned, with a release of heat and neutrons, and the production of fission products (such as Cesium-134 and -137, Strontium-89 and -90, Krypton-85, and various Iodine and Xenon species). Some of the fission products are stable, but others are unstable (radioactive). Spent fuel is, therefore, highly radioactive.

In addition to the radionulides formed as described above, some of the metallic structural components of the fuel assemblies also undergo transmutation by absorption of

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3 Until approximately 1975, it was planned that spent fuel from nuclear power reactors would be stored in spent fuel pools for an interim period, following which the spent fuel would be transported to a reprocessing plant for recovery and recycle of fissile and fertile uranium and plutonium isotopes. Accordingly, nuclear power plants, including the Prairie Island facility, were constructed with considerably less spent fuel capacity than would be required in order to store all of the spent fuel that could be generated assuming that the plants operated until the end of the time period permitted under the provisions of their NRC facility operating licenses. (From an operational standpoint, it is prudent to maintain the capability to fully offload the reactor core to the spent fuel pool.) Commercial reprocessing of spent reactor fuel has not developed as had been anticipated for a variety of reasons (among them being the economics of reprocessing). As a result of limited spent fuel storage provisions in the original design of nuclear power plants, including Prairie Island, applications were made to the NRC and subsequently approved to permit the expansion of pool storage through the use of reracking of the spent fuel storage pools to permit a greater density of spent fuel storage. Reracking has been accomplished twice at Prairie Island.

neutrons. Such radioactive materials are often referred to as activation products. Chief among these radionulides is the production of Cobalt-60 from Iron-59.

Table 5 (at the end of Section II) provides a listing of the radioactivity content of pressurized water reactor spent fuel irradiated to 33,000 megawatt-days per metric ton of uranium and decayed for 10 years (which is the minimum decay period intended for placement in the Prairie Island ISFSI). Of the many nuclides listed in Table 5, however, only a small number are of primary concern due to their presence in significant quantities, their gaseous form or their solubility in water, and their biological mobility. These species are: Krypton-85 (gas); Cesium-134; Cesium-137; and Iodine-129.[4]

Q: Would you please briefly describe the nature of the proposed Prairie Island ISFSI?

A: The Company has requested permission from the Commission for authority to construct an ISFSI at the Prairie Island Nuclear Station for the purpose of dry storage of spent nuclear fuel. In particular, the ISFSI would consist of 48 dry metal casks designed by Transnuclear, Inc., stored upright on two reinforced concrete pads. The cask consists of a fully sealed metal cask with an internal basket for holding spent pressurized water reactor fuel assemblies. The cask is 16 feet, 10 inches tall and 8 feet, 6 inches in diameter, and weighs approximately 122 tons fully loaded. The cask walls consist of steel approximately 9.5 inches thick. The casks are filled with helium gas.

The storage facility itself consists of two reinforced concrete pads, enclosed by a security fence. Each of the pads will hold two parallel rows of 12 dry storage casks (a total of 24 casks per storage pad).

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4 U.S. Nuclear Regulatory Commission, Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel, NUREG-0575, Vol. 1, August 1979, page 4-14.

Cooling of the spent fuel is by entirely passive means (radiant and convective cooling which transfer heat from the spent fuel to the atmosphere).[5] Since there are no active systems required to assure adequate cooling, only environmental externalities (environmental stresses, such as temperature, pressure, etc., which exceed the design specifications) and errors in the design, fabrication, or installation of the dry storage casks can result in accidents with consequences to the public.[6] This greatly simplifies the task of estimating the risk posed by operation of the ISFSI as compared with, for example, estimating the risk posed by operation of a nuclear power plant, which relies on numerous active safety and some non-safety systems in order to protect public health and safety.[7]

The spent fuel casks are stored upright on the concrete pads. The specific cask design planned for use at the Prairie Island ISFSI is Transnuclear TN-40 metal cask, each of which can hold 40 spent fuel assemblies of the type used at the Prairie Island facility. Each fuel assembly, weighing about 1300 pounds, consists of 179 fuel rods. Only spent fuel which has been stored in the existing spent fuel pool for at least 10 years is proposed to be stored at the ISFSI. Under NRC regulations (10 CFR Part 72), a license for an ISFSI is

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5 NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, page 3.1-2.

6 This is in contrast with a spent fuel pool or an existing nuclear reactor, both of which rely on actively functioning systems in order to maintain an adequate degree of safety. For example, the spent fuel pool relies on a spent fuel pool cooling system to remove decay heat from the spent fuel pool water and transfer this heat to a service water system for ultimate dissipation into the environment. Similarly, existing nuclear reactors (such as the Prairie Island nuclear reactors) rely on a variety of active systems to ensure plant safety (such as the reactor protection system, the emergency core cooling system, the containment spray system, etc.). In these more mechanically and electrically complex systems, the assessment of risks requires a more sophisticated form of analysis (primarily consisting of the use of formal fault tree and event tree analyses).

7 Just for perspective, MHB is currently reviewing a risk assessment of a nuclear power plant which is documented in twenty 3-inch ring binders. Eighteen of these volumes consist of detailed fault trees for the various plant systems. The fault trees are a graphical depiction of the various failure modes and mechanisms for the systems. In order to perform the risk assessment, these fault trees are linked with system event trees by reducing these graphical displays to Boolean algebraic mathematical expressions. In this manner, the sequences of system failures which can result in a severe accident can be logically identified, and the likelihood of these sequences occurring can be calculated. It is the wide variety of systems, and their interactions with one another, which result in the complexity of a nuclear power plant risk assessment. In comparison, a risk assessment of an ISFSI would be expected to be considerably simpler due to the general lack of active safety systems (for activities other than loading and unloading the casks and transporting them to the ISFSI and placing them on the concrete pads).

limited to 20 years, with the possibility of renewal upon timely application. The Company filed its Part 72 license application with the NRC in August 1990, and expects that NRC will act on the application in early 1992.

II.B Cost Quantification Method

Q: How did you conduct your analysis of this proposed ISFSI?

A: I separately considered normal operational doses (including both doses to the public and doses to personnel at the Prairie Island facility) and doses arising from postulated design basis and beyond design basis accidents (including consideration of the probability of occurrence of these accidents). I then normalized the doses to a per-year basis for comparison and cost quantification.

Q: How did you quantify the cost of the dose consequences arising from normal operation of and accidents involving the ISFSI?

A: I used typical nuclear industry practice of quantification based on a dollars per person-rem calculation. Specifically, I used a range from $1,000 to $10,000 per person-rem in my calculations.

Q: What is the basis for the range of $1,000 to $10,000 per person-rem?

A: It has been typical industry practice, dating from the mid-1970s, to quantify dose consequences at the rate of $1,000 per person-rem. That figure has been kept through the present in the NRC's safety goals program (NRC, Implementation of Safety Goal Policy, SECY-89-102, March 30, 1989, page 16 and Enclosure 2) and in various other NRC uses (cost-benefit analysis, regulatory analysis, ALARA regulations at Appendix I to 10 CFR Part 50, etc.).

My associates and I at MHB Technical Associates (and others) have been critical of the continued use of the $1,000 per person-rem figure as being out of date and not representative of current understandings of dose-response relationships. A recent paper by a senior scientist at Brookhaven National Laboratory (a copy of which is provided as Attachment 2) corrected for inflation since 1975 when the $1,000 per person-rem value was first adopted, and also corrected for more recent understandings of dose-response risk estimates. After making these corrections, a new estimate of $10,000 per person-rem was derived. This estimate was confirmed by comparing it with an independently-derived estimate (also described in Attachment 2) of what the industry actually spends to avoid doses in the implementation of NRC ALARA ("*As Low As Reasonably Achievable*") guidance.

For the purposes of this testimony, I have used the $10,000 per person-rem estimate as providing a better estimate of the value of a person-rem of exposure in today's regulatory environment. I also note that NSP uses the $10,000/person-rem figure in its radiation protection ("*ALARA*")[8] calculations.[9]

II.C Normal ISFSI Operations

Q: How is the ISFSI cask loaded and placed onto the ISFSI concrete pad?

A: The basic process is summarized in NSP's safety analysis of the ISFSI:[10]

> *Each cask will be handled with a lifting yoke, the 125 ton capacity auxiliary building crane, a transport vehicle, or other appropriate equipment. The crane will lift the cask from the spent fuel pool, in the spent fuel pool enclosure, move the cask laterally through an access door, and lower the cask to ground level in the rail bay of the Auxiliary Building. The cask will then be picked up by the*

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8 ALARA is an acronym for "*As Low As Reasonably Achievable*", and refers to NRC radiation protection regulatory requirements. NRC regulatory guidance for maintaining radiation exposures ALARA is set forth in Appendix I to 10 CFR Part 50.

9 NSP Response to DPS Information Request No. 207.

10 NSP, ISFSI Safety Analysis Report, August 1990, Rev. 0, page 3.1-2.

> *transport vehicle which will be pulled to the ISFSI by a tow vehicle. After the transport vehicle has been maneuvered to locate the cask in its storage position, the cask will be set down.*

Q: What exposures result from normal operation?

A: The public is exposed to radiation due to low level "*sky shine*" doses resulting from gamma radiation penetrating the dry storage casks. Plant personnel are also exposed in this manner, as well as during other aspects of ISFSI operations (cask loading, cask transport, cask placement, monitoring the casks during storage, security inspections, etc.). Under normal conditions, where the casks are maintained within their design envelope, there are no other exposures since cask integrity is maintained and the radioactive contents of the cask do not escape to the environment.

Q: What are the dose consequences of normal operation of the proposed ISFSI?

A: NSP has a calculation of the dose consequences of normal operation of the ISFSI. NSP (conservatively) estimates the annual population dose to the general offsite population arising from normal ISFSI operation to be 0.007 person-rem per year.[11] In addition, NSP estimates the annual population dose to onsite personnel arising from normal ISFSI to be 1.11 person-rem per year.[12] Accordingly, the total population dose arising from normal operations of the ISFSI is 1.12 person-rem per year (that is, 1.11 person-rem per year for the on-site population plus 0.007 person-rem per year for the offsite population).

II.D <u>Postulated ISFSI Accidents</u>

Q: What sorts of accident conditions form the design basis for the TN-40 spent fuel casks?

A: The cask design criteria are fully described in the Company's Certificate of Need Application and in the Company's NRC license application. Briefly, the casks have a

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11 NSP Response to DPS Information Request No. 69.

12 NSP Response to DPS Information Request No. 68.

nominal 25 year design lifetime (i.e., about five years longer than the 20-year license period for the ISFSI).[13] The casks are designed to account for ambient temperatures ranging from minus 40° (considering snow and ice loads) to plus 120° F. (considering the maximum solar heat load incident on the cask). The casks are designed for an internal pressure of 100 pounds per square inch (psi). The casks are also designed to withstand the following environmental externalities:[14]

(a) A tornado with a total windspeed of 360 miles per hour (300 mph rotational and 60 mph translational) without tipping the cask from its vertical storage position.

(b) The impact of tornado missiles (projectiles produced as a result of a tornado) consisting of a 12 foot, 4 inch by 12 inch plank travelling at 300 mph, and a 4,000 pound automobile travelling at 50 mph, without tipping the cask from its vertical storage position or penetrating the cask.

(c) Earthquakes with a ground acceleration of 0.12g horizontal and 0.08g vertical without tipping the cask from its vertical storage position.

(d) Dropping or tipping the cask onto an ISFSI pad.

Q: Is it possible that extreme environmental conditions could produce impacts larger than those considered in the dry cask design?

A: This is possible, however there must be account taken of the likelihood of such extreme environmental conditions. For example, take the case of tornado winds in excess of 360 miles per hour combined windspeed. Although tornadoes are capable of producing high windspeeds, only the most severe tornadoes are capable of producing winds of such severity. In the standard tornado severity classification system, only Intensity F-6 tornadoes (with windspeeds in excess of 277 mph) are capable of such windspeeds.

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13 In reality, the casks would be expected (absent any significant error in design, fabrication, and installation) to last longer than this, but how much longer is not accurately known. The cask manufacturer (Transnuclear) has indicated that the casks should have an "*economic life*" of 40 years (NSP Response to DPS Information Request No. 62).

14 NSP, Revised Application, 6/10/91, pages 36-37 and 56-59.

Based on data compiled by the Electric Power Research Institute ("*EPRI*"), Intensity F-6 tornadoes have frequencies no greater than approximately 1×10^{-7} per year per square mile even in the highest risk tornado region of the U.S. (and this region is not the one in which the Prairie Island site is located).[15] This information is confirmed by a study done for the NRC, which concluded that the tornado wind speeds expected at a frequency of 1×10^{-7} per year in the United States range from less than 153 mph to 332 mph.[16] Data for Minnesota covering the two-decade period from 1964 to 1983 indicate that tornadoes of Intensity F-5 and F-6 are very unusual (accounting for only two out of the total of 535 tornadoes to strike the state in this 20-year period, or only about 0.37% of the total).[17]

Moreover, it must be recognized that the target area presented by the ISFSI pads is considerably smaller than a square mile (the two pads -- each being 36 feet by 216 feet -- together have a surface area of 15,552 square feet, or about 0.0006 square miles). In addition, NSP calculates that a wind speed of 549 mph (rather than the 360 mph corresponding to the 1×10^{-7} probability cited above) would be required to tip over the cask.[18] Such a wind speed would have a frequency far less than 1×10^{-7} per year per square mile (if indeed such a high windspeed is even physically possible in a tornado). Even then, should a wind speed of 549 mph occur, cask tip over would occur and NSP indicates that cask tip over events would not breach the integrity of the casks. (As

15 EPRI data summarized in ABB Combustion Engineering Nuclear Power, System 80+ Standard Design, Amendment H, August 31, 1990, Table B.4.2.1-1. NSP, based on an NRC report, provides an identical estimate specifically for the Prairie Island site. NSP also indicates that the probability of occurrence for a 549 mph tornado is "*infinitesimally small*" (NSP Response to DPS Information Request No. 218). I concur in NSP's assessment.

16 J.V. Ramsdell and G.L. Andrews, Tornado Climatology of the Contiguous United States, Pacific Northwest Laboratory, Richland, Washington, prepared for the U.S. Nuclear Regulatory Commission, Washington, D.C., NUREG/CR-4461 (PNL-5697), May 1986, page v. The lower bound of the cited wind speeds are for the western U.S., while the upper bound is for Kansas and Nebraska.

17 Id., page C.41.

18 NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, pages 3.2-3 to 3.2-4.

discussed below, I have in my analysis allowed for the possibility that a cask could be defective to the point where tip over could breach cask integrity.)

Taken together, this information clearly indicates that it is extremely unlikely that tornadoes could contribute to the risk posed by the Prairie Island ISFSI -- that is, the probability of tornado winds of sufficient severity to tip over the dry storage casks is far less than 1×10^{-7} per year -- probably of the order of 7×10^{-10} per year.[19,20] Even taking the largest consequence event for a 48-cask tip over event at 42,500 person-rem (see below). this works out to an annual consequence of about 0.00003 person-rem per year. Even this calculation fails to account for the probability that all of the casks are defective (since otherwise there is no radiological release). Accordingly, the risks posed by the ISFSI as a result of tornado events is negligible.

Q: Could a cask tip over event result in a radiological release?

A: Yes. However, it must first be noted that NSP's analysis of cask tip over events indicates that even if a cask tip over occurs, the integrity of the cask will not be compromised.[21] Thus, assuming that cask tip over results in a breach of the cask is (in the absence of some

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19 NSP reports that the probability of any tornado (irrespective of total wind speed) striking a 1° square (that is, one degree of latitude by one degree of longitude) centered on the plant site per year is between 3.8×10^{-3} and 5.8×10^{-3} (NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, page 2.3-1). A 1° square in the vicinity of the plant site encompasses an area of approximately 3948 square miles (47 x 84 miles). According, the probability of any tornado hitting any particular square mile in the plant region is between 9.6×10^{-7} and 1.5×10^{-6} per year. Considering that the ISFSI target area is 6×10^{-4} square miles, the probability of any tornado striking the ISFSI is between 5.8×10^{-10} and 9.0×10^{-10} per year. Such probabilities are so low that they are meaningless in any real sense, except to indicate that the probability of such an event is very low.

20 An alternative calculation would be as follows. The area of Minnesota is 84,068 square miles (Hammond Citation World Atlas, Hammond, Inc. (Maplewood, New Jersey), 1978, page 253). There were two F-5 Intensity tornadoes in twenty years in Minnesota between 1964 and 1983. This gives a strike frequency for F-5 Intensity tornadoes (or worse) of 1.2×10^{-6} per square mile per year. The area of the ISFSI is 0.0006 square miles. Thus, the strike frequency of an F-5 Intensity tornado at the ISFSI is approximately 7.1×10^{-10} per year. Again, this is a very low probability number.

21 NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, page 8.2-16.

significant error in design, fabrication, or installation) excessively conservative and not a reasonable basis for calculating risk.

Of course, it is possible that one or more casks could be defective as a result of an error in design, fabrication, or installation, and be subject to failure during cask tip over events. I am aware of no data concerning the rate at which such defective (i.e., defective) casks are produced. A study prepared for the Electric Power Research Institute by NUS Corporation estimated the probability of having at least one defective cask out of 100 to be 0.20.[22] Lacking any other evidence, and considering that 48 casks will be used at Prairie Island, I would estimate that the probability of having at least one weak cask out of 48 to be 0.10 (using the NUS estimate as a guide).

It is also possible that the defective cask results from some sort of egregious error affecting all 48 casks -- essentially, a common-mode error (that is, a single error affecting more than one structure, system, or component). Unfortunately, there are even less data available to estimate the probability that all casks will be defective as a result of a common-mode error. There are data available for the probability of common-mode failures for nuclear power plant safety systems; such data are used to estimate what is referred to as a "*Beta factor*" in PRA analyses.[23]

Since nuclear plant safety system components and spent fuel cask components are designed, fabricated, installed, operated, and maintained under arguably similar circumstances (e.g., formal quality assurance programs are used), common-mode failure data for nuclear plant safety system components may provide the best currently available

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22 NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984, page 4-57.

23 A "*Beta factor*" is a generic factor which is multiplied together with the failure rate of one train of a safety system to estimate the joint failure probability of a two-train (or sometimes greater redundant) system when the failure is the result of a common cause. For example, if the failure probability for one train of a two-train system is 0.001, and the Beta factor is 0.1, the failure probability for the two-train system for common mode failure contributions is 0.0001 (i.e., the product of the single-train failure probability and the Beta factor).

information on the likelihood of a common-mode error affecting cask integrity. A generic Beta factor of 0.10 has been derived from operating experience for nuclear power plant safety systems. The component-specific Beta factors comprising this overall average range from 0.03 to 0.22.[24] Design, manufacturing, and construction errors are important contributors to these Beta factors.[25] This evidence suggests that it would be reasonable to assume, for the purposes of this analysis, that there is a probability of 0.1 of one weak cask out of 48, and that there is also a probability of 0.1 times 0.22 (using the highest Beta factor suggested by the evidence) or about 0.022 that all 48 casks are weak as a result of a common cause error.

I hasten to point out that this estimate is speculative, being based on essentially no data on dry casks. For the purposes of a bounding assessment, however, I consider the estimate to be a reasonable working value until more experience is obtained with dry storage casks.

Q: Could earthquakes provide a mechanism for damaging the dry storage casks.

A: Yes. The casks are designed for a 0.12g horizontal acceleration. NSP calculates that a horizontal acceleration of 0.37g would be required to tip over the casks.[26]

Site-specific seismic ground acceleration calculations have been prepared for the NRC for all U.S. nuclear power plant sites east of the Rocky Mountains by Lawrence Livermore National Laboratories ("*LLNL*"). The results for the Prairie Island site using the arithmetic mean seismic hazard curve[27] indicate that the frequency of the ISFSI design

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[24] Pickard, Lowe and Garrick, Inc., Classification and Analysis of Reactor Operating Experience Involving Dependent Events, prepared for the Electric Power Research Institute, EPRI-NP-3967, Interim Report, June 1985, page 5-3.

[25] Id., pages 5-6 to 5-7.

[26] NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, page 3.2-8.

[27] As recommended by the NRC for use in seismic accident probability calculations for Individual Plant Examinations for External Events in Generic Letter 88-20, Supp. 4, June 28, 1991.

basis horizontal acceleration (0.12g) is about 7 x 10^{-4} per year, and that the frequency of the "tip over" acceleration (0.37g) is about 8 x 10^{-5}.[28,29,30]

These estimates are not exact. However, these estimates represent the result of state-of-the-art assessments and are quantities whose mean values are substantially driven by the tails of the probability distributions (that is, the distributions are highly skewed by a small number of very high values). In essence, most of the uncertainty in the frequency of various ground accelerations lies in the direction of lower -- not higher -- frequencies of occurrence. That is, most of the uncertainty lies in the direction of lower estimated risks.

As noted above, the casks are normally expected to survive a cask tip over event without failure. The conditional probability of one cask being defective and therefore failing as a result of an earthquake is 0.1. Thus, a probability of 8 x 10^{-6} per year is estimated for a single cask failure event.[31] The conditional probability for simultaneous

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28 Lawrence Livermore National Laboratory, Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains: Results and Discussion for the Batch 4 Sites, NUREG/CR-5250, Vol. 5, January 1989, page 143.

29 A site-specific analysis of seismically-initiated reactor accidents has not yet been made publicly available for Prairie Island (although such an analysis will be done as part of NSP's response to NRC Generic Letter 88-20, Supplement 4, Individual Plant Examination for External Events). It is worth noting, however, for the sake of perspective, that for the Surry plant (like Prairie Island, an early Westinghouse PWR) an earthquake with a ground acceleration in the range of 0.35g-0.45g has roughly a 25% chance of producing a severe accident (Sandia National Laboratories, Analysis of Core Damage Frequency: Surry Power Station, Unit 1, External Events, NUREG/CR-4550, Vol. 3, Part 3, Rev. 1, December 1990, page 4-96). It is possible that in addition to producing an ISFSI accident, the earthquake discussed above could produce a reactor accident (involving one or both Prairie Island units) as well. The dose consequences of the reactor accident could be substantially greater depending upon the mode and timing of containment failure.

30 NSP has produced estimates of the frequency of earthquakes producing a ground acceleration of 0.12g based on the LLNL study and a similar study prepared for EPRI. At an 85th percentile value (which roughly corresponds to a mean value, but not precisely), NSP estimates the frequency of a 0.12g earthquake at 1 x 10^{-3} (0.001) per year (based on LLNL) to 7 x 10^{-5} (0.00007) per year (based on EPRI). *See*, NSP Response to DPS Information Request No. 213. Similarly, NSP has estimated the frequency of 0.37g earthquakes. Based on using the 85th percentile value, NSP estimates the frequency of such earthquakes to be 8 x 10^{-5} (0.00008) per year (based on LLNL) to 9 x 10^{-6} (0.000009) per year (based on EPRI). *See*, NSP Response to DPS Information Request No. 214. These values are very similar or identical to the estimates I obtained independently from a review of the LLNL study.

31 This number represents the frequency of the cask tip-over ground acceleration times the probability of one defective cask.

failure of all 48 casks is 0.022. Thus, the probability of a seismically-initiated event leading to the failure of all 48 casks is 1.8×10^{-6} per year.[32]

Q: What would be the consequences of such events?

A: NSP has performed an analysis of the worst case single cask accident by assuming the release of all of the Krypton-85 contained in the spent fuel stored in the cask.[33] Krypton-85 is readily susceptible to being released because it is in gaseous form at normal temperature and pressure, and a significant fraction of the Krypton-85 inventory is available for immediate release from the fuel rods should they fail because it is present as a gas in the "*gap*" between the fuel pellets and the fuel cladding. The assumption of a 100% release is, however, conservative, since a cask tip-over event (even if it breached the cask) would be unlikely to simultaneously rupture all of the spent fuel rods. Even if all of the cladding ruptured, not all of the Krypton-85 would be released unless there were some mechanism (such as heat or mechanical disruption) present for forcing the release of the Krypton-85 that was not initially present in the cladding gap. Assuming a complete Krypton-85 release from one cask with 10-year old fuel, NSP calculated a dose of 120 millirem at the site boundary.[34]

Q: Have you performed your own analysis of cask tip-over events?

A: Yes, I have performed a separate analysis of the "worst-case" consequences of a single cask breach accident because NSP's estimate considers only the release of Krypton-85, when in fact other materials might also be released. The NRC regularly estimates the consequences of ISFSI accidents in environmental assessments ("*EAs*") which it prepares in

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32 This number represents the frequency of the cask tip-over ground acceleration times the probability of all of the casks being defective simultaneously (8×10^{-5} times 2.2×10^{-2}, or about 1.8×10^{-6} per year).

33 NSP, Application, April 29, 1991, pages 126-129; NSP, Revised Application, June 10, 1991, pages 126-129; NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, page 8.2-17.

34 NSP, Revised Application, June 10, 1991, page 129.

support of the issuance of ISFSI licenses. I have estimated the radiological contents of a Prairie Island ISFSI cask based on data supplied by NSP (supplemented on generic data contained in NRC and EPRI publications).[35] This estimate is summarized in Table 1, at the end of Section II of this testimony. Then, using the methodology adopted by the NRC in its ISFSI EAs, I performed similar "*licensing basis*" calculations for Prairie Island, using the worst-case X/Q downwind dispersion value for the site boundary.[36] These results are set forth in Table 3 (at the end of Section II of this testimony), and indicate that the single cask event produces an estimated dose at the site boundary of 0.25 rem (i.e., 250 millirem).

In order to estimate the population dose resulting from such an event, I used X/Q data corresponding to a ground level release with a wind speed of 1 meter per second under conditions of moderate stability (generally thought of as the worst-case for radiological releases, corresponding to 5th percentile values) and 50-mile population data provided by NSP to calculate the population dose. Calculations were carried out for each of 16 compass sectors, with the resulting doses weighted by the probability of the wind blowing into each sector.[37] The weighted doses were then summed to obtain the average dose resulting from the postulated release.

For a single cask event, the average population dose is 193 person-rem. For perspective, the low and high values for a single cask event are 11 and 885 person-rem,

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35 U.S. Nuclear Regulatory Commission, Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel, NUREG-0575, Vol. 1, August 1979; NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984.

36 X/Q ("*chi over Q*") is a measure of dispersion of a pollutant downwind from a release point. The worst-case X/Q value was obtained from NSP's ISFSI Safety Analysis Report.

37 The wind rose data (probability of wind blowing into each compass sector) was obtained from Sandia National Laboratories, Technical Guidance for Siting Criteria Development, NUREG/CR-2239, November 1982, page A-24.

respectively.[38] On a person-rem per year basis, the seismically-initiated single cask event produces an annual risk of 0.002 person-rem per year.[39]

Multiplying the radiation release quantities for a single-cask event by 48 to account for all casks being involved at once (resulting in a dose of 12 rem at the site boundary), and performing population dose calculations as described above, the average population dose for the 48-cask event is 9,282 person-rem (with low and high values of 561 and 42,500 person-rem, respectively). On a person-rem per year basis, the seismically-initiated 48-cask event produces a risk of 0.02 person-rem per year.[40,41]

Q: Are there any other environmental externalities which could affect the ISFSI dry casks?

A: Yes. There are at least two other environmental externalities which could affect the casks -- flooding and aircraft crash.

Q: Please address the consequences of flooding events.

A: Concerning flooding, the Company states that for the 100-year flood water levels would remain below the base of the cask. For the so-called Probable Maximum Flood (PMF), water levels would rise above this level, but remain below the level of the cask seals. The

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38 These high and low values represent the highest and lowest calculated population dose for the 16 compass sectors surrounding the ISFSI site. The average value cited above reflects the contributions of all 16 sector values (more specifically, it is the sum of each of the 16 sector values, each weighted by the probability of the wind blowing into that sector).

39 This number is the product of the frequency of the single-cask seismic initiating event (8×10^{-6} per reactor-year) and the average population dose resulting from this event (193 person-rem).

40 This number is the product of the frequency of the 48-cask seismic initiating event (1.8×10^{-6} per reactor-year) and the average population dose resulting from this event (9,282 person-rem).

41 For perspective, even were one to assume that the probability of 0.37g earthquake (which would cause cask tip-over under NSP's analysis) were as high as 1×10^{-4} per year, and to assume that such an event would tip over and breach all 48 casks (which is a very conservative assumption as discussed above, equivalent to assuming that all 48 casks are defective with a probability of one), the risk on a person-rem per year basis would still be less than 1 person-rem per year (i.e., 9282 person-rem x 1×10^{-4} per year, or 0.9 person-rem per-year) based on these very conservative assumptions. Even going one step further and assuming that the highest estimate sector dose (42,500 person-rem) represents the consequence, the estimated risk (assuming the earthquake frequency is 1×10^{-4} per year and that all 48 of the casks are defective), the risk would be less than 5 person-rem per year. Of course, these assumptions are unrealistically conservative.

Company states flood water velocity would not cause the casks to tip over, that cask seal integrity would be maintained, and that no water would leak into the cask.[42] Even considering flood debris in the river, it does not appear possible to cause cask tip-over during a flood due to the cask's large mass and relatively low center of gravity.

Floods beyond the PMF are possible, although less likely than the PMF. The probability of the PMF at Prairie Island is uncertain, but generally the PMF has a frequency of about 1×10^{-6} per year.[43] There is considerable variability in the frequency of the PMF, however, as is illustrated by the case of the Three Mile Island Unit 1 plant (which is located on an island in the middle of the Susquehanna River about 10 miles south of Harrisburg). The PRA of that plant estimated the frequency of the PMF at 1×10^{-5} per year. An NRC-sponsored review of the TMI-1 PRA estimated the probability of the PMF at a higher value of 5×10^{-4} per year.[44,45]

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42 NSP, Revised Application, 6/10/91, page 57.

43 NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984, page 4-67.

44 Idaho National Engineering Laboratory, A Review of the Three Mile Island-1 Probabilistic Risk Assessment, NUREG/CR-5457, November 1989, pages 48-51.

45 NSP estimates that the flood which causes water to reach the bottom of the casks corresponds to a 1,000-year flood. NSP states, regarding the likelihood of floods larger than the PMF (NSP Response to DPS Information Request No. 204):

> *The probable maximum flood is the hypothetical flood that would result if all the factors that contribute to generation of the flood were to concurrently reach their most critical values that could occur. The probable maximum flood is derived from hydrometeorological and hydrological studies and is independent of flood frequency. It is the estimate of the boundary between possible floods and impossible floods. Therefore, it would have a return period approaching infinity and a probability of occurrence in any particular year approaching zero.*

While this is a textbook response to a question regarding the probability of the PMF, in fact floods previously estimated as the PMF have been exceeded in actual experience. This can occur as a result of a number of factors, such as rainfall at a greater rate than anticipated or greater runoff than anticipated (due to increased urban development, for example). As an illustration, the previously calculated PMF was exceeded at the Three Mile Island site in 1972 as a result of Tropical Storm Agnes. Accordingly, I adopt the position in this testimony that floods in excess of the PMF are possible. As can be seen from the results of my analysis, however, such floods (even when their likelihood is conservatively estimated) are not important contributors to risk.

NSP states that the storage casks are designed for an external pressure of 25 psi, which corresponds to a static head of water of approximately 56 feet.[46] Such a flood level is much more severe than the PMF (which would place water only 7 feet up the side of the 16 feet, 10 inch high cask).

Moreover, it should be recognized that a flood which places the casks under slightly more water than the design basis will not necessarily result in failure of the cask -- there is some factor of safety incorporated into the design of such containers such that there is high assurance that integrity will be maintained at the design pressure. In actuality, assuming that the cask is correctly designed, fabricated, and installed, the cask will survive a greater pressure (corresponding to a greater depth) without failing. This greater depth is not precisely known, but a factor of two or three margin against failure would not be surprising.

Lacking a precise frequency estimate for either the PMF or the design basis flood equivalent for the casks, I have performed what I consider to be a bounding analysis of the possible consequences of flooding events. For the purposes of a bounding estimate, I take the probability of the PMF to be 5×10^{-4} per year (1 chance in 2,000 per year), and assume that the PMF is of sufficient severity to breach defective casks, with the breach location above the water line to maximize the dose (otherwise, only Krypton-85 will be released to the atmosphere; the remaining materials will be dissolved in the river water which, being at such a large flow rate, will rapidly dilute the concentration of these materials to very low values). This results in a population dose of 0.9 person-rem. For a single defective cask case (at a conditional probability of 0.1, as estimated above), this works out on a person-rem per year basis to 0.00005 person-rem/year (5×10^{-4} flood probability, 0.1 probability of defective cask, 0.9 person-rem).

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[46] NSP, <u>ISFSI Safety Analysis Report</u>, Rev. 0, August 1990, page 3.2-7.

For a 48-defective cask case (at a conditional probability of 0.022 as discussed above), this works out to 0.0005 person-rem/year (5×10^{-4} flood probability, 0.022 all defective cask probability, 48 times 0.9 person-rem per cask, or 43.2 person rem). In reality, the probability of a flood sufficient to damage even defective casks will be very much lower in probability than 5×10^{-4} per year, and the actual annual risk from such events will be very much less than estimated above. For the purposes of this testimony, however, I will adopt the risk estimate calculated above for flooding events, recognizing that they represent a bounding case. Moreover, it is clear that the risk from the ISFSI in the flooding scenario discussed above would pale by comparison to the very considerable damage caused by the flood itself, which would certainly run to many millions of dollars (not to mention the impact of such a flood on the Prairie Island plant).[47]

Q: Please address the consequences of aircraft crash events at the ISFSI facility.

A: Aircraft crash onto the ISFSI pads could cause a cask to tip over, as acknowledged by the Company. The Company states, however, that most of the collision energy would be expended in disintegrating the aircraft, and that the cask would not be breached as a result of the collision.[48] This might be considered to be dispositive of the situation, however, NSP's analysis does not appear to have extended to a consideration of the impact of a post-crash fire on cask integrity.

Severe fires are acknowledged to have the potential to breach dry storage casks.[49] In any event, however, the likelihood of an aircraft crash at the ISFSI site is very low, as indicated by available aircraft crash statistics (sabotage or deliberate "*kamikaze*" crashes

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47 It should be noted that even if one postulates a frequency well above 5×10^{-4} per year for the extremely serious flood discussed in this testimony, the risk posed by such events would not be significant. Even if I were to postulated a flood frequency of 1 per year (which is clearly absurd), the risk in terms of person-rems per year would be a little over 1 person-rem per year (accounting for both single and multiple cask failure events).

48 NSP, Revised Application, 6/10/91, pages 58-59.

are not considered here, both because the probability of occurrence cannot be reasonably estimated and because the 16-foot high berm surrounding the ISFSI would make such an attack very difficult to perform successfully).

As indicated previously, the target area represented by the ISFSI is small, amounting to only a surface area of 15,552 square feet, or about 0.0006 square miles. The nearest airport to the ISFSI is the Red Wing airport, which is about seven miles from the ISFSI.[50] If we were to assume, for the sake of illustration, that a major airport was located instead at five miles from the ISFSI site or that the site was in the immediate vicinity of a heavily travelled airway, the crash rate for such circumstances is 1×10^{-4} per square mile per year for all aircraft.[51] Given the ISFSI site target area of 0.0006 square miles, this produces a crash rate at the ISFSI site of approximately 6×10^{-8} per year.[52]

Even this calculation ignores any consideration of the weight distribution of the aircraft and the size of aircraft required to crash at the ISFSI site in order to either tip-over a dry storage cask or cause a sufficiently large fire as to pose a threat to the integrity of the casks (it is likely that only a small fraction of the total crashes would involve a sufficiently large aircraft to accomplish this). Moreover, this calculation ignores the fact that the airport is actually 7 miles away, not 5 miles away, and that the airport serves only small aircraft. Finally, this calculation ignores the fact that there would be a distribution of values of the number of casks affected by the crash (it is not likely that all casks would be affected).

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49 Edwin L. Wilmot, Transportation-Accident Scenarios for Commercial Spent Fuel, Sandia National Laboratories, SAND80-2121, February 1981, page 17.

50 NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, page 2.2-1.

51 Argonne National Laboratory, Evaluation of Aircraft Crash Hazards Analyses for Nuclear Power Plants, NUREG/CR-2859, June 1982, page 2.

52 NSP has developed no independent estimates of the frequency of an aircraft crash at the ISFSI site (NSP Response to DPS Information Request No. 215).

In the case of cask failure resulting from a severe fire (or resulting from cask tip over aggravated by a severe fire), the radiological releases from the casks will be larger than the previous "*worst-case*" discussed above.[53] The previous case considered a radiological release resulting from impact damage. The current circumstances require the consideration of other release mechanisms.

I have performed a separate analysis of the "*worst-case*" consequences of an accident involving a very severe fire, which results in larger radiological release fractions than typically assumed by the NRC in their licensing basis calculations, particularly for Cesium-134 and Cesium-137 (which tend to dominate dose effects for ISFSI accidents). Performing the calculation using these larger release fractions,[54] I have calculated an accident dose for the aircraft crash/fire scenario of 118 rem whole body at the site boundary for a single cask. Assuming that all 48 cases are involved, this would raise the whole body dose at the site boundary to 5,664 rem. These doses are quite conservative because they are calculated assuming a ground level release with no plume due to lofting the hot air resulting from the fire which produces the release. In addition, these doses assume no emergency response (i.e., no sheltering, no evacuation, etc.).

I performed population-dose calculations for this case in the same manner as discussed earlier. The average population dose for the 48-cask aircraft crash/fire event is 4.4 million person-rem (with low and high values of 265,000 to 20.1 million person-rem). On an annual basis, however, this extremely severe event (it would be difficult to imagine

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53 NSP has stated that the ISFSI will fall under the existing fire protection capabilities of the Prairie Island nuclear station. NRC regulations require that a mininum fire brigade of 5 trained persons be onsite at all times. In addition, the Red Wing fire department responds to fires at the Prairie Island site (NSP Response to DPS Information Request No. 216).

54 Edwin L. Wilmot, Transportation-Accident Scenarios for Commercial Spent Fuel, Sandia National Laboratories, SAND80-2124, February 1981; and NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984.

circumstances which could lead to a more severe release from a dry storage ISFSI facility) results in an estimated risk of 0.26 person-rem per year.[55,56]

It is important to keep this risk estimate in perspective. The aircraft crash frequency portion of the risk estimate is very conservative -- it assumes that only large aircraft are involved and that an airport serving these aircraft is located five miles from the ISFSI site (or that a major air travel route passes over the site). Neither of these assumptions is accurate for the ISFSI site. The closest airport is seven miles away, and serves only small aircraft. Accordingly, the actual crash frequency at the ISFSI site for aircraft sufficiently large to result in the accident which I have postulated is much lower than 6×10^{-8} per year. Since such low frequency numbers begin to lack physical meaning and are very uncertain, I have adopted a conservative assumption here to calculate the risk arising from aircraft crashes at the ISFSI site. The resulting risk, even when calculated very conservatively as I have done, is a minor contribution to the overall risks posed by the ISFSI.

Q: Are there any other events which could produce a release from the ISFSI facility?

A: A sufficiently large explosion could in principle breach the dry storage casks. In addition, transporter accidents could have the potential to result in accidents involving the dry storage casks. Apart from these two events, I have not been able to identify any other events capable of breaching the dry storage casks which are credible for the Prairie Island site. In fact, some of the events discussed above are so conservatively presented as to stretch credibility.

In order to provide assurance that I have considered all possible events, I consulted the NRC-sponsored PRA Procedures Guide which contains a detailed listing of possible

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55 I have not included risk estimates for less severe fires (i.e., those involving fewer than 48 casks) since those events would produce lower levels of risk because the crash frequency does not change.

56 This estimate is the product of 4.4×10^6 person-rems and the crash frequency of 6×10^{-8} per year.

external events.[57] The listing of events in the PRA Procedures Guide, along with my comments as to their credibility as threats to the Prairie Island ISFSI, are provided as Table 6 (at the end of Section II of this testimony).

Q: Please discuss events involving explosions and their impact on the ISFSI.

A: Certainly one could ultimately postulate a sufficiently large explosion which would breach the casks. For example, in the extreme case a very large explosive device exploded nearby could breach the casks. The issue becomes one of credibility in terms of evaluating the likelihood (frequency) of such an explosion. Moreover, should some individual or entity gain access to such an explosive device, there is no reason to suppose that the Prairie Island ISFSI would necessarily represent a likely target. Indeed, there are other potentially higher consequence targets available for such purposes. (It is true, however, that the ISFSI represents a large radionulide inventory for such purposes, however so do other similar facilities, including existing reactors and reactor spent fuel pools.)

The most severe explosion (apart from a military attack or a severe sabotage event) would likely arise as a result of a munitions barge explosion in the river near the Prairie Island plant (2600 feet from the ISFSI). Such a barge explosion has been estimated by NSP to result in a peak overpressure of less than 2.5 psi, which is well below the cask design pressure of 25 psi.[58,59]

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57 American Nuclear Soceity andInstitute for Electrical and Electronics Engineers, PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, prepared for the U.S. Nuclear Regulatory Commission, NUREG/CR-2300, January 1983, pages 10-8 to 10-9.

58 NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, pages 3.2-14 and 8.2-2.

59 As in indication of just how conservative the ISFSI cask design is in terms of resistance to external pressure, the design basis external pressure of 25 psi corresponds roughly to the detonation of a one kiloton nuclear device at a distance of less than 550 feet (Samuel Glasstone and Philip J. Dolan, eds., The Effects of Nuclear Weapons, U.S. Department of Defense and U.S. Department of Energy, Third Edition, 1977, page 117).

No hydrocarbon fuel is stored at the ISFSI site. The quantity of fuel carried in the transporter vehicle is limited so that only a small fire of short duration would be possible. There are no other combustible sources located within the ISFSI security fence.[60] Accordingly, I have identified no explosion events as risk contributors in my analysis.

Q: Are there any possible accidents with possible offsite consequences associated with moving of the spent fuel from the existing spent fuel storage pool to the dry storage casks, transporting the casks to the ISFSI pads, and then preparing the spent fuel for shipment to a federal repository?

A: Yes. There are three types of accident events to be considered: accidents involving movement of spent fuel from the existing pool storage racks to the dry storage cask; accidents involving movement of the dry storage cask from the spent fuel pool to the transporter; and accidents involving the transport of the casks to the ISFSI pad and placement on the pad.[61]

Q: Would you please set forth your analysis of these accidents?

A: Certainly.

Spent Fuel Assembly Damage

It is possible that spent fuel assemblies could be mechanically damaged during the process of removal from their existing storage locations in the spent fuel pool racks and their movement (under water) to a dry storage cask. Such a process takes place entirely under water, and this significantly mitigates the consequences of accidents occurring during spent fuel handling operations in the spent fuel pool. Only one assembly at a time is

60 NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, page 3.3-16.

61 The reverse order of these actions will also have to be considered should it be necessary to move the spent fuel from the TN-40 dry storage casks to another cask design for transport to a federal

moved, so that the maximum hypothetical consequences for the accident would be the release of noble gases as a result of mechanical damage to a spent fuel assembly. Assuming drop of a 10-year old assembly and 100% release of Krypton-85, this results in a dose at the site boundary of 0.0011 rem as set forth in Table 7 (at the end of Section II). Performing population dose calculations as discussed above, this event results in a population dose of 0.9 person-rem (with low and high values of 0.05 to 3.9 person-rem, respectively).[62]

Little actual data concerning the failure rates for such activities is available. A study prepared for EPRI suggests a failure rate for single fuel assembly movements of 2 x 10^{-3} per movement.[63] Since there are up to 1920 fuel assemblies which could be loaded into the dry storage casks for the Prairie Island ISFSI (48 casks with 40 assemblies each), this suggests that four assemblies might be dropped during operations leading to the placement of the spent fuel in dry storage casks. It is important to recognize that these "*drop*" events involve quite a range of severity, including everything from a very short drop back into the spent fuel pool rack with no fuel damage whatsoever to a significant event resulting in mechanical damage to the fuel assembly. Considering that operations involving the ISFSI will take place over 20 years, I estimate the probability of such an event to be 0.2 per year (4 events in 20 years). On an annual basis, therefore, these events pose a risk of 0.9 person-rem x 0.2/year, or 0.18 person-rem/year.[64]

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repository. These actions will occur after the 20-year ISFSI license period, and at an undefined time period in the future.

62 NSP has calculated the doses arising from spent fuel handling accidents involving freshly discharged spent fuel (100 hours after shutdown for the highest rated assembly, assuming the breach of all fuel rods in the assembly). NSP estimates the dose at the site boundary from such an event to be 0.518 rem whole-body and 1.04 rem thyroid (NSP Response to DPS Information Request No. 208). These doses would be extremely conservative for the case considered here since I am discussing a fuel assembly which has been in the spent fuel pool for at least 10 years (87,600 hours, instead of 100 hours as in NSP's calculation).

63 NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984, page 4-17.

64 This is a conservative estimate of risk since not every cask drop event results in failure of the fuel cladding and release of Krypton-85 gas from the cladding gap. Even in those incidents where

Cask Drop With Spent Fuel Pool Structural Damage

Accidents involving dropping of the dry storage casks could be quite serious depending upon the circumstances of the accident. Studies sponsored by the NRC of such events indicate that there is a potential, should a dry storage cask be dropped from a significant height while in the pool area, for the spent fuel storage pool to be damaged, resulting in drainage of water from the pool and a resulting severe accident involving the spent fuel in the storage pool. NRC estimates that there is a 0.001 conditional probability of pool failure given a cask drop.[65] The frequency of cask drop accidents (from any height) is estimated to be a maximum of 1 x 10^{-5} per year for crane design which complies with NRC heavy load requirements.[66] Thus the frequency of cask drop accidents resulting in spent fuel pool severe accidents is 1 x 10^{-8} per year.

Few analyses have been done of spent fuel pool severe accidents, and performing a detailed site-specific analysis here is beyond the resources available for preparation of this testimony. The NRC has sponsored an analysis of spent fuel pool accidents which I will adopt as a reasonable approximation of the population dose arising from a cask drop accident resulting in spent fuel pool damage.[67] Using the same weighting factors set forth below for circumstances within 90 days of a refueling outage and beyond 90 days from a refueling outage, this estimation method yields a population dose of about 13.1 million

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some release occurs, not all of the fuel rods in the fuel assembly necessarily experience cladding failure. Indeed, the estimate is even more conservative considering that the consequence estimates assume a complete release of Krypton-85 (i.e., not just the cladding release, but the release of all Krupton-85 in the fuel).

65 Brookhaven National Laboratory, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82, NUREG/CR-4982, July 1987, pages 27-28.

66 NRC, Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36, NUREG-0612, July 1980, page 5-16.

67 U.S. Nuclear Regulatory Commission, Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools", NUREG-1353, February 1989, page 4-41.

person-rem. On an annualized basis, this results in an estimated risk of 0.13 person-rem/year.

Cask Drop Without Spent Fuel Pool Structural Damage

Even if the pool does not fail, however, dropping the cask into the spent fuel pool could result in the failure of spent fuel stored in the pool racks. NRC estimates that between 220 and 440 fuel assemblies could be mechanically damaged (i.e., by crushing) in such an accident.[68] Given that such an event would occur under water, the only releases should be noble gases for old fuel (primarily Krypton-85), and noble gases (Krypton-85 and isotopes of Xenon) and some iodine for relatively fresh fuel (NRC concludes that 99% of the iodine would remain in the pool water). For the purposes of this analysis, spent fuel is considered relatively fresh until it has remained in the pool for 90 days.[69] Assuming that the Prairie Island reactors are operated on an 18-month refueling interval, this works out to an average of 1.3 refuelings per year for the two reactors,[70] or a "*fresh fuel vulnerability period*" of 117 days per year, or about 32.5% of the time. NRC has already calculated the consequences of such accidents, so I will rely on NRC's calculations. I will use the whole body dose for the midpoint of the vulnerability period as representing the average conditions during the vulnerability period, or an exclusion area dose of 4.4 rem.[71] Assuming this dose may be propogated through population dose calculations as done previously, the average population dose is 3,403 person-rem (with low and high values of

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68 U.S. Nuclear Regulatory Commission, Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36, NUREG-0612, July 1980, page 2-6.

69 This is supported by analyses in NUREG-0612 which indicate that after 90 days, the consequences of load drop accidents at the exclusion area boundary become negligible due to radioactive decay of the iodine and xenon isotopes (NRC, Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36, NUREG-0612, July 1980, page 2-4).

70 With an 18-month refueling interval over 40 years, each plant would be refueled 26 times, for a total of 52 refueling events in 40 years. This averages out to 1.3 refuelings per year over the long term.

71 This is derived from 0.01 rem per assembly damaged, and assuming a maximum number of assemblies damaged (440 assemblies, according to NRC), and assumes no filters are present to reduce the iodine dose.

205 and 15,587 person-rem, respectively). These conditions obtain about a third of the time.

The rest of the time, the consequences would be limited to the Krypton-85 dose. Table 7 below contains a dose calculation for a single fuel assembly assuming 10-years cooling time and a complete release of Krypton-85. In order to obtain a dose estimate pertinent to the current situation, I multiplied by a factor of 440 to obtain results for 440 fuel assemblies (the maximum number damaged), and multiplied by a factor of 2 to account for a greater concentration of Krypton-85 (due to less decay time).[72] This yields an estimated dose of 0.001 rem/assembly x 440 assemblies x 2, or 1 rem. Again, propogating this dose through the population dose calculations, the average population dose is 773 person-rem (with low and high values of 47 and 3,543 person-rem, respectively). This dose would be obtained two-thirds of the time.

Weighting these results accordingly, I obtain an average population dose of 1,536 person-rem. On an annual basis, this yields a risk of 0.015 person-rem per year (for cask drop events in which the pool is not structurally damaged).[73]

Cask Drop Outside the Spent Fuel Pool

A cask could also be dropped outside the spent fuel pool. The cask drop frequency, as explained above, remains at 1×10^{-5} per year. NSP has stated that the cask should survive the 50-60 foot drop from the crane to the auxiliary building floor, although it would sustain minor damage. NSP concludes that at most a somewhat elevated seal leakage rate could occur, but that this was acceptable.[74] Assuming, as before, that there is a 0.1 conditional probability of a defective cask, and that this would result in cask breach, we

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72 The half-life of Krypton-85 is 10.7 years (NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984, page 5-5).

73 This estimated risk is conservative since it assumes that 440 fuel assemblies are damaged in every case.

would have a consequence of 193 person-rem at a frequency of 1×10^{-6} per year, or a risk contribution of 0.0002 person-rem per year.

Dry Cask Transporter Accidents

Very little analysis has been documented concerning the frequency of accidents involving dry cask transporters. A study prepared for EPRI in 1984 identified five scenarios of concern: (a) extreme weather; (b) venting of the transport cask without fuel failure; (c) collisions during transport without fires; (d) fires involving the transport vehicle, with or without collisions; and (e) accidents involving placement of the cask on the storage pad.[75]

The EPRI study judged that extreme weather scenarios were negligible contributors during transport operations. I concur since it is unlikely that transport of a dry storage cask would be attempted during actual or threatening severe weather. Similarly I judge that spurious venting of the cask without a collision during transport is very unlikely. Moreover, unless there is fuel failure, the consequences of such an event would be limited to localized contamination and the dose consequences offsite would be extremely small.

Accidents during transport are a possibility. The EPRI study estimated the frequency of accidents without fires to be 5×10^{-5} per trip,[76] and estimated that the conditional probability that the accident would involve a significant fire is 0.01.[77,78] Given

74 NSP Response to DPS Information Request No. 21.

75 NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984, pages 4-19, 4-25, and 4-28.

76 Statistics from the early 1970s suggest a much lower accident rate for trucks -- 6×10^{-9} per vehicle mile with fires involved (AEC, Environmental Survey of Transportation of Radioactive Materials to and From Nuclear Power Plants, December 1972, pages 65-66).

77 NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984, pages 4-20, 4-21, and 6-3.

78 This statistic (1% of accidents involving fires) is consistent with truck accident statistics cited in an early U.S. Atomic Energy Commission ("AEC", the predecessor agency of the NRC and Department of Energy) study of transportation of radioactive materials (AEC, Environmental Survey of Transportation of Radioactive Materials to and From Nuclear Power Plants, December

that there will be a maximum of 96 trips over the 20-year lifetime of the ISFSI license (48 round trips),[79] this yields an average probability per year of 2.4 x 10^{-4} per year for an accident without a fire and an average probability of 2.4 x 10^{-6} per year for an accident with a fire. It should be noted that the EPRI results were calculated assuming a transport distance of one mile,[80] whereas the transport distance for Prairie Island is 0.45 mile.[81] To the extent that the frequency of transporter accidents is distance dependent, this makes the use of the EPRI data conservative for Prairie Island. In addition, as noted early, NSP states that the amount of hydrocarbon fuel carried by the transporter vehicle is limited so that only a small fire of short duration would be possible (unless, of course, a collision occurs with another vehicle carrying a larger load of combustible materials).

The dose consequences of transporter accidents are approximated by assuming that the accident without a fire is bounded by the NRC licensing basis value for impact releases, or a population dose of 193 person-rem (see previous discussion of seismic events for details of this single-cask dose calculation). On an annualized basis, this results in a dose consequence of 0.046 person-rem/year for transport accidents without fires. For transport accidents with fires, it will be assumed that such accidents are bounded by the worst-case results in Table 2 below, or a dose of 118 rem. The population dose in this case is calculated using the same methods as discussed before, with a result of 91,200 person-rem. On an annualized basis, this results in a dose consequence of 0.22 person-rem/year for transport accidents with a fire.

1972, pages 65-66). The AEC study notes, moreover, that most fires involve only the fuel from the transport vehicle fuel tank and 89% of the fires last less than 30 minutes (another 10% last 30-60 minutes, and less than 1% last longer than 60 minutes).

79 Round trips will be necessary since the casks will be transported back to the plant for placement in a shipping cask or for placement on the shipping vehicle if the TN-40 cask is later certified for transport.

80 NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984, page 4-20.

81 Minnesota Environmental Quality Board, Final Environmental Statement, Prairie Island Independent Spent Fuel Storage Installation, April 12, 1991, page 4.20.

Cask Placement Accidents

Accidents during placement on the ISFSI pad are another possibility. The EPRI study evaluated the likelihood of such events based on an assumed number of movements per year, calculating an annual probability of occurrence. This presentation requires modification to the specific circumstances of the Prairie Island ISFSI. There will be 48 operations involving placement of a cask on the storage pad, and 48 involving removal. The EPRI study calculated a probability of 1×10^{-5} per year for an accident during such an operation,[82] based on an assumption that 60 fuel assemblies would be moved to dry storage annually, following five years of storage. The EPRI study considered a cask design holding 24 fuel assemblies, thus the EPRI estimate appears to be based on an assumption of 3 operations per year. To obtain the conditional probability per operation, I divide the rate of 1×10^{-5} per year by 3, yielding 3.3×10^{-6} per operation. With 96 operations at Prairie Island over 20 years, this results in an average of 5 operations per year, or an annual frequency of 1.7×10^{-5} for a cask placement accident. The EPRI study concluded that if a cask is dropped or knocked against another cask already in storage, little should happen unless one or both of the casks is out of design specification; this possibility is included in the estimate above. However, assuming for the purpose of illustration that two casks are involved (one cask is knocked against another during placement), and the dose is that calculated for NRC licensing basis purposes, the dose would be 0.5 rem (for two casks). Thus, the population dose would be double that calculated above for a single cask accident, or about 386 person-rem. On an annualized basis, this results in a dose consequence of 0.006 person-rem/year.

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82 NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984, page 4-28.

II.E Cost Quantification

Q: Combining your previous results, do you have an estimate of the annual person-rems of exposure resulting from operation of the Prairie Island ISFSI?

A: Yes. Summarizing the above results (as set forth in Table 4, at the end of Section II), the total dose consequence per year (for normal operation and accidents) is conservatively estimated at 2 person-rem per year.[83] Quantifying this value at a rate ranging from $1,000 to $10,000 person-rem per year,[84] I obtain an annual cost of $2,000 to $20,000 per year arising from normal operation and a range of postulated accidents (both likely and unlikely).

II.F Risk Comparison

Q: Some of the accidents which you have evaluated have rather large dose consequences, in the range of hundreds of thousands to millions of person-rems. Would you please place these accidents into perspective in comparison with other radiological accidents?

A: Certainly. There are two key factors to keep in mind when reviewing the results calculated above. First, the calculations are crude and very conservative in most cases. Second, many of the events evaluated have very low probabilities of occurrence.

Notwithstanding this, however, there are radiological accidents associated with nuclear power plant operation with more significant consequences and at higher probabilities which have been identified in other studies. For example, NRC-sponsored studies of spent fuel pool storage accidents have identified the possibility that seismic events could lead to a loss of pool water and an accident scenario involving a spent fuel

83 For the purpose of comparison, the 42,000 residents of Goodhue County (in which the Prairie Island facility is located) collectively receive approximately 8,400 person-rem per year from background radiation (i.e., from environmental sources other than the nuclear power industry).

84 *See* discussion in Section II.B, above, for the derivation of these cost values.

cladding fire. Such accidents, because they involve large quantities of long-lived radionuclides (especially cesium and strontium species) can have significant consequences. NRC estimates that a spent fuel pool accident could result in population doses ranging from 8-26 million person-rem (depending upon the timing of the accident with respect to how recently refueling occurred). These consequences account for the exposure within a 50-mile radius of the reactor. It should be observed, however, that such accidents are not as improbable as one might initially believe (particularly compared to some of the accident probabilities considered in this testimony). NRC estimates for two plants indicate that seismically-initiated spent fuel pool accidents have likelihoods in the range of 2×10^{-6} to 7×10^{-6} per reactor-year. Other potential causes of severe spent fuel pool accidents total approximately 1.5×10^{-7} per reactor-year.[85]

Whether the Prairie Island results (were they to be calculated) would be higher or lower than these values is not currently known. However, a comparison of the seismic hazard curves for the plants in the NRC study and the curves for Prairie Island indicates that the curves are similar. Thus, unless the Prairie Island plant is less able to withstand earthquakes, the seismically-initiated portion of the spent fuel pool accident risks should be equal to or less than those calculated by the NRC for the Vermont Yankee and H.B. Robinson facilities. Assuming this is true, it is worth observing that the worst case/high consequence accident arising from the ISFSI (aircraft crash at the ISFSI site with a large fire, resulting in the breach of all 48 casks and an enhanced release of radioactivity due to the fire) is a probability of about 100 less than the probability of the large consequence spent fuel pool accidents.

Comparing the risks on a person-rem per year basis provides additional insight. The risk posed by spent fuel pool accidents can be approximated by using the NRC-estimated

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85 U.S. Nuclear Regulatory Commission, Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools", NUREG-1353, February 1989, page 4-

consequence range of 8-26 million person-rem at a spent fuel pool accident frequency of 4 x 10^{-6} to 8 x 10^{-6} per year, or a risk estimate range of 32 to 208 person-rem per year. This level of risk is significantly higher than what I have estimated for the Prairie Island ISFSI -- 2 person-rem per year. This comparison indicates that dry storage has a significant safety margin compared with pool storage of spent fuel.[86]

Reactor accidents have been calculated to result in population doses ranging from several million to about 100 million person-rem,[87] depending upon the severity of the accident, the population density in the region of the facility, and other variables. Such very severe accidents have estimated likelihoods of the order of 9 x 10^{-7} to 5 x 10^{-5} per year.[88] Using these release frequencies as a guide, and assuming a range of consequences between 5 million and 100 million person-rem, resulting in annual risk estimates spanning the range from 5 to 5,000 person-rem per year.[89] The ISFSI large consequence accident frequency is less (by a factor of ten and probably a great deal more) than is the case for current generation reactors (6 x 10^{-8} for the aircraft crash/fire scenario, which is itself a conservative overestimate, compared with the range of 9 x 10^{-7} to 5 x 10^{-5} for large reactor

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86 The numbers used above indicate a safety factor of 16-104, but the actual safety margin is probably greater since the Prairie Island ISFSI risk estimate is very conservatively calculated compared with the spent fuel pool risk estimate).

87 For perspective, estimates of the population dose due to the 1986 Chernobyl accident range from 60-120 million person-rem (Gordon Thompson, Institute for Resource and Security Studies, Cambridge, Massachusetts, personal communication, August 1991).

88 This is the range of large release frequencies from the recent NRC five-reactor risk assessment study, NUREG-1150. These results are summarized in a paper presented by an MHB colleague; see, Steven C. Sholly, "*Driving Forces Shaping Advanced Reactor Designs: Near-Term and Long-Term Prospects*", in Proceedings of the First MIT International Conference on the Next Generation of Nuclear Power Technology, Massachusetts Institute of Technology, Cambridge, Massachusetts, MIT-ANP-CP-001, proceedings published June 1991, page 1-17.

89 The actual person-rem per year estimates for the five NUREG-1150 plants span a range from 6 to 296 person-rem per year. At the low end of this spectrum, however, the PRA of plant involved (Grand Gulf) did not include external events (such as earthquakes, fires, floods, etc.) The Department of Energy has prepared a risk assessment of the K-Reactor, which is a tritium production reactor. The risk posed by the K-Reactor is estimated by DOE to be 900 person-rem per year (U.S. Department of Energy, Final Environmental Impact Statement: Continued Operation of the K-, L-, and P-Reactors, Savannah River Site, Aiken, South Carolina, DOE/EIS-0147, December 1990, page 4-98).

accidents), and there is a comfortable margin (perhaps very large) between the risks of ISFSI accidents and the risks of reactor accidents estimated on a per year basis.

It requires multiple and extremely conservative assumptions to produce large consequences (millions of person-rem) for a dry spent fuel storage installation -- that is, a crash of a very large aircraft directly on the site, a very large fire following the crash, involvement of all 48 casks, conservative dose estimation procedures, <u>and</u> a complete lack of emergency response. In contrast, the sets of circumstances required to produce consequences of this order as a result of reactor accidents are not so far fetched, and are routinely evaluated in nuclear power plant risk assessment studies.

Q: Are there any methods of storing spent reactor fuel apart from pool storage and dry cask storage which have been described in the literature?

A: Yes. An EPRI report describes conceptual designs for vault and caisson storage systems for use at individual reactor sites.[90] In the vault storage concept, spent fuel assemblies are stored inside a sealed metal cannister that is placed inside a concrete vault or canyon. Cooling is provided by internal air convection with heat transferred through the concrete to the ambient air by heat pipes. In the caisson storage concept, each fuel assembly is sealed in a metal cannister and placed in a steel-lined concrete well or caisson. Cooling is provided by conduction through the ground to the air at the ground surface.

In addition, it is possible to store spent fuel from a number of plants at a Monitored Retrievable Storage (MRS) facility. Indeed, the possibility of constructing an MRS is recognized in the 1987 Nuclear Waste Policy Act amendments.

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90 NUS Corporation, <u>Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment</u>, EPRI-NP-3365, February 1984.

Q: Are there risk estimates available for vault and caisson storage systems?

A: Yes, but only comparative risk estimates are available. The EPRI report indicates that the vault storage concept poses a similar level of risk compared with dry cask storage, while caisson storage poses less risk of high consequence accidents resulting from earthquakes than either dry cask or vault storage.[91] Briefly considering vault and caisson storage based on the results of my testimony, I would observe that these concepts might be less vulnerable to high consequence aircraft crash accidents compared with dry cask storage. However, it should be observed that the aircraft crash accident has such a very low likelihood that it would make little difference to the risk estimates if vault or caisson storage concepts were utilized at Prairie Island.

Q: Are risk estimates for an MRS available?

A: I have not seen any such estimates, but I have not searched extensively for such documentation in preparing this testimony.

Q: Based on the analysis which you have prepared for this testimony, do you have any perspective on what risks might be posed by an MRS?

A: Yes. In the broadest sense, an MRS trades off having a number of smaller facilities, each with varying degrees of risk posed by externalities such as earthquakes and aircraft crash, for a single larger facility. Provided that the location of a geological repository is known and that the MRS is located at or near the repository,[92] it might be possible to achieve some overall societal risk reduction if the MRS site is far from an airport and if the MRS site has a low seismic hazard. The validity of this observation depends upon the degree to

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91 NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984, page S-3.

92 Otherwise one might actually be increasing the risk posed by transportation accidents.

which transportation risks for spent fuel are important and, if they are, the degree to which these transportation risks are impacted by the age of the spent fuel.[93]

93 It might be, for example, that if transportation accidents posed a significant fraction of the risk, these risks could be lessened by waiting until the geologic repository is in operation to ship the spent fuel, since the spent fuel will be older than if it is shipped to an MRS facility which would presumably be in operation sooner than the geologic respository.

TABLE 1

Radionuclide Content of Prairie Island Dry Spent Fuel Storage Cask Assuming 10-Year Cooling Period Before Storage (in Curies)

Nuclide	Content of 1 Fuel Assembly	Content of 40 Fuel Assemblies
H-3 (tritium)	162	6,480
Kr-85	2,379	95,160
Sr-90	28,538	1,141,520
Ru-106	298	11,920
Cs-134	8,900	356,000
Cs-137	112,000	4,480,000
I-129	0.02	0.8

TABLE 2

Worst-Case Dry Spent Fuel Storage Radiological Dose Analysis (Using SAND80-2124 Release Fractions for Burst and Oxidation Release Mechanisms)*
(Whole Body Exposure)

Nuclide	Cask Quantity (μCi)	Respir. Aerosol Release Fraction	X/Q (sec/m^3)	Breathing Rate (m^3/sec)	Whole Body Inhal. Dose Convers. Factor (Rem/μCi)	Dose at Controlled Area Boundary (Rem)
H-3	6.48E9	5E-1	1.40E-3	2.54E-4	1.20E-4	0.14
Kr-85	9.52E10	5E-1	1.40E-3	N/A	3.34E-10**	0.022
I-129	8E5	5E-1	1.40E-3	2.54E-4	1.80E-1	0.25
Cs-134	3.56E11	2.2E-3	1.40E-3	2.54E-4	4.40E-2	12
Cs-137	4.48E12	2.2E-3	1.40E-3	2.54E-4	3.00E-2	105
Sr-90	6.89E11	2E-6	1.40E-3	2.54E-4	1.3	0.64
Ru-106	1.2E10	2.2E-6	1.40E-3	2.54E-4	4.70E-1	0.0044
Rh-106	4.3E11***	2E-6	1.40E-3	N/A	4.3E-8**	5.2E-5
Co-60	1.1E9***	2.5E-2	1.40E-3	N/A	6.0E-7**	0.023
Y-90	1.7E12***	2E-6	1.40E-3	N/A	0**	0
Pu-238	7.7E11***	2E-6	1.40E-3	N/A	5.3E-11**	1.1E-7
Pu-239	9.1E9***	2E-6	1.40E-3	N/A	2.3E-11**	5.9E-10
Pu-241	3.0E12***	2E-6	1.40E-3	N/A	4.2E-16**	3.5E-12
Cm-244	4.5E10***	2E-6	1.40E-3	N/A	1.4E-9**	1.8E-9
Ce-144	3.0E11***	2E-6	1.40E-3	N/A	4.3E-9**	3.6E-6
TOTAL	---	---	---	---	---	118

TABLE 2

Worst-Case Dry Spent Fuel Storage Radiological Dose Analysis (Using SAND80-2124 Release Fractions for Burst and Oxidation Release Mechanisms)*
(Whole Body Exposure)

(continued)

* Assumes severe impact followed by severe fire
** Units of Rem-m^3/sec-Ci
*** Estimated from EPRI study by NUS (NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984, page A-23); linearly scaled from 33,000 to 45,000 MWD/MTU; multiplied by 40 to account for the number of assemblies (40); converted to μCi (i.e., multiplied by 54.5 and converted μCi)

All release fractions as per NUS, Table A-14, page A-23, except tritium, iodine, and actinides. Tritium and iodine are assumed to be released to the same extent as krypton. Actinide release fractions as per Wilmot (Edwin L. Wilmot, Transportation-Accident Scenarios for Commercial Spent Fuel, Sandia National Laboratories, SAND80-2121, February 1981, Table III, page 4).

Dose conversion factors from NRC licensing basis calculations and from NUS (NUS Corporation, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, EPRI-NP-3365, February 1984, page A-9).

TABLE 3

NRC Licensing Basis Dry Spent Fuel Storage Radiological Dose Analysis (Using SAND80-2124 Release Fractions for Impact Release Only Mechanism) (Whole Body Exposure)

Nuclide	Cask Quantity (μCi)	Respir. Aerosol Release Fraction	X/Q (sec/m^3)	Breathing Rate (m^3/sec)	Whole Body Inhal. Dose Convers. Factor (Rem/μCi)	Dose at Control. Area Boundary (Rem)
H-3	6.48E9	3E-1	1.40E-3	2.54E-4	1.20E-4	0.086
Kr-85	9.52E10	3E-1	1.40E-3	N/A	3.34E-10*	0.013
I-129	8E5	3E-1	1.40E-3	2.54E-4	1.80E-1	0.15
Cs-134	3.56E11	5E-10	1.40E-3	2.54E-4	4.40E-2	2.8E-6
Cs-137	4.48E12	5E-10	1.40E-3	2.54E-4	3.00E-2	2.4E-5
Sr-90	6.89E11	5E-10	1.40E-3	2.54E-4	1.3	1.6E-4
Ru-106	1.2E10	5E-10	1.40E-3	2.54E-4	4.70E-1	1.0E-6
TOTAL	---	---	---	---	---	0.25

* Units of Rem-m^3/sec-Ci

TABLE 4

Summary of Dose Consequences Arising from Normal Operation And Postulated Accidents: Prairie Island ISFSI Activities

Dose Source	Dose Consequence (person-rem/yr)
NORMAL OPERATIONS	
Normal ISFSI Operations (Dose to Public)	0.007
Normal ISFSI Operations (Dose to Site Personnel)	1.11
NORMAL OPERATIONS SUBTOTAL	1.12
POSTULATED ACCIDENTS	
Earthquakes	
Single-Cask Events	0.002
48-Cask Events	0.020
Aircraft Crash W/Fire	0.26
Severe Floods	
Single-Cask Events	0.00005
48-Cask Events	0.0005
Tornadoes	Negligible
Fuel Assembly Drop	0.18
Cask Drop (Pool Damaged)	0.13
Cask Drop (Fuel Damaged)	0.015
Cask Drop (Outside Pool)	0.0002
Cask Transport Accident (Without Fire)	0.046
Cask Transport Accident (With Fire)	0.22
Cask Placement Accident	0.006
POSTULATED ACCIDENTS SUBTOTAL	0.88
GRAND TOTAL, ALL CAUSES	**2.00**

TABLE 5

Typical Radionuclide Content of PWR Spent Fuel Irradiated to 33,000 MWD/MTU and Cooled for 10 Years (NUREG-0575, Tables G.6, G.9 & G.12; Units of Curies per Metric Ton of Uranium; Species With More Than One Curie/MTU)

Radioactive Species	**Activity After 10 Years Decay (Ci/MTU)**
FISSION PRODUCTS	
Tritium (H-3)	3.23E+02
Krypton-85	5.02E+03
Strontium-90	6.15E+04
Y-90	6.15E+04
Zirconium-93	2.93E+00
Niobium-93m	1.35E+00
Technicium-99	1.43E+01
Ruthenium-106	5.70E+02
Rhodium-106	5.70E+02
Cd-113m	2.19E+01
Sb-125	7.85E+02
Tellurium-125m	1.92E+02
Iodine-129	3.33E-02
Cesium-134	1.09E+04
Cesium-137	8.66E+04
Barium-137m	8.19E+04
Cerium-144	1.62E+02
Pr-144	1.62E+02
Pr-144m	1.94E+00
Pm-147	6.86E+03
Sm-151	1.08E+03
Europium-152	6.23E+00
Europium-154	6.76E+03
Europium-155	7.50E+02
FISSION PRODUCTS SUBTOTAL	3.26E+05
TRANSURANICS	
Uranium-234	1.05E+00
Uranium-237	2.33E+00
Neptunium-239	2.03E+01
Plutonium-238	3.19E+03
Plutonium-239	3.75E+02
Plutonium-240	4.64E+02
Plutonium-241	9.71E+04
Plutonium-242	1.85E+00

TABLE 5

Typical Radionuclide Content of PWR Spent Fuel Irradiated to 33,000 MWD/MTU and Cooled for 10 Years (NUREG-0575, Tables G.6, G.9 & G.12; Units of Curies per Metric Ton of Uranium; Species With More Than One Curie/MTU)

Radioactive Species	Activity After 10 Years Decay (Ci/MTU)
Americium-241	2.11E+03
Americium-242m	1.31E+01
Americium-242	1.31E+01
Americium-243	2.03E+01
Cm-242	1.07E+01
Cm-243	3.92E+00
Cm-244	1.55E+03
TRANSURANICS SUBTOTAL	1.05E+05
LIGHT ELEMENTS AND MATERIALS OF CONSTRUCTION	
Iron-55	1.25E+02
Cobalt-60	1.90E+03
Ni-59	3.36E+00
Ni-63	4.58E+02
Sb-125	3.13E+00
Tellurium-125m	1.30E+00
LIGHT ELEMENTS AND MATERIALS OF CONSTRUCTION SUBTOTAL	2.50E+03

* Iodine-129 listed despite being present in less than one curie quantity due to its radiological importance.

TABLE 6

Summary of External Events Considered in Risk Assessment Studies and Their Applicability to the Assessment of the Risks Posed by the Prairie Island ISFSI

External Event	Remarks
Aircraft Crash	Considered in this testimony
Avalanche	Not credible; cannot occur close enough to the ISFSI or with sufficient severity to affect it
Coastal Erosion	Not credible; cannot occur quickly enough to be a realistic threat to the ISFSI
Drought	Not credible; no impact on the ISFSI which does not rely on water for cooling
External Flooding	Considered in this testimony
Extreme Winds/Tornadoes	Considered in this testimony
Fires	Considered in this testimony
Fog	Only possible impact is on frequency of aircraft and transporter accidents; such impacts presumed to be reflected in crash statistics
Forest Fires	Not credible; no means of spread of forest fire to ISFSI due to lack of combustibles within ISFSI facility and presence of 16-foot high berm around the facility
Frost	Not considered; within cask design basis
Hail	Not considered; less severe than tornado missiles which are within cask design basis
High Tide, High Lake Level, or High River Level	Included within definition of external flooding
High Summer Temperature	Not considered; within cask design basis
Hurricane	Not considered; not credible for Prairie Island site; effects considered under flooding and toranadoes; wind speeds within cask design basis

TABLE 6 (continued)

Summary of External Events Considered in Risk Assessment Studies and Their Applicability to the Assessment of the Risks Posed by the Prairie Island ISFSI

EXTERNAL EVENT	REMARKS
Ice Cover	Ice blockage included within definition of flooding; ice on cask within cask design basis
Industrial or Military Facility Accident	Not considered; none sufficiently close as to pose possibility of exceeding cask design basis (NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, page 2.2-1)
Internal Flooding	Not considered; cask design precludes internal flooding
Landslide	Not considered; not credible for Prairie Island site
Lightning	Not considered; within cask design basis
Low Water Level	Not considered; irrelevant since ISFSI does not rely on water cooling
Low Temperature	Not considered; within cask design
Meteorite	Not considered; extremely low probability
Pipeline Accident	Not considered; none close to plant (NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, page 2.2-1)
Intense Precipitation	Included under definition of external flooding
Railroad Accident	Not considered; encompassed within effects of river barge explosion shown not to impact ISFSI (NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, pages 2.2-1 to 2.2-2)
Release of Chemicals	Not considered; not credible for Prairie Island site
River Barge Accident	Not considered; within cask design basis for 1.4 kiloton explosion at 2600 feet
River Diversion	Not considered; irrelevant (NSP, ISFSI Safety Analysis Report, Rev. 0, August 1990, pages 2.2-1 to 2.2-2)
Sandstorm	Not considered; not credible for Prairie Island site

TABLE 6 (continued)

Summary of External Events Considered in Risk Assessment Studies and Their Applicability to the Assessment of the Risks Posed by the Prairie Island ISFSI

EXTERNAL EVENT	REMARKS
Seiche	Not considered; included under definition of external flooding
Seismic Activity	Considered in this testimony
Snow	Not considered; within cask design basis
Soil Stability	Not considered; considered in ISFSI pad design basis
Storm Surge	Not considered; included within definition of external flooding
Transportation Accidents	Considered in this testimony
Tsunami	Not considered; not credible for Prairie Island site
Toxic Gas	Not considered; irrelevant
Turbine Missiles	Not considered; ISFSI too far from turbine to be of concern
Volcanic Activity	Not considered; not credible for Prairie Island site
Waves	Not considered; included under defintion of external flooding

TABLE 7

Site Boundary Dose Calculation
For Single Rod Drop Accident

Nuclide	Fuel Assembly Quantity (μCi)	Respir. Aerosol Release Fraction	X/Q (sec/m^3)	Breathing Rate (m^3/sec)	Whole Body Inhal. Dose Convers. Factor (Rem/μCi)	Dose at Control. Area Boundary (Rem)
Kr-85	2.38E9	1	1.40E-3	N/A	3.34E-10*	0.0011

* Units of Rem-m^3/sec-Ci

III. ADDITIONAL ISFSI RISK CONSIDERATIONS

III.1. ISFSI Risk Implications of Plant Life Extension

Q: What is "plant life extension" in the context of Prairie Island?

A: Plant life extension refers to the possibility of operating a nuclear power plant (such as Prairie Island) for a period of time beyond the 40-year operating license granted by the NRC. Under the provisions of NRC regulations (specifically 10 CFR §50.51), a license can be issued by the NRC for a duration not to exceed 40 years. Licenses can be renewed under the provisions of 10 CFR §54.31 for an additional period up to 20 years beyond the expiration date of the initial license. In addition, if a license renewal application is made at least five years prior to the expiration of the initial license, the existing license is not deemed to have expired until the renewal application has been finally determined.[94] Thus, some period of extended operation could result even if a license renewal application is ultimately denied by the NRC.

Q: What would be the impact on spent fuel generation of operation of Prairie Island for an additional 20 years beyond the initial license period?

A: Roughly speaking, I would expect the total quantity of spent fuel to increase by 50%. Thus, assuming storage of this spent fuel in TN-40 dry casks (as NSP has requested for the existing spent fuel and the spent fuel to be generated during the remainder of the 40-year licenses for Prairie Island Units 1 and 2), this would increase the number of casks from a total of 48 to a total of 72 casks. In addition, the ISFSI license would have to be renewed (or a new ISFSI license issued for the additional 24 casks) in order to account for the longer period of power generation.

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94 10 CFR §2.109(b); <u>see</u>, NRC memorandum dated 15 May 1991 from James M. Taylor to The Commissioners (SECY-91-138), Subject: "*Final Rule on Nuclear Power Plant License Renewal*", Enclosure 1.

Q: If stored as you have indicated, what impact would this additional spent fuel have on risk?

A: The principal impact on risk would be to increase risk due to the increased amounts of radioactive materials potentially available for release to the environment in the event of ISFSI accidents and/or due to the increased number of operations involving the casks. The increase would not be an automatic 50% increase in every case, however, since radioactive decay during the additional 20-years of operation of the plant would reduce the risk from the spent fuel already put into ISFSI storage before the license extension period is entered. However, the risk increase can be bounded by simply (but conservatively) assuming a 50% increase in risk per year for the license extension period. This would increase the risk to 3 person-rems per year, or a cost of $3,000 to $30,000 per year for the license extension period.

III.2. <u>ISFSI Risk Implications of Cask Storage Beyond 20 Years</u>

Q: Are there any risk implications of storage of spent fuel in the dry casks beyond the 20-year license period of the ISFSI?

A: Yes. At a minimum, the period over which risks are incurred increases beyond 20 years. At the same time, however, radioactive decay of the spent fuel means that risk will be on a general decline.[95]

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95 The dose calculations which I have performed in support of this testimony clearly indicate that the dose (and the risk) from the worst ISFSI accidents is governed by doses resulting from exposure to Cesium-134 and Cesium-137. Cesium-134 has a half life of 2.06 years; Cesium-137 has a half life of 30.1 years. For the Prairie Island spent fuel at 10 years decay, these radionuclides account for 99% of the dose. Thus, as time goes on, and Cesium-135 decays away, doses from the most severe ISFSI accidents are largely controlled by Cesium-137 and are thus reduced by roughly half every 30 years.

Similarly, the doses for lesser ISFSI accidents are governed by the doses resulting from exposure to Iodine-129, Tritium (H-3), and Krypton-85. Tritium has a half life of 12.3 years; Krypton-85 has a half life of 10.76 years; Iodine-129 has a very long half life (17 million years). After 30 years, Iodine-129 will dominate doses from the less severe ISFSI accidents, but the total dose will have declined by about 40%.

There is an additional consideration, however. The casks are exposed to environmental conditions for an additional period, which theoretically increases the possibility of cask failure. The most vulnerable aspect of the casks as a result of aging is the cask seals. It is unlikely that the casks themselves would be vulnerable to failure due to aging for periods considerably beyond the 20-year duration of the ISFSI license. The seals will be subject to inspections during ISFSI operation and could, if necessary, be replaced by transporting the casks from the ISFSI to the spent fuel pool, removing the old seal, replacing the seal with a new seal, and transporting the cask back to the ISFSI pad. This would entail some increase in risk due to additional cask movements and additional movements of the casks over the spent fuel pool.

The increase in risk could be bounded by simply doubling those risks associated with cask movement over the pool, cask transport, and cask placement on the ISFSI pad. Doing so (referring to Table 4 at the end of Section II of my testimony), I obtain an additional risk of about 0.6 person-rem per year.[96] This is not a significant impact (i.e., it would increase risk from 2 to 2.6 person-rem/year).

Q: If one nevertheless assumes that a cask fails during storage on the ISFSI pad, what could be done to mitigate such an accident?

A: Without performing a detailed study of the matter, I believe that substantial steps could be taken to mitigate the risks posed by such a hypothetical accident. It is important to note, as a preamble to this discussion, that failure of the cask does not necessarily imply failure of the fuel cladding or complete emptying of the cask contents onto the concrete pad. Thus, the consequences arising from such an event could be quite minor (as trivial as an increased leak rate from the cask).

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96 In obtaining this estimate, I doubled the risks associated with Cask Drop (Pool Damaged), Cask Drop (Fuel Damaged), Cask Transport Accident (Without Fire), Cask Transport Accident (With Fire), and Cask Placement Accident.

Hypothetically assuming the worst, however, and postulating fuel assemblies on the concrete ISFSI pad in the open air, it seems evident that some form of shielding could be erected around the spent fuel to reduce dose to plant personnel and the public from direct (shine) exposures. Such shielding would protect against significant exposures until the fuel assemblies could be loaded into a storage container. At this point, the fuel could be transported to another location for more secure disposition (perhaps loading back into another dry storage cask). In short, spurious failure of a cask (i.e., not due to impact or a fire) should not be considered to be a catastrophic event. Cleanup would require reasonable care and involve some expense, but it is my opinion that such a cleanup process is well within the capabilities of current technology and radiation protection practices to handle such an event.

III.3. ISFSI Risk Implications of Extended Delays in DOE Acceptance of Spent Fuel for Geologic Storage

Q: Are there any risk implications for the ISFSI of extended delays in DOE acceptance of spent fuel for geologic storage?

A: Yes, but the risk implications are insignificant. The risk implications are principally a longer period of storage of spent fuel at the ISFSI and possibly the need to periodically transport the casks to a water pool structure for replacement of the cask seals. The casks themselves should last well beyond the 20-year lifetime of the ISFSI. Of course, it is possible to replace the casks themselves should this prove to be necessary. None of these activities should result in a significant increase in risk.

IV. SUMMARY AND CONCLUSIONS

Q: Please summarize your findings and conclusions.

A: The purpose of this testimony is to evaluate the cost impact of normal and accidental exposures to radioactivity arising from operation of the proposed Prairie Island ISFSI. I have considered the normal operational offsite and onsite doses resulting from the ISFSI. I have also considered postulated accidents involving the ISFSI. Performing conservative bounding calculations of dose consequence and estimating probabilities for these events as described above, I have estimated that the annual dose consequence arising from normal operation and accidents at the ISFSI are approximately 2 person-rem per year, as summarized in Table 4, above. Quantifying this at a rate ranging from $1,000 to $10,000 per person-rem, I estimate the costs arising from normal and accidental radiation exposures associated with operating the ISFSI at Prairie Island to be in the range of $2,000 to $20,000 per year.

It should be understood that I have used very conservative estimates of consequences in preparing this estimate, and it is likely that this estimate is an overestimate of the consequences which would be calculated on the basis of a full scope risk assessment study. Such a study, however, would have required far more resources than were available for the preparation of this testimony. Accordingly, I believe that the consequence estimates above represent a bounding result. Nonetheless, notwithstanding their extreme conservatism, these results indicate that the cost consequences of normal operation and postulated accidents involving the proposed ISFSI do not represent a significant additional cost.

Q: Do you claim to have identified every possible type of accident which could result in risk to offsite areas resulting from the ISFSI?

A: Certainly not. Indeed, it is not possible to be absolutely certain that all risk contributors have been identified and appropriately considered in any type of risk assessment. Risk assessments can only deal with those things which have either happended or have been imagined or analyzed as being possible to happen. Risk assessments are not capable of formally accounting for the unknown in any meaningful way. Neither, however, is any other type of safety assessment. The strength of risk assessment is its ability to structure what is known into a useful format for consideration by decision makers. It is in this spirit that my analysis is offerred. The analysis indicates that, with a considerable margin, the costs of normal operational and accident-related radiation exposures do not add significantly to the cost of operation.

Q: Does this conclude your testimony?

A: Yes.

EXHIBIT OF

GREGORY C. MINOR

MINNESOTA DEPARTMENT OF PUBLIC SERVICE

* * *

BEFORE THE

MINNESOTA PUBLIC UTILITIES COMMISSION

* * *

NORTHERN STATES POWER COMPANY

DOCKET NO. E002/CN-91-19

* * *

SEPTEMBER 30, 1991

ATTACHMENT 1

PROFESSIONAL QUALIFICATIONS OF GREGORY C. MINOR

GREGORY C. MINOR
MHB Technical Associates
1723 Hamilton Avenue
Suite K
San Jose, California 95125
(408) 266-2716

EXPERIENCE:

1976 to PRESENT

Vice-President - MHB Technical Associates, San Jose, California

Engineering and energy consultant to state, federal, and private organizations and individuals. Major activities include studies of safety and risk involved in energy generation, providing technical consulting to legislative, regulatory, public and private groups and expert witness in behalf of state organizations and citizens' groups. Was co-editor of a critique of the Reactor Safety Study (WASH-1400) for the Union of Concerned Scientists and co-author of a risk analysis of Swedish reactors for the Swedish Energy Commission. Served on the Peer Review Group of the NRC/TMI Special Inquiry Group (Rogovin Committee). Actively involved in the Nuclear Power Plant Standards Committee work for the Instrument Society of America (ISA).

1972 - 1976

Manager, Advanced Control and Instrumentation Engineering, General Electric Company, Nuclear Energy Division, San Jose, California

Managed a design and development group of thirty-four engineers and support personnel designing systems for use in the measurement, control and operation of nuclear reactors. Involved coordination with other reactor design organizations, the Nuclear Regulatory Commission, and customers, both overseas and domestic. Responsibilities included coordinating and managing and design and development of control systems, safety systems, and new control concepts for use on the next generation of reactors. The position included responsibility for standards applicable to control and instrumentation, as well as the design of short-term solutions to field problems. The disciplines involved included electrical and mechanical engineering, seismic design and process computer control/programming, and equipment qualification.

1970 - 1972

Manager, Reactor Control Systems Design, General Electric Company, Nuclear Energy Division, San Jose, California

Managed a group of seven engineers and two support personnel in the design and preparation of the detailed system drawings and control documents relating to safety and emergency systems for nuclear reactors. Responsibility required coordination with other design organizations and interaction with the customer's engineering personnel, as well as regulatory personnel.

1963 - 1970

Design Engineer, General Electric Company, Nuclear Energy Division, San Jose, California

Responsible for the design of specific control and instrumentation systems for nuclear reactors. Lead design responsibility for various subsystems of instrumentation used to measure neutron flux in the reactor during startup and intermediate power operation. Performed lead system design function in the design of a major system for measuring the power generated in nuclear reactors. Other responsibilities included on-site checkout and testing of a complete reactor control system at an experimental reactor in the Southwest. Received patent for Nuclear Power Monitoring System.

1960 - 1963

Advanced Engineering Program, General Electric Company; Assignments in Washington, California, and Arizona

Rotating assignments in a variety of disciplines:

- Engineer, reactor maintenance and instrument design, KE and D reactors, Hanford, Washington, circuit design and equipment maintenance coordination.

- Design engineer, Microwave Department, Palo Alto, California. Work on design of cavity couplers for Microwave Traveling Wave Tubes (TWT).

- Design engineer, Computer Department, Phoenix, Arizona. Design of core driving circuitry.

- Design engineer, Atomic Power Equipment Department, San Jose, California. Circuit design and analysis.

- Design engineer, Space Systems Department, Santa Barbara, California. Prepared control portion of satellite proposal.

- Technical Staff - Technical Military Planning Operation. (TEMPO), Santa Barbara, California. Prepare analyses of missile exchanges.

During this period, completed three-year General Electric program of extensive education in advanced engineering principles of higher mathematics, probability and analysis. Also completed courses in Kepner-Tregoe, Effective Presentation, Management Training Program, and various technical seminars.

EDUCATION

University of California at Berkeley, BSEE, 1960.

Advanced Course in Engineering - three-year curriculum, General Electric Company, 1963.

Stanford University, MSEE, 1966.

HONORS AND ASSOCIATIONS

- Tau Beta Pi Engineering Honorary Society
- Co-holder of U.S. Patent No. 3,565-760, "Nuclear Reactor Power Monitoring System," February, 1971.
- Member: American Association for the Advancement of Science.
- Member: Nuclear Power Plant Standards Committee, Instrument Society of America.

PUBLICATIONS AND TESTIMONY

1. G. C. Minor, S. E. Moore, "Control Rod Signal Multiplexing," IEEE Transactions on Nuclear Science, Vol. NS-19, February 1972.

2. G. C. Minor, W. G. Milam, "An Integrated Control Room System for a Nuclear Power Plant," NEDO-10658, presented at International Nuclear Industries Fair and Technical Meetings, October, 1972, Basle, Switzerland.

3. The above article was also published in the German Technical Magazine, NT, March, 1973.

4. Testimony of G. C. Minor, D. G. Bridenbaugh, and R. B. Hubbard before the Joint Committee on Atomic Energy, Hearing held February 18, 1976, and published by the Union of Concerned Scientists, Cambridge, Massachusetts.

5. Testimony of G. C. Minor, D. G. Bridenbaugh, and R. B. Hubbard before the California State Assembly Committee on Resources, Land Use, and Energy, March 8, 1976.

6. Testimony of G. C. Minor and R. B. Hubbard before the California State Senate Committee on Public Utilities, Transit, and Energy, March 23, 1976.

7. Testimony of G. C. Minor regarding safety issues at the Grafenrheinfeld Nuclear Plant, March 16-17, 1977, Wurzbuerg, Germany.

8. Testimony of G. C. Minor regarding Reactor Safety and the Long-Term Implications of Uranium Mining, before the Cluff Lake Board of Inquiry, Regina, Saskatchewan, Canada, September 21, 1977.

9. The Risks of Nuclear Power Reactors: A Review of the NRC Reactor Safety Study WASH-1400 (NUREG-75/014), H. Kendall, et al, edited by G. C. Minor and R. B. Hubbard for the Union of Concerned Scientists, August, 1977.

10. Swedish Reactor Safety Study: Barseback Risk Assessment, MHB Technical Associates, January, 1978. (Published by Swedish Department of Industry as Document DsI 1978:1)

11. Testimony by G. C. Minor before the Wisconsin Public Service Commission, February 13, 1978, Loss of Coolant Accidents: Their Probability and Consequence.

12. Testimony by G. C. Minor regarding Reactor Safety before the California Legislature Assembly Committee on Resources, Land Use, and Energy, AB 3108, April 26, 1978, Sacramento, California.

13. Presentation by G. C. Minor before the Federal Ministry for Research and Technology (BMFT), Meeting on Reactor Safety Research, Man/Machine Interface in Nuclear Reactors, August 21, and September 1, 1978, Bonn, Germany.

14. Testimony of G. C. Minor, D. G. Bridenbaugh, and R. B. Hubbard, before the Atomic Safety and Licensing Board, September 25, 1978, in the matter of Black Fox Nuclear Power Station Construction Permit Hearings, Tulsa, Oklahoma.

15. Testimony of G. C. Minor, ASLB Hearings Related to TMI-2 Accident, Rancho Seco Power Plant, on behalf of Friends of the Earth, September 13, 1979.

16. Testimony of G. C. Minor before the Michigan State Legislature, Special Joint Committee on Nuclear Energy, Implications of Three Mile Island Accident for Nuclear Power Plants in Michigan, October 15, 1979.

17. A Critical View of Reactor Safety, by G. C. Minor, paper presented to the American Association for the Advancement of Science, Symposium on Nuclear Reactor Safety, January 7, 1980, San Francisco, California.

18. The Effects of Aging on Safety of Nuclear Power Plants, paper presented at Forum on Swedish Nuclear Referendum, Stockholm, Sweden, March 1, 1980.

19. Minnesota Nuclear Plants Gaseous Emissions Study, MHB Technical Associates, September 1980, prepared for the Minnesota Pollution Control Agency, Roseville, MN.

20. Testimony of G. C. Minor and D. G. Bridenbaugh before the New York State Public Service Commission, Shoreham Nuclear Plant Construction Schedule, in the matter of Long Island Lighting Company Temporary Rate Case, case # 27774 September 22, 1980.

21. Direct testimony of Dale G. Bridenbaugh and Gregory C. Minor before the New York State Public Service Commission, Kaiser Engineers Power Corporation Review, Shoreham Nuclear Power Station Costs and Schedule, in the matter of Long Island Lighting Company Temporary Rate Case, Case Number 27774, September 29, 1980.

22. Systems Interaction and Single Failure Criterion, MHB Technical Associates, January, 1981, prepared for and available from the Swedish Nuclear Power Inspectorate, Stockholm, Sweden.

23. Testimony of G. C. Minor and D. G. Bridenbaugh before the New Jersey Board of Public Utilities, Oyster Creek 1980 Refueling Outage Investigation, in the matter of the Petition of Jersey Central Power and Light Company for approval of an increase in the rates for electrical service and adjustment clause and factor for such service, OAL Docket No. PUC 3518-80, BPU Docket Nos. 804-285, 807-488, February 19, 1981.

24. Testimony of G. C. Minor and D. G. Bridenbaugh on PORV's and Pressurizer Heaters, Diablo Canyon Operating License hearing before ASLB, in the matter of Pacific Gas and Electric Company (Diablo Canyon Nuclear Power Plant, Units 1 and 2), Docket Nos. 50-275-OL, 50-323-OL, January 11, 1982.

25. Testimony of G. C. Minor and R. B. Hubbard on Emergency Response Planning, Diablo Canyon Operating License hearing before ASLB, Docket Nos. 50-275-OL, 50-323-OL, January 11, 1982.

26. Systems Interaction and Single Failure Criterion: Phase II Report, MHB Technical Associates, February 1982, prepared for and available from the Swedish Nuclear Power Inspectorate, Stockholm, Sweden.

27. Testimony of G. C. Minor, R. B. Hubbard, M. W. Goldsmith, S. J. Harwood on behalf of Suffolk County, before the Atomic Safety and Licensing Board, in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station, Unit 1, regarding Contention 7B, Safety Classification and Systems Interaction, Docket No. 50-322-OL, April 13, 1982.

28. Testimony of G. C. Minor and D. G. Bridenbaugh on behalf of Suffolk County, before the Atomic Safety and Licensing Board, in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station, Unit 1, regarding Suffolk County Contention 11, Passive Mechanical Valve Failure, Docket no. 50-322-OL, April 13, 1982.

29. Testimony of G. C. Minor and R. B. Hubbard on behalf of Suffolk County, before the Atomic Safety and Licensing Board, in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station, Unit 1, regarding Suffolk County Contention 27 and SOC Contention 3, Post-Accident Monitoring, Docket No. 50-322-OL, May 25, 1982.

30. Testimony of G. C. Minor and D. G. Bridenbaugh on behalf of Suffolk County, before the Atomic Safety and Licensing Board, in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station, Unit 1, regarding Suffolk County Contention 22, SRV Test Program, Docket No. 50-322-OL, May 25, 1982.

31. Testimony of G. C. Minor and D. G. Bridenbaugh on behalf of Suffolk County, before the Atomic Safety and Licensing Board, in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station, Unit 1, regarding Reduction of SRV Challenges, Docket No. 50-322-OL, June 14, 1982.

32. Testimony of G. C. Minor on behalf of Suffolk County, before the Atomic Safety and Licensing Board, in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station Unit 1, regarding Environmental Qualification, Docket No. 50-322-OL, January 18, 1983.

33. Testimony of G. C. Minor and D. G. Bridenbaugh before the Pennsylvania Public Utility Commission, on behalf of the Office of Consumer Advocate, Regarding the Cost of Constructing the Susquehanna Steam Electric Station, Unit I, Re: Pennsylvania Power and Light, Docket No. R-822189, March 18, 1983.

34. Supplemental testimony of G. C. Minor, R. B. Hubbard, and M. W. Goldsmith on behalf of Suffolk County, before the Atomic Safety and Licensing Board, in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station, Unit 1, regarding Safety Classification and Systems Interaction (Contention 7B), Docket No. 50-322, March 23, 1983.

35. Verbal testimony before the District Court Judge in the case of Sierra Club et. al. vs. DOE regarding the Clean-up of Uranium Mill Tailings. June 20, 1983.

36. Systems Interaction and Single Failure Criterion: Phase 3 Report, MHB Technical Associates, June, 1983, prepared for and available from the Swedish Nuclear Power Inspectorate, Stockholm, Sweden.

37. Systematic Evaluation Program: Status Report and Initial Evaluation, MHB Technical Associates, June, 1983, prepared for and available from the Swedish Nuclear Power Inspectorate, Stockholm, Sweden.

38. Testimony of G. C. Minor, F. C. Finlayson, and E. P. Radford before the Atomic Safety and Licensing Board, in the Matter of Long Island Lighting Company, Shoreham Nuclear Power Station, Unit 1, regarding Emergency Planning - Evacuation Times and Doses (Contentions 65, 23.D and 23.H), Docket No. 50-322-OL-3, November 18, 1983.

39. Testimony of G. C. Minor, Sizewell 'B' Power Station Public Inquiry, Proof of Evidence Regarding Safety Issues, December, 1983.

40. Testimony of D. G. Bridenbaugh, L. M. Danielson, R. B. Hubbard and G. C. Minor before the State of New York Public Service Commission, PSC Case No. 27563, in the matter of Long Island Lighting Company Proceeding to Investigate the Cost of the Shoreham Nuclear Generating Facility -- Phase II, on behalf of County of Suffolk, February 10, 1984.

41. Testimony of Fred C. Finlayson, Gregory C. Minor and Edward P. Radford before the Atomic Safety and Licensing Board, in the Matter of Long Island Lighting Company, Shoreham Nuclear Power Station, Unit 1, on behalf of Suffolk County Regarding Emergency Planning - Sheltering (Contention 61), Docket No. 50-322-OL, March 21, 1984.

42. Testimony of G. Dennis Eley, C. John Smith, Gregory C. Minor and Dale G. Bridenbaugh before the Atomic Safety and Licensing Board, in the matter of Long Island Lighting company, Shoreham Nuclear Power Station Unit 1, regarding EMD Diesel Generators and 20 MW Gas Turbine, Docket No. 50-322-OL, March 21, 1984.

43. Revised Testimony of Gregory C. Minor before the Atomic Safety and Licensing Board, in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station Unit 1, on behalf of Suffolk County regarding Emergency Planning - Recovery and Reentry (Contentions 85 and 88), Docket No. 50-322-OL, July 30, 1984.

44. Testimony of Dr. Christian Meyer, Dr. Jose Roesset, and Gregory C. Minor before the Atomic Safety and Licensing Board, in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station Unit 1, on behalf of Suffolk County, regarding <u>Low Power Hearings - Seismic Capabilities of AC Power Sources</u>, Docket No. 50-322-OL, July 1984.

45. Surrebuttal Testimony of Dale G. Bridenbaugh, Lynn M. Danielson, Richard B. Hubbard, and Gregory C. Minor, Before the New York State Public Service Commission, PSC Case No. 27563, Shoreham Nuclear Station, Long Island Lighting Company, on behalf of Suffolk County and New York State Consumer Protection Board, regarding <u>Investigation of the Cost of the Shoreham Nuclear Generating Facility</u>, October 4, 1984.

46. Direct Testimony of Dale G. Bridenbaugh, Lynn M. Danielson and Gregory C. Minor on behalf of Massachusetts Attorney General, DPU 84-145, before the Massachusetts Department of Public Utilities, regarding <u>Prudence of Expenditures by Fitchburg Gas and Electric Light Company for Seabrook Unit 2</u>, November 23, 1984, 84 pgs.

47. Direct Testimony of Dale G. Bridenbaugh, Lynn M. Danielson and Gregory C. Minor on behalf of Maine Public Utilities Commission Staff regarding <u>Prudence of Costs of Seabrook Unit 2</u>, Docket No. 84-113, December 21, 1984.

48. Direct Testimony of Dale G. Bridenbaugh and Gregory C. Minor on behalf of Suffolk County regarding <u>Shoreham Emergency Diesel Generator Loads</u>, Docket No. 50-322-OL, January 25, 1985.

49. Direct Testimony of Dale G. Bridenbaugh, Lynn M. Danielson, and Gregory C. Minor on behalf of the Vermont Department of Public Service, PSB Docket No. 5030, regarding <u>Prudence of Central Vermont Public Service Corporations Costs for Seabrook 2</u>, November 11, 1985.

50. Surrebuttal testimony of Gregory C. Minor on behalf of the Vermont Department of Public Service, PSB Docket No. 5030, <u>Prudence of Central Vermont Public Service Corporations Costs for Seabrook 2</u>, December 13, 1985.

51. <u>Report on Almaraz Steam Generator Problems</u>, MHB Technical Associates, 1985, prepared for Urbanismo Y Medio Ambiente, Junta De Extremadura, Caceres (Badajoz) Spain.

52. Direct Testimony of Dale G. Bridenbaugh, Gregory C. Minor, Lynn K. Price, and Steven C. Sholly on behalf of State of Connecticut Department of Public Utility Control Prosecutorial Division and Division of Consumer Counsel regarding the <u>Prudence of Expenditures on Millstone Unit 3</u>, Docket No. 83-07-03, February 18, 1986.

53. Direct Testimony of Dale G. Bridenbaugh and Gregory C. Minor on behalf of Massachusetts Attorney General regarding the <u>Prudence of Expenditures by New England Power Co. for Seabrook Unit 2</u>, Docket Nos. ER-85-646-000, ER-85-647-000, February 21, 1986.

54. Direct Testimony of Gregory C. Minor on behalf of the Prosecutorial Division of CDPUC regarding <u>CL&P Construction Prudence for Millstone Unit 3</u>, Docket No. ER-85-720-001 March 19, 1986.

55. Direct Testimony of Dale G. Bridenbaugh and Gregory C. Minor on behalf of Massachusetts Attorney General regarding <u>WMECo Construction Prudence for Millstone Unit 3</u>, Docket No. 85-270, March 19, 1986.

56. Direct Testimony of Dale G. Bridenbaugh and Gregory C. Minor on behalf of Massachusetts Attorney General regarding WMECo's Commercial Operating Dates and Deferred Capital Additions on Millstone Unit 3, Docket No. 85-270, March 19, 1986.

57. Rebuttal Testimony of Dale G. Bridenbaugh and Gregory C. Minor on behalf of Massachusetts Attorney General regarding Rebuttal to New England Power Company's Seabrook 2, Docket Nos. ER-85-646-001, ER-85-647-001, April 2, 1986.

58. Direct Testimony of Dale G. Bridenbaugh and Gregory C. Minor on behalf of State of Maine Staff of Public Utilities Commission regarding Construction Prudence of Millstone Unit 3, in the matter of Maine Power Company Proposed Increase in Rates, Docket No. 85-212, April 21, 1986.

59. Implications of the Accident at Chernobyl-4 for Nuclear Emergency Planning, prepared by MHB Technical Associates for Amici della Terra, Rome, Italy, for Conferenza Internazionale, May 21, 1986.

60. Implications of the Chernobyl-4 Accident for Nuclear Emergency Planning for the State of New York, prepared for the State of New York Consumer Protection Board, by MHB Technical Associates, June 1986.

61. Direct Testimony of Dale G. Bridenbaugh and Gregory C. Minor on behalf of the Vermont Department of Public Service, regarding Prudence of Costs by Central Vermont Public Service Corporation for Millstone 3, Docket No. 5132, August 25, 1986.

62. Surrebuttal Testimony of Gregory C. Minor in the matter of Jersey Central Power and Light Company, regarding TMI Restart and Performance Incentives, (Oral testimony), OAL Docket No. PUC 7939-85, BPU Docket No. ER851116, September 11, 1986.

63. Surrebuttal Testimony of Gregory C. Minor on behalf of State of Vermont Department of Public Service, regarding CVPS/NU Construction Prudence related to Millstone Unit 3, Docket No. 5132, November 6, 1986.

64. Direct Testimony of Gregory C. Minor and Lynn K. Price on behalf of State of Vermont Department of Public Service, regarding Prudence of Expenditures for Seabrook 1, Docket No. 5132, December 31, 1986.

65. Direct Testimony of Gregory C. Minor on behalf of Suffolk County, before the Atomic Safety and Licensing Board, concerning Shoreham - Protective Action Recommendations (Contention EX 36), in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station, Unit 1, Docket No. 50-322-OL-5, February 27, 1987.

66. Surrebuttal Testimony of Gregory C. Minor and Lynn K. Price, on behalf of the State of Vermont Department of Public Service, regarding Tariff Filing of Central Vermont Public Service Corporation Requesting a 12 Percent Increase in Rates to Take Effect June 2, 1986, Docket No. 5132, March 3, 1987.

67. Direct Testimony of Gregory C. Minor et. al. on behalf of the State of New York and Suffolk County, before the Atomic Safety and Licensing Board, regarding The Scope of the Emergency Planning Exercise (Contentions EX 15 and 16), in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station, Unit 1, Docket No. 50-322-OL-5, April 6, 1987.

68. Direct Testimony of Gregory C. Minor regarding Emergency Planning Reception Centers - Monitoring and Decontamination, Shoreham Docket 50-322-OL-3 (Emergency Planning), April 13, 1987.

69. Testimony of Gregory C. Minor, Steven C. Sholly et. al. on behalf of Suffolk County, regarding LILCO's Reception Centers - Planning Basis, before the Atomic Safety and Licensing Board, in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station Unit 1, Docket No. 50-322-OL-3, April 13, 1987.

70. Rebuttal Testimony of Gregory C. Minor and Steven C. Sholly on behalf of Suffolk County regarding LILCO's Reception Centers (Rebuttal to Testimony of Lewis G. Hulman), in the matter of Long Island Lighting Company, Shoreham Nuclear Power Station, Unit 1, Docket No. 50-322-OL-3, May 27, 1987.

71. Direct Testimony of Dale G. Bridenbaugh and Gregory C. Minor on behalf of Massachusetts Attorney General, before the Federal Energy Regulatory Commission, regarding Canal Electric Company Prudence Related to Seabrook Unit 2 Construction Expenditures, Docket No. ER86-704-001, July 31, 1987.

72. Direct Testimony of Dale G. Bridenbaugh and Gregory C. Minor before the Pennsylvania Public Utility Commission, Regarding Beaver Valley Unit 1 1979 Outage, Docket No. 1-79070318, OCA Statement No. 2, August 31, 1987.

73. Oral testimony of Gregory C. Minor Before the Illinois Pollution Control Board on behalf of Reed-Custer Community Unit School District No. 255-U, re: Braidwood Cooling Pond September 8, 1988, Case PCB 87-209.

74. Testimony of Gregory C. Minor in the U. S. District Court, Brooklyn, New York, September 31, 1988, re: RICO Litigation, County of Suffolk vs. LILCO et. al., Case CV 87-646.

75. GE Reed Report Safety Issue Reviews, Issues 5, 10, and 24, prepared by MHB Technical Associates for The Ohio State University Nuclear Engineering Program Expert Review Panel, Public Utility Commission of Ohio, October 1988.

76. Direct Testimony and Exhibits of Dale G. Bridenbaugh, Gregory C. Minor and Steven C. Sholly on Behalf of Massachusetts Department of the Attorney General, Re: Pilgrim Nuclear Power Station, Investigation of Pilgrim Outage, DPU 88-28, November 30, 1988, PROTECTED INFORMATION.

77. Supplemental Testimony of Dale G. Bridenbaugh, Gregory C. Minor and Steven C. Sholly on Behalf of Massachusetts Department of the Attorney General, Re: Pilgrim Nuclear Power Station, Investigation of Pilgrim Outage, DPU 88-28, January 20, 1989, Exhibit AG-2.

78. Testimony of Gregory C. Minor, U. S. District Court, Brooklyn, New York, February 3, 1989, re: RICO Litigation, County of Suffolk vs. LILCO et. al., Case 87 CIV. 646 (JBW).

79. Surrebuttal Testimony of Dale G. Bridenbaugh, Gregory C. Minor and Steven C. Sholly on Behalf of Massachusetts Department of the Attorney General, Re: Pilgrim Nuclear Power Station, Investigation of Pilgrim Outage, DPU 88-28, February 13, 1989, Exhibit AG-74.

80. Surrebuttal Testimony of Dale G. Bridenbaugh, Gregory C. Minor and Steven C. Sholly on Behalf of Massachusetts Department of the Attorney General, Re: Pilgrim Nuclear Power Station, Investigation of Pilgrim Outage, DPU 88-28, February 17, 1989, Exhibit AG-93.

81. Direct Testimony of Gregory C. Minor on Behalf of the Minnesota Department of Public Service, before the Minnesota Public Utilities Commission, regarding Pathfinder Nuclear Plant Decommissioning, Application for Rate Increase by Northern States Power Company, Docket No. E002/GR-89-867, February 26, 1990.

82. Surrebuttal Testimony of Gregory C. Minor on Behalf of the Minnesota Department of Public Service, before the Minnesota Public Utilities Commission, regarding Pathfinder Nuclear Plant Decommissioning, Application for Rate Increase by Northern States Power Company, Docket No. E002/GR-89-867, April 5, 1990.

83. Advanced Reactor Study, prepared for the Union of Concerned Scientists, Cambridge, Massachusetts, July 1990.

84. "Advanced Reactors: Are We Ready For Them?", paper and presentation at 1990 California Clean Air and New Technologies Conference, Los Angeles, California, October 16, 1990.

85. Braidwood Station Cooling Pond: Function and Purpose, prepared for Will County State's Attorney, Joliet, Illiois, March 25, 1991.

86. Testimony of Gregory C. Minor on Behalf of the Office of Public Utility Counsel, before the Public Utility Commission of Texas, Application of HL&P for Authority to Change Rates, Hearing on Contested Settlement, Revenue Requirement Issues, Docket No. 9850, April 23, 1991.

87. "Advanced Reactors: How Do They Fit In Our National Energy Strategy?", paper and presentation at NASUCA Conference, Seattle, Washington, May 21, 1991.

ATTACHMENT 2

Valuation of Dose Avoided at U.S. Nuclear Power Plants

By John W. Baum, Brookhaven National Laboratory.

Introduction

Modern day radiation protection standards are based on the International Commission on Radiological Protection (ICRP) system of dose limitation (ICRP, 1977), which includes as an essential element that doses be kept "as low as reasonably achievable" (ALARA). Both social and economic factors are to be included in the ALARA process, which is referred to as optimization of radiation protection by the ICRP.

Implementation of the ALARA principle at nuclear power plants presents a continuing challenge for health physicists at utility, corporate, and plant levels; for plant designers; and for regulatory agencies. The relatively large collective doses at some plants are being addressed through a variety of dose reduction techniques. A monetary value ($/person-cSv or $/person-rem) is needed to complete the quantitative evaluations that are important in the decision process.

Limits on ALARA?

The ALARA (optimization) process is applicable throughout the entire range of doses below the dose limits, even into areas of background radiation. Throughout this application, one should always consider both differential costs and differential benefits. It is the ratio of these two values that determines cost-effectiveness, which is then compared to the monetary value of dose reduction. Even doses below regulatory concern or below negligible individual risk levels should be considered. If the cost or effort is negligible, even a negligible (comparable) risk should be avoided.

The process will be self-limiting if costs of doing evaluations are included in the total because when the collective doses are small, the costs will be large in comparison and one soon reaches a point of no net benefit, or excessively large cost-effectiveness values ($/cSv). At this point, the process should stop.

Regulatory agencies are required to do cost-benefit evaluations in arriving at below regulatory concern (BRC), exempt, or trivial levels. However, there may still be need for some consideration of ALARA by those exempt from regulatory pressures. This can be the case, for example, if large numbers of individuals may be exposed and if simple (low cost) efforts could be implemented to avoid these small doses.

Monetary Values of Dose Reduction

Application of quantitative methods in the ALARA process is essential if consistent, rational, documentable, and coherent decisions are to be made. The level of effort must, of course, bear some reasonable relationship to potential dose savings that may be made. To apply quantitative thinking to the decision process, a monetary value for dose reduction is needed. This value can be used in cost-benefit studies as suggested

Table I. Monetary Value of Dose Reduction Based on ICRP 1973 Summary

Author	Dollars/person-cSv in ICRP-22	1990 Equivalent* Dollars/person-cSv	1990 Values Adjusted for New Risk Estimates**
Dunster/ McLean	10 - 25	34 - 85	140 - 340
Hedgran/ Lindell	100 - 250	340 - 850	1,400 - 3,400
Otway	200	680	2,700
Lederberg	100 - 250	340 - 850	1,400 - 3,400
Cohen	250	850	3,400
Sagan	30	100	410
		Mean	≈ $1,900/person-cSv
		Median	≈ $2,400/person-cSv

* 1990 values adjusted for inflation are estimated as 3.4 times the 1970 values based on purchasing power of the dollar as reflected in consumer prices (U.S. Bureau of Census 1989).

** 1990 values were increased by a factor of four to account for higher 1990 risk estimates (BEIR V 1990) compared to a value of 10^{-4} commonly used in the 1970's.

Author

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by the ICRP in its Publications 22, 26, 27, 37, 45, and 55 (see references). It can also be used as a cost-effectiveness guide in comparing and prioritizing various options for dose control in the design or operational phases of facilities (Baum and Matthews 1985).

A review of previous thinking on the value of dose reduction and the related value of risk reduction has been made to provide a basis for recommendations of an appropriate value for dose avoided. Results are summarized in Tables I through V and discussed below.

Table I summarizes information available in the early 1970s. The values cited are from ICRP Publication 22 (ICRP 1973). Values were adjusted for inflation to reflect 1990 costs and adjusted (increased) for higher 1990 risk estimates (BEIR V 1990). This latter adjustment is based on the assumption that larger values would have (or should be) used if risks are found to be higher. Adjusted values range from $140 to $3,400 per person-cSv. These early values were based on rather little data or analysis and were specifically for doses low in comparison to dose limits. A medium value based on these findings would be about $2,400/person-cSv (1990 risk adjusted values).

In the early 1970s, the Atomic Energy Commission (now the U.S. Nuclear Regulatory Commission [NRC]) suggested the use of $1,000/person-cSv be used in evaluating costs and benefits of off-site exposures during design of nuclear power plants (AEC 1971). The same value was utilized by the NRC in 10 CFR 50 Appendix I (NRC 1975). This latter value and other values that have been used in U.S. nuclear facilities are summarized in Table II. Original values have been adjusted for inflation and new risk estimates to provide in 1990 dollars an equivalent monetary value per unit risk reduction or life saved. The studies of U.S. Department of Energy (DOE) contractor facilities by Gilchrist, et al. (Gilchrist 1978) revealed that values between $1,000 and $10,000 per person-cSv were being employed in the 1970s. Discussions at a recent workshop (Baum, et al. 1989) revealed a similar range ($1,000 - $20,000) was being employed at U.S. nuclear power plants in 1989, with most plants using about $5,000. A 1989-90 study of major

(Continued on next page)

Table II. Monetary Values of Dose Reduction Used at U.S. Nuclear Facilities

Locations	Value Employed (dollar/person-cSv)	Approximate 1990 Equivalent Value Adjusted for Inflation and new risk estimates
Environs of Nuclear Power Plants (10CFR50, Appendix I)	$1,000 (1975)	$10,000*
DOE Facilities (1970's)	$1,000 (minimum)	$10,000
DOE Facilities (89-90)	$2,000 (minimum)	$4,000**
Nuclear Power Plants (89)	$1,000 - $20,000	$10,000**(avg.)
	Mean	≈ $7,000
	Median	≈ $10,000

*Adjusted for inflation using a 2.5 factor since 1975 and adjusted for higher 1990 risk estimates using a factor of four over 1970's values.

**Adjusted for risk estimates by a factor of only two since the 1989 values employed may have included some adjustments in anticipation of higher risk estimates.

Table III. Value of Risk Reduction Based On Compensating Wage Differentials (1990 U. S. Dollars)*

Author(s)	Study Year (Country)	Estimated Value of Risk Reduction In 1990 U.S. Dollars
Thaler and Rosen (1973)	1967 (USA)	$800,000
Smith, R.S. (1973)	1973 (USA)	$15,000,000
Melinek (1974)	1971 (UK)	$1,900,00
Smith, R.S. (1976)	1976 (USA)	$4,700,000
Viscusi (1978)	1969 (USA)	$4,900,000
Veljanovksi (1978)	1970 (UK)	$8,700,000
Dillingham (1979)	1970 (USA)	$760,000
Brown (1980)	1967 (USA)	$2,400,000
Needleman (1980)	1968 (UK)	$250,000
Olson (1981)	1973 (USA)	$10,000,000
Maria & Psacharopoulos (1982)	1975 (UK)	$3,600,000
Smith, V.K. (1983)	1978 (USA)	$1,100,000
Arnould & Nichols (1983)	1970 (USA)	$780,00
Weiss et al. (1986)	1981 (Austria)	$6,200,000
	Mean	≈$4,360,000
	Median	≈$3,000,000

$$\text{Implied Value of dose reduction} = \frac{\$3{,}000{,}00}{\text{risk}} \times \frac{4\times10^{-4}\ \text{risk}}{\text{person-cSv}}$$

$$= \$1{,}200/\text{person-cSv}$$

*After Jones-Lee 1989, adjusted for inflation since study year.

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DOE facilities (Dionne, et al. 1990) revealed several plants were using a range from $2,000 to $60,000 as suggested in a DOE guide (Kathren, et al. 1980). It is important to note that the more recent values reflect not only the possible health effects detriments, but also some costs associated with operations such as hiring and training additional crews especially for high dose jobs. They may also reflect a trend toward greater acceptance of the "willingness to pay" approach to valuation of detriment rather than the older "human capital" and medical costs approach, and greater public and worker perception and concerns with safety, especially radiation.

The median value obtained from the four sources listed in Table II is $10,000 per person-cSv. All values seem to reflect the earlier $1,000 per person-cSv value which was an upper limit on values being proposed at that time.

Information from several studies on compensating wage differentials has been summarized recently (Jones-Lee 1989). In this approach, wage differentials are compared to risk differentials for various job categories to arrive at an implied value of risk reduction. The value thus derived is, of course, biased and reflects more than just risk of death. Many of the higher risk jobs are in lower wage brackets and thus may lead to underestimates of the average worker's willingness to accept risk for compensation. Counteracting this bias is the fact that these higher risk jobs often involve discomfort, stress, or other disadvantages. These other factors presumably account for some of the wage differential.

Results of 9 U.S., 4 U.K., and 1 Austrian study are summarized in Table III. Values (in 1990 dollars) per unit risk (mortality) range from $250,000 for a study of differentials in the U.K. construction industry to $15,000,000 for a study of various U.S. industries. The median value for all 14 studies was $3,000,000 per unit risk. Using a radiation risk coefficient of 4×10^{-4} (BEIR V 1990) risk/person-cSv (serious genetic effects plus fatal cancer) yields equivalent monetary value of dose reduction of $3,000,000/risk x 4×10^{-4} risk/person-cSv = $1,200 per person-cSv.

Another approach to arrive at a value that reflects the average person's willingness to pay for risk reduction is through use of questionnaires. Cohen surveyed about 100 students in a course on energy and environment at the University of Pittsburgh in two successive years. The results of these surveys are shown in Table IV. Student answers yielded values from $40,000/life saved for safer cigarettes to $50,000,000 in electric rates per life saved by reductions of 1 in a million risk from a nuclear power plant. This set of results yielded a mean value of $2,300,000 per life saved and a median of about $3,800,000 (both in 1990 dollars).

A number of major studies were summarized by Jones-Lee (Jones-Lee 1989). Results based on these are com-

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Table IV Results of Student Questionnaire on Willingness to pay for risk reduction (Cohen 1980)*

Proposed Safety Action	1990 Dollars/Statistical Life
10^{-6} reduction of nuclear risks	$125,000,000
10^{-3} reduction of coal plant risks	$300,000
Gov. Health Plan to save 1,000 lives	$6,250,000
Air bags in autos	$1,250,000
Safer cigarettes	$100,000
Safer transportation	$6,500,000
Mean = $2,300,000	
Median = $3,800,000	

*Values in Cohen were increased by a factor 2.5 to adjust for inflation since 1975.

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pared with the combined result from the six smaller questionnaire studies reported by Cohen (Cohen 1980). There is good agreement between Cohen's median value and the median value obtained from all values listed in Table V. These median values are also very consistent with the large and most recent study by Jones-Lee (Jones-Lee 1989) of willingness to pay for transport safety in the U.K. (the last item in Table V).

In contrast to the above examples of societies' willingness to pay fairly high values for risk reduction, many highly cost-effective yet not fully implemented health and safety options have been cited by various reviewers (e.g., Cohen 1980; Siddall 1981; Graham and Vaupel 1981). Graham and Vaupel cite several options that would not only save lives but also save in costs (e.g., medical and/or property savings exceed costs of implementation). These include several traffic and auto safety actions such as mandatory air bags, mandatory passive seat belts, 55 mph speed limit, roadside hazard removal, vehicle inspection, traffic enforcement, and compulsory helmet usage by motorcyclists. Other examples in the area of home safety include a clothing flammability law and mandatory smoke detectors. The wide range of costs per life saved in medical screening, traffic safety and home safety options reveals a lack of consistency in how society spends its health and safety dollars. This inconsistency has many causes including strong influences of public perception and the difficulty of judging values and probabilities when small risks are involved. Knowing the cost-effectiveness of many of the other options, one tends to avoid excessive expenditures in any given area in hopes that at least a portion of the money thus saved would be used for more effective measures. Since these other options are so numerous and lacking in robustness, they are not included in the listings employed here.

Summary

The U.S. nuclear industry is currently spending about $10,000/person-cSv for dose reduction efforts, or about $25,000,000 per cancer plus major genetic effects averted. This is about ten times higher than would be expected based on wage differential studies and societies' willingness to pay based on questionnaire studies. This high value reflects the fact that persons working on the high-dose jobs may receive doses that approach the administrative dose limits (usually set at 4 or 5 cSv/yr). To avoid exceeding these limits, additional workers or contractors must be hired and trained at costs of $30,000 to $40,000/yr per person. Dividing these costs by the dose limit yields values of $6,000 to $10,000/person-cSv. The high value is also reflecting insurance, litigation, and worker and public relations concerns.

References

1. Baum, J. W.; Matthews, G. R. Compendium of Cost-Effectiveness Evaluations of Modifications for Dose Reduction at Nuclear Power Plants. NUREG/CR-4373. Brookhaven National Laboratory, Upton, New York 11973; 1985.
2. Baum, J. W.; Dionne, B. J.; Khan, T. A. Proceedings of the International Workshop on New Developments in

(Continued on page 46)

Valuation of Dose,
Continued from page 44

Occupational Dose Control and ALARA Implementation at Nuclear Power Plants and Similar Facilities. NUREG/CP-0110, BNL-NUREG-52226. Brookhaven National Laboratory, Upton, New York 11973; 1989.

3. Cohen, B. L. Society's Valuation of Life Saving in Radiation Protection and Other Contexts. Health Physics, 38:33-51, Pergamon Press Ltd.; 1980.
4. Dionne, B. J.; Meinhold, C. B.; Khan, T. A.; Baum, J. W. Occupational Dose Reduction at Department of Energy Contractor Facilities: Study of ALARA Programs - Status 1989. Brookhaven National Laboratory, Upton, New York 11973; 1990.
5. Gilchrist, R. L.; Selby, J. M.; Wedlick, H. L. Technical Guidelines for Maintaining Occupational Exposures As Low as Practicable: Phase I - Summary of Current Practices. PNL-2663. Pacific Northwest Laboratories, Richland, WA 99352; August 1978.
6. Graham, J. D.; Vaupel, J. W. Value of Life: What Difference Does It Make? Risk Analysis, No. 1, 1:89-95; 1981.
7. Hallenbeck, W. H.; Cunningham, K. M. Quantitative Risk Assessment for Environmental and Occupational Health. Lewis Publishers, Inc., Chelsea, MI; 1986.
8. International Commission on Radiological Protection. Implications of Commission Recommendations that Doses Be Kept as Low as Readily Achievable. ICRP Publication 22. Pergamon Press, Oxford, England; 1973.
9. International Commission on Radiological Protection. Recommendations of the International Commission on Radiological Protection. ICRP Publication 26. Pergamon Press, Oxford, England; January 17, 1977a.
10. International Commission on Radiological Protection. Problems Involved in Developing an Index of Harm. ICRP Publication 27. Annals of the ICRP, Volume 1, No. 4. Pergamon Press, Oxford, England; 1977b.
11. International Commission on Radiological Protection. Cost-Benefit Analysis in the Optimization of Radiation Protection. ICRP Publication 37. Annals of the ICRP, Volume 10, No. 2/3. Pergamon Press, Oxford, England; 1983.
12. International Commission on Radiological Protection. Quantitative Bases for Developing a Unified Index of Harm. ICRP Publication 45. Annals of the ICRP, Volume 15, No. 3. Pergamon Press, Oxford, England; 1989.
13. International Commission on Radiological Protection. Optimization and Decision-Making in Radiological Protection. ICRP Publication 55. Annals of the ICRP, Volume 20, No. 1. Pergamon Press, Oxford, England; 1989.
14. Jones-Lee, M. W. The Economics of Safety and Physical Risk. Basil Blackwell Ltd., Oxford, England; 1989.
15. Kathren, R. L.; Selby, J. M.; Vallario, E. J. A Guide to Reducing Radiation Exposure to As Low As Reasonably Achievable (ALARA). DOE/EV/1830-T5. U.S. Department of Energy, Washington, D.C. 20545; 1980.
16. National Council on Radiation Protection and Measurements. Recommendations on Limits for Exposure to Ionizing Radiation. NCRP Report No. 91; 1987.
17. National Research Council, Committee on the Biological Effects of Ionizing Radiation. Health Effects of Exposure to Low Levels of Ionizing Radiation (BEIR V). National Academy Press, Washington, D.C; 1990.
18. Siddall, E. Risk, Fear and Public Safety. AECL-7404. Atomic Energy of Canada Limited, Ontario; 1981.
19. U.S. Atomic Energy Commission, "Light Water-Cooled Nuclear Power Reactors," Federal Register 35 FR 18385, U.S. Government Printing Office, Washington, D.C.; 1971.
20. U.S. Bureau of Census. Statistical Abstract of the United States 1989. 109th Edition; 1989.

(Continued on page 62, column 3)

Table V. Questionnaire estimates of the value of risk reduction (based on Jones-Lee 1989 and Cohen 1980)

Authors	Nature of Study	Estimated Value of Risk Reduction in 1990 U.S. Dollars
Acton (1973)	Small non-random sample survey (n=93) of willingness to pay for heart attack ambulance (USA)	93,000
Melinek et al. (1973)	Non-random sample survey (n=873) of willingness to pay for domestic fire safety (UK)	480,000
Melinek et al. (1973)	Non-random sample survey (n=873) of willingness to pay for hypothetical "safe" cigarettes (UK)	150,000
Cohen (1975)	Student surveys	3,800,000
Maclean (1979)	Quota sample survey (n=325) of willingness to pay for domestic fire safety (UK)	4,700,000
Frankel (1979)	Small, non-random sample survey (n=169) of willingness to pay for elimination of small airline risk (USA)	22,000,000
Frankel (1979)	Small, non-random sample survey (n=169) of willingness to pay for elimination of large airline risk (USA)	95,000
Jones-Lee et al. (1985)	Large, random sample survey (n=1,150) of willingness to pay for transport safety (UK)	3,500,000
	Mean	≈ 7,300,000
	Median	≈ 3,500,000

Implied value of dose reduction = \$3,500,000/risk x 4 x 10^{-4} risk/person-cSv
= \$1,400/person-cSv

Teledosimetry,

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to a worker, group of workers, or one part of a worker may increase ALARA effectiveness. Work in the transfer canal or reactor cavity, removal or replacement of the reactor head, resin transfers, removal of incore detectors, boiling water reactor (BWR) control rod drive removal, and steam generator maintenance are a few examples. Whenever there are high dose rates and inhomogeneous radiation fields, the advantages of teledosimetry are obvious. In areas with multiple sources, such as valve nest or a BWR drywell, teledosimetry may also improve monitoring capability. A measured dose is always preferable to an estimated dose. Today's miniaturized telemetry equipment provides the means to that end.

Another aspect of teledosimetry is the potential for reducing the dose to the health physics technician. The ICRP will soon officially recommend a reduction in allowable dose to 2 rem per year. The use of teledosimetry could allow the health physics technician to monitor more work from a distance. To this end, we are participating in the development of a miniature continuous air monitor (CAM). This mini CAM will weigh 30 pounds and feature local readout, local alarm, and telemetry of air monitoring results to the same receiver used for teledosimetry. Future developments will include a digital telephone to replace the present analog radio transmitter. This device will have a range of approximately one mile in air and allow the health physics control point to converse with any radiation worker via the dosimeter. Add closed circuit video to this combination (Figure 2) and the health physics technician will be able to accurately monitor radiological work at a distance. In addition, the use of local area networks will also allow the ALARA office or radiation protection manager to monitor a job in real time. The video can be recorded for training purposes. Video stills can be placed in the ALARA files.

This equipment would allow the health physics control point to constantly monitor worker that would otherwise have intermittent coverage. When direct visual oversight of workers is difficult, or where there are multiple sources of exposure in close proximity, such as a BWR drywell, the ability to monitor each EDRD remotely may be of benefit.

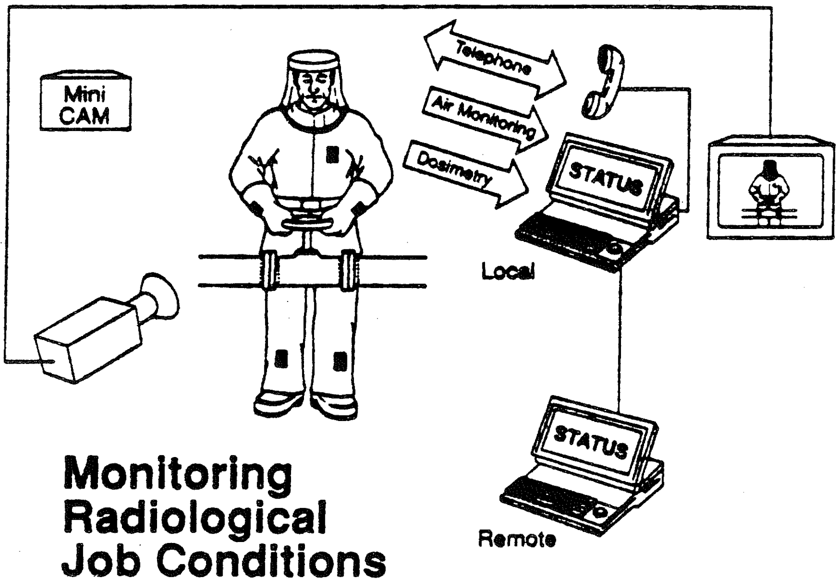

Figure 2. An Example of Remote Coverage.

Contact

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Valuation of Dose,

Continued from page 46

21. U.S. Department of Energy. DOE Order 5480.11. Radiation Protection for Occupational Workers; 1988.
22. U.S. Nuclear Regulatory Commission. 10 CFR 50, Appendix I. Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents. 40 FR 19442; May 5, 1975.
23. U.S. Nuclear Regulatory Commission. Below Regulatory Concern. NUREG/BR-0157. U.S. Government Printing Office, Box 37082, Washington, D.C. 20013-7082; 1990.

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The Cover

Palo Verde is a triple-unit commercial nuclear facility operated by Arizona Public Service Company, which owns 29.1% of the plant. The other co-owners are Salt River Project, Southern California Edison, Public Service Company of New Mexico, El Paso Electric Company, Los Angeles Department of Water & Power, and the Southern California Public Power Authority. All three units are operating at 100% power. Unit 1 returned to service on February 16, 1991, after a 39-day testing and maintenance outage. **Photo courtesy of Arizona Public Service Company.**

ATTACHMENT 3

	A	B	C	D	E	F	G
1	DISTANCE	X/Q CORR. FACTOR	ONE CASK NO FIRE (DOSE - R)	ONE CASK WITH FIRE (DOSE - R)	ALL CASKS NO FIRE (DOSE - R)	ALL CASKS WITH FIRE (DOSE - R)	ONE ASSEMBLY DROPPED UNDER WATER
2	MILE 0-1	1.00	0.25	118	12	5664	0.0011
3	MILE 1-2	0.053	0.25	118	12	5664	0.0011
4	MILE 2-3	0.032	0.25	118	12	5664	0.0011
5	MILE 3-4	0.023	0.25	118	12	5664	0.0011
6	MILE 4-5	0.018	0.25	118	12	5664	0.0011
7	MILE 5-10	0.01	0.25	118	12	5664	0.0011
8	MILE 10-20	0.0049	0.25	118	12	5664	0.0011
9	MILE 20-30	0.004	0.25	118	12	5664	0.0011
10	MILE 30-40	0.0031	0.25	118	12	5664	0.0011
11	MILE 40-50	0.0026	0.25	118	12	5664	0.0011
12							
13	UNWEIGHTED SECTOR DOSE (PERSON-REM)						
14	WIND ROSE WEIGHT						
15	WEIGHTED SECTOR DOSE (PERSON-REM)		193.37	91,157.91	9,281.87	4,381,040.72	0.85

	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	V	W
1	N (POPULATION NORTH SECTOR, VS DISTANCE)	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
2	3	0	0	0	0	0	0	6	0	0	0	0	18	101	6	40
3	50	123	10	8	3	0	0	6	3	3	6	0	41	34	3	0
4	81	23	18	13	32	0	0	6	12	34	35	25	10	32	3	0
5	13	18	21	11	53	106	0	795	43	68	13	41	19	10	6	5
6	21	51	32	64	34	215	209	829	307	9	26	26	26	22	3	0
7	289	492	708	438	279	1405	11132	692	188	370	305	271	531	1330	469	271
8	10700	1634	1713	1598	1294	1539	1120	1290	1834	1291	1616	4428	2146	12575	14564	2056
9	3840	5212	4174	2945	2920	3191	3995	2224	6125	2258	3418	16225	8867	17135	120508	21879
10	11259	3192	4987	9864	4943	4851	4206	7041	11225	3577	17277	8046	13363	342639	675752	75936
11	6738	7275	4193	11229	5662	2931	3676	46030	35469	5259	26552	8380	13116	150662	339696	19039
12																
13																
14																
15																

	X	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK	AL	AM	AN
1	N DOSE (ONE CASK, NO FIRE)	NNE DOSE	NE DOSE	ENE DOSE	E DOSE	ESE DOSE	SE DOSE	SSE DOSE	S DOSE	SSW DOSE	SW DOSE	WSW DOSE	W DOSE	WNW DOSE	NW DOSE	NNW DOSE	WEIGHTED AVERAGE POPULATION DOSE (ONE CASK , NO FIRE)
2	0.75	0.00	0.00	0.00	0.00	0.00	0.00	1.50	0.00	0.00	0.00	0.00	4.50	25.25	1.50	10.00	
3	0.66	1.63	0.13	0.11	0.04	0.00	0.00	0.08	0.04	0.04	0.08	0.00	0.54	0.45	0.04	0.00	
4	0.65	0.18	0.14	0.10	0.26	0.00	0.00	0.05	0.10	0.27	0.28	0.20	0.08	0.26	0.02	0.00	
5	0.07	0.10	0.12	0.06	0.30	0.61	0.00	4.57	0.25	0.39	0.07	0.24	0.11	0.06	0.03	0.03	
6	0.09	0.23	0.14	0.29	0.15	0.97	0.94	3.73	1.38	0.04	0.12	0.12	0.12	0.10	0.01	0.00	
7	0.72	1.23	1.77	1.10	0.70	3.51	27.83	1.73	0.47	0.93	0.76	0.68	1.33	3.33	1.17	0.68	
8	13.11	2.00	2.10	1.96	1.59	1.89	1.37	1.58	2.25	1.58	1.98	5.42	2.63	15.40	17.84	2.52	
9	3.84	5.21	4.17	2.95	2.92	3.19	4.00	2.22	6.13	2.26	3.42	16.23	8.87	17.14	120.51	21.88	
10	8.73	2.47	3.86	7.64	3.83	3.76	3.26	5.46	8.70	2.77	13.39	6.24	10.36	265.55	523.71	58.85	
11	4.38	4.73	2.73	7.30	3.68	1.91	2.39	29.92	23.05	3.42	17.26	5.45	8.53	97.93	220.80	12.38	
12																	
13	33.01	17.79	15.17	21.50	13.47	15.83	39.79	50.84	42.36	11.70	37.36	34.56	37.05	425.45	885.64	106.33	
14	0.07	0.03	0.03	0.03	0.07	0.10	0.13	0.07	0.05	0.02	0.02	0.02	0.06	0.11	0.13	0.08	
15	2.15	0.55	0.38	0.67	0.98	1.61	4.97	3.30	1.95	0.27	0.71	0.66	2.04	45.95	118.68	8.51	193.37

	AO	AP	AQ	AR	AS	AT	AU	AV	AW	AX	AY	AZ	BA	BB	BC	BD	BE
1	N DOSE (ONE CASK, WITH FIRE)	NNE DOSE	NE DOSE	ENE DOSE	E DOSE	ESE DOSE	SE DOSE	SSE DOSE	S DOSE	SSW DOSE	SW DOSE	WSW DOSE	W DOSE	WNW DOSE	NW DOSE	NNW DOSE	WEIGHTED AVERAGE POPULATION DOSE (ONE CASK, WITH FIRE)
2	354.00	0.00	0.00	0.00	0.00	0.00	0.00	708.00	0.00	0.00	0.00	0.00	2124.00	11918.00	708.00	4720.00	
3	312.70	769.24	62.54	50.03	18.76	18.76	0.00	37.52	18.76	18.76	37.52	0.00	256.41	212.64	18.76	0.00	
4	305.86	86.85	67.97	49.09	120.83	120.83	0.00	22.66	45.31	128.38	132.16	94.40	37.76	120.83	11.33	0.00	
5	35.28	48.85	56.99	29.85	143.84	143.84	0.00	2157.63	116.70	184.55	35.28	111.27	51.57	27.14	16.28	13.57	
6	44.60	108.32	67.97	135.94	72.22	72.22	443.92	1760.80	652.07	19.12	55.22	55.22	55.22	46.73	6.37	0.00	
7	341.02	580.56	835.44	516.84	329.22	329.22	13135.76	816.56	221.84	436.60	359.90	319.78	626.58	1569.40	553.42	319.78	
8	6186.74	944.78	990.46	923.96	748.19	748.19	647.58	745.88	1060.42	746.46	934.37	2560.27	1240.82	7270.87	8420.90	1188.78	
9	1812.48	2460.06	1970.13	1390.04	1378.24	1378.24	1885.64	1049.73	2891.00	1065.78	1613.30	7658.20	4185.22	8087.72	56879.78	10326.89	
10	4118.54	1167.63	1824.24	3608.25	1808.15	1808.15	1538.55	2575.60	4106.11	1308.47	6319.93	2943.23	4888.19	125337.35	247190.08	27777.39	
11	2067.22	2231.97	1286.41	3445.06	1737.10	1737.10	1127.80	14122.00	10881.89	1613.46	8146.15	2570.98	4023.99	46223.10	104218.73	5841.17	
12																	
13	15578.44	8398.27	7162.15	10149.06	6356.55	6356.55	18779.25	23996.37	19994.10	5521.57	17633.84	16313.36	17489.76	200813.77	418023.66	50187.57	
14	0.07	0.03	0.03	0.03	0.07	0.10	0.13	0.07	0.05	0.02	0.02	0.02	0.06	0.11	0.13	0.08	
15	1012.60	260.35	179.05	314.62	464.03	648.37	2347.41	1559.76	919.73	127.00	335.04	309.95	961.94	21687.89	56015.17	4015.01	91157.91

	BF	BG	BH	BI	BJ	BK	BL	BM	BN	BO	BP	BQ	BR	BS	BT	BU	BV
1	N DOSE (ALL CASKS, NO FIRE)	NNE DOSE	NE DOSE	ENE DOSE	E DOSE	ESE DOSE	SE DOSE	SSE DOSE	S DOSE	SSW DOSE	SW DOSE	WSW DOSE	W DOSE	WNW DOSE	NW DOSE	NNW DOSE	WEIGHTED AVERAGE POPULATION DOSE (ALL CASKS, NO FIRE)
2	36.00	0.00	0.00	0.00	0.00	0.00	0.00	72.00	0.00	0.00	0.00	0.00	216.00	1212.00	72.00	480.00	
3	31.80	78.23	6.36	5.09	1.91	0.00	0.00	3.82	1.91	1.91	3.82	0.00	26.08	21.62	1.91	0.00	
4	31.10	8.83	6.91	4.99	12.29	0.00	0.00	2.30	4.61	13.06	13.44	9.60	3.84	12.29	1.15	0.00	
5	3.59	4.97	5.80	3.04	14.63	29.26	0.00	219.42	11.87	18.77	3.59	11.32	5.24	2.76	1.66	1.38	
6	4.54	11.02	6.91	13.82	7.34	46.44	45.14	179.06	66.31	1.94	5.62	5.62	5.62	4.75	0.65	0.00	
7	34.68	59.04	84.96	52.56	33.48	168.60	1335.84	83.04	22.56	44.40	36.60	32.52	63.72	159.60	56.28	32.52	
8	629.16	96.08	100.72	93.96	76.09	90.49	65.86	75.85	107.84	75.91	95.02	260.37	126.18	739.41	856.36	120.89	
9	184.32	250.18	200.35	141.36	140.16	153.17	191.76	106.75	294.00	108.38	164.06	778.80	425.62	822.48	5784.38	1050.19	
10	418.83	118.74	185.52	366.94	183.88	180.46	156.46	261.93	417.57	133.06	642.70	299.31	497.10	12746.17	25137.97	2824.82	
11	210.23	226.98	130.82	350.34	176.65	91.45	114.69	1436.14	1106.63	164.08	828.42	261.46	409.22	4700.65	10598.52	594.02	
12																	
13	1584.25	854.06	728.35	1032.11	646.43	759.86	1909.75	2440.31	2033.30	561.52	1793.27	1658.99	1778.62	20421.74	42510.88	5103.82	
14	0.07	0.03	0.03	0.03	0.07	0.10	0.13	0.07	0.05	0.02	0.02	0.02	0.06	0.11	0.13	0.08	
15	102.98	26.48	18.21	32.00	47.19	77.51	238.72	158.62	93.53	12.91	34.07	31.52	97.82	2205.55	5696.46	408.31	9281.87

	BW	BX	BY	BZ	CA	CB	CC	CD	CE	CF	CG	CH	CI	CJ	CK	CL	CM
1	N DOSE (POPULATION DOSE, ALL CASKS, WITH FIRE)	NNE DOSE	NE DOSE	ENE DOSE	E DOSE	ESE DOSE	SE DOSE	SSE DOSE	S DOSE	SSW DOSE	SW DOSE	WSW DOSE	W DOSE	WNW DOSE	NW DOSE	NNW DOSE	WEIGHTED AVERAGE POPULATION DOSE (ALL CASKS, WITH FIRE)
2	16992	0	0	0	0	0	0	33984	0	0	0	0	101952	572064	33984	226560	
3	15010	36924	3002	2402	901	0	0	1801	901	901	1801	0	12308	10207	901	0	
4	14681	4169	3262	2356	5800	0	0	1087	2175	6162	6344	4531	1812	5800	544	0	
5	1694	2345	2736	1433	6904	13809	0	103566	5602	8858	1694	5341	2475	1303	782	651	
6	2141	5200	3262	6525	3466	21920	21308	84518	31299	918	2651	2651	2651	2243	306	0	
7	16369	27867	40101	24808	15803	79579	630516	39195	10648	20957	17275	15349	30076	75331	26564	15349	
8	296964	45349	47542	44350	35913	42713	31084	35802	50900	35830	44850	122893	59559	349002	404203	57061	
9	86999	118083	94566	66722	66156	72295	90511	50387	138768	51157	77438	367594	200891	388211	2730229	495691	
10	197690	56046	87564	173196	86791	85176	73851	123629	197093	62806	303356	141275	234633	6016193	11865124	1333315	
11	99226	107135	61748	165363	83381	43163	54134	677856	522331	77446	391015	123407	193151	2218709	5002499	280376	
12																	
13	747765	403117	343783	487155	305115	358655	901404	1151826	959717	265036	846424	783041	839508	9639061	20065136	2409003	
14	0.07	0.03	0.03	0.03	0.07	0.10	0.13	0.07	0.05	0.02	0.02	0.02	0.06	0.11	0.13	0.08	
15	48605	12497	8595	15102	22273	36583	112676	74869	44147	6096	16082	14878	46173	1041019	2688728	192720	4381041

	CN	CO	CP	CQ	CR	CS	CT	CU	CV	CW	CX	CY	CZ	DA	DB	DC	DD
1	N DOSE (ONE ASSEMBLY DROPPED UNDER WATER)	NNE DOSE	NE DOSE	ENE DOSE	E DOSE	ESE DOSE	SE DOSE	SSE DOSE	S DOSE	SSW DOSE	SW DOSE	WSW DOSE	W DOSE	WNW DOSE	NW DOSE	NNW DOSE	WEIGHTED AVERAGE POPULATION DOSE
2	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.02	0.11	0.01	0.04	
3	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
4	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
5	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.02	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
6	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.02	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
7	0.00	0.01	0.01	0.00	0.00	0.02	0.12	0.01	0.00	0.00	0.00	0.00	0.01	0.01	0.01	0.00	
8	0.06	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.02	0.01	0.07	0.08	0.01	
9	0.02	0.02	0.02	0.01	0.01	0.01	0.02	0.01	0.03	0.01	0.02	0.07	0.04	0.08	0.53	0.10	
10	0.04	0.01	0.02	0.03	0.02	0.02	0.01	0.02	0.04	0.01	0.06	0.03	0.05	1.17	2.30	0.26	
11	0.02	0.02	0.01	0.03	0.02	0.01	0.01	0.13	0.10	0.02	0.08	0.02	0.04	0.43	0.97	0.05	
12																	
13	0.15	0.08	0.07	0.09	0.06	0.07	0.18	0.22	0.19	0.05	0.16	0.15	0.16	1.87	3.90	0.47	
14	0.07	0.03	0.03	0.03	0.07	0.10	0.13	0.07	0.05	0.02	0.02	0.02	0.06	0.11	0.13	0.08	
15	0.01	0.00	0.00	0.00	0.00	0.01	0.02	0.01	0.01	0.00	0.00	0.00	0.01	0.20	0.52	0.04	0.85