**AP1000 Oversight Group, Bellefonte Efficiency and Sustainability Team,
Blue Ridge Environmental Defense League,
Citizens Allied for Safe Energy, Friends of the Earth,
Georgia Women's Action for New Directions, Green Party of Florida,
North Carolina Waste Awareness and Reduction Network,
Nuclear Information and Resource Service, Nuclear Watch South,
SC Chapter - Sierra Club, Southern Alliance for Clean Energy**

<u>VIA MAIL AND EMAIL</u>

April 21, 2010

Dr. Said Abdel-Khalik, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Re: PETITION TO INITIATE SPECIAL INVESTIGATION ON SIGNIFICANT AP1000 DESIGN DEFECT

Dear Chairman:

The above local, regional and national organizations are requesting that the Advisory Committee on Reactor Safeguards ("ACRS") initiate a special investigation on an unreviewed safety issue which fundamentally calls into question the adequacy of the AP1000 reactor design to protect public health and safety in the event of an accident.

The basis for our concern is described in the attached report by Arnold Gundersen, Chief Engineer, Fairewinds Associates, Inc., "Post Accident AP1000 Containment Leakage: An Unreviewed Safety Issue," April 7, 2010 ("Fairewinds Report"). This report is further supported by an affidavit from Dr. Rudolf H. Hausler, Corro Consulta.

As the Fairewinds Report states, one of the design features in the Westinghouse AP1000 reactors is that in a post accident event, radioactive leakage from a containment failure could be deliberately wafted out into the environment. The result of this potential design flaw is that during containment breach, a significant volume of radionuclides will be released into the air with the potential for a significant public health catastrophe.

As the Fairewinds Report states, rather than resolve the real world impacts resulting from this unique design weakness, the Westinghouse analysis relies on several significant and extraordinary assumptions to "minimize" its impact. Westinghouse has failed in its efforts to prove that there is no need to modify the AP1000 containment and shield building in order to eliminate the possibility of releases directly into the environment and to protect public health and safety. In fact, containment failure through only a small hole similar to that at Beaver Valley is likely to exist when the design basis event occurs.

While your committee is investigating the potential defect in the AP1000 design, we have appealed to Chairman Jaczko for the NRC staff to immediately review it also.

We will be glad to meet with the ACRS to assist you in your investigation. Please contact me at the address below and I will inform the organizations that have joined in this petition of the scope of your investigation and what assistance we can provide.

Sincerely,

John D. Runkle
Attorney at Law
Counsel for the AP1000 Oversight Group
Post Office Box 3793
Chapel Hill, NC 27515
jrunkle@pricecreek.com

ENC. Fairewinds Report

cc. Chairman Gregory B. Jaczko

Post Accident AP1000 Containment Leakage

An Unreviewed Safety Issue

Fairewinds Associates, Inc, April 7, 2010

A Report by Arnold Gundersen, March 26, 2010
Chief Engineer, Fairewinds Associates, Inc

Affidavit by Rudolf H. Hausler, PhD, Corro-Consulta
Re. Post Accident AP1000 Containment Leakage:
An Un-reviewed Safety Issue

Attachments**:**
Attachment 1 – Curriculum Vitae
Attachment 2 – Table 1 from *Detection of Aging Nuclear Power Plant Structures*
Attachment 3 – Table 35-4 *Summary Of Release Category Definitions*
Attachment 4 – *Declaration Of Arnold Gundersen Supporting Citizen Power's Petition*
Attachment 5 – *Declaration Of Arnold Gundersen Supporting Connecticut Coalition Against Millstone In Its Petition For Leave To Intervene, Request For Hearing, And Contentions*

Post Accident AP1000 Containment Leakage
An Unreviewed Safety Issue

A Report by Arnold Gundersen[1]
March 26, 2010

1. Introduction

The AP1000 design has no secondary containment to provide for fission product control following a design basis accident. The purpose of this report is to describe the basis for concerns regarding an apparently unreviewed safety issue raised by the AP1000 containment system design (Revision 18).

My four concerns are:

- Recent experience with the current generation of nuclear reactors shows that containment corrosion, cracking, and leakage are far more prevalent and serious than anticipated by the U.S. Nuclear Regulatory Commission (NRC) in establishing its regulatory program for the safe operation of nuclear reactors.

- By design, the AP1000 containment has an even higher vulnerability to corrosion than containment systems of current reactor designs because the outside of the AP1000 containment is subject to a high-oxygen and high-moisture environment conducive to corrosion and is prone to collect moisture in numerous inaccessible locations that are not available for inspection.

- By design, the AP1000 containment has an even higher vulnerability to unfiltered, unmonitored leakage than the current generation containment system designs, and it lacks the defense in depth of existing structures. While the AP1000 is called an *advanced passive system*, in fact the containment design and structures immediately outside the containment are designed to create a chimney-like effect and draw out any radiation that leaks through the containment into the

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[1] Arnold Gundersen is the Chief Engineer with Fairewinds Associates, Inc., a paralegal and expert witness firm that specializes in nuclear safety, engineering, and reliability issues. Mr. Gundersen holds a bachelor's and master's degree in nuclear engineering and has more than 38 years of experience in nuclear power plant operation, management and design. A copy of his curriculum vitae is attached.

environment. Such a system will also facilitate the more efficient release of unfiltered, unmonitored radiation from any cracks or holes that might develop in the containment.

- Finally, a leakage path exists that is not bounded by any existing analysis and will be more severe than those previously identified by Westinghouse in its AP1000 application and various revisions.

The potential consequences of a radiation release to the environment from a small hole or crack in the AP1000 containment are significant. A containment hole approximately ¾" by ¼", like the one discovered at Beaver Valley in 2009, would create exposure to the public well in excess of the 25 rem limit in 10 CFR 100.11(2) for the entire period of the accident. A hole that is the size of the hole in Beaver Valley's containment is not a low probability event, as several through-wall liner holes have already occurred in existing nuclear containments. Therefore, it is not a concept to be pushed off into the severe accident category. Yet, to my knowledge, neither Westinghouse nor the NRC has adequately analyzed this significant safety issue for the AP1000 design.

2. Background of Containment Design

2.1 General. All nuclear power reactor containment systems are designed to contain the radiation and energy that would be released during a Loss Of Coolant Accident (LOCA). In the absence of a containment system, post accident exposures to the public would be unacceptably high. "A containment building, in its most common usage, is a steel or concrete structure enclosing a nuclear reactor. It is designed to contain the escape of radiation… during any emergency. The containment is the final barrier to radioactive release, the first being the fuel ceramic itself, the second being the metal fuel cladding tubes, the third being the reactor vessel and coolant system."[2]

2.2 Current Reactor Containment Designs. According to H.L. Graves, III, NRC, and D.J. Naus, Oak Ridge National Laboratories, there are two main types of

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[2] http://encyclopedia.thefreedictionary.com/containment+structure

containment designs currently in operation: freestanding containments and concrete containments with liners.[3]

> <u>Freestanding Containments are:</u>
> "freestanding, welded steel structures that are enclosed in a reinforced concrete reactor or shield building. The reactor or shield buildings are not part of the pressure boundary and their primary function is to provide protection for the containment from external missiles and natural phenomena (e.g., tornadoes or site-specific environmental events). Thirty-two of the NPPs licensed for commercial operation in the US employ a metal containment."[4]

> <u>Concrete Containments With Liner are:</u>
> "metal lined, reinforced concrete pressure-retaining structures that in some cases may be post-tensioned. The concrete vessel includes the concrete shell and shell components, shell metallic liners, and penetration liners that extend the containment liner through the surrounding shell concrete. The reinforced concrete shell, which generally consists of a cylindrical wall with a hemispherical or ellipsoidal dome and flat base slab, provides the necessary structural support and resistance to pressure-induced forces. Leak-tightness is provided by a steel liner fabricated from relatively thin plate material (e.g., 6-mm thick) that is anchored to the concrete shell by studs, structural steel shapes, or other steel products… Seventy-two of the NPPs licensed for commercial operation in the US employ either a reinforced concrete (37 plants) or post-tensioned concrete (35 plants) containment.[5]"

2.3 AP1000 Containment Design. The proposed AP1000 reactors use concepts common to both types of containment system designs to create a wholly *new hybrid containment* that has had no prior operational history. While the AP1000 is a PWR that uses a dry containment system similar to that which most other existing PWRs use, unlike most currently operating PWRs, the AP1000 design proposes to use a freestanding steel containment and no secondary containment.

2.4 Existing freestanding containment systems are normally surrounded by a reactor building that also acts as a filtered enclosure in the case of a design-basis accident. In the AP1000 design, the freestanding steel containment is surrounded by a

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[3] Naus, D.J. and Graves, III, H.L., <u>*Detection of Aging Nuclear Power Plant Structures,*</u> Proceedings of the OECD-NEA Workshop on the Instrumentation and Monitoring of Concrete Structures, NEA/CSNI/ R(2000)15, Organization for Economic Cooperation and Development – Nuclear Energy Agency, ISSY-les-Moulineaux, France, 2001.
[4] *Id.,* page 3.
[5] *Id.,* pages 3-4.

shield building that is not intended or designed to filter exhaust gases that may leak from the steel containment in the event of an accident.

The AP1000 containment has another unique feature: following an accident it serves a role as a heat exchanger. Unlike any previous containment system ever built, the AP1000 uses a large tank of water above the shield building to pour water directly onto the outside of the steel containment shell. After an accident, the falling water then cools the containment shell, which then cools the radioactive steam inside the containment via two processes known as thermal conduction and convection during which the steel shell evaporates the water that is sprayed from above. As stated in a Westinghouse report:

> "The steel containment vessel provides the heat transfer surface that removes heat from inside the containment and transfers it to the atmosphere. Heat is removed from the containment by the continuous, natural circulation of air. During an accident, air cooling is supplemented by water evaporation. The water drains by gravity from a tank located on top of the containment shield building."[6]

The process of falling water effectively converts the containment into a heat exchanger rather than the passive containment building that is the hallmark of the original PWR containment system design.

2.5 History of NRC Containment Analysis. One of the hallmarks of NRC regulation is that licensees and applicants must apply either *conservative assumptions* or *conservative estimates* in order to meet the NRC's statutory requirement to protect public health and safety. The dictionary defines "*conservative*" *as "Moderate: cautious: a conservative estimate"*. The pattern of recently uncovered weakness in the overall integrity of the current operating containment system design methodology proves that presumptions made for the AP1000 containment system considered in the containment design bases lack the level of prudence and caution as required to protect public health and safety.

3. Discussion

3.1 History of Containment Corrosion and Leakage A recent string of failures in

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[6] W.E. Cummins, et al, *Westinghouse AP1000 Advanced Passive Plant,* Proceedings of ICAPP '03, Cordoba, Spain, May 4-7, 2003, Paper 3235.

the current generation of containment systems strongly indicates that these current containment systems are not as impervious to the post accident environment as was anticipated and calculated by NRC and the nuclear industry in conducting design basis analysis for nuclear reactors. As discussed below in paragraph 3.1.8, this disturbing trend calls for a new analysis of the potential for containment corrosion and leakage. As further discussed in Section 3.2 below, the need for such an analysis is all the more pronounced with respect to the AP1000 design, which appears to invite corrosion through the establishment of a moist oxygenated environment.

For Example:

3.1.1 Beaver Valley. The NRC and the ACRS have received expert witness testimony concerning three pitting indications at Beaver Valley in 2006 and a through-wall hole at Beaver Valley in 2009 as delineated in the April 23, 2009 NRC Event Notification Report 45015. Moreover, the Beaver Valley NRC Event Notification Report clearly shows that visual inspections have proven inadequate to discover leaks before the leaks penetrate the entire metal surface. Below is a picture taken in April 2009 of a through-wall hole in the Beaver Valley containment that was undetected until complete penetration of the liner had occurred.

BEAVER VALLEY UNIT 1 LINER HOLE

3.1.2 European PWRs. Weld anomalies in the containment liner of the latest generation European Pressurized Reactor at Framanville 3 have caused construction delays and setbacks.[7] Weld anomalies may lead to crevices that create through-wall corrosion if they occurred in the unique AP1000 containment design. While there is a significant amount of European data, the data cited in this report is limited to United States nuclear power plants.

3.1.3 Naus and Graves Study. In their treatise, _Detection of Aging Nuclear Power Plant Structures_, Naus and Graves have created a lengthy and comprehensive list of 66 containment system failures beginning as early as 1970 and following through to the end of their published research in 1999. According to their report:

> "As nuclear plant containments age, degradation incidences are starting to occur at an increasing rate, primarily due to environmental-related factors. There have been at least 66 separate occurrences of degradation in operating containments (some plants may have more than one occurrence of degradation). One-fourth of all containments have experienced corrosion, and nearly half of the concrete containments have reported degradation related to either the reinforced concrete or post- tensioning system. Since 1986, there have been over 32 reported occurrences of corrosion of steel containments or liners of reinforced concrete containments. In two cases, thickness measurements of the walls of steel containments revealed areas that were below the minimum design thickness. Two instances have been reported where corrosion has completely penetrated the liner of reinforced concrete containments. There have been four additional cases where extensive corrosion of the liner has reduced the thickness locally by nearly one-half (10)."[8]

Naus and Graves also report that: "Since the early 1970's, at least 34 occurrences of containment degradation related to the reinforced concrete or post-tensioning systems have been reported."[9]

More disturbingly, Naus and Graves chronicled 32 reported incidences of steel containment or liner degradation that are particularly germane to anticipated problems

[7] Oliver, Anthony and Owen, Ed, *New Civil Engineer Magazine* June 18, 2009
[8] *Id.,* page 5.
[9] *Id.,* page 6.

with the proposed AP1000 containment system. While some of the problems detailed by Naus and Graves are corrosion or pitting that did not completely penetrate the containment system, *their report also uncovered complete containment system failures of either the liner or the steel containment shell*. Table 1, labeled Attachment 2, from Detection of Aging Nuclear Power Plant Structures identifies through-wall containment cracks that occurred in 1984 at Hatch 2,in 1985 at Hatch 1, and in 1999, North Anna 2 also experienced a through-wall hole in its containment.

Naus and Graves also identify significant problems with containment inspections in locations where inspections are difficult due to inaccessibility. It is stated on Page 18 of their report that:

> "**Inaccessible Area Considerations**
> Inspection of inaccessible portions of metal pressure boundary components of nuclear power plant containments (e.g., fully embedded or inaccessible containment shell or liner portions, the sand pocket region in Mark I and II drywells, and portions of the shell obscured by obstacles such as platforms or floors) requires special attention. Embedded metal portions of the containment pressure boundary may be subjected to corrosion resulting from groundwater permeation through the concrete; a breakdown of the sealant at the concrete-containment shell interface that permits entry of corrosive fluids from spills, leakage, or condensation; or in areas adjacent to floors where the gap contains a filler material that can retain fluids. Examples of some of the problems that have occurred at nuclear power plants include corrosion of the steel containment shell in the drywell sand cushion region, shell corrosion in ice condenser plants, corrosion of the torus of the steel containment shell, and concrete containment liner corrosion. In addition there have been a number of metal pressure boundary corrosion incidents that have been identified in Europe (e.g., corrosion of the liner in several of the French 900 MW(e) plants and metal containment corrosion in Germany). Corrosion incidences such as these may challenge the containment structural integrity and, if through-wall, can provide a leak path to the outside environment." [10]

Not only do Naus and Graves identify inspection problems with containments in the United States, but also in Europe. The data they collected, however, only reflect containment problems in the United States. While their report was written in 1999, the

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[10] *Id*., Page 18

inspection problems have actually accelerated in severity since that time, with the most recent containment problem reviewed occurring at Beaver Valley in April 2009.

3.1.4 Reports in NRC Information Notice. The 66 incidences of containment system degradation occurring between 1970 and 1999 and reported by Naus and Graves appear to be comprehensive for that specific period of time. While my research to date has not uncovered a comprehensive and all-inclusive list for the current decade from 1999 to present, my review of *USNRC Information Notice 2004-09* identified another eight additional episodes of containment system degradation including a through-wall hole in the containment liner at D.C. Cook in 2001, three through-wall holes through the liner at Brunswick in late 1999, and 60 areas of pitting at D.C. Cook (Ice Containment) in 1998 where the liner was not penetrated but the thickness of the pitting was below the minimum design value[11].

According to the evidence reviewed, at least 77 instances of containment system degradation have occurred at operating US reactors since 1970, including two through-wall cracks in steel containments (Hatch 1 & 2), six through-wall holes in containment liners (Cook, North Anna 2, Beaver Valley 1, and three at Brunswick), and at least 60 instances of liners pitting to below allowable minimum wall thickness (minimum design value).

3.1.5 Citizens Power Report. In its May 2009 filing regarding Beaver Valley's application for a 20-year license extension, Citizen Power recently informed the NRC's Advisory Committee on Reactor Safeguards (ACRS) of the increased likelihood of containment system leakage failures. The expert witness declaration, entitled *Declaration Of Arnold Gundersen Supporting Citizen Power's Petition* and attached herein as Attachment 3 and contained within Citizen Power's filing to the ACRS, identified the *industry-wide* significance of the containment liner hole at Beaver Valley. The declaration detailed potential causes of containment through-wall liner failure and the currently existing weaknesses in inspection techniques on PWR containment systems.

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[11] The minimum standard upon which the licensing design of this specific nuclear power plant was predicated and upon which risk assessment data was factored.

The _Declaration Of Arnold Gundersen Supporting Citizen Power's Petition_ also addresses United States patents on containment design that clearly state that concrete containment structures are considered porous to radioactive gases and no credit for retention of radiation in concrete may be allowed.[12]

3.1.6 ACRS 2008 Meeting with Connecticut Coalition Against Millstone.

Following my July 9, 2008 testimony to ACRS regarding potential problems with Dominion Nuclear Connecticut Inc.'s Millstone Unit 3's sub-atmospheric *containment system*, the ACRS questioned a *containment specialist staff member of NRC* as to whether the NRC even has the capability to analyze a sub-atmospheric containment. According to the NRC *containment specialist*, the NRC cannot accurately analyze containment systems.

The NRC *containment specialist* and staff member said:

> "It's sort of difficult for us to do an independent analysis. It takes time. We're not really set up to do it. The other thing you have to realize, too, for containment, which isn't as true in the reactor systems area, is that **we don't have the capability**."[13]

To date, the NRC ACRS has met at least twice to discuss Citizen Power's concerns regarding liner failures and the transcripts of those meetings contain key details for containment system failure that should be of concern to the entire nuclear industry.

The most informed discussion of the probability of significant leakage from a PWR containment system may be found in the July 8, 2009 ACRS transcript regarding the Citizen Power petition alerting the NRC to the magnitude and significance of the failure of the containment system. The specific text relating to probability of gross containment leakage is addressed on Page 40 of the July 8, 2009 ACRS transcript:

> "MEMBER RAY: At which point the condition of the concrete can't be taken credit for. So I guess I just think that **the idea that the leakage is**

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[12] According to one of Stone and Webster's patents, "A Sub-atmospheric double containment system is a reinforced concrete double wall nuclear containment structure with each wall including an essentially impervious membrane or liner and **porous concrete** filling the annulus between the two walls." US Patent 4081323 Issued on March 28, 1978 to Stone & Webster Engineering Corp. [Emphasis Added]

[13] ACRS Transcript, July 9, 2008, page 88 lines 6-11 [Emphasis added]

> **going to be small from a small hole, from a hole this size, as small as Dan says, in the design-basis conditions isn't logically supportable because the concrete, you can't -- you, yourself said, you can't take credit for the concrete and the reason is because it's condition in the design-basis event can't be predicted, can't be credited. The only thing you can credit is the membrane itself**.
>
> MEMBER SHACK: From a deterministic basis, you're correct. From a probabilistic basis, which is what they use and can take credit based on –
>
> MEMBER RAY: I don't think so.
>
> MEMBER SHACK: Well, that's the way it is.
>
> MEMBER RAY: That's not right."[14]

The July 8, 2009 ACRS discussion between ACRS members Ray and Shack regarding the probability of significant leakage from a PWR containment system occurred after failure of the containment liner at Beaver Valley.

- Ray emphasizes that deterministically the steel containment liner is the only leakage barrier that protects the public.
- Shack implies that the if the liner fails, radiation leaks would be delayed by the concrete containment behind it and therefore a probabilistic risk assessment credit should be given for that reduction in dose release.

My 2008 testimony to ACRS contradicts Shack's assessment and directs one to the original patent delineating the fact that concrete is porous. [See footnote 12]. In the case of the AP1000 design, there is no porous concrete secondary barrier suggested by Shack. Therefore, in regards to the AP1000 design, Ray's position is both deterministically and probabilistically correct.

These ACRS discussions, and further correspondence submitted to the ACRS by Citizen Power indicate that the ACRS has developed an increased awareness of the newly uncovered weaknesses in PWR containment designs. Moreover, a more detailed discussion, including my analysis of the containment issues at Millstone, is detailed within my expert report entitled _Declaration Of Arnold Gundersen Supporting Connecticut Coalition Against Millstone In Its Petition For Leave To Intervene, Request For Hearing, And Contentions,_ herewith filed as Attachment 4.

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[14] Transcript, page 40 [emphasis added].

Furthermore, the ACRS wrote a letter to NRC Executive Director for Operation R. W. Borchart on September 21, 2009 entitled _Request By The ACRS For A Future Briefing By NRR On Current Containment Liner Corrosion Issues And Actions Being Taken By The Staff To Address Them_ in which the ACRS said:

> "During the 565th meeting of the Advisory Committee on Reactor Safeguards, September 10-12, 2009, the Committee indicated the need for a future briefing by NRR on the topic of containment liner corrosion. **In recent years liner corrosion issues have been identified on a few of the operating nuclear power reactors. The Committee would like to hear from NRR about current staff efforts to address these issues generically.** Please let us know about a proper date and time for this briefing to take place.[15]

3.1.7 Petrangeli Report. The ACRS is not the only organization expressing concern regarding the overall integrity of PWR containments. In his book _Nuclear Safety_, Dr. Gianni Petrangeli, a nuclear engineering professor at the University of Pisa in Italy, also reported his concern regarding the likelihood of *containment* breaches *and the probability of severe post-accident leakage from a PWR containment*. In his book, Dr. Petrangeli noted:

> "There is a tendency in the design phase to specify for the containments a figure for the maximum admissible leakage rate which is close to that which is technically obtainable in ideal conditions… In the course of plant operation however, even if at the start the leak rate was the specified one or lower, a certain deterioration in the containment leak rate takes place and then in the case of an accident, the leak rate would probably be higher than that measured in the last leakage test…. In depth studies ... were performed on the deterioration probability of the leak proofing in real containment systems. The picture that emerges is not very reassuring… The probability of overcoming the specification values in the case of an accident is 15 per cent for BWR's and 46 percent for PWRs"[16].

Using US NRC data gathered from 1965 through 1988 and NUREG-1273 on containment leakage from a variety of sources, Dr. Petrangeli presents the probability that a containment system will exceed its technical specification limits during an accident in Table 14-2 reproduced below.

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[15] Meeting Transcript, page 40 [Emphasis Added]

[16] Petrangeli, Gianni, Nuclear Safety, Butterworth-Heinemann, 2006, ISBN 10: 0-7506-6723-0, Page 141.

Table 14-2. Measured containment leaks (USNRC 1988)

Leak measured relative to the specifications	BWRs*	PWRs*
From 1 to 10 times	0.10	0.31
From 10 to 100 times	0.04	0.08
Higher than 100	0.01	0.07

* These columns represent the probability of exceeding the technical specification leakage rates.

In my review of the more comprehensive data from the 1999 Naus and Graves study, as well as significant liner failures between 2000 and 2010 after Naus and Graves collected their data, the leakage rates in Table 14-2 of Dr. Petrangeli's 2006 book may in fact underestimate the post-accident containment system leakage risk.

Dr. Petrangeli further expressed his concerns based on his review of this data as it pertains to the new containment designs including the AP1000 when he said:

> "It is surprising that this issue does not receive much attention in the field of safety studies… This issue has been dealt with here because, for plants now under construction and for future ones, the tendency is to restrict the important consequences of severe accidents to within a very small distance from the plant possibly to avoid the need to evacuate the population. From this perspective, the real leakage of the containment system becomes very important."[17]

Dr. Petrangeli then continues by suggesting as a solution the exact opposite approach to that taken in the AP1000 containment design. Rather than act as a chimney and draw unfiltered gases from the gap between the containment and shield building as the AP1000 does, Petrangeli suggests as a possible solution for severe accident dose mitigation would be "… systems with a double containment with filtering of the effluents from the annulus between the containments…" when a secondary containment can be constructed. I note that the AP1000 shield building is not designed to "contain" any gases, and that Westinghouse has stated, "There is no secondary containment provided for the fission product control following a design basis accident." (AP1000 DCD, Rev. 16, Section 6.5.3.2).

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[17] *Id.,* page 142.

3.1.8 Conclusions Regarding Containment Degradation and Leakage.
As discussed above, the recent history of nuclear reactor operation shows a disturbing, unanticipated and unanalyzed trend of containment corrosion and leakage. This trend is seen in both standard containments and in containment designs such as the sub-atmospheric design used at Millstone and six other plants, and the ice containment system that has a litany of serious safety related containment failures. And clearly, the newfound containment liner hole at Beaver Valley creates a dilemma for both the industry and regulators in that it shows the increased likelihood of gross leakage by a PWR containment system that would significantly compromise public health and safety.

In my professional opinion, this disturbing trend calls for a new analysis of the potential for containment corrosion and leakage in the existing fleet of operating reactors. As further discussed in Section 3.2 below, the need for such an analysis is all the more pronounced with respect to the AP1000 design, which appears to invite corrosion through the establishment of a moist environment.

3.2 The Unique AP1000 Design Introduces An Unanalyzed Vulnerability

3.2.1 General. In the event the AP1000 containment leaks radioactive material into the annular gap between it and the shield building, the AP1000 is specifically designed to immediately act as a chimney and draw those vapors directly into the environment without filtration. The design of the AP1000 containment also has a greater potential to leak than existing containments with an increased likelihood that the leakage will exceed dose exposure limits at the Low Population Zone.

3.2.2 AP1000 Integrity and Corrosive Attacks. Well before the discovery of pitting (2006) or the through wall leak (2009) at Beaver Valley, the NRC expressed concerns about the integrity of the AP1000 containment to resist a corrosive attack. In 2003 the NRC wrote:

> "The staff's review of the containment shell design identified a concern that the 4.44 cm (1.75 in.) thickness of the cylindrical shell just meets the minimum thickness requirement of 4.4336 cm (1.7455 in.) of the 1998 ASME Code, Section III, Subsection NE, Paragraph NE-3324.3(a), based on a 406.8 kPa (59 psi) design pressure, a 148.9 °C (300 °F) design temperature, allowable stress, S = 182 MPa (26.4 ksi), and a containment vessel radius, R = 1981.2 cm (780 in.). **The staff noted that there is no**

> **margin in the nominal design thickness for corrosion allowance.** Of particular concern is the embedment transition region of the cylinder, which has been prone to corrosion in operating plants. Paragraph NE-3121 specifically requires that the need for a corrosion allowance be evaluated. Consequently, the staff requested the applicant to provide justification for (1) making no provision, in defining the nominal design thickness, for general corrosion of the containment shell over its 60-year design life, and (2) not specifying a corrosion allowance in the embedment transition region. In its response to RAI 220.002 (Revision 1), the applicant submitted the following information to address the corrosion allowance for the AP1000 containment shell:
>
>> The ASME Code of record has been updated to the 2001 Edition including 2002 Addenda. (The applicant has revised the DCD to incorporate this change.) Per the revised Code of record, S = 184.09 MPa (26.7 ksi) and tmin = 4.38 cm (1.726 in.), which provides a nominal margin for corrosion of 0.06 cm (0.024 in.).
>>
>> The design has been changed to add a corrosion allowance for the embedment transition region, as was provided for the AP600. The nominal thickness of the bottom cylinder section is increased to 4.76225 cm (1.875 in.) and the vertical weld joints in the first course will be post-weld, heat-treated per ASME Code requirements. Design of Structures, Components, Equipment, and Systems
>>
>> Corrosion protection has been identified as a safety-related function for the containment vessel coating in DCD Tier 2, Section 6.1.2.1.1, "General (Protection Coatings)." The COL applicant will provide a program to monitor the coatings, as described in DCD Tier 2, Section 6.1.3.2, "Coating Program."
>>
>> On the basis that enough corrosion allowance and proper corrosion protection were provided, the staff found the applicant's response acceptable, pending (1) incorporation of the design change in the cylinder embedment transition region in a future revision, and (2) designation of the "inhibit corrosion" function as "safety" for coatings on the outside surface of the containment vessel in a future revision of DCD Tier 2, Table 6.1-2. This was Confirmatory Item 3.8.2.1-1 in the DSER." [18]

The use of the term *corrosion allowance* refers to situations during which the containment experiences general corrosion over a large area. This general corrosion is a structural problem because it is a broad attack upon the entire structure rather than a pinhole, and therefore the NRC staff concern regarding a general corrosion issue with the

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[18] Page 3-106 AP1000 SER

AP1000 does not address the potential for the through-wall pitting problem reviewed and analyzed in this report. The unique features of the AP1000 exacerbate the likelihood of through-wall pitting corrosion that would increase post accident leakage.

The NRC requirements for increasing the thickness of the AP1000 containment by only one-eighth of an inch and by adding field applied protective coatings do not provide adequate assurance to mitigate potential pitting. The proposed NRC remedies are inadequate in light of industry experience and the unique features of the AP1000 containment design. One needs only to review the 3/8”-thick hole at Beaver Valley which occurred on a field coated surface and other through-wall failures discussed above to conclude that the 1/8 inch corrosion allowance in the AP1000 design is simply not adequate to address pitting.

3.2.3 Vulnerability To Hole Propagation. As discussed in 3.1.3 above, Naus and Graves have already identified the difficulty of thoroughly inspecting inaccessible locations in any containment system. The data reviewed show that such inspections will be more problematic in the AP1000 where abundant air, moisture and corrosive chemicals may allow holes to continue to grow over extended periods of time thereby forming unlimited pockets of corrosion in crevasses at inaccessible locations. This action would likely be especially true in the vicinity of non heat-treated or poorly heat-treated welds of high strength steels. In comparison, the corrosion at Beaver Valley and other existing PWRs has not progressed quite as rapidly as what is projected to occur in the AP1000 because there was no constant replenishment of oxygen and moisture on the outside of the containment liner shell. However, in the event that a corrosion site begins on the outside of the AP1000 containment, unlimited amounts of oxygen, moisture and corrosive chemicals are available for the corrosion to propagate and eventually result in broad weakening of the shell by deep grooves.

The annular gap outside the AP1000 containment is continually subjected to air, is subject to moisture buildup from humidity and condensation in the air, and subject to corrosive chemicals creating the ideal incubator for crack propagation and the creation of holes. The AP1000 containment design effectively continuously "breathes" in air, moisture and contaminants into the annular gap between the shield building and the containment. “Breathing” in this case is what engineers would call natural convection. For example, at Turkey Point and other saltwater sites, that air would also contain salt and other minerals that give ocean air its familiar *ocean smell* and corrosivity of the salt water. On cooling tower sites, the AP1000 would "breath" in cooling tower drift (fine water droplets in the vapor cloud), containing chlorides and biocides and accumulated minerals in the cooling water. The net effect is that these chemicals are corrosive agents traveling immediately next to the outside of the steel containment.

Furthermore, the 8,000,000 gallon (8 million gallon) water tank situated above the containment may leak over extended periods of time thereby providing additional moisture to aid in the propagation of holes.

In addition to the possibility of holes or pitting in the wall of the AP1000 containment due to the factors previously discussed, there is also an additional failure mode due to corrosion that must be addressed. Since concrete cannot bond to steel, a gap or pocket will be formed at the interface between the containment wall and the concrete containment floor. History has proven that over time moisture and contamination will enter this gap and cause corrosion to begin. Once again, as Naus and Graves suggest, it is at just such an inaccessible location that pitting can grow to cause either complete failure of the containment system or deterioration of the containment wall thickness to below the Code Allowable.

A second method of containment integrity failure would also be possible at the junction between the concrete floor and steel wall. In this inaccessible location, it is most likely that corrosion would first form as numerous pits ultimately coalescing into a grove that would present a mechanism of loss of structural integrity called *buckling*. If devolved pitting were to occur at the junction between the concrete floor and steel wall, then the low margin of safety for the overall thickness of the AP1000 containment actually becomes a serious structural issue and not just a hole that causes increased leakage.

The net effect of all these parameters upon the AP1000 design is that through-wall holes or flaws below minimum allowable wall thickness are at least as vulnerable to develop in the new AP 1000 design as compared to the existing PWR containments in which the industry has already witnessed failures.

3.2.4 Inspection Of The AP1000 Containment. Current visual inspections of the containment from easily accessible areas within existing containments have a history of failing to identify any corrosion until the containment barrier itself has been penetrated. Visual inspection on the inside of all containments therefore relies upon a hole fully penetrating the containment in order to be detected.

My experience as a Senior Vice President of an ASME Section XI non-destructive testing division and my review of the AP1000 containment design has led me to conclude that the AP1000 design presents similar obstacles to visual and ultrasonic inspection techniques, and also introduces more locations that are inaccessible to inspection and prone to corrosive attack. Moisture buildup and corrosive agent attack in small crevasses between the containment and the shield building will most likely increase the likelihood of hole-propagation at exactly the locations that are most difficult or impossible to inspect.

3.2.5 Field Welding and Coatings on the AP1000. The AP1000 containment is not a single piece of steel but rather many sheets welded together in the field. These numerous field-welded connections to the containment provide ideal locations both for pitting and crevice corrosion to develop and horizontal surfaces for moisture to collect. In addition, an Idaho National Laboratories Report entitled *Study Of Cost Effective Large Advanced Pressurized Water Reactors That Employ Passive Safety Features* states that, "The containment vessel supports most of the containment air baffle. …Flow distribution weirs are welded to the dome as part of the water distribution system…"[19]

In addition to field-welds, coatings will also be applied to the containment in the field. According to the Idaho National Labs report, "The containment vessel is coated with an inorganic zinc coating".[20] While coatings can provide some protection when properly applied, there is no assurance that field application can be completely successful and will

[19] Pages 2-11 and 2-12 of an Idaho National Laboratories Report entitled Study Of Cost Effective Large Advanced Pressurized Water Reactors That Employ Passive Safety Features (DOE/SF/22170) dated November 12, 2003

[20] *Id.,* page 2-12.

last for the 40 to 60 years of projected operating life. In fact, field quality assurance problems during the construction of existing containments have been determined to be the root cause of many of the containment degradation issues identified earlier in this report. Moreover, there are oil and gas facilities where components have completely corroded even though they were protected by galvanic coatings. A galvanic coating protects only as long as the zinc is present as a metal. For protection, the zinc corrodes and thereby prevents the underlying iron from corroding. However, when the zinc is gone the iron corrodes.

Given that moisture and corrosive chemicals will be drawn into the gap between the shield building and the containment and that various welded connections will provide locations for pit and crevasse corrosion to initiate, it is possible that intergranular corrosion in weldments could propagate at a rate of 0.15inches per year of faster, and in locations that are under stress, cracks could form. In my opinion a small crack could create a hole that would remain undetected and completely penetrate the AP1000 containment in a through-wall leak within approximately ten years or less.

3.2.6 AP1000 Chimney Effect. The AP1000's containment design is uniquely designed to act like a chimney and draw air and moisture out of the annular gap between the containment and the shield building. In the event a containment hole develops, the pressure inside the containment will push any radioactivity into the annular gap and then that radioactivity will immediately be drawn out into the air above the reactor by this chimney effect.

3.2.7 Increased Radiation Exposure From A Leak Into Annular Gap. Based upon my experience in Integrated Leak Rate Testing, the industry expectation is that a ¼ inch hole in the containment will produce leakage in excess of 100 Standard Cubic Feet per Hour (SCFH) resulting in an off-site exposure of approximately 25-rem at the Low Population Zone (LPZ). The hole at Beaver Valley was significantly larger than the aforementioned industry standard and would have resulted in approximately ten times that exposure, as leakage increases with the square of the hole diameter. However, as noted earlier in the conversation between ACRS members Ray and Shack, the existing steel liner at Beaver Valley was also backed up by a concrete containment. No such redundancy is incorporated in the AP1000 design. A hole the size of Beaver Valley's would clearly exceed the NRC's Low Population Zone (LPZ) dose limits. Admittedly the AP1000 containment is thicker than Beaver Valley's, but hole propagation is not self-limiting in the AP1000 design as previously described.

3.2.8 Implications To The AP1000 Design. The ACRS concern regarding containment integrity following the discovery of the Beaver Valley hole, Dr. Petrangeli's concern with respect to new containment design leakage rates, and the detailed history of at least 77-containment system failures nationwide, demand a wholly new analysis to determine exactly how the newly proposed AP1000 design accommodates leakage through the wall of its unique hybrid containment system.

Containment system leakage from through-wall holes in steel has already occurred at North Anna, Beaver Valley, Hatch 1, Hatch 2, Cook and Brunswick. However, in each of these circumstances ACRS member Shack articulated the fact that there was another potential barrier by which to collect and filter the airborne radiation that leaked from the containment system. Previous freestanding steel containments with holes were enclosed within a reactor building into which the leakage entered and was controlled. The liner failures appeared to be backed up by a concrete containment building.

In the event of an accident at a proposed AP1000 reactor, leakage through the freestanding steel containment will pass directly into the gap between the steel and the shield building. Therefore, the proposed AP1000 containment design is inherently less safe than current reactors presently licensed and operating.

The following four pages contain accident sequence illustrations.

- Figure 1 – AP1000 in normal operation.
- Figure 2 – AP1000 design basis accident begins.
- Figure 3 – AP1000 containment hole opens as containment fills with radioactive gases.
- Figure 4 – AP1000 chimney effect draws radioactivity directly into the environment.

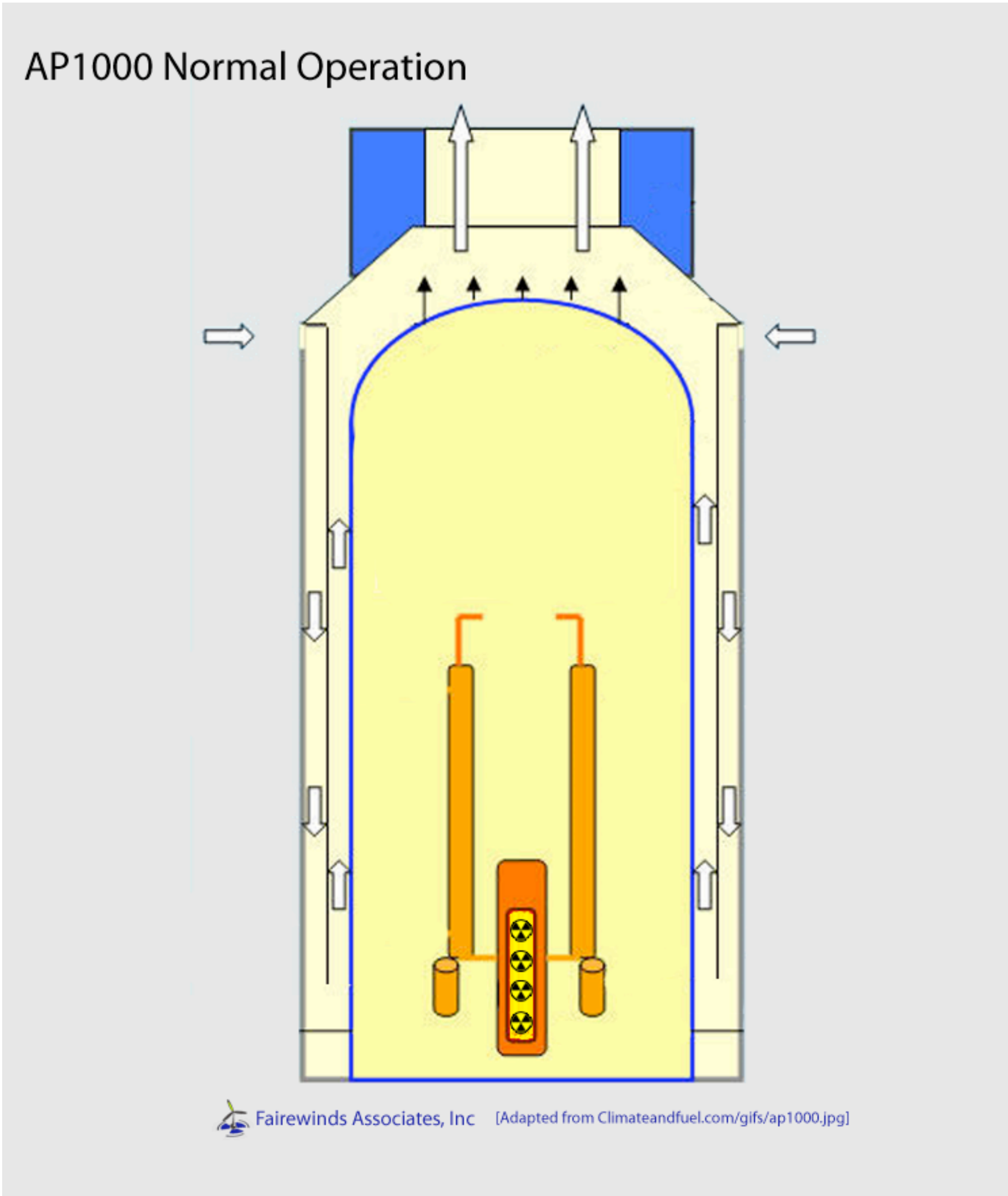
AP1000 Normal Operation
Fairewinds Associates, Inc
[Adapted from Climateandfuel.com/gifs/ap1000.jpg]

Figure 1

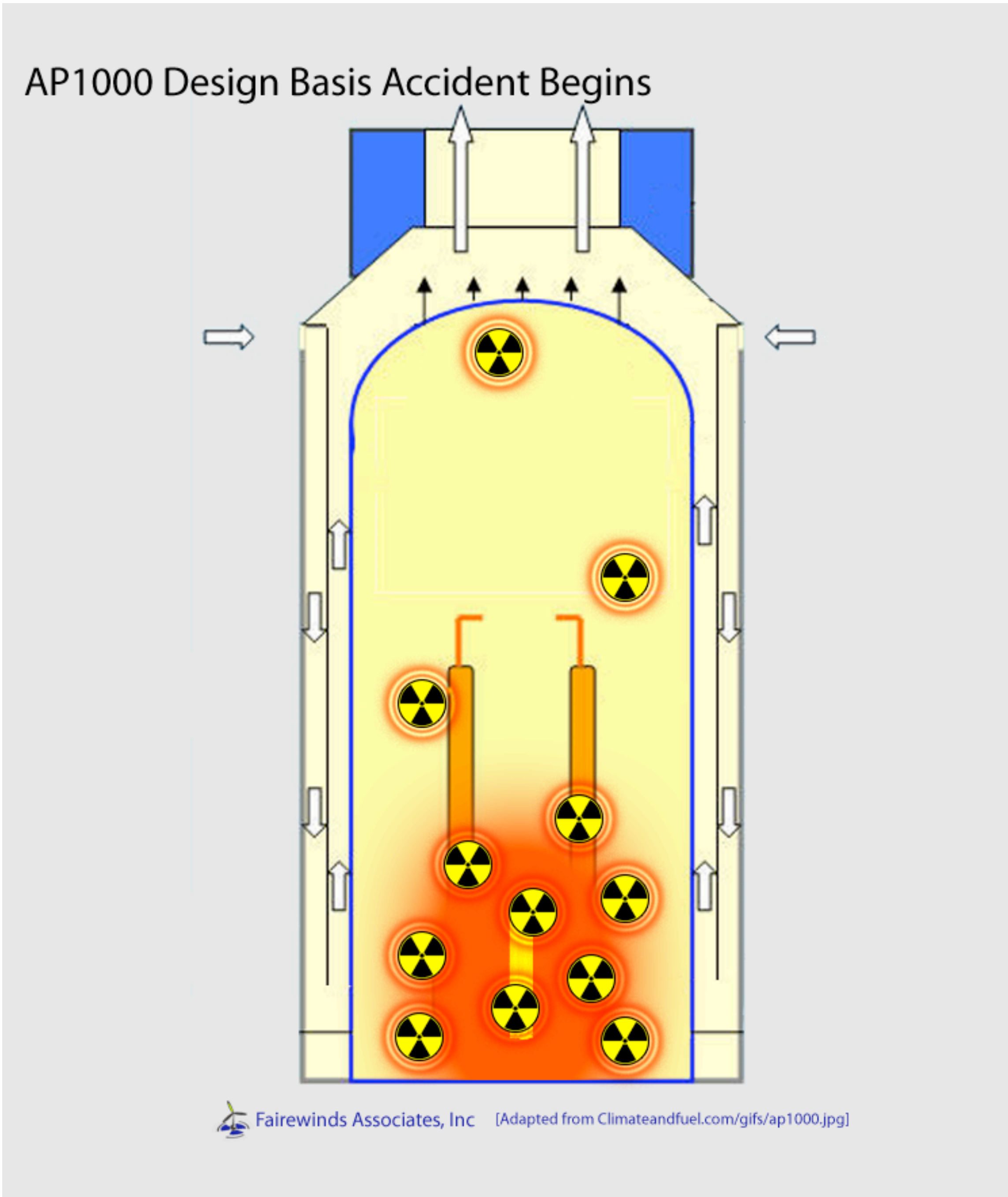
AP1000 Design Basis Accident Begins
Fairewinds Associates, Inc
[Adapted from Climateandfuel.com/gifs/ap1000.jpg]

Figure 2

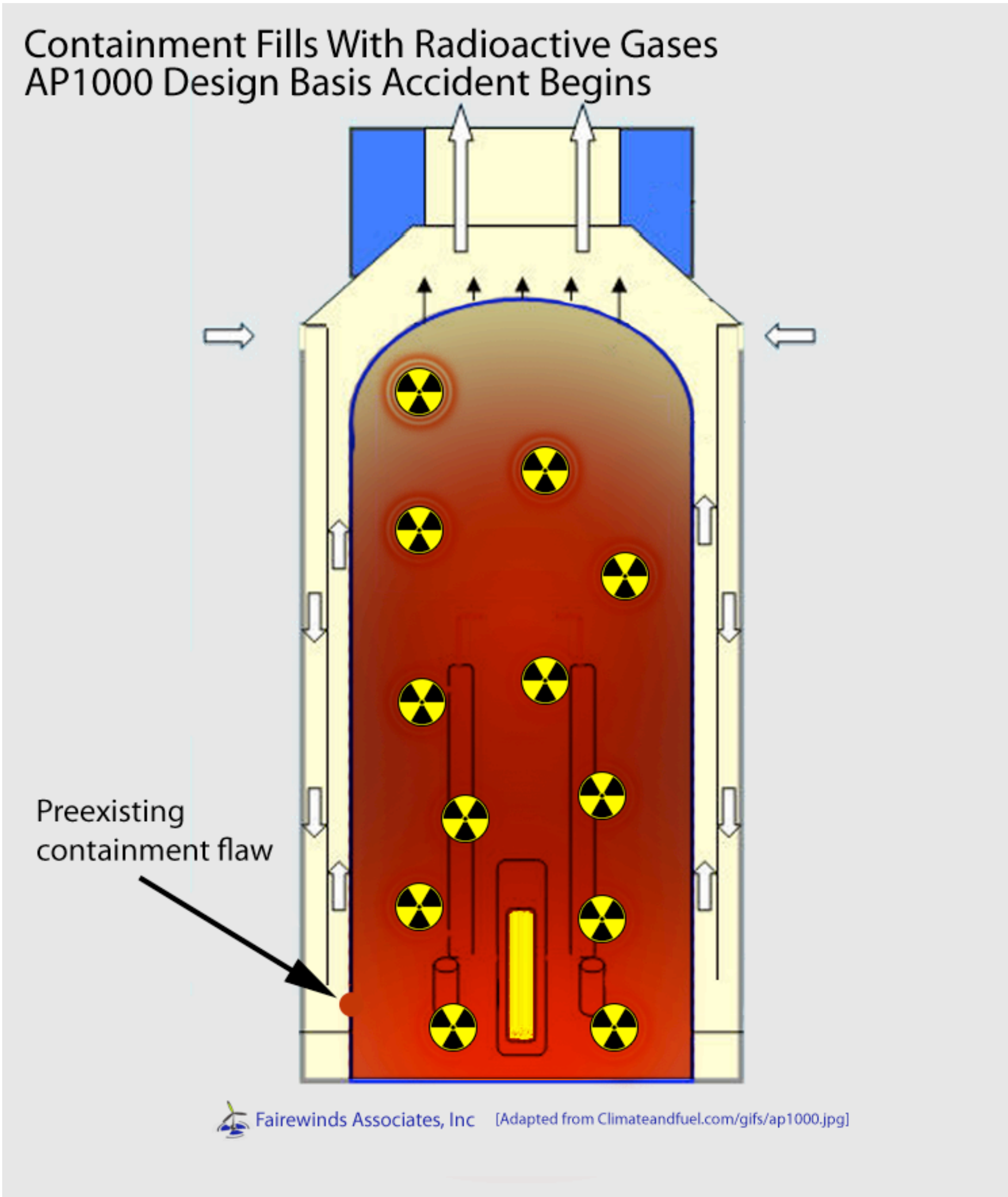
Containment Fills With Radioactive Gases
AP1000 Design Basis Accident Begins
Preexisting
containment flaw
Fairewinds Associates, Inc
[Adapted from Climateandfuel.com/gifs/ap1000.jpg]

Figure 3

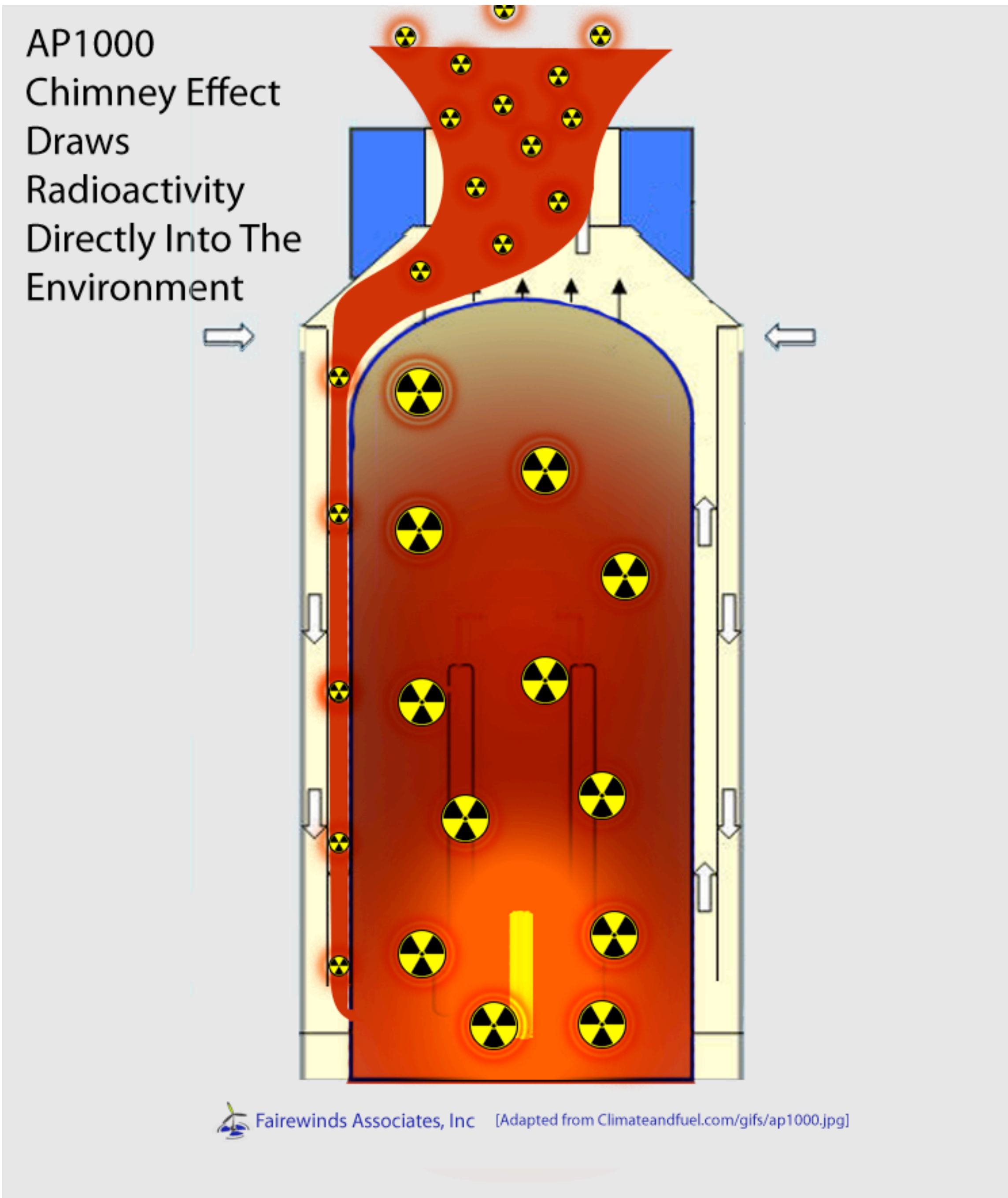
AP1000
Chimney Effect
Draws
Radioactivity
Directly Into The
Environment
Fairewinds Associates, Inc
[Adapted from Climateandfuel.com/gifs/ap1000.jpg]

Figure 4

Concernedly, the hybrid AP1000 containment system appears to lack any of the redundancy or defense in depth[21] in containment system design that was present in earlier designs reviewed in this report and upon which *design bases events* are predicated.

The hole in the Beaver Valley containment confirms Dr. Petrangeli's analysis about the increased likelihood of severe containment leakage. In his analysis, Dr. Petrangeli shows that there is at least a 10-percent likelihood and potentially a 31-percent likelihood of leakage from the AP1000 containment system being 10-times higher than that specified in the AP1000 Design Basis and Technical Specifications. This significant variation in potential leakage corresponds roughly to the size of the hole in the Beaver Valley Containment. See Table 14-2 on Page 12 for comparative chart.

Incongruously, the purpose of *the gap between the steel and the shield building* in the design has ***NOT*** been created to collect and treat radiation as Dr. Petrangeli suggests would be appropriate, but rather to allow air and moisture to cool the containment itself and then to act as a chimney allowing those gases to be siphoned directly out into the environment.

Consequently, the design of the proposed AP1000 containment and its shield building might actually cause the occurrence of a larger leakage rate and a higher probability of a through-wall leakage than the currently existing containment system failures discussed above due to the active role of the AP1000 shield building in acting as a chimney which draws radioactively contaminated air into the environment.

Specifically, the outside of the containment is designed to be wetted and for that reason it has millions of gallons of water suspended above it in order to provide moisture following an accident. More specifically, containment holes and leaks in existing

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[21] **Defense in depth** is an approach to nuclear power plant safety that builds-in layers of defense against release of radioactive materials so that no one layer by itself, no matter how good, is completely relied upon. To compensate for potential human and mechanical failures, *defense in depth* is based upon several layers of protection with successive barriers to prevent the release of radioactivity to the environment. This approach includes protection of the barriers to avert damage to the plant and to the barriers themselves. It includes further measures to protect the public, workers, and the environment from harm in case these barriers are not fully effective. *Defense in depth* is a hallmark of nuclear regulation and risk assessment to meet the statutory requirements inherent in the NRC responsibility to protect public health and safety.

containment systems were previously self-limiting because they ran out of moisture and oxygen. Moisture, oxygen and corrosive chemicals would be plentiful in the annular gap surrounding the containment and would promote the propagation of holes in normal AP1000 operational scenarios.

Existing data shows that containment system failures occur with moisture and oxygen. Therefore, it is clear that for the AP1000 design, leakage from the water tank, water from testing the tank, and/or atmospheric moisture due to the condensation on the water tank will create a constant environment of moisture and oxygen that may in fact provoke a through-wall containment failure in locations that are difficult and/or impossible to inspect.

Consequently, by looking at the historical record of containment system failures detailed in NRC records and in this report, and given the lack of a bond between the concrete floor and steel containment wall, and the inspection difficulty within crevasses in the annular gap between the AP1000 containment and the shield building, it is very likely that corrosion will develop that will limit the containment's effectiveness in the event of an accident.

4. Severe Accident Scenario or Design Basis Event?

4.2.1 General. Published reports indicate that the NRC already considers a breach of existing containments to be a plausible accident scenario. Emergency planning exercises at Oyster Creek and Callaway have already been based upon containment failure. My concern is that the potential for a breach of the AP1000 containment as discussed in this report is not a remote probability event, and may in fact occur prior to a design basis accident, and may remain undetected until the accident occurs.

4.2.2 AP1000 PRA. According to Chapter 35 of the Westinghouse AP1000 Probabilistic Risk Assessment on file with the NRC, Westinghouse has not assessed the possibility of radioactive gasses moving through the annular gap between the steel containment and the shield building and then directly out into the environment.

In Chapter 35 of the Westinghouse AP1000 probabilistic risk assessment, which is entitled _CONTAINMENT EVENT TREE ANALYSIS_, ***none*** of the seven AP1000 accident scenarios assumed containment leaks into the an annular gap of the shield building that would then move radiation out into the environment without filtration.

Moreover, in Table 35-4 entitled _SUMMARY OF RELEASE CATEGORY DEFINITIONS_ on page 35-24 of the report (reproduced as Attachment 5), only seven possible "*Release Categories*" have been defined and identified by Westinghouse as possible candidates for releasing gases into the environment following an accident. None of these release categories identified by Westinghouse include steel containment failure directly into the annular gap created by the shield building.

4.2.3 Severe Accident Mitigation Design Alternatives (SAMDA). As part of the AP1000's *Severe Accident Mitigation Design Alternatives (SAMDA)* analysis, Westinghouse claims to have considered and rejected the need for "Secondary Containment Filtered Ventilation". In its Revision 9 of the AP1000 Design Control Document, Page 1B-6 Westinghouse said:

> **"Secondary Containment Filtered Ventilation**
> This SAMDA consists of providing the middle and lower annulus… of the secondary concrete containment with a passive annulus filter system for filtration of elevated releases. The passive filter system is operated by drawing a partial vacuum on the middle annulus through charcoal and HEPA filters. The partial vacuum is drawn by an eductor with motive flow from compressed gas tanks. The secondary containment would then reduce particulate fission product release from any failed containment penetrations (containment isolation failure). In order to evaluate the benefit from such a system, this design change is assumed to eliminate the CI release category."

I have no understanding of why, in the above quotation, Westinghouse uses the term "*secondary concrete containment*" to refer to the AP1000 Shield Building. The Shield Building is proposed to be of modular construction and will not serve the purpose of containing radiation. It is not designed to contain anything, but rather is designed to disperse air and moisture used to cool the containment. *Westinghouse's use of the term "secondary concrete containment" is a misnomer.*

The starting point (base case) for all the AP1000 containment scenarios is the "Intact Containment". The intact containment is explained as "Release Category IC" on Page 1B-10:

> **"Release Category IC – Intact Containment**
> If the containment integrity is maintained throughout the accident, then the release of radiation from the containment is due to nominal leakage and is expected to be within the design basis of the containment. This is the "no failure" containment failure mode and is termed intact containment. The main location for fission-product leakage from the containment is penetration leakage into the auxiliary building where significant deposition of aerosol fission products may occur."

In addition to this base case scenario, the SAMDA analysis then postulates several extremely low probability events on Pages 1B-10 and 1B-11:

> "**Release Category CFE – Early Containment Failure**
> Early containment failure is defined as failure that occurs in the time frame between the onset of core damage and the end of core relocation. During the core melt and relocation process, several dynamic phenomena can be postulated to result in rapid pressurization of the containment to the point of failure. The combustion of hydrogen generated in-vessel, steam explosions, and reactor vessel failure from high pressure are major phenomena postulated to have the potential to fail the containment. If the containment fails during or soon after the time when the fuel is overheating and starting to melt, the potential for attenuation of the fission-product release diminishes because of short fission-product residence time in the containment. The fission products released to the containment prior to the containment failure are discharged at high pressure to the environment as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Containment failures postulated within the time of core relocation are binned into release category CFE."

> "**Release Category CFI – Intermediate Containment Failure**
> Intermediate containment failure is defined as failure that occurs in the time frame between the end of core relocation and 24 hours after core damage. After the end of the in-vessel fission- product release, the airborne aerosol fission products in the containment have several hours for deposition to attenuate the source term. The global combustion of hydrogen generated in-vessel from a random ignition prior to 24 hours can be postulated to fail the containment. The fission products in the containment atmosphere are discharged at high pressure to the environment as the containment blows down. Containment failures postulated within 24 hours of the onset of core damage are binned into release category CFI."

“**Release Category CFL – Late Containment Failure**
Late containment failure is defined as containment failure postulated to occur later than 24 hours after the onset of core damage. Since the probabilistic risk assessment assumes the dynamic phenomena, such as hydrogen combustion, to occur before 24 hours, this failure mode occurs only from the loss of containment heat removal via failure of the passive containment cooling system. The fission products that are airborne at the time of containment failure will be discharged at high pressure to the environment, as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Accident sequences with failure of containment heat removal are binned in release category CFL.”

“**Release Category CI – Containment Isolation Failure**
A containment isolation failure occurs because of the postulated failure of the system or valves that close the penetrations between the containment and the environment. Containment isolation failure occurs before the onset of core damage. For such a failure, fission-product releases from the reactor coolant system can leak directly from the containment to the environment with diminished potential for attenuation. Most isolation failures occur at a penetration that connects the containment with the auxiliary building. The auxiliary building may provide additional attenuation of aerosol fission-product releases. However, this decontamination is not credited in the containment isolation failure cases. Accident sequences in which the containment does not isolate prior to core damage are binned into release category CI.”

“**Release Category BP – Containment Bypass**
Accident sequences in which fission products are released directly from the reactor coolant system to the environment via the secondary system or other interfacing system bypass the containment. The containment failure occurs before the onset of core damage and is a result of the initiating event or adverse conditions occurring at core uncovery. The fission-product release to the environment begins approximately at the onset of fuel damage, and there is no attenuation of the magnitude of the source term from natural deposition processes beyond that which occurs in the reactor coolant system, in the secondary system, or in the interfacing system. Accident sequences that bypass the containment are binned into release category BP.”

4.2.4 Analysis of SAMDA Assumptions. A brief examination of the SAMDA assumptions Westinghouse applied to the AP1000 containment beyond its design basis (*Intact Containment*) scenario shows many non-conservative assumptions.

- For Release Category CLF (Late Containment Failure), Westinghouse assumes that the postulated containment failure occurs only 24-hours after the accident has begun and that the failure is due to the inability of the containment to remove decay heat. Westinghouse has simply made an arbitrary choice of the 24-hour number and the causative action.

- For Release Category CI (Containment Isolation), Westinghouse first assumes that the containment fails to properly isolate. Secondly, Westinghouse assumes that the isolation failure occurs at a containment penetration from which any additional leakage then enters the auxiliary building. Leakage into another building then provides additional filtration and delay. Westinghouse **does *not* assume** that the failure might occur at a location in the containment that directly exhausts into the annular ring between the containment and the shield building. Any leakage into this annular gap would then leak directly into the environment, which has not been factored into either the Westinghouse assessment or the NRC review of the Westinghouse data.

- For Release Category BP (Containment Bypass) Westinghouse has assumed that the containment is bypassed through an open piping system. Once again, Westinghouse fails to consider or factor in to its analysis that the containment failure might occur at a location in the containment that directly exhausts into the annular ring between the containment and the shield building. Any leakage into this annular gap would then leak directly into the environment. As delineated before, the Westinghouse assessment has not considered all the pertinent data.

Westinghouse has ignored the long history of previous containment and containment liner failures that indicate there is an unacceptably high risk that the AP1000 containment might be in a failed condition at the onset of an accident. Inspection results of existing PWR containments have shown numerous occasions when containment liners have completely failed or experienced holes below minimum allowable wall thickness. Therefore, there is a significant probability that leakage from the AP1000 containment would begin immediately and most likely **will *not* occur** at the site of containment penetration. This potential AP1000 leakage is not related to an extraordinary SAMDA event, but may be anticipated to exist at the beginning of the accident due to uninspected corrosion of the containment as discussed in this report. The leakage problem in the AP1000 design is exacerbated because it is the only containment design that has an annular gap specifically created to act as a chimney and draw air directly into the environment.

4.2.5 SAMDA Summation. In every case Westinghouse chose to analyze, it ignored the likelihood that radioactive leakage would move directly into the annular gap between the containment and the shield building.

Moreover, in the design *features* of the Westinghouse AP1000 reactor, this leakage would *be deliberately* wafted out into the environment. Furthermore, there are several significant and extraordinary assumptions within the Westinghouse analysis that has the net effect of minimizing the AP1000's unique design weakness.

These non-conservative SAMDA assumptions include:

- The likelihood of containment failure is minimized.
- The timing of the failure is delayed, hence reducing radionuclide concentrations.
- The location of the failure is chosen to avoid the annular gap.
- The likelihood of significant leakage is minimized.
- And, the dose consequences are therefore also minimized.

With these five erroneous assumptions, Westinghouse has failed in its efforts to *prove* that there is no need to modify the AP1000 Containment and Shield building in order to eliminate the possibility of releases directly into the environment and to protect public health and safety. In fact, containment failure through only a small hole similar to that at Beaver Valley should not be a SAMDA event, but is likely to exist when the design basis event occurs.

5. Conclusion

Given the newly discovered Beaver Valley containment system failure and a litany of other containment failures identified throughout this report, the facts show that it is unreasonable to assume that the AP1000 containment design for the proposed AP1000 reactors will not leak radiation directly into the annular gap created by the shield building.

In conclusion, the potential for containment leakage directly through holes in the steel shell creates an unanalyzed safety risk to the public from the proposed AP1000 containment design. Releases from this potential leakage path are not bounded by any existing analysis and will be more severe than those previously identified by Westinghouse in its AP1000 applications and various revisions.

Four contributing factors will increase the consequences of an accident in which the containment leaks radiation directly into the annular gap.

- First, more radiation is likely to be released than previously analyzed.
- Second, radiation will be released sooner than in other scenarios because the hole or leakage path exists prior to the accident.
- Third, radioactive gases entering this gap are not filtered or delayed.
- Fourth, moisture and oxygen, routinely occurring between the containment and the shield building in the AP1000 design, exacerbates the likelihood of larger than design basis containment leaks.

Filtration of the air leaving the annular gap between the containment and the shield building was previously rejected by Westinghouse's SAMDA analysis. However, in my opinion, this issue should be reconsidered because it is a design basis event and not a low probability SAMDA occurrence. Finally, because the NRC and Westinghouse have not analyzed the containment system for the design of the proposed AP1000 reactors in light of these flaws, the public is presented with an *unreviewed safety issue* that creates a potential accident with much more severe consequences than previously analyzed.

Attachments:

Attachment 1 – Curriculum Vitae

Attachment 2 – Table 1 from *Detection of Aging Nuclear Power Plant Structures*

Attachment 3 – Table 35-4 *Summary Of Release Category Definitions*

Attachment 4 – *Declaration Of Arnold Gundersen Supporting Citizen Power's Petition*

Attachment 5 – *Declaration Of Arnold Gundersen Supporting Connecticut Coalition Against Millstone In Its Petition For Leave To Intervene, Request For Hearing, And Contentions*

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Affidavit

Re.

Post Accident AP1000 Containment Leakage: An Un-reviewed Safety Issue

By
Arnold Gundersen, March 26, 2010

I, Rudolf H. Hausler, Corrosion Engineer, NACE Corrosion Specialist, recipient of the NACE Technical Achievement Award, and NACE Fellowship, dipl. Chemical Engineer and PhD in Technical Sciences, hereby assert that I have read subject report in detail.

I agree with the assessment that the construction of the containment building of the AP1000 leaves the reactor containment (carbon steel shell) subject to various modes of corrosion attack. Even though both the inside and the outside of the containment may be coated for corrosion protection (it is not clear that they are because heavy protective paint coat layers will reduce the necessary heat transfer rate) there are always pinholes in any paint layer where corrosion processes may be initiated. Inaccessible areas will be most vulnerable to defects and hence corrosion.

In recent years coatings for applications in nuclear energy plants have been given much attention. However, with all the testing in salt spray cabinets supplemented by irradiation, there are no manufacturers who will give assurances beyond the life expectancies based on intuitive extrapolations.

It turns out that the paint manufactures develop paints and perform test procedures according to industry standards but leave the final selection of a paint schedule to the operating engineer at the respective generating plants. Clearly in this case the blind are leading the seeing.

Because of the impossibility of ruling out defects in the protective coating, the uncertainty of the fitness for purpose of coatings beyond the customarily guaranteed 10 years, the further uncertainty of the performance of the natural convection cooling scheme of the AP-1000, it would appear extremely risky to deny and rule out need for secondary containment.

I therefore agree with Arnold Gundersen's assessment in its entirety.

Signed

Rudolf H. Hausler

March 29, 2010

CURRICULUM VITAE
Arnold Gundersen
Chief Engineer, Fairewinds Associates, Inc
April 2010

Education and Training

ME NE	Master of Engineering Nuclear Engineering Rensselaer Polytechnic Institute, 1972 U.S. Atomic Energy Commission Fellowship Thesis: Cooling Tower Plume Rise
BS NE	Bachelor of Science Nuclear Engineering Rensselaer Polytechnic Institute, Cum Laude, 1971 James J. Kerrigan Scholar
RO	Licensed Reactor Operator, U.S. Atomic Energy Commission License # OP-3014

Qualifications – including and not limited to:

- Chief Engineer, Fairewinds Associates, Inc
- Nuclear Engineering, Safety, and Reliability Expert
- Federal and Congressional hearing testimony and Expert Witness testimony
- Former Senior Vice President Nuclear Licensee
- Former Licensed Reactor Operator
- 39-years of nuclear industry experience and oversight
    - Nuclear engineering management assessment and prudency assessment
    - Nuclear power plant licensing and permitting – assessment and review
    - Nuclear safety assessments, source term reconstructions, dose assessments, criticality analysis, and thermohydraulics
    - Contract administration, assessment and review
    - Systems engineering and structural engineering assessments
    - Cooling tower operation, cooling tower plumes, thermal discharge assessment, and consumptive water use
    - Nuclear fuel rack design and manufacturing, nuclear equipment design and manufacturing, and technical patents
    - Radioactive waste processes, storage issue assessment, waste disposal and decommissioning experience
    - Reliability engineering and aging plant management assessments, in-service inspection
    - Employee awareness programs, whistleblower protection, and public communications
    - Quality Assurance (QA) & records

Publications

Co-author — *DOE Decommissioning Handbook, First Edition*, 1981-1982, invited author.

Co-author — *Decommissioning the Vermont Yankee Nuclear Power Plant: An Analysis of Vermont Yankee's Decommissioning Fund and Its Projected Decommissioning Costs*, November 2007, Fairewinds Associates, Inc.

Co-author — *Decommissioning Vermont Yankee – Stage 2 Analysis of the Vermont Yankee Decommissioning Fund – The Decommissioning Fund Gap*, December 2007, Fairewinds

Associates, Inc. Presented to Vermont State Senators and Legislators.
Co-author — *Vermont Yankee Comprehensive Vertical Audit – VYCVA – Recommended Methodology to Thoroughly Assess Reliability and Safety Issues at Entergy Nuclear Vermont Yankee*, January 30, 2008 Testimony to Finance Committee Vermont Senate
Co-author — *Act 189 Public Oversight Panel Report*, March 17, 2009, to the Vermont State Legislature by the Vermont Yankee Public Oversight Panel.
Author — Fairewinds Associates, Inc *First Quarterly Report to the Joint Legislative Committee*, October 19, 2009.
Co-author — The Second Quarterly Report by Fairewinds Associates, Inc to the Joint Legislative Committee regarding buried pipe and tank issues at Entergy Nuclear Vermont Yankee and Entergy proposed Enexus spinoff. See two reports: *Fairewinds Associates 2nd Quarterly Report to JFC* and *Enexus Review by Fairewinds Associates*.

Patents
Energy Absorbing Turbine Missile Shield – U.S. Patent # 4,397,608 – 8/9/1983

Committee Memberships
Vermont Yankee Public Oversight Panel – appointed 2008 by President Pro-Tem Vermont Senate
National Nuclear Safety Network – Founding Board Member
Three Rivers Community College – Nuclear Academic Advisory Board
Connecticut Low Level Radioactive Waste Advisory Committee – 10 years, founding member
Radiation Safety Committee, NRC Licensee – founding member
ANSI N-198, Solid Radioactive Waste Processing Systems

Honors
U.S. Atomic Energy Commission Fellowship, 1972
B.S. Degree, Cum Laude, RPI, 1971, 1st in nuclear engineering class
Tau Beta Pi (Engineering Honor Society), RPI, 1969 – 1 of 5 in sophomore class of 700
James J. Kerrigan Scholar 1967–1971
Teacher of the Year – 2000, Marvelwood School
Publicly commended to U.S. Senate by NRC Chairman, Ivan Selin, in May 1993 – "It is true...everything Mr. Gundersen said was absolutely right; he performed quite a service."

Nuclear Consulting and Expert Witness Testimony
Vermont State Legislature House Natural Resources – April 5, 2010
Testified to the House Natural Resources Committee regarding discrepancies in Entergy's TLG Services decommissioning analysis. See *Fairewinds Cost Comparison TLG Decommissioning* (http://www.leg.state.vt.us/JFO/Vermont%20Yankee.htm).

Vermont State Legislature Joint Fiscal Committee Legislative Consultant Regarding Entergy Nuclear Vermont Yankee – February 22, 2010
The Second Quarterly Report by Fairewinds Associates, Inc to the Joint Legislative Committee regarding buried pipe and tank issues at Entergy Nuclear Vermont Yankee and Entergy proposed Enexus spinoff. See two reports: *Fairewinds Associates 2nd Quarterly Report to JFC* and *Enexus Review by Fairewinds Associates*.
(http://www.leg.state.vt.us/JFO/Vermont%20Yankee.htm).

Vermont State Legislature Senate Natural Resources – February 16, 2010
Testified to Senate Natural Resources Committee regarding causes and severity of tritium leak in unreported buried underground pipes, status of Enexus spinoff proposal, and health effects of tritium.

Vermont State Legislature Senate Natural Resources – February 10, 2010
Testified to Senate Natural Resources Committee regarding causes and severity of tritium leak in unreported buried underground pipes. http://www.youtube.com/watch?v=36HJiBrJSxE

Vermont State Legislature Senate Finance – February 10, 2010
Testified to Senate Finance Committee regarding *A Chronicle of Issues Regarding Buried Tanks and Underground Piping at VT Yankee.*
(http://www.leg.state.vt.us/JFO/Vermont%20Yankee.htm)

Vermont State Legislature House Natural Resources – January 27, 2010
A Chronicle of Issues Regarding Buried Tanks and Underground Piping at VT Yankee.
(http://www.leg.state.vt.us/JFO/Vermont%20Yankee.htm)

Eric Epstein, TMI Alert – January 5, 2010
Expert Witness Report Of Arnold Gundersen Regarding Consumptive Water Use Of The Susquehanna River By The Proposed PPL Bell Bend Nuclear Power Plant In the Matter of RE: Bell Bend Nuclear Power Plant Application for Groundwater Withdrawal Application for Consumptive Use BNP-2009-073.

U.S. Nuclear Regulatory Commission Atomic Safety and Licensing Board (NRC-ASLB)
Declaration of Arnold Gundersen Supporting Supplemental Petition of Intervenors Contention 15: Detroit Edison Cola Lacks Statutorily Required Cohesive QA Program, December 8, 2009.

U.S. NRC Region III Allegation Filed by Missouri Coalition for the Environment
Expert Witness Report entitled: *Comments on the Callaway Special Inspection by NRC Regarding the May 25, 2009 Failure of its Auxiliary Feedwater System,* November 9, 2009.

Vermont State Legislature Joint Fiscal Committee Legislative Consultant Regarding Entergy Nuclear Vermont Yankee
Oral testimony given to the Vermont State Legislature Joint Fiscal Committee October 28, 2009.
See report: *Quarterly Status Report - ENVY Reliability Oversight for JFO*
(http://www.leg.state.vt.us/JFO/Vermont%20Yankee.htm).

Vermont State Legislature Joint Fiscal Committee Legislative Consultant Regarding Entergy Nuclear Vermont Yankee
The First Quarterly Report by Fairewinds Associates, Inc to the Joint Legislative Committee regarding reliability issues at Entergy Nuclear Vermont Yankee, issued October 19, 2009.
See report: *Quarterly Status Report - ENVY Reliability Oversight for JFO*
(http://www.leg.state.vt.us/JFO/Vermont%20Yankee.htm).

Florida Public Service Commission (FPSC)
Gave direct oral testimony to the FPSC in hearings in Tallahassee, FL, September 8 and 10, 2009 in support of Southern Alliance for Clean Energy (SACE) contention of anticipated licensing and construction delays in newly designed Westinghouse AP 1000 reactors proposed by Progress Energy Florida and Florida Power and Light (FPL).

Florida Public Service Commission (FPSC)
NRC announced delays confirming my original testimony to FPSC detailed below. My supplemental testimony alerted FPSC to NRC confirmation of my original testimony regarding licensing and construction delays due to problems with the newly designed Westinghouse AP 1000 reactors in *Supplemental Testimony In Re: Nuclear Plant Cost Recovery Clause By The Southern Alliance For Clean Energy*, FPSC Docket No. 090009-EI, August 12, 2009.

Florida Public Service Commission (FPSC)
Licensing and construction delays due to problems with the newly designed Westinghouse AP 1000 reactors in *Direct Testimony In Re: Nuclear Plant Cost Recovery Clause By The Southern Alliance For Clean Energy*, FPSC Docket No. 090009-EI, July 15, 2009.

Vermont State Legislature Joint Fiscal Committee Expert Witness Oversight Role for Entergy Nuclear Vermont Yankee (ENVY)
Contracted by the Joint Fiscal Committee of the Vermont State Legislature as an expert witness to oversee the compliance of ENVY to reliability issues uncovered during the 2009 legislative session by the Vermont Yankee Public Oversight Panel of which I was appointed a member along with former NRC Commissioner Peter Bradford for one year from July 2008 to 2009. Entergy Nuclear Vermont Yankee (ENVY) is currently under review by Vermont State Legislature to determine if it should receive a Certificate for Public Good (CPG) to extend its operational license for another 20-years. Vermont is the only state in the country that has legislatively created the CPG authorization for a nuclear power plant. Act 160 was passed to ascertain ENVY's ability to run reliably for an additional 20 years. Appointment from July 2009 to May 2010.

U.S. Nuclear Regulatory Commission
Expert Witness Declaration regarding Combined Operating License Application (COLA) at North Anna Unit 3 *Declaration of Arnold Gundersen Supporting Blue Ridge Environmental Defense League's Contentions* (June 26, 2009).

U.S. Nuclear Regulatory Commission
Expert Witness Declaration regarding Through-wall Penetration of Containment Liner and Inspection Techniques of the Containment Liner at Beaver Valley Unit 1 Nuclear Power Plant *Declaration of Arnold Gundersen Supporting Citizen Power's Petition* (May 25, 2009).

U.S. Nuclear Regulatory Commission
Expert Witness Declaration regarding Quality Assurance and Configuration Management at Bellefonte Nuclear Plant *Declaration of Arnold Gundersen Supporting Blue Ridge Environmental Defense League's Contentions in their Petition for Intervention and Request for Hearing*, May 6, 2009.

Pennsylvania Statehouse
Expert Witness Analysis presented in formal presentation at the Pennsylvania Statehouse, March 26, 2009 regarding actual releases from Three Mile Island Nuclear Accident. Presentation may be found at: http://www.tmia.com/march26

Vermont Legislative Testimony and Formal Report for 2009 Legislative Session
As a member of the Vermont Yankee Public Oversight Panel, I spent almost eight months examining the Vermont Yankee Nuclear Power Plant and the legislatively ordered Comprehensive Vertical Audit. Panel submitted Act 189 Public Oversight Panel Report March 17, 2009 and oral testimony to a joint hearing of the Senate Finance and House Natural Resources March 19, 2009. (See: http://www.leg.state.vt.us/JFO/Vermont%20Yankee.htm)

Finestone v FPL (11/2003 to 12/2008) Federal Court
Plaintiffs' Expert Witness for Federal Court Case with Attorney Nancy LaVista, from the firm Lytal, Reiter, Fountain, Clark, Williams, West Palm Beach, FL. This case involved two plaintiffs in cancer cluster of 40 families alleging that illegal radiation releases from nearby nuclear power plant caused children's cancers. Production request, discovery review, preparation of deposition questions and attendance at Defendant's experts for deposition, preparation of expert witness testimony, preparation for Daubert Hearings, ongoing technical oversight, source term reconstruction and appeal to Circuit Court.

U.S. Nuclear Regulatory Commission Advisory Committee Reactor Safeguards (NRC-ACRS)
Expert Witness providing oral testimony regarding Millstone Point Unit 3 (MP3) Containment issues in hearings regarding the Application to Uprate Power at MP3 by Dominion Nuclear, Washington, and DC. (July 8-9, 2008).

Appointed by President Pro-Tem of Vermont Senate to Legislatively Authorized Nuclear Reliability Public Oversight Panel
To oversee Comprehensive Vertical Audit of Entergy Nuclear Vermont Yankee (Act 189) and testify to State Legislature during 2009 session regarding operational reliability of ENVY in relation to its 20-year license extension application. (July 2, 2008 to present).

U.S. Nuclear Regulatory Commission Atomic Safety and Licensing Board (NRC-ASLB)
Expert Witness providing testimony regarding *Pilgrim Watch's Petition for Contention 1 Underground Pipes* (April 10, 2008).

U.S. Nuclear Regulatory Commission Atomic Safety and Licensing Board (NRC-ASLB)
Expert Witness supporting *Connecticut Coalition Against Millstone In Its Petition For Leave To Intervene, Request For Hearing, And Contentions Against Dominion Nuclear Connecticut Inc.'s*

Millstone Power Station Unit 3 License Amendment Request For Stretch Power Uprate (March 15, 2008).

U.S. Nuclear Regulatory Commission Atomic Safety and Licensing Board (NRC-ASLB)
Expert Witness supporting *Pilgrim Watch's Petition For Contention 1*: *specific to issues regarding the integrity of Pilgrim Nuclear Power Station's underground pipes and the ability of Pilgrim's Aging Management Program to determine their integrity*. (January 26, 2008).

Vermont State House – 2008 Legislative Session

- House Committee on Natural Resources and Energy – Comprehensive Vertical Audit: *Why NRC Recommends a Vertical Audit for Aging Plants Like Entergy Nuclear Vermont Yankee (ENVY)*
- House Committee on Commerce – Decommissioning Testimony

Vermont State Senate – 2008 Legislative Session

- Senate Finance – testimony regarding Entergy Nuclear Vermont Yankee Decommissioning Fund
- Senate Finance – testimony on the necessity for a Comprehensive Vertical Audit (CVA) of Entergy Nuclear Vermont Yankee
- Natural Resources Committee – testimony regarding the placement of high-level nuclear fuel on the banks of the Connecticut River in Vernon, VT

U.S. Nuclear Regulatory Commission Atomic Safety and Licensing Board (NRC-ASLB)
MOX Limited Appearance Statement to Judges Michael C. Farrar (Chairman), Lawrence G. McDade, and Nicholas G. Trikouros for the "Petitioners": Nuclear Watch South, the Blue Ridge Environmental Defense League, and Nuclear Information & Resource Service in support of *Contention 2: Accidental Release of Radionuclides, requesting a hearing concerning faulty accident consequence assessments made for the MOX plutonium fuel factory proposed for the Savannah River Site*. (September 14, 2007).

Appeal to the Vermont Supreme Court (March 2006 to 2007)
Expert Witness Testimony in support of *New England Coalition's Appeal to the Vermont Supreme Court Concerning: Degraded Reliability at Entergy Nuclear Vermont Yankee as a Result of the Power Uprate*. New England Coalition represented by Attorney Ron Shems of Burlington, VT.

State of Vermont Environmental Court (Docket 89-4-06-vtec 2007)
Expert witness retained by New England Coalition to review Entergy and Vermont Yankee's analysis of alternative methods to reduce the heat discharged by Vermont Yankee into the Connecticut River. Provided Vermont's Environmental Court with analysis of alternative methods systematically applied throughout the nuclear industry to reduce the heat discharged by nuclear power plants into nearby bodies of water and avoid consumptive water use. This report included a review of the condenser and cooling tower modifications.

U.S. Senator Bernie Sanders and Congressman Peter Welch (2007)
Briefed Senator Sanders, Congressman Welch and their staff members regarding technical and engineering issues, reliability and aging management concerns, regulatory compliance, waste storage, and nuclear power reactor safety issues confronting the U.S. nuclear energy industry.

State of Vermont Legislative Testimony to Senate Finance Committee (2006)
Testimony to the Senate Finance Committee regarding Vermont Yankee decommissioning costs, reliability issues, design life of the plant, and emergency planning issues.

U.S. Nuclear Regulatory Commission Atomic Safety and Licensing Board (NRC-ASLB)
Expert witness retained by New England Coalition to provide Atomic Safety and Licensing Board with an independent analysis of the integrity of the Vermont Yankee Nuclear Power Plant condenser (2006).

U.S. Senators Jeffords and Leahy (2003 to 2005)
Provided the Senators and their staffs with periodic overview regarding technical, reliability, compliance, and safety issues at Entergy Nuclear Vermont Yankee (ENVY).

10CFR 2.206 filed with the Nuclear Regulatory Commission (July 2004)
Filed 10CFR 2.206 petition with NRC requesting confirmation of Vermont Yankee's compliance with General Design Criteria.

State of Vermont Public Service Board (April 2003 to May 2004)
Expert witness retained by New England Coalition to testify to the Public Service Board on the reliability, safety, technical, and financial ramifications of a proposed increase in power (called an uprate) to 120% at Entergy's 31-year-old Vermont Yankee Nuclear Power Plant.

International Nuclear Safety Testimony
Worked for ten days with the President of the Czech Republic (Vaclav Havel) and the Czech Parliament on their energy policy for the 21st century.

Nuclear Regulatory Commission (NRC) Inspector General (IG)
Assisted the NRC Inspector General in investigating illegal gratuities paid to NRC Officials by Nuclear Energy Services (NES) Corporate Officers. In a second investigation, assisted the Inspector General in showing that material false statements (lies) by NES corporate president caused the NRC to overlook important violations by this licensee.

State of Connecticut Legislature
Assisted in the creation of State of Connecticut Whistleblower Protection legal statutes.

Federal Congressional Testimony
Publicly recognized by NRC Chairman, Ivan Selin, in May 1993 in his comments to U.S. Senate, "It is true...everything Mr. Gundersen said was absolutely right; he performed quite a service." Commended by U.S. Senator John Glenn for public testimony to Senator Glenn's NRC Oversight Committee.

<u>PennCentral Litigation</u>
Evaluated NRC license violations and material false statements made by management of this nuclear engineering and materials licensee.

<u>Three Mile Island Litigation</u>
Evaluated unmonitored releases to the environment after accident, including containment breach, letdown system and blowout. Proved releases were 15 times higher than government estimate and subsequent government report.

<u>Western Atlas Litigation</u>
Evaluated neutron exposure to employees and license violations at this nuclear materials licensee.

<u>Commonwealth Edison</u>
In depth review and analysis for Commonwealth Edison to analyze the efficiency and effectiveness of all Commonwealth Edison engineering organizations, which support the operation of all of its nuclear power plants.

<u>Peach Bottom Reactor Litigation</u>
Evaluated extended 28-month outage caused by management breakdown and deteriorating condition of plant.

<u>Special Remediation Expertise</u>**:**
Director of Engineering, Vice President of Site Engineering, and the Senior Vice President of Engineering at Nuclear Energy Services (NES).

- NES was a nuclear licensee that specialized in dismantlement and remediation of nuclear facilities and nuclear sites. Member of the radiation safety committee for this licensee.
- Department of Energy chose NES to write *DOE Decommissioning Handbook* because NES had a unique breadth and depth of nuclear engineers and nuclear physicists on staff.
- Personally wrote the “Small Bore Piping” chapter of the DOE’s first edition Decommissioning Handbook, personnel on my staff authored other sections, and I reviewed the entire Decommissioning Handbook.
- Served on the Connecticut Low Level Radioactive Waste Advisory Committee for 10 years from its inception.
- Managed groups performing analyses on dozens of dismantlement sites to thoroughly remove radioactive material from nuclear plants and their surrounding environment.
- Managed groups assisting in decommissioning the Shippingport nuclear power reactor. Shippingport was the first large nuclear power plant ever decommissioned. The decommissioning of Shippingport included remediation of the site after decommissioning.
- Managed groups conducting site characterizations (preliminary radiation surveys prior to commencement of removal of radiation) at the radioactively contaminated West Valley site in upstate New York.
- Personnel reporting to me assessed dismantlement of the Princeton Avenue Plutonium Lab in New Brunswick, NJ. The lab’s dismantlement assessment was stopped when we uncovered extremely toxic and carcinogenic underground radioactive contamination.

- Personnel reporting to me worked on decontaminating radioactive thorium at the Cleveland Avenue nuclear licensee in Ohio. The thorium had been used as an alloy in turbine blades. During that project, previously undetected extremely toxic and carcinogenic radioactive contamination was discovered below ground after an aboveground gamma survey had purported that no residual radiation remained on site.

Teaching and Academic Administration Experience

Rensselaer Polytechnic Institute (RPI) – Advanced Nuclear Reactor Physics Lab
Community College of Vermont – Mathematics Professor – 2007 to present
Burlington High School
 Mathematics Teacher – 2001 to June 2008
 Physics Teacher – 2004 to 2006
The Marvelwood School – 1996 to 2000
 Awarded Teacher of the Year – June 2000
 Chairperson: Physics and Math Department
 Mathematics and Physics Teacher, Faculty Council Member
 Director of Marvelwood Residential Summer School
 Director of Residential Life
The Forman School & St. Margaret's School – 1993 to 1995
 Physics and Mathematics Teacher, Tennis Coach, Residential Living Faculty Member

Nuclear Engineering 1970 to Present

Vetted as expert witness in nuclear litigation and administrative hearings in federal, international, and state court and to Nuclear Regulatory Commission, including but not limited to: Three Mile Island, US Federal Court, US NRC, NRC ASLB & ACRS, Vermont State Legislature, Vermont State Public Service Board, Florida Public Service Board, Czech Senate, Connecticut State Legislature, Western Atlas Nuclear Litigation, U.S. Senate Nuclear Safety Hearings, Peach Bottom Nuclear Power Plant Litigation, and Office of the Inspector General NRC.

Nuclear Engineering, Safety, and Reliability Expert Witness 1990 to Present

- Fairewinds Associates, Inc – Chief Engineer, 2005 to Present
- Arnold Gundersen, Nuclear Safety Consultant and Energy Advisor, 1995 to 2005
- GMA – 1990 to 1995, including expert witness testimony regarding the accident at Three Mile Island.

Nuclear Energy Services, Division of PCC (Fortune 500 company) 1979 to 1990
 Corporate Officer and Senior Vice President - Technical Services
 Responsible for overall performance of the company's Inservice Inspection (ASME XI), Quality Assurance (SNTC 1A), and Staff Augmentation Business Units – up to 300 employees at various nuclear sites.

<u>Senior Vice President of Engineering</u>
Responsible for the overall performance of the company's Site Engineering, Boston Design Engineering and Engineered Products Business Units. Integrated the Danbury based, Boston based and site engineering functions to provide products such as fuel racks, nozzle dams, and transfer mechanisms and services such as materials management and procedure development.

<u>Vice President of Engineering Services</u>
Responsible for the overall performance of the company's field engineering, operations engineering, and engineered products services. Integrated the Danbury-based and field-based engineering functions to provide numerous products and services required by nuclear utilities, including patents for engineered products.

<u>General Manager of Field Engineering</u>
Managed and directed NES' multi-disciplined field engineering staff on location at various nuclear plant sites. Site activities included structural analysis, procedure development, technical specifications and training. Have personally applied for and received one patent.

<u>Director of General Engineering</u>
Managed and directed the Danbury based engineering staff. Staff disciplines included structural, nuclear, mechanical and systems engineering. Responsible for assignment of personnel as well as scheduling, cost performance, and technical assessment by staff on assigned projects. This staff provided major engineering support to the company's nuclear waste management, spent fuel storage racks, and engineering consulting programs.

New York State Electric and Gas Corporation (NYSE&G) — 1976 to 1979

<u>Reliability Engineering Supervisor</u>
Organized and supervised reliability engineers to upgrade performance levels on seven operating coal units and one that was under construction. Applied analytical techniques and good engineering judgments to improve capacity factors by reducing mean time to repair and by increasing mean time between failures.

<u>Lead Power Systems Engineer</u>
Supervised the preparation of proposals, bid evaluation, negotiation and administration of contracts for two 1300 MW NSSS Units including nuclear fuel, and solid-state control rooms. Represented corporation at numerous public forums including TV and radio on sensitive utility issues. Responsible for all nuclear and BOP portions of a PSAR, Environmental Report, and Early Site Review.

Northeast Utilities Service Corporation (NU) — 1972 to 1976

<u>Engineer</u>
Nuclear Engineer assigned to Millstone Unit 2 during start-up phase. Lead the high velocity flush and chemical cleaning of condensate and feedwater systems and obtained discharge permit for chemicals. Developed Quality Assurance Category 1 Material, Equipment and Parts List. Modified fuel pool cooling system at Connecticut Yankee, steam generator blowdown system and diesel generator lube oil system for Millstone. Evaluated Technical Specification Change Requests.

Associate Engineer
Nuclear Engineer assigned to Montague Units 1 & 2. Interface Engineer with NSSS vendor, performed containment leak rate analysis, assisted in preparation of PSAR and performed radiological health analysis of plant. Performed environmental radiation survey of Connecticut Yankee. Performed chloride intrusion transient analysis for Millstone Unit 1 feedwater system. Prepared Millstone Unit 1 off-gas modification licensing document and Environmental Report Amendments 1 & 2.

Rensselaer Polytechnic Institute (RPI) — 1971 to 1972
Critical Facility Reactor Operator, Instructor
Licensed AEC Reactor Operator instructing students and utility reactor operator trainees in start-up through full power operation of a reactor.

Public Service Electric and Gas (PSE&G) — 1970
Assistant Engineer
Performed shielding design of radwaste and auxiliary buildings for Newbold Island Units 1 & 2, including development of computer codes.

Public Service, Cultural, and Community Activities

2005 to Present – Public presentations and panel discussions on nuclear safety and reliability at University of Vermont, NRC hearings, Town and City Select Boards, Legal Panels, Television, and Radio

2007-2008 – Created Concept of Solar Panels on Burlington High School; worked with Burlington Electric Department and Burlington Board of Education Technology Committee on Grant for installation of solar collectors for Burlington Electric peak summer use

Vermont State Legislature – Ongoing Public Testimony to Legislative Committees

Certified Foster Parent State of Vermont – 2004 to 2007

Mentoring former students – 2000 to present – college application and employment application questions and encouragement

Tutoring Refugee Students – 2002 to 2006 – Lost Boys of the Sudan and others from educationally disadvantaged immigrant groups

Designed and Taught Special High School Math Course for ESOL Students – 2007 to 2008

Featured Nuclear Safety and Reliability Expert (1990 to present) for Television, Newspaper, Radio, & Internet
Including, and not limited to: CNN (Earth Matters), NECN, WPTZ VT, WTNH, VPTV, WCAX, Cable Channel 17, The Crusaders, Front Page, Mark Johnson Show, Steve West Show, Anthony Polina Show, WKVT, WDEV, WVPR, WZBG CT, Seven Days, AP News Service, Houston Chronicle, Christian Science Monitor, New York Times, Brattleboro Reformer, Rutland Herald, Times-Argus, Burlington Free Press, Litchfield County Times, The News Times, The New Milford Times, Hartford Current, New London Day, evacuationplans.org, Vermont Daily Briefing, Green Mountain Daily, and numerous other national and international blogs

NNSN – National Nuclear Safety Network, Founding Advisory Board Member, meetings with and testimony to the Nuclear Regulatory Commission Inspector General (NRC IG)

Berkshire School Parents Association, Co-Founder

Berkshire School Annual Appeal, Co-Chair

Sunday School Teacher, Christ Episcopal Church, Roxbury, CT

Washington Montessori School Parents Association Member
Episcopal Marriage Encounter National Presenting Team with wife Margaret
  Provided weekend communication and dialogue workshops weekend retreats/seminars
  Connecticut Episcopal Marriage Encounter Administrative Team – 5 years
Northeast Utilities Representative Conducting Public Lectures on Nuclear Safety Issues

End

Table 1. Instances of containment pressure boundary component degradation at commercial nuclear power plants in the United States.

Plant Designation (Occurrence Date) Plant Type (Source)*	Containment Description (No. of Similar Plants)	Degradation Description	Detection Method
Vermont Yankee (1978) BWR/4 (Ref. 52)	Mark I Steel drywell and wetwell (22)	Surface cracks in the overlay weld-to-torus base metal heat-affected zone	Visual examination (As part of modifications to restore the originally intended design safety margins)
Hatch 2 (1984) BWR/4 (Refs. 53, 54, and 55)	Mark I Steel drywell and wetwell (22)	Through-wall cracks around containment vent headers within the containment torus (Brittle fracture caused by injection of cold nitrogen into torus during inerting)	Visual examination of torus interior
Hatch 1 (1985) BWR/4 (Ref. 55)	Mark I Steel drywell and wetwell (22)	Through-wall crack in nitrogen inerting and purge line (Brittle fracture caused by injection of cold nitrogen during inerting)	In-service inspection testing using magnetic particle method
Monticello (1986) BWR/3 (Ref. 56)	Mark I Steel drywell and wetwell (22)	Polysulfide seal at the concrete-to-shell interface became brittle allowing moisture to reach the steel shell	Visual examination (A small portion of the drywell shell was excavated as a part of a life extension study)
Dresden 3 (1986) BWR/3 (Ref. 57)	Mark I Steel drywell and wetwell (22)	Coating degradation due to exposure to fire with peak metal temperatures of 260°C (500°F) and general corrosion of metal shell by water used to extinguish fire	Visual examination (Polyurethane between the drywell shell and concrete shield wall was ignited by arc-air cutting activities producing smoke and heat)
Oyster Creek (1986) BWR/2 (Refs. 58, 59, and 60)	Mark I Steel drywell and wetwell (22)	Defective gasket at the refueling pool allowed water to eventually reach the sand cushion region causing drywell shell corrosion	Visual examination of uncoated areas and ultrasonic inspection
Fitzpatrick (1987) BWR/4 (Refs. 56 and 61)	Mark I Steel drywell and wetwell (22)	Degradation of torus coating with associated pitting	Visual examination of uncoated areas and ultrasonic inspection (Technical specification surveillance performed during outage)
Millstone 1 (1987) BWR/3 (Ref. 61)	Mark I Steel drywell and wetwell (22)	Degradation of torus coating	Visual examination of uncoated areas and ultrasonic inspection (The torus had been drained for modifications)
Oyster Creek (1987) BWR/2 (Ref. 61)	Mark I Steel drywell and wetwell (22)	Degradation of torus coating with associated pitting	Visual examination of uncoated areas and ultrasonic inspection

Table 1. Instances of containment pressure boundary component degradation at commercial nuclear power plants in the United States (cont.).

Plant Designation (Occurrence Date) Plant Type (Source)*	Containment Description (No. of Similar Plants)	Degradation Description	Detection Method
Brunswick 1 (1987) BWR/4 (Ref. 62)	Reinforced concrete with steel liner (9)	Corrosion of steel liner	General visual examination of coated areas
Nine Mile Point 1 (1988) BWR/5 (Ref. 63)	Steel drywell and wetwell (22)	Corrosion of uncoated torus surfaces	Visual examination of uncoated areas and ultrasonic inspection
Pilgrim (1988) BWR/3 (Ref. 61)	Steel drywell and wetwell (22)	Degradation of torus coating	Visual examination of uncoated areas and ultrasonic inspection (Licensee inspection as a result of occurrences at similar plants)
Brunswick 2 (1988) BWR/4 (Ref. 62)	Reinforced concrete with steel liner (9)	Corrosion of steel liner	General visual examination of coated areas
Dresden 2 (1988) BWR/3 (Ref. 64)	Steel drywell and wetwell (22)	Coating, electrical cable, and valve operator component degradation due to excessive operating temperatures	Visual examination of uncoated areas and ultrasonic inspection (Ventilation hatches in the drywell refueling bulkhead inadvertently left closed)
Hatch 1 and 2 (1989) BWR/4 (Ref. 65)	Steel drywell and wetwell (22)	Bent anchor bolts in torus supports (due to weld induced radial shrinkage)	Visual examination
McGuire 2 (1989) PWR (Ref. 66)	Ice Condenser Reinforced concrete with steel liner (4)	Corrosion on outside of steel cylinder in the annular region at the intersection with the concrete floor	General visual examination prior to Type A leakage rate test
McGuire 1 (1989) PWR (Ref. 66)	Ice Condenser Reinforced concrete with steel liner (4)	Corrosion on outside of steel cylinder in the annular region at the intersection with the concrete floor	General visual examination (Inspection initiated as a result of corrosion detected at McGuire 2)
Catawba 1 (1989) PWR (Refs. 66 and 67)	Ice Condenser Steel cylinder (5)	Corrosion on outside of steel cylinder in the annular region	General visual examination (Inspection initiated as a result of corrosion detected at McGuire 2)
Catawba 2 (1989) PWR (Ref. 66)	Ice Condenser Steel cylinder (5)	Corrosion on outside of steel cylinder in the annular region	General visual examination (Inspection initiated as a result of corrosion detected at McGuire 2)

Table 1. Instances of containment pressure boundary component degradation at commercial nuclear power plants in the United States (cont.).

Plant Designation (Occurrence Date) Plant Type (Source)*	Containment Description (No. of Similar Plants)	Degradation Description	Detection Method
McGuire 1 (1990) PWR (Ref. 68, 69, and 70)	Ice Condenser Reinforced concrete with steel liner (4)	Corrosion on inside surface of coated containment shell under the ice condenser and between the floors near the cork filler material	Visual examination and ultrasonic inspection (Degradation possibly caused by moisture from the ice condenser or condensation)
Quad Cities 1 (1991) BWR/3 (Refs. 71, 72, and 82)	Steel drywell and wetwell (22)	Two-ply containment penetration bellows leaked due to transgranular stress-corrosion cracking	General visual examination (Excessive leakage detected)
Quad Cities 2 (1991) BWR/3 (Refs. 71 and 72)	Steel drywell and wetwell (22)	Two-ply containment penetration bellows leaked due to transgranular stress-corrosion cracking	General visual examination (Excessive leakage detected)
Dresden 3 (1991) BWR/3 (Ref. 72)	Steel drywell and wetwell (22)	Two-ply containment penetration bellows leaked due to transgranular stress-corrosion cracking	General visual examination (Excessive leakage detected)
Point Beach 2 (1992) PWR (Ref. 73)	Post-tensioned concrete cylinder with steel liner (35)	Liner plate separated from concrete	General visual examination
H. B. Robinson (1992) PWR (Ref. 73)	Post-tensioned concrete cylinder (vertical only) with steel liner (35)	Degradation of liner coating	General visual examination
Cooper (1992) BWR/4 (Ref. 73)	Steel drywell and wetwell (22)	Corrosion of interior torus surfaces and corrosion stains on exterior torus surface in one area	General visual examination
Beaver Valley 1 (1992) PWR (Refs. 73 and 74)	Subatmospheric Reinforced concrete cylinder with steel liner (7)	Corrosion of steel liner, degradation of liner coating, and instances of liner bulging	General visual examination prior to Type A leakage rate test
Salem 2 (1993) PWR (Ref. 75)	Reinforced concrete cylinder with steel liner (13)	Corrosion of steel liner	General visual examination prior to Type A leakage rate test

Table 1. Instances of containment pressure boundary component degradation at commercial nuclear power plants in the United States (cont.).

Plant Designation (Occurrence Date) Plant Type (Source)*	Containment Description (No. of Similar Plants)	Degradation Description	Detection Method
Sequoyah 1 (1993) PWR (Ref. 76)	Ice Condenser Steel cylinder with concrete shield building (5)	Degradation of moisture barriers resulting in corrosion of the steel shell	General visual examination and visual examination of coated areas
Sequoyah 2 (1993) PWR (Ref. 76)	Ice Condenser Steel cylinder with concrete shield building (5)	Degradation of moisture barriers resulting in corrosion of the steel shell	General visual examination and visual examination of coated areas
Brunswick 2 (1993) BWR (Refs. 62 and 77)	Reinforced concrete drywell and wetwell with steel liner (9)	Corrosion of steel liner	General visual examination and visual examination of coated areas (Follow-up inspection based on conditions noted in 1988)
Brunswick 1 (1993) BWR/4 (Ref. 77)	Reinforced concrete drywell and wetwell with steel liner (9)	Corrosion of steel liner	General visual examination and visual examination of coated areas (Inspection initiated as a result of corrosion detected at Brunswick 2)
McGuire 1 (1993) PWR (Ref. 78)	Ice Condenser Reinforced concrete with steel liner (4)	Main steam isolation line bellows leakage	Leakage testing conducted on bellows following successful Type A leakage rate test
Braidwood 1 (1994) PWR (Ref. 79)	Post-tensioned concrete cylinder with steel liner (35)	Liner leakage detected but not located	Type A leakage rate test
North Anna 2 (1999) PWR (Ref. 80)	Subatmospheric Reinforced concrete with steel liner (7)	6-mm-diameter hole in liner due to corrosion	General visual examination and visual examination of coated areas
Brunswick 2 (1999) BWR/4 Ref. 81)	Reinforced concrete drywell and wetwell with steel liner (9)	Corrosion of liner ranging from clusters of surface pitting corrosion to a 2-mm-diameter hole	General visual examination and visual examination of coated areas (Inspection initiated as a result of corrosion detected at Surry)

Attachment 3
Summary of Release Category Definitions
Extracted from Chapter 35 AP1000 PRA

Table 35-4

SUMMARY OF RELEASE CATEGORY DEFINITIONS

Release Category	Definition	Release Category Description	Release Magnitude	Release Timing
IC	Intact Containment	Containment integrity is maintained throughout the accident, and the release of radiation to the environment is due to nominal leakage.	Normal Leakage	-
BP	Containment Bypass	Fission products are released directly from the RCS to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to onset of core damage	Large Release	Time Frame 1
CI	Containment Isolation Failure	Fission-product release through a failure of the system or valves that close the penetrations between the containment and the environment. Containment failure occurs prior to onset of core damage.	Large Release	Time Frame 1
CFE	Early Containment Failure	Fission-product release through a containment failure caused by severe accident phenomenon occurring after the onset of core damage but prior to core relocation. Such phenomena include hydrogen combustion phenomena, steam explosions, and vessel failure.	Large Release	Time Frame 2
CFV	Containment Venting	Fission-product release through a containment vent line during intentional depressurization of the containment	Controlled Release	Time Frame 3
CFI	Intermediate Containment Failure	Fission-product release through a containment failure caused by severe accident phenomenon, such as hydrogen combustion, occurring after core relocation but before 24 hours.	Large Release	Time Frame 3
CFL	Late Containment Failure	Fission-product release through a containment failure caused by severe accident phenomenon, such as a failure of passive containment cooling, occurring after 24 hours.	Large Release	Time Frame 4

DOCKET NOS. 50-334 and 50-412
CITIZEN POWER
EXHIBIT ONE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the matter of

FirstEnergy Nuclear Operating Co.	)	May 25, 2009
Beaver Valley Power Station Unit 1	)	Docket No. 50-334 and 50-412
License Renewal for Beaver Valley Units 1 and 2	)	

DECLARATION OF ARNOLD GUNDERSEN
SUPPORTING CITIZEN POWER'S PETITION

I, Arnold Gundersen, declare as follows:

1. My name is Arnold Gundersen. I am sui juris. I am over the age of 18-years-old.

2. Citizen Power has retained me as an expert witness in the above captioned matter, and my declaration is intended to support the Petition of Citizen Power.

3. I have a Bachelor's and a Master's Degree in Nuclear Engineering from Rensselaer Polytechnic Institute (RPI) cum laude.

4. I began my career as a reactor operator and instructor in 1971 and progressed to the position of Senior Vice President for a nuclear licensee. A copy of my Curriculum Vitae is attached. (*Exhibit 3*)

5. I have qualified as an expert witness before the Nuclear Regulatory Commission (NRC) Atomic Safety and Licensing Board (ASLB) and Advisory Committee on Reactor Safeguards (ACRS), in Federal Court, before the State of Vermont Public Service Board and the State of Vermont Environmental Court.

6. I am an author of the first edition of the Department of Energy (DOE) Decommissioning Handbook.

7. I have more than 35-years of professional nuclear experience including and not limited to: Nuclear Plant Operation, Nuclear Management, Nuclear Safety Assessments, Reliability Engineering, In-service Inspection, Criticality Analysis, Licensing, Engineering Management, Thermohydraulics, Radioactive Waste Processes, Decommissioning, Waste Disposal, Structural Engineering Assessments, Cooling Tower Operation, Cooling Tower Plumes, Consumptive Water Loss, Nuclear Fuel Rack Design and Manufacturing, Nuclear Equipment Design and Manufacturing, Prudency Defense, Employee Awareness Programs, Public Relations, Contract Administration, Technical Patents, Archival Storage and Document Control, Source Term Reconstruction, Dose Assessment, Quality Assurance and Records, Configuration Management, Whistleblower Protection, and NRC Regulations and Enforcement.

8. My declaration is intended to support the Petition by Citizen Power and is specific to issues regarding FirstEnergy Nuclear Operating Company's application to extend Beaver Valley Unit 1 Power Station's operating license for an additional 20 years.

9. Beaver Valley Unit 1 is a Westinghouse three loop Nuclear Steam Supply System with a Stone & Webster designed "sub-atmospheric containment." It received its operating license to generate electricity on July 2, 1976.[1]

10. According to NUREG/CR 5640, the *Nuclear Power Plant System Sourcebook*:

> "Sub-atmospheric containments are only found at seven Westinghouse PWR plants, six 3-loop plants, and one 4-loop plant."

11. Stone & Webster Engineering Corporation designed all sub-atmospheric containment systems. The six three-loop sub-atmospheric units are Beaver Valley 1 and 2, North Anna 1 and 2, and Surry l and 2. Stone & Webster's last sub-atmospheric containment is at Millstone Unit 3, a Westinghouse four-loop unit.

12. As a former Northeast Utilities employee who worked on the Millstone Unit 3 engineering, design, and construction, I have personal knowledge of Stone &

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[1] http://www.nrc.gov/info-finder/reactor/bv1.html

Webster's sub-atmospheric design. Moreover, in 2008, I provided written testimony to the NRC regarding Millstone Unit 3 sub-atmospheric containment. *(Exhibit 2)*

13. Furthermore, I briefed the NRC ACRS on the problems and contradictions associated with the NRC's analysis of sub-atmospheric containments.

14. As the lead licensing engineer for Northeast Utilities' Millstone Power Station Unit 3 during the 1970's, I was responsible for coordinating the analysis for the PSAR (Preliminary Safety Analysis Report), which formed the original design basis of the Millstone Power Station Unit 3 including its Containment. This interface was among Millstone's structural mechanical, electrical, construction, and operations personnel as well as the architect Stone & Webster and the NSSS vendor Westinghouse. Millstone Power Station Unit 3 was originally designed to be a "Sub-Atmospheric Containment." [In this instance my testimony is that of a fact witness[2] in addition to my overall testimony as an expert witness in my Millstone Unit 3 Declaration (Exhibit 2).]

15. In my 2008 expert witness report to the NRC ACRS, I identified generic issues with sub-atmospheric containments. The issues of critical concern to both the engineering and operations staff regarding the Sub-Atmospheric Containment were:

    15.1. Members of the operations staff, who worked within the Containment, were repeatedly subjected to the adverse effects of high temperature and low oxygen.

    15.2. The small size of the Containment Building severely limited space for equipment and also complicated accident analysis.

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[2] According to the Department of Justice United States Attorneys' Manual Title 3, Chapter 3-19.111 An expert witness qualifies as an expert by knowledge, skill, experience, training or education, and may testify in the form of an opinion or otherwise. (See Federal Rules of Evidence, Rules 702 and 703). The testimony must cover more than a mere recitation of facts. It should involve opinions on hypothetical situations, diagnoses, analyses of facts, drawing of conclusions, etc., all which involve technical thought or effort independent of mere facts. And according to Chapter 3-19.112 Fact Witness A fact witness is a person whose testimony consists of the recitation of facts and/or events, as opposed to an expert witness, whose testimony consists of the presentation of an opinion, a diagnosis, etc
http://www.usdoj.gov/usao/eousa/foia_reading_room/usam/title3/19musa.htm#3-19.111

15.3. Significant construction problems relating to the placement of concrete and rebar were caused by the Containment's small size.

15.4. Minimal analytical data regarding the long-term strength of the building's concrete and its continual exposure to the combination of high temperatures, low pressure, and low specific humidity within its sub-atmospheric Containment as it has aged has led to doubts and questions regarding the strength of this critical safety-related structure in the event of a nuclear accident.

16. Following my ACRS testimony, the ACRS questioned a *containment specialist* staff member of NRC as to whether the NRC even has the capability to analyze a sub-atmospheric containment. According to the NRC *containment specialist*, the NRC cannot accurately analyze Containment systems.

The NRC staff member *containment specialist* said,

> "It's sort of difficult for us to do an independent analysis. It takes time. We're not really set up to do it. The other thing you have to realize, too, for containment, which isn't as true in the reactor systems area, is that **we don't have the capability**." (Page 88, ACRS Transcript, July 9, 2008, lines 6-11.) [*Emphasis added*]

17. From 1976 until 2002, Beaver Valley Unit 1 (BV1) was operated with a sub-atmospheric containment building. In my opinion, Stone & Webster's similar patents[3] provide two important considerations that apply directly to Beaver Valley's design. Those two considerations are that concrete is considered

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[3] According to one of S&W's patents, "A Sub-atmospheric double containment system is a reinforced concrete double wall nuclear containment structure with each wall including an essentially impervious membrane or liner and porous concrete filling the annulus between the two walls. The interior of the structure is maintained at sub-atmospheric pressure, and the annulus between the two walls is maintained at a sub-atmospheric pressure intermediate between that of the interior and the surrounding atmospheric pressure, during normal operation. In the event of an accident within the containment structure the interior pressure may exceed atmospheric pressure, but leakage from the interior to the annulus between the double walls will not result in the pressure of the annulus exceeding atmospheric pressure so that there is no net outleakage from the containment structure. US Patent 4081323 Issued on March 28, 1978 to Stone & Webster Engineering Corp.

porous and all boundaries leak to some extent. On page 1 of the footnoted patent, Stone & Webster considers the concrete to be "***porous***", and on page 8 of the cited patent, Stone and Webster stated, "…***all boundaries leak to some extent***…".

18. In a sub-atmospheric containment, the air pressure in the containment is approximately 4 psi[4] below the pressure outside the containment liner.

19. During the past four years the evidence I reviewed shows that several age related corrosion problems have impacted BV1's containment system.

20. According to Beaver Valley Senior Resident Inspector David Werkheiser[5], May 19, 2009, the first documented containment liner problem at BV1 was uncovered during the BV1 2006 steam generator replacement outage.

    20.1. Specifically, NRC Senior Resident Inspector Werkheiser said that when the containment liner was cut and removed to allow the steam generator replacement, Beaver Valley personnel noticed three locations or pockets on the "outside" of the cut portion of the liner where significant corrosion was present.

    20.2. According to Werkheiser, FirstEnergy's BV1 attributed these "pockets" to construction problems dating back to the early 1970's. Werkheiser also noted that in FirstEnergy's analysis, the "pockets" or voids appear to have been caused by improper vibration of the concrete as it was being poured.

    20.3. Furthermore, Werkheiser noted that FirstEnergy's analysis showed that over time these "pockets" had allowed moisture to accumulate and gradually corrode the "outside" of the liner.

    20.4. Finally, Werkheiser confirmed that the three corrosion locations were analyzed and repaired prior to start-up in 2006 in accordance with:

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[4] pounds per square inch

[5] Telephone conversation between Beaver Valley Senior Site Resident Inspector David Werkheiser and Arnold Gundersen, expert witness nuclear engineer, May 19, 2009 12:33 pm.

- Duquesne Light Company Calculation 8700-DSC-156W, 2/26/91;
- Liner Minimum Wall Thickness S&W Calculation 11700-EA-41, 11/3/71;
- Duquesne - Beaver Valley Unit 1 – Reactor Containment Liner Stress Analysis and repaired before the Unit started up in 2006.

21. In my opinion, the data I reviewed from the FirstEnergy BV1 SER and outage report indicates problems with the BV1 inspection techniques. For more than 30-years, BV1's visual, ultrasonic and integrated leak-rate inspection techniques were unable to detect these three voids and their associated corrosion until 2006, though the voids and corrosion clearly existed well before then.

22. When the steam generator was replaced in 2006, the 17' x 21' piece of liner which was removed represents, according to my calculations, approximately three percent of the total containment liner.

    22.1. Given that the voids are randomly positioned, when I applied a ratio of the containment surface area to the piece removed, a basic statistical analysis showed that if three voids were found behind a 17'x 21' section, there may be as many as 99 (ninety-nine) more voids that are similarly impacted by corrosion, but remain hidden behind the residual containment liner.

    22.2. By failing to reexamine the full liner in 2006 after detecting three corrosion sites, I believe that FirstEnergy and the NRC made analytical errors by not analyzing whether the sampling density is sufficient to make a reasonably valid conclusion. By not inspecting for more corrosion, in other words, not looking for evidence of the corrosion problem does not prove that corrosion does not exist and that the containment system is sound.

23. BV1 documented a second containment liner problem on April 23, 2009, when the company filed event report 45015 with the NRC. According to BVI event report 45015 *Damaged Area In Containment Liner*:

    "On April 21, 2009 during the Beaver Valley Power Station Unit No.1

> (BEAVER VALLEY PS-1) refueling outage, an ASME XI Section IWE General Visual examination was performed on the interior containment liner. A suspect area was identified at the 738 foot elevation level of containment. This area was approximately 3 inches in diameter and exhibited blistered paint and a protruding rust product. At approximately 1015 hours on April 23, 2009 after cleaning the area and removal of the corrosion products, a rectangular area approximately 1 inch (horizontal) by 3/8 inch (vertical) was discovered that penetrated through the containment steel liner plate (nominal .375 inch thickness). The BEAVER VALLEY PS-1 containment design consists of an internal steel liner that is surrounded by reinforced concrete."
>
> "With the plant currently shutdown and in Mode 6, the containment as specified in Technical Specification 3.6.1 is not required to be operable. The cause of this discrepancy is currently being evaluated.
>
> "This is reportable pursuant to 10 CFR 50.72(b)(3)(ii)(A) as a condition of the principal safety barrier (i.e., containment) being seriously degraded."

23.1. In my opinion, it is important to note once again that all visual, ultrasonic and integrated leak-rate inspection techniques at BV1 *failed to detect the incipient passive failure of a key safety structure before the full perforation of the steel liner.*

24. FirstEnergy claims that the "root cause" of both the BV1 2006 containment liner corrosion and the 2009 gross containment liner failure may be related to construction problems that occurred more than 33-years ago. However, the evidence I examined shows that this purported *root cause* analysis is simplistic for several reasons:

24.1. In the National Association of Corrosion Engineers (NACE) book[6] *Corrosion Basics,* Pierre R. Roberge defines the electrochemistry of corrosion as resulting "from the overwhelming tendency of metals to react electrochemically with oxygen, water, and other substances in the aqueous environment".

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[6] *Corrosion Basics: An Introduction*, 2nd Edition, by Pierre R. Roberge, 2006 by NACE Press Book, 364 pages, 77 tables, 292 figures hardbound, ISBN: 1-57590-198-0

24.2. Therefore, in order for any corrosion to occur, there must be both moisture and oxygen present during which the corrosion reaction would occur. In my expert opinion, if this corrosion issue were solely due to construction problems that occurred more than 33-years ago, there would not have been enough oxygen to cause the identified corrosion. Thus, there must be a secondary source of oxygen.

24.3. Neither the construction voids between the liner and the concrete, which was the purported BV1 2006 reason for containment corrosion, nor BV1's 2009 claim, that a block of wood left from construction, is the *cause* of this recent gross containment failure, because neither accounts for the significant oxygen and moisture buildup that must have occurred. I believe that both FirstEnergy and the NRC have failed to address the underlying issue, which is how did the accumulated moisture and oxygen infiltrate the containment system for such an extensive period of time as to perpetuate a serious corrosion reaction.

25. No root cause analysis to date has addressed moisture and oxygen buildup behind the liner, or why such a buildup occurred at only four very specific locations. The failure to conduct a root cause analysis implies that the four sites of corrosion identified during the past three years may be an anomaly. Rather, I believe that a root cause analysis must investigate in an in-depth fashion the possibility of systemic corrosion issues which may be even greater than 99 corrosion "pockets" on the "outside" of the containment liner rather than limited to these four recently discovered random sites.

26. As discussed above, BV1's sub-atmospheric containment design is unique. In my opinion, it is possible that the pressure differential between the outside moist air and the sub-atmospheric conditions within the containment could act as the driving force to draw moisture and oxygen through the porous concrete into construction voids and wood adjacent to the liner. Therefore, I believe this sub-atmospheric design may be the *root cause* of the oxygen and moisture buildup behind the liner. A thorough *root cause analysis* must consider what impact the sub-atmospheric containment had upon the accumulation of oxygen and moisture between the liner and the porous concrete.

27. In summation, I found the incomplete analytical evidence in the FirstEnergy BV1 and the NRC assessments of BV1's containment failures to be simplistic and believe such incomplete analysis puts an undue risk on public health and safety. In my opinion, an in-depth analysis of the corrosion problems that exists between the liner and the porous concrete may uncover systemic failure mechanisms.

28. Moreover, I believe the breach of this containment liner with no prior warning following repeated and various types of containment inspections which occurred for more than 33-years has broad nuclear policy and safety ramifications, for BV1, Beaver Valley Unit 2 and the other sub-atmospheric containments nationwide.

29. The evidence I reviewed also shows significant problems, therefore, I believe that corrective actions are appropriate, including, but not limited to:

29.1. The prompt 100% ultrasonic inspection of the entire liner at BV1 due to the fact that more than 33-years of visual inspection and fractional ultrasonic testing failed to detect the 2009 corrosion until the liner failed.

29.1.1. In my opinion, the liner failure implies that visual and partial ultrasonic techniques are inappropriate for liner inspections under any conditions.

29.1.2. In my assessment, the Beaver Valley liner degradation and/or failures of both 2006 and 2009 indicate a gross breakdown in Quality Assurance (QA) procedures during the construction phase of BV1.

29.1.3. Based upon my knowledge of the construction processes involved in pouring a sub-atmospheric containment, the QA process applied during the BV1 construction repeatedly missed opportunities for this piece of wood to have been discovered and removed.

29.1.4. If the failure discovered in 2009 existed in 2006, an Integrated Leak rate Test in 2006 failed to detect incipient failure implying that slow, controlled pressurization of the containment in that test is inadequate to detect incipient failure.

29.2. It is my position that the 20-year life extension of the Beaver Valley Units 1 and 2 should be put on hold until these significant programmatic Aging Management problems have been analyzed and resolved.

29.2.1. The visual, ultrasonic and integrated leak test inspection failures show programmatic weakness in the aging management systems upon which FirstEnergy has relied upon for its Beaver Valley Units' license extensions.

29.3. In my opinion, if the 100% UT inspection process discovers other construction voids, then the containment liner should be reanalyzed to determine the operability BV1 in order to ascertain any overall weakening of the liner.

29.3.1. An analysis of the Containment liner will ascertain its ability to withstand seismic stress and limit radiation releases, and the NRC has informed the ACRS of its inability to perform a containment analysis, I believe that an independent National Lab should perform this analysis.

29.4. Likewise, I believe that Beaver Valley Unit 2 (BV2) should also be inspected using 100% ultrasonic techniques, given that BV1 and BV2 have the same design, were built by the same contractor, have the same inspection program, and the same Aging Management Program.

30. Furthermore, it is my conclusion that these events at BV1 also have critical ramifications for the entire U.S. nuclear industry, but especially for PWRs.

30.1. In my opinion, the Containment Breach at BV1 in 2009 was the *Passive Failure* of one of the most important safety barriers in a nuclear power plant.

30.1.1. The nuclear industry has heretofore considered such containment liner failures virtually impossible.

30.1.2. NRC Risk Informed Decision Making does not take the likelihood of

Passive Failure of the Containment into consideration.

30.1.3. Given the generic nature and risk to public health and safety due to *containment breach*, I believe that the NRC should order 100% Ultrasonic Testing of all PWR containment liners.

31. In my opinion, FirstEnergy's inability to detect the most recent failure (2009) of the containment liner prior to perforation, as well as its inability to detect three other corrosion sites discovered in 2006, may indicate one of two possible failure scenarios.

31.1. If the 2006 and 2009 corrosion events grew slowly and began during construction, I believe this implies that during the 35-years since construction, neither the visual, ultrasonic, nor integrated leak rate testing have been adequate to detect incipient containment liner failure.

31.2. The second possibility is that visual, ultrasonic and integrated leak rate testing do indeed work, but that through wall liner failure can propagate much more quickly than anticipated between inspection intervals.

31.3. Both of these scenarios are equally troubling to me, as one indicates that ANY existing inspection regime has been inadequate, and the second indicates rapid failures are possible between inspections whose corrosion growth mechanisms have yet to be determined.

32. Given either scenario, it is my professional opinion that the NRC must modify the Beaver Valley SER and AMP to include a full ultrasonic inspection and root cause analysis prior to license extension.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Executed this day, May 25, 2009 at Burlington, Vermont.

Arnold Gundersen, MSNE

STATE OF VERMONT)
COUNTY OF CHITTENDEN) ss.

I HEREBY CERTIFY that on this 25th day of May 2009, personally appeared Arnold Gundersen resident of Burlington Vermont, who is personally known to me or who produced the following identification, and he swore, subscribed, and acknowledged before me that he executed the foregoing as his free act and deed as an expert witness of said case, for the uses and purposes therein mentioned, and that he did take an oath.

In witness whereof, I have hereunto set my hand in the County and State aforesaid:

OFFICIAL NOTARY ____________

NOTARY PUBLIC STATE OF VERMONT

MY COMMISSION EXPIRES: 2/2010

EXHIBIT A

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

DOMINION NUCLEAR CONNECTICUT INC.	)	
MILLSTONE POWER STATION UNIT 3	)	Docket No. 50-423
LICENSE AMENDMENT REQUEST	)	
STRETCH POWER UPRATE	)	

<u>DECLARATION OF ARNOLD GUNDERSEN SUPPORTING CONNECTICUT COALITION AGAINST MILLSTONE IN ITS PETITION FOR LEAVE TO INTERVENE, REQUEST FOR HEARING, AND CONTENTIONS</u>

I, Arnold Gundersen, declare as follows:

1. My name is Arnold Gundersen. I am sui juris. I am over the age of 18-years-old. I have personal knowledge of the facts contained in this Declaration.

2. I reside at 376 Appletree Point Road, Burlington, Vermont.

3. The Connecticut Coalition Against Millstone has retained me as an expert witness in the above captioned matter.

4. I have a Bachelor's and a Master's Degree in Nuclear Engineering from Rensselaer Polytechnic Institute (RPI) cum laude.

5. I began my career as a reactor operator and instructor at RPI in 1971 and progressed to the position of Senior Vice President for a nuclear licensee. I am a vetted expert witness on nuclear safety and engineering issues. My more than 37-years of professional nuclear experience include and are not limited to: nuclear

safety expert witness testimony; nuclear engineering management and nuclear engineering management assessment; prudency assessment; nuclear power plant licensing, licensing and permitting assessment, and review; nuclear safety assessments, public communications, contract administration, assessment and review; systems engineering, structural engineering assessments, cooling tower operation, cooling tower plumes, nuclear fuel rack design and manufacturing, nuclear equipment design and manufacturing, in-service inspection, criticality analysis, thermohydraulics, radioactive waste processes and storage issue assessment, decommissioning, waste disposal, source term reconstructions, thermal discharge assessment, reliability engineering and aging plant management assessments, archival storage and document control technical patents, federal and congressional hearing testimony, and employee awareness programs.

6. My Curriculum Vitae delineating my qualifications is attached.

7. My Declaration is intended to support Connecticut Coalition Against Millstone's Petition For Leave To Intervene, Request For Hearing, and Contentions.

8. The Five Contentions my Declaration supports are:

   A. The proposed power level for which Dominion Nuclear has applied to uprate Millstone Power Station Unit 3 exceeds the NRC Stretch Power Uprate (SPU) regulatory criteria.

B. The design margins for the Millstone Unit 3 Containment, which help to protect public health and safety, have been significantly reduced by license amendments granted in 1991, and Dominion's proposed power increase, if granted, will further reduce Containment margins designed for safety.

C. When compared to all other Westinghouse Reactors, Millstone Unit 3 is an outlier or anomaly. Dominion's proposed uprate is the largest percent power increase for a Westinghouse reactor. Additionally, Millstone Unit 3 also has the smallest Containment for any Westinghouse reactor of roughly comparable output.

D. Construction problems due to the unique Sub-Atmospheric Containment Design, coupled with the impact upon the Containment concrete by the operation of the Containment Building at very low pressure, very high pressure and very low specific humidity, place the calculations used to predict the stress on that concrete Containment in uncharted analytical areas.

E. The impact of flow-accelerated corrosion at Dominion Nuclear's proposed higher power level for Millstone Unit 3 have not been adequately analyzed and addressed.

9. As an expert witness, who happens to hold both a Bachelor's and Master's degree in Nuclear Engineering, have more than 35-years of nuclear industry engineering experience, and as a former Northeast Utilities employee worked on Millstone Nuclear Power Station Unit 3, in my professional opinion the Dominion Nuclear application fails to satisfy *any of the NRC criteria* to be accepted as a Stretched Power Uprate. A thorough review of the evidence presented by Dominion Nuclear and compared and contrasted with NRC Stretched Power Uprate requirements clearly shows that the Dominion Nuclear Stretched Power Uprate application should in fact be treated as an Extended Power Uprate (EPU) application.

10. According to the NRC, there are two criteria[1] that must be met for a licensee to be considered for a Stretch Power Uprate (SPU):

A. An increase in the reactor power that is "**up to 7 percent**"
and

B. "… **are within the design capacity of the plant**"

C. Furthermore, the NRC states that achieving a Stretch Power Uprate "**depends on the operating margins included in the design of a particular plant**". [Emphasis added]

11. In my opinion, the magnitude of Dominion Nuclear's proposed power increase, the uniqueness of the initial Millstone 3 Power Plant Containment design, the Containment's unusually small size, and the fact that the design margins of the Containment have already been dramatically reduced by changes made to

[1] www.nrc.gov/reactors/operating/licensing/power-uprates

Millstone 3 in 1990 by Northeast Utilities, makes it necessary for the NRC to conduct the more thorough and intensive Extended Power Uprate review.

12. Dominion Nuclear has characterized this proposed increase in power at Millstone Unit 3 (Millstone Power Station Unit 3) as a Stretch Power Uprate (SPU), and Dominion Nuclear claims that Millstone 3 meets all the criteria for a Stretched Power Uprate. According to Dominion's letter filing for the power increase:

> "DNC developed this LAR utilizing the guidelines in NRC Review Standard, RS- 001, "Review Standard for Extended Power Uprates." In addition, requests for additional information (RAIs) regarding SPU and Extended Power Uprate (EPU) applications for other nuclear units were reviewed for applicability. Information that addresses many of those RAIs is included in this MPS3 SPU LAR. RS-001 states that a SPU is **characterized by power level increases up to 7 percent and does not generally involve major modifications**. Plant modifications are addressed in Section 1.0 of the License Report (LR) (Attachment 5) and are not considered to be major. Since the requested uprate is 7 percent and does not involve major plant modifications, it is considered to be a Stretched Power Uprate."[2] [emphasis added]

13. Contention 1: To begin with, the Dominion Nuclear application fails to satisfy the first NRC criteria[3] that the NRC has set the power limit for SPU's at "**… up to 7% …**". Yet Dominion Nuclear notifies its acceptance of the NRC's specific criteria in stating "**…a SPU is characterized by power level increases up to 7 percent** *…*". Most importantly, Dominion's proposed power increase at Millstone Unit 3 in fact exceeds the seven percent limit established by the NRC and accepted by Dominion Nuclear.

[2] Letter, Dominion Nuclear to NRC, SPU Filing, February 2007
[3] www.nrc.gov/reactors/operating/licensing/power-uprates

14. Millstone Power Station Unit 3 is currently licensed to operate at 3411 thermal megawatts (MWt). This number signifies how much heat the reactor is generating and is accurate to four significant figures (numbers).

- The proposed power level of 3650, for which Dominion Nuclear has applied, exceeds the NRC 7% limit that would qualify the power uprate for the less rigorous review of a Stretched Power Uprate.

- Dominion Nuclear has applied for a power increase to 3650 MWt, which is a full 300 KW above what is allowable by the NRC regulations for a Stretch Power Uprate.

- Let's look at the math. Multiply the current licensed power by the NRC's maximum allowable 7% SPU increase. The calculation total equals 3649.7 MWt, which is below the reactor power level of 3650 MWt for which Dominion Nuclear has applied. <u>3411 x 1.07 < 3650</u>

- The 7% NRC limit is accurate to two significant figures. When multiplying a two significant figure number by a four significant figure number *mathematical methodology demands the calculation be rounded down <u>not up</u>* as Dominion Nuclear has done in its application.

- By rounding its proposed reactor power level to a higher power level the requested Dominion Nuclear reactor power increase exceeds the regulatory limit for a Stretched Power Uprate (SPU). Thus, this unscientific rounding up of the thermal megawatt power to a higher power level causes the reactor power to exceed the legal Stretched Power Uprate limit of "up to 7 %" by a full 300 KW.

15. The mathematical evidence shows that Dominion Nuclear proposed power level increase for its Millstone Power Station Unit 3 exceeds the 7% regulatory limit clearly established by the NRC. Therefore, it is my opinion that the Dominion Nuclear's Millstone Unit 3 *is disqualified* for a Stretched Power Uprate.

16. Moreover, while on the face, this mathematical discrepancy may not appear to be a huge number, the 300 KW discrepancy between the NRC 7% limit and Dominion Nuclear's application for a 3650 megawatt thermal increase at Millstone 3 is a significant number that will yield approximately an additional $1 Million in profit for each additional electric megawatt produced per year.

- In other words, industry data[4] shows that the profit from each megawatt of electricity generated from uprated power increases the profit yield to each electric generating corporation by approximately $1,000,000 per year.
- Therefore the data show us that by rounding up the power level increase at Millstone 3 in excess of 7%, Dominion Nuclear's Millstone Power Station Unit 3 will earn additional profits of approximately $330,000 each year until 2045.
- Stated in total dollars, the round up to a power increase in excess of 7% will yield Dominion Nuclear an extra $10,000,000 during the

[4] *Condenser Long Term Plan*, Enrico Betti, Vermont Yankee, Memo FILE UND2002-042 07; MSD 2002/002.

uprated license extension to 2045.

17. In the first place, according to the NRC document *Approved Applications for Power Uprates*[5], the NRC has never allowed a Westinghouse reactor to be licensed for a Stretched Power Uprate with a power level increase as great as that proposed for Millstone Unit 3 by Dominion Nuclear. In the second place, no other Dry Containment[6] Westinghouse reactor with a reactor power level greater than 2000 MWt has been granted a Stretched Power Uprate beyond 6.9 percent.

18. Table 1, inserted below, which is entitled Westinghouse Uprates Ranked in Ascending Order, is a list of all Westinghouse Dry Containment reactors whose thermal power exceeds 2000 MWt.

19. Table 1 ranks the Stretched Power Uprate from smallest to largest, and the NRC data provided in Table 1 shows that no other reactor of this type has ever been granted a Stretched Power Uprate in excess of seven percent like Dominion Nuclear has proposed for Millstone Power Station Unit 3.

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5 NRC *Approved Applications for Power Uprates* http://www.nrc.gov/reactors/operating/licensing/power-uprates/approved-applications.html

6 A Dry Containment is a cylindrical structure with a hemispherical dome that relies solely on its large volume to contain the initial release of radioactive steam after an accident, and to reduce the peak accident pressure. It is a robust passive structure without any additional active mechanical means by which to mitigate immediate post accident pressure. Dry Containment does not rely upon ice or water suppression, nor is it maintained at a large sub-atmospheric pressure in order to reduce the peak accident pressure.

Westinghouse Uprates Ranked in Ascending Order

Name	Initial power	Power Uprate %	Current Power
Indian Point 2	2758	1.4	2797
Commanche Peak 1	3425	1.4	3473
Commanche Peak 2	3425	1.4	3473
STP 1	3800	1.4	3853
STP 2	3800	1.4	3853
Diablo Canyon 1	3338	2	3405
Diablo Canyon 2	3338	2	3405
Salem 1	3411	3.4	3527
Salem 2	3411	3.4	3527
Robinson 2	2300	4.5	2403
Shearon Harris	2775	4.5	2900
Vogtle 1	3411	4.5	3564
Vogtle 2	3411	4.5	3564
Wolf Creek	3411	4.5	3564
Turkey Point 3	2200	4.5	2300
Turkey Point 4	2200	4.5	2300
Callaway	3565	4.5	3725
Braidwood 1	3411	5	3581
Braidwood 2	3411	5	3581
Byron 1	3411	5	3581
Byron 2	3411	5	3581
Farley 1	2652	5	2785
Farley 2	2652	5	2785
Indian Point 3	3025	6.2	3213
Seabrook	3411	6.9	3646
Millstone 3	3411	7.01	3650

Table 1

20. Contention 2: The current application by Dominion Nuclear fails to meet the NRC's second criteria for a Stretched Power Uprate application, because the Millstone Power Station Unit 3 already had its design margins dramatically reduced.

21. According to the NRC, achieving a Stretch Power Uprate "…**depends on the operating margins included in the design of a particular plant**."[7] [emphasis added] Dominion has stated that since the Millstone Power Station Unit 3 application "…does not involve major plant modifications, it is considered to be a SPU". Dominion has erroneously neglected to consider the significant reduction in structural **operating margins** already in place at Millstone Unit 3 prior to its application for a power uprate.

22. The Millstone Power Station Unit 3 Containment structure and its requisite systems have already been "stretched" by previous changes to its design basis when the Containment was converted from Sub-Atmospheric Containment to Dry Containment more than a decade ago. I believe that the proposed changes to Containment systems and structures that have already been reanalyzed and fine tuned once over a decade ago constitutes a dramatic decrease in "…the **operating margins** included in the design of a particular plant."

23. The Containment is the safety related building, which houses the nuclear reactor. As such, it "contains", or in other words collects, the steam and

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7 NRC *Approved Applications for Power Uprates* http://www.nrc.gov/reactors/operating/licensing/power-uprates/approved-applications.html

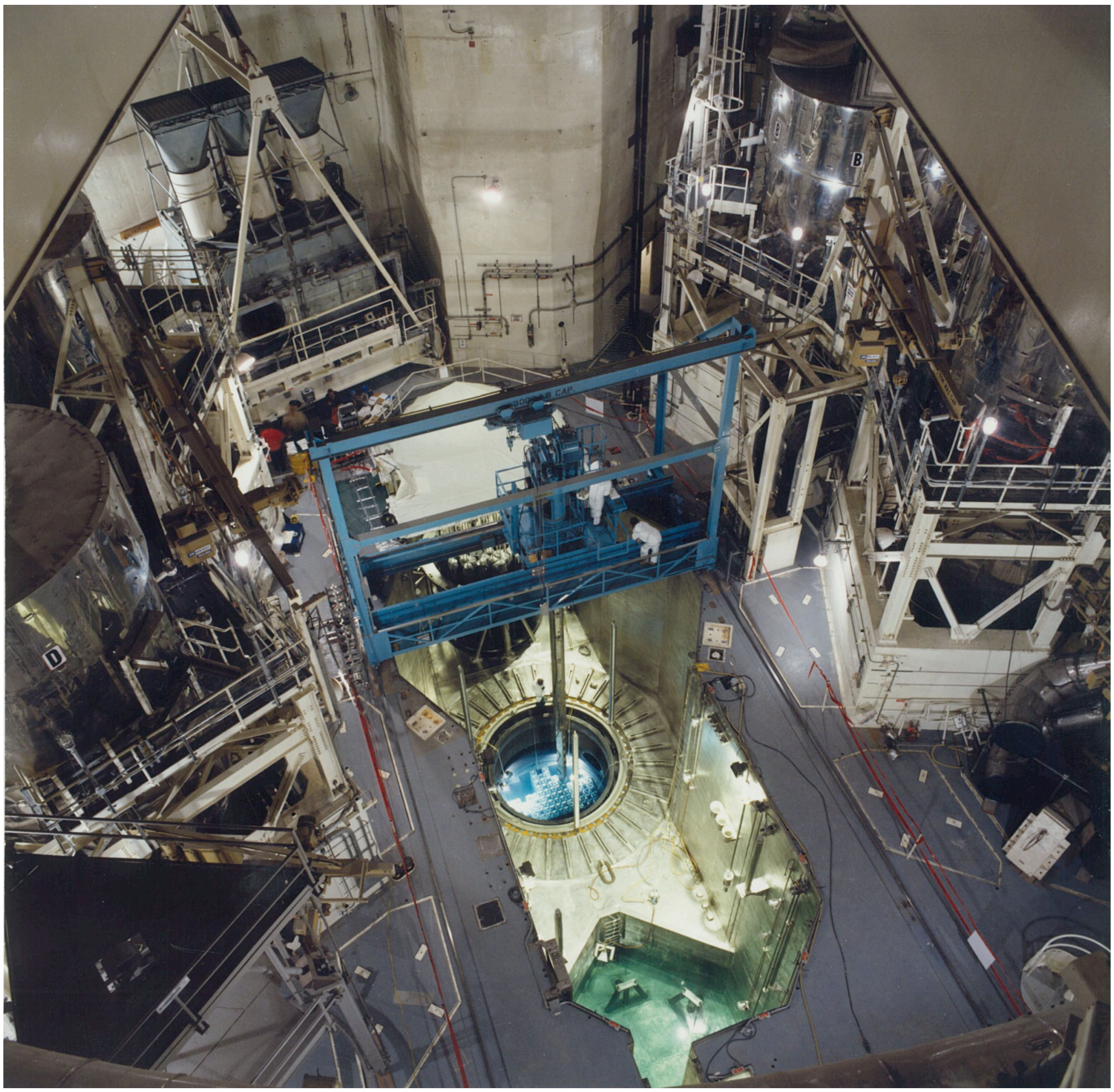
radioactive material that may be released from the reactor after an accident. Please see the photo below of the inside of the Millstone Power Station Unit 3 Containment during initial fuel load in 1986.

24. As the Northeast Utilities lead licensing engineer on Millstone Power Station Unit 3 during the 1970s, I was responsible for coordinating all of the analysis for the PSAR (Preliminary Safety Analysis Report), which formed the original design basis of the Millstone Power Station Unit 3 including its Containment. This interface was among Millstone's structural mechanical, electrical, construction, and operations personnel as well as the architect Stone & Webster and the NSSS vendor Westinghouse. Millstone Power Station Unit 3 was originally designed to be "Sub-Atmospheric Containment." [In this instance my testimony is that of a fact witness[8] in addition to my overall testimony as an expert witness in this Declaration.]

25. The unique design approach of the Sub-Atmospheric Containment maintained the pressure inside the Containment at a "negative pressure" with respect to the atmosphere. Thus the difference between the pressure outside the Containment and inside the Containment (pressure differential) was approximately four pounds. Speaking as an expert witness nuclear engineer, this pressure

[8] According to the Department of Justice United States Attorneys' Manual Title 3, Chapter 3-19.111 An expert witness qualifies as an expert by knowledge, skill, experience, training or education, and may testify in the form of an opinion or otherwise. (See Federal Rules of Evidence, Rules 702 and 703). The testimony must cover more than a mere recitation of facts. It should involve opinions on hypothetical situations, diagnoses, analyses of facts, drawing of conclusions, etc., all which involve technical thought or effort independent of mere facts. And according to Chapter 3-19.112 Fact Witness A fact witness is a person whose testimony consists of the recitation of facts and/or events, as opposed to an expert witness, whose testimony consists of the presentation of an opinion, a diagnosis, etc http://www.usdoj.gov/usao/eousa/foia_reading_room/usam/title3/19musa.htm#3-19.111

differential is quite dramatic for a structure of this size. According to the NRC Sourcebook[9], page 4-26,paragraph B, Sub-atmospheric Containment, Millstone Unit 3 was the only Westinghouse four-loop plant in the nation to have Sub-Atmospheric Containment.



26. Due to critical engineering and operations concerns during my employment as

[9] NRC Sourcebook, page 4-26, paragraph B

the lead licensing engineer for Northeast Utilities on Millstone Power Station Unit 3, both the engineering and operations staff at Northeast Utilities (NU) expressed sincere regret as early as 1975 regarding NU's decision to design and build this unique Sub-Atmospheric Containment.

27. Critical issues of concern to both the engineering and operations staff regarding the Sub-Atmospheric Containment were:

A. The operations staff working within the Containment was repeatedly subjected to the adverse effects of the high temperature and low oxygen.

B. The small size of the Containment Building severely limited space for equipment and also complicated accident analysis.

C. Significant construction problems relating to the placement of concrete and rebar were caused by the Containment's small size.

D. Minimal analytical data regarding the long-term strength of the building's concrete and its continual exposure to the combination of high temperatures, low pressure, and low specific humidity within the sub-atmospheric Containment as it aged lead to doubts and questions regarding the strength of this critical safety-related structure in the event of a nuclear accident.

28. Despite these major concerns, NU decided in 1976 to continue with the licensing process for Millstone Unit 3 as a Sub-atmospheric Containment rather than risk delaying the license by changing the design. At the same time, the company made the strategic decision to modify Millstone Unit 3's license to

operate, by converting the Containment to a standard "Dry" Containment, but only after the nuclear power plant became operational because it is easier to amend a power plant license after a plant is operational.

29. Millstone Power Station Unit 3 began generating power in 1986, and at that time had Sub-Atmospheric Containment. However, Millstone Unit 3's original design basis with its one-of-a-kind four loop Sub-Atmospheric Containment was modified after it became operational in 1986.

30. The purpose of this one-of-a-kind four loop Sub-Atmospheric Containment was to lower peak design pressure[10] in case of a nuclear accident and to rapidly reduce out-leakage[11] after an accident.

   A. More specifically, the Containment Building is designed to capture steam, energy, and radiation after an accident. In order to capture this post-accident energy, the Containment pressure increases. Thus, Containment Buildings are designed to specific pressure levels that must be considered during all power level design changes.

   B. At Millstone Unit 3 the 1975 initial peak Containment design pressure was 39.4 psig[12].

   C. However, prior to Millstone Unit 3's start-up[13], NU reanalyzed the peak pressure and dropped it to 36.1 psig.

   D. Then on February 26, 1990, NU applied to modify the Millstone Power

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10 Maximum pressure inside the Containment after a design basis accident
11 Leakage out of the Containment
12 psig - pounds per square inch, gauge
13 Amendment 17 to FSAR

Station Unit 3 license by changing the design basis pressure of the Containment from 9.8 psia to 14.0 psia[14].

31. When NU applied for the 1990 license change, it claimed that the sole basis for the change was to reduce the risk of injury to operations personnel who struggled to work at the reduced pressures inside this unique Containment. Such an environment is roughly equivalent to working at the top of the Grand Teton Mountains in temperatures in excess of 100 degrees.

A. On page 2 of the initial application, NU stated, "… very little is known about the health effects of people working in high-temperature, low pressure environments."

B. While it is true that this was indeed a staff concern dating back to 1975, it was only ONE of other equally important concerns.

C. Another major staff concern was the fact that the Containment concrete is being exposed to these very same conditions and there is no data to review regarding the ability of concrete to withstand such a unique high-temperature low-pressure environment. Disturbingly, NU was silent on this major concern throughout its application to modify its license and convert the Sub-Atmospheric Containment to Dry Containment.

32. These changes to the design of Millstone Unit 3's one-of-a-kind Containment actually changed the design basis for the plant.

A. From the time the initial PSAR was filed with the NRC, the peak accident pressure of Millstone Unit 3 was repeatedly *fine tuned* by NU.

[14] psia - pounds per square inch, absolute

B. From a nuclear engineering standpoint, the critical concern in my mind is that each time a new Containment pressure analysis was derived, NU applied less conservative assumptions in order to achieve more operational flexibility and decidedly increasing public exposure to radiation if there were an accident.

C. In order to accomplish the 1990 modification of Millstone Unit 3, NU changed numerous design criteria and further reduced design margins by taking further credits for systems that were in the original accident scenario design basis.

33. On page 5 of the application to increase Millstone Unit 3's Containment pressure, Northeast Utilities acknowledged that these modifications to the original design "…constitute an Unreviewed Safety Question."[15]

A. In this February 26, 1990 application to the NRC, NU requested to increase the design basis for the normal pressure inside the Containment from 9.8 psia to 14.0 psia, which resulted in the increase of the post-accident peak Containment pressure from 36.0 to 38.57 psig.

B. Since Millstone Unit 3 was originally designed with this unique Sub-Atmospheric Containment Design, in the event of an accident the Containment was designed to leak radiation to the environment for only an hour until it was able to drop the pressure back down and once again

[15] An unreviewed safety question means a change which involves any of the following: (1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; (2) A possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (3) The margin of safety as defined in the basis for any technical safety requirement is reduced. *http://www.nuclearglossary.com*

contain any radiation releases inside the Containment Building.

C. The 1990 modifications changed the ability of the Containment Building to release radiation for only an hour and instead allowed the Containment to leak at 0.65 weight percent per day after an accident.

D. Bypass leakage was also increased from 0.01 to 0.042 weight percent per day as a result of the change, and the modification to the Containment pressure increased the calculated exposure to a person at the Exclusion Area Boundary from 16.8 rem to 19.5 rem.

34. Contention 3: Earlier in this Declaration, I also mentioned that the Millstone Power Station Unit 3 Containment has what is considered a *small* Containment. To illustrate the fact that Millstone Unit 3's Containment is small in comparison to other Westinghouse designed nuclear reactors, I evaluated data from the publicly available "NRC Sourcebook" and compiled information regarding 25 Westinghouse Reactors, which all have "Dry" Atmospheric Containment[16].

35. Table 2, inserted below, shows, in ascending order by size, the free Containment volume (in millions of cubic feet) of these 25 Westinghouse Reactors.

A. The Containment for Millstone Unit 3 clearly stands out as one of the smallest such Containment Buildings in the country.

[16] Since they are not comparable with Dominion Nuclear's Millstone Power Station Unit 3, I have not included the Westinghouse Reactors with Ice Containments, or several three-loop Reactors with Sub-Atmospheric Containment in the compilation. Also, not included for the same reason are decommissioned reactors and reactors whose thermal power is less than 2000 MWt.

B. For that matter, the only nuclear power plants with a Reactor Containment that is smaller than Millstone Power Station Unit 3 have power outputs that are 800 to 1200 MWt less than the power output of Millstone Unit 3 *prior to the Dominion's proposed uprate*.

C. Moreover, of the 11 identical 3411 MWt Westinghouse four-loop Reactors, Millstone is smaller by as much as half a million cubic feet.

36. The ratio of the initial licensed power level to the Containment Volume at each of the same 25 nuclear reactors is clearly shown in Table 3. This ratio comparison is the real indicator of Millstone Unit 3's small Containment. By applying these ratio criteria in comparison with all 25 reactors, Table 3 clearly shows that Millstone Power Station Unit 3 has the smallest Power to Volume ratio of any Dry Containment Westinghouse reactor in the nation.

37. Dominion Nuclear's proposed 7+% power increase to Millstone Power Station Unit 3 widens even further the size gap between Millstone Unit 3 and the other reactors, thus making Millstone Power Station Unit 3's Containment even "smaller" in comparison to every other Dry Containment Westinghouse reactor in the country.

38. Table 4 shows how the initial licensed power levels of all 25 reactors adjusted as a result of NRC approved "stretch" increases.

A. Accordingly, I have adjusted the power level number for Millstone Unit 3 in order to reflect the amount proposed by Dominion Nuclear's application to uprate Millstone 3's power.

Ascending Comparison of Containment Volumes

Name	Volume xE6	Initial power
Turkey Point 3	1.55	2200
Turkey Point 4	1.55	2200
Farley 1	2.03	2652
Farley 2	2.03	2652
Robinson 2	2.1	2300
Millstone 3	2.35	3411
Shearon Harris	2.5	2775
Wolf Creek	2.5	3411
Callaway	2.5	3565
Indian Point 2	2.6	2758
Indian Point 3	2.6	3025
Salem 1	2.6	3411
Salem 2	2.6	3411
Vogtle 1	2.7	3411
Vogtle 2	2.7	3411
Seabrook	2.7	3411
Diablo Canyon 1	2.83	3338
Diablo Canyon 2	2.83	3338
Braidwood 1	2.9	3411
Braidwood 2	2.9	3411
Byron 1	2.9	3411
Byron 2	2.9	3411
Commanche Peak 1	2.98	3425
Commanche Peak 2	2.98	3425
STP 1	3.3	3800
STP 2	3.3	3800

Table 2

Containment Volume Compared to Initial Power

Name	Volume xE6	Initial power	Initial Power/ Volume
Indian Point 2	2.6	2758	1,060.8
Robinson 2	2.1	2300	1,095.2
Shearon Harris	2.5	2775	1,110
Commanche Peak 1	2.98	3425	1,149.3
Commanche Peak 2	2.98	3425	1,149.3
STP 1	3.3	3800	1,151.5
STP 2	3.3	3800	1,151.5
Indian Point 3	2.6	3025	1,163.5
Braidwood 1	2.9	3411	1,176.2
Braidwood 2	2.9	3411	1,176.2
Byron 1	2.9	3411	1,176.2
Byron 2	2.9	3411	1,176.2
Diablo Canyon 1	2.83	3338	1,179.5
Diablo Canyon 2	2.83	3338	1,179.5
Vogtle 1	2.7	3411	1,263.3
Vogtle 2	2.7	3411	1,263.3
Seabrook	2.7	3411	1,263.3
Farley 1	2.03	2652	1,306.4
Farley 2	2.03	2652	1,306.4
Salem 1	2.6	3411	1,311.9
Salem 2	2.6	3411	1,311.9
Wolf Creek	2.5	3411	1,364.4
Turkey Point 3	1.55	2200	1,419.4
Turkey Point 4	1.55	2200	1,419.4
Callaway	2.5	3565	1426
Millstone 3	2.38	3411	1,433.2

Table 3

Containment Volume Compared to Uprate License Power

Name	Volume xE6	Initial power	Power Uprate %	Current Power	Current Power/V
Indian Point 2	2.6	2758	1.4	2797	1,075.76923
Robinson 2	2.1	2300	4.5	2403	1,144.28571
Shearon Harris	2.5	2775	4.5	2900	1,160
Commanche Peak 1	2.98	3425	1.4	3473	1,165.43624
Commanche Peak 2	2.98	3425	1.4	3473	1,165.43624
STP 1	3.3	3800	1.4	3853	1,167.57576
STP 2	3.3	3800	1.4	3853	1,167.57576
Diablo Canyon 1	2.83	3338	2	3405	1,203.18021
Diablo Canyon 2	2.83	3338	2	3405	1,203.18021
Braidwood 1	2.9	3411	5	3581	1,234.82759
Braidwood 2	2.9	3411	5	3581	1,234.82759
Byron 1	2.9	3411	5	3581	1,234.82759
Byron 2	2.9	3411	5	3581	1,234.82759
Indian Point 3	2.6	3025	6.2	3213	1,235.76923
Vogtle 1	2.7	3411	6.2	3564	1,320
Vogtle 2	2.7	3411	6.2	3564	1,320
Seabrook	2.7	3411	6.9	3646	1,350.37037
Salem 1	2.6	3411	3.4	3527	1,356.53846
Salem 2	2.6	3411	3.4	3527	1,356.53846
Farley 1	2.03	2652	5	2785	1,371.92118
Farley 2	2.03	2652	5	2785	1,371.92118
Wolf Creek	2.5	3411	4.5	3564	1,425.6
Turkey Point 3	1.55	2200	4.5	2300	1,483.87097
Turkey Point 4	1.55	2200	4.5	2300	1,483.87097
Callaway	2.5	3565	4.5	3725	1,490
Millstone 3	2.35	3411	7.01	3650	1,553.19149

Table 4

39. An examination of Table 4, inserted above, shows that the new Power to Volume ratio created by the proposed uprate indicates that Millstone Unit 3's Containment would be even "smaller" if Dominion's proposed power increase is approved.

40. A smaller Containment does not mean that the physical Containment has shrunk in size, but rather that more reactor power, and, in the case of an accident, more radioactive releases are being squeezed by volume into the same small Containment Building as a result of this proposed power increase.

41. If approved, Dominion's power increase to Millstone Unit 3 would be the largest ever power uprate approved to Millstone 3's unique Containment with the "smallest" volume ever licensed as discussed above.

42. What is the net effect of increasing the reactor power in this unique very small Sub-Atmospheric designed Containment? I believe that the proposed power increase at Millstone Power Station Unit 3 means that in the event of a nuclear accident at Unit 3, more than 7% additional energy must be absorbed into this one-of-a-kind Containment.

43. I believe that Core samples from within the Containment should be analyzed to assure that the Containment's integrity has not been jeopardized by operating Millstone Unit 3 under these conditions during the first four years of its operational life during the time period while concrete curing shrinkage is

known to occur.

44. In addition to my concerns regarding Millstone Unit 3's operation beyond its design basis due to the analytical tweaking of its one-of-a-kind Sub-Atmospheric Containment, I am also concerned about the reactor power level Dominion has applied in its new analysis in order to support the proposed increase application.

A. Specifically, Dominion Nuclear used a 7.01 percent increase as the basis for energy added to the Containment during an accident. As I have already shown in this Declaration, that 7.01 percent exceeds the NRC limits for consideration for a Stretched Power Uprate.

B. More importantly, Millstone Power Station Unit 3 already has a history of exceeding its licensed reactor power. According to the NRC Integrated Inspection Report on Millstone[17], Dominion Nuclear was cited for:

> "failure to maintain reactor core thermal power less than or equal to 3411 megawatts thermal (MGTH). Specifically, during performance of turbine overspeed protection system testing, the Unit 3 reactor's four minute power average exceeded 3479 MWTH." [Unit 3's license limit is 3411 MGTH also written MWt]

C. This higher power level, for which Dominion Nuclear was cited, is a full 2% higher than level of power Millstone Unit 3 is licensed to produce.

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[17] Inspection Report on Millstone, ML 080380599, February 7, 2008 for the period 10/012007 to 12/31/2007, Pages 4, 5, 21, and 22

D. Such a power level increase would also increase the energy available in an accident scenario by the same additional two percent.

E. Given Dominion's history of exceeding its licensed power level, it is my opinion that any analysis of Millstone Unit 3's Containment should use a 9% additional power level in order to most accurately reflect the condition of this one-of-a-kind Containment to withstand any additional pressures during an accident.

45. <u>Contention 4</u>: In its 1990 licensing application to change its Containment pressure, NU never mentioned its staffs' previous concerns about possible stress to the Containment's concrete due to the impact of its operation at high temperatures, low pressures, and low specific humidity. While it is a well known fact throughout the industry that concrete continues to shrink for up to 30-years as it matures after being poured, I was unable to uncover any NU or Dominion studies the long term impact Millstone Unit 3's concrete Containment due to its unique high temperature, low pressure, and low specific humidity environment.

46. Since nothing about this proposed change is either simple or standard, it is therefore my professional opinion that an Extended Power Uprate (EPU) review is more appropriate than a Stretched Power Uprate (SPU) review.

47. Furthermore, the Containment analysis for Millstone Unit 3 is further complicated by the fact that for the first four years of its operation, Millstone Power Station Unit 3 operated at the high, temperature, low pressure, low specific humidity unique to its Sub-Atmospheric Containment and therefore which may have compromised the structural integrity of the concrete.

48. In addition to being the lead licensing engineer at for NU at its Millstone Unit 3 nuclear plant during the 1970s, I have also been both a vice president and the senior vice president of a company that provided goods and services to Millstone 3 during the 1980s.

    A. In my capacity as an officer of the firm contracted to conduct structural analytical support to Millstone Unit 3 during its construction phase, I oversaw a group of sixty structural engineers at the Millstone Unit 3 site in 1984.

    B. Engineers reported to me during the construction phase informed me of other structural problems involving Millstone Unit 3's unique Containment.

    C. Due to the design of this Containment, the size and amount of rebar near major Containment penetrations created strategic geometry problems in the ability of the construction contractors to pour adequate amounts of concrete around the rebar in this tight configuration.

    D. This unique Containment design placed an enormous amount of rebar in several different directions around the Containment penetrations[18], making it extraordinarily difficult for concrete to slip by the rebar. Concrete voids between the rebar were a major concern. To "solve" this problem, NU qualified a procedure for the construction workers to apply long vibrating shafts into the rebar to get the concrete to slide around the rebar and create a heterogeneous block without voids.

E. This vibration method caused the sand to separate from the concrete if applied too long, and would create voids if applied for too short of a time.

F. While the procedure was qualified and construction workers were trained in how to operate the vibrating rods, my structural engineers were concerned that there was no way to test the Containment penetrations after the concrete had hardened to assure there where no voids.

G. The complex geometry at penetrations and the presence of concrete and steel intertwined made any ultrasonic exam impossible.

H. Core drilling was, of course, impossible, as it would weaken the Containment.

I. Given the structural limitations of the original design, and given that licensing changes in 1990 modified the Containment, it is imperative that this license modification be given a more thorough investigation than what is normally provided during a *Stretch* Power Uprate approval

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[18] Containment penetrations - Locations through the Containment wall where pipes like steam lines and feedwater lines enter and exit the Containment.

process.

49. Contention 5: Flow Accelerated Corrosion is another critical issue that should be considered the review of Dominion's proposed power increase application.

A. Dominion's proposed power uprate will change Millstone Power Station Unit 3's reactor coolant flow by approximately 7%.

B. It will impact the flow in and out of the reactor and the steam and condensate/feedwater flow on the secondary side of the plant will also be increased by 7%.

C. These flow increases in turn increase "Flow Accelerated Corrosion" thus causing pipes to wear out much faster.

D. This Flow Accelerated Corrosion is a non-linear phenomenon, and in my opinion is a significant risk due to the application of a 7% power increase on a plant that is already in the second-half of its engineered design life.

E. Disturbingly, in its application, Dominion did not propose hiring any new personnel at Millstone Power Station Unit 3 to deal with *flow accelerated corrosion* following the unit's proposed power uprate. This despite the fact that components will require more inspections because an uprate will cause those components to wear out much faster.

F. In general, Flow Accelerated Corrosion increases the likelihood of pipe failure.

G. Equally important, given Millstone Power Station Unit 3 exceeded licensed power less than a year ago, is the concern that pipe already worn thin by the seven percent power increase might break when power is increased further.

H. I saw no evidence that the Containment has been analyzed to withstand this increased energy.

50. I believe that Millstone Unit 3's program for assessing Flow Accelerated Corrosion in Dominion's proposed uprate of the plant fails to comply with 10 CFR50 Appendix B, XVI which states:

<u>10 CFR Appendix B to Part 50</u> – Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, XVI. Corrective Action that reads:

> "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management."

51. The power increase at Millstone Power Station Unit 3 will be accomplished by increasing the flow of water through both the primary and secondary sides of

the power plant. This increased flow through the pipes causes pipes to wear out faster by a phenomenon called Flow Accelerated Corrosion (FAC).

52. The basic two causes of FAC are erosion-corrosion of the pipe walls and cavitation- corrosion of the pipe wall. Electrolytic attack may also occur. Wall thinning from FAC is non-linear and is a local issue, caused by local geometry like Elbows and flow restrictions, local turbulence, and local metallurgical conditions (welds and impurities) in the pipe. Once local corrosion has started, changes in turbulence in the local area can intensify the corrosive attack. This localized nature of the corrosion is evident in a FAC pipe failure at the Surry plant in 1986. There a feed-water elbow had holes in one area, yet the nearby pipe wall was much less worn. Similar FAC piping failures have occurred at San Onofre in 1991 and 1993, Fort Calhoun in 1997, and Mihama in Japan in 2004. While this is an *old issue*, it has not been resolved, and instead has continued to plague the nuclear industry for more than three decades.

53. Due to the localized nature of the FAC, it is difficult to predict where and when a piping component might fail. The difficulty in developing accurate predictive models for FAC is the reason why, as recently as 2004, several workers were killed at Japan's Mihama I nuclear power plant. While prediction of what might fail is difficult, it is certain, however, to say that the rate at which piping components will wear out as a result of the proposed increase in power at Millstone 3 will exceed the 7 percent power increase due to the non-linear nature of FAC.

54. In my opinion, Dominion's application does not adequately address the guidance of NRC NUREG-1800, which requires that a FAC program address the scope, analytical tools, benchmarking of the computer model, preventative activities, what is monitored, what is inspected, trend analysis, acceptance criteria, operating experience, inspection techniques as well as data collection.

55. Furthermore, I believe Dominion's proposed License amendment for Millstone Power Station Unit provides inadequate information to determine if Millstone Nuclear Power Station Unit 3 has the management systems and staff in place to properly evaluate FAC if NRC approves Dominion's proposed power increase to the plant.

    A. The application did not discuss the increases in staff necessitated in order to maintain the plant in a safe condition if the proposed power increase is approved.

    B. Clearly the increase in the increased corrosion rates caused by the proposed 7% power level increase will require extra analysis, extra inspection, and extra maintenance, yet the application is silent on the need to increase Millstone Unit 3's inspection and maintenance staff.

56. Without such programmatic and staffing information, I am unable to further assess the adequacy of any actions Dominion Nuclear might have to mitigate

the consequences of Flow Accelerated Corrosion caused by the proposed power uprate at Millstone Nuclear Power Station Unit 3.

57. In conclusion: following a complete review of the evidence presented and by relying upon my nuclear safety and nuclear engineering experience in my review of the documents referenced herein above, it is my professional opinion that the issues discussed above are serious safety considerations germane to the subject of the license application in this case. Similarly after reviewing all the evidence presented, it is my professional opinion that Dominion Nuclear is ill prepared to increase the power at Millstone Nuclear Power Station Unit 3. Finally, since Dominion's proposed power increase is above NRC regulatory criteria and given the new stresses upon the one-of-a-kind formerly Sub-Atmospheric Containment, I believe that the evidence clearly shows the entire application should be given the more rigorous review of the Extended Power Uprate License Evaluation.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this day, March 15, 2008 at Burlington, Vermont.

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Arnold Gundersen, MSNE