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NEW AND SIGNIFICANT INFORMATION FROM THE FUKUSHIMA DAIICHI ACCIDENT IN THE CONTEXT OF FUTURE OPERATION OF THE PILGRIM NUCLEAR POWER PLANT

by
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Abstract

In March 2011 an earthquake and tsunami initiated an accident involving four nuclear power plants (NPPs) on the Fukushima Daiichi (Number 1) site in Japan. That accident is ongoing. Publicly available information about the accident in English language – and probably in Japanese as well – is incomplete and inconsistent at this time. Nevertheless, information has become available that is new and significant in the context of a proceeding before the US Nuclear Regulatory Commission (NRC) regarding an extension of the operating license for the Pilgrim NPP in Massachusetts. Additional information of this type is likely to become available over the coming months. This report discusses a selected set of Pilgrim-relevant issues that are affected by new and significant information from Fukushima. For the selected issues, findings are presented here that are designated as either Conclusive or Provisional. The Conclusive findings are fully supported by information that is already available. The Provisional findings await final confirmation from information that is likely to become available during coming months. For both types of finding, this report discusses the implications of the new and significant information for future operation of the Pilgrim plant. Findings in this report are consistent with a May 2006 report by the same author that addressed risk issues at the Pilgrim and Vermont Yankee NPPs.

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Contents

(See list of Tables, Figures, and Appendix on next page.)

List of Tables, Figures, and Appendix
(Tables and Figures appear after Sections I through VIII of the report, and are followed by the Appendix.)

Tables

Figures

Appendix

I. Introduction

The Pilgrim nuclear power plant (NPP) in Massachusetts features a boiling-water reactor (BWR) with a Mark 1 containment. The licensee (Entergy) has applied to the US Nuclear Regulatory Commission (NRC) for a 20-year extension of the plant's operating license. An NRC proceeding to consider this application is ongoing. The present license expires in 2012.

In March 2011 an earthquake and tsunami initiated an accident involving four NNPs on the Fukushima Daiichi (Number 1) site in Japan.[1] That accident is ongoing. Substantial amounts of radioactive material have been released to the atmosphere and the ocean. The four affected NPPs are BWR plants with Mark 1 containments, and are similar in design to the Pilgrim plant.

The Fukushima accident has already yielded information that is new and significant in the context of the Pilgrim license extension proceeding. However, information from Fukushima that has entered the public domain to date in English language – and probably in Japanese as well – is incomplete and inconsistent. Moreover, the accident is ongoing. Thus, it is not possible at this time to provide a full accounting of information from Fukushima that is new and significant in the context of the Pilgrim proceeding.

At present, indications are that conditions at the four damaged plants at Fukushima are becoming stabilized, and that future radioactive releases to atmosphere and ocean will be small in comparison to past releases. If this expectation proves correct, then a full accounting of information that is new and significant in the Pilgrim context will become possible when thorough, retrospective investigations of the Fukushima accident have been completed. Various investigations of the accident and its international implications are under way or anticipated. Their thoroughness remains to be seen. If the Fukushima accident takes a turn for the worse – which could involve events such as structural failure, renewed onset of fuel damage, and/or substantial radioactive release – then the clock will have to be re-started in terms of identifying information that is new and significant in the Pilgrim context.

This report discusses a selected set of issues that are relevant to the Pilgrim license extension proceeding, and that are affected by information from Fukushima that is new and significant in the Pilgrim context. For the selected issues, findings are presented here that are designated as either Conclusive or Provisional. The Conclusive findings are fully supported by information that is available to date. The Provisional findings are generally supported by available information, but await final confirmation from information that is likely to become available during coming months. For both types of finding, this report

[1] This site has six NPPs, all of which were operable prior to the accident. There were plans to construct two additional NPPs on the site. It appears that those plans have been canceled.

discusses the implications of the new and significant information for future operation of the Pilgrim plant.

The issues that are discussed here do not constitute a comprehensive set. Other issues that are relevant to the Pilgrim license extension proceeding, and that are affected by new and significant information from Fukushima, may be identified in the future. Also, in discussing a particular issue, this report does not purport to provide an exhaustive analysis. The purpose of this report is to identify selected issues, and to discuss each issue in sufficient detail to demonstrate that it should be thoroughly addressed in the Pilgrim license extension proceeding.

This report builds upon a previous report by the same author, completed in May 2006, that discussed risks and risk-reducing options associated with pool storage of spent fuel at the Pilgrim and Vermont Yankee NPPs.[2] The present report focuses on the Pilgrim plant, and addresses risks and risk-reducing options associated with both the reactor and the spent-fuel pool.[3] Findings in the present report are consistent with those in the May 2006 report.

There are eight narrative sections of this report, as shown on the Contents page. Conclusions are presented in Section VII, and a bibliography appears in Section VIII. Tables and figures appear after Section VIII, and are followed by an appendix. Documents and websites cited in the report are described in the footnotes or the bibliography.

II. The Fukushima Daiichi Accident: Status of Knowledge

To date, much of the information about the Fukushima accident that has entered the public domain in English language has become accessible through non-governmental sources. These sources include newspapers and the websites of trade associations and public-interest groups. Many of these sources work hard to be accurate. Nevertheless, the record of information available to date is incomplete and inconsistent. To an extent that is not yet clear, this situation reflects an absence of relevant knowledge, even by the Japanese officials directly involved.

Over time, the state of knowledge is likely to improve. It can be expected that English-language technical reports will emerge from governmental agencies, national and industry laboratories, academia, and professional societies. Interactions among the

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[2] Thompson, 2006.

[3] Some analysts define "risk" as the arithmetic product of two quantitative indicators: a consequence indicator; and a probability indicator. That definition is arbitrary and simplistic. NRC employs a broader definition of risk in its Glossary. See: http://www.nrc.gov/reading-rm/basic-ref/glossary/risk.html, accessed on 30 May 2011. NRC has cited a similar, broad definition by ASME, at: NRC, 2008, Footnote 4. This report employs a similar definition. Here, the risk of an activity is defined as a set of quantitative and qualitative information that describes the potential adverse outcomes from the activity and the probabilities of occurrence of those outcomes.

various authors will weed out inconsistencies and spur investigations to provide missing information. Nevertheless, some questions will never be fully answered.

One current source of information is a periodically updated status report issued by the Japan Atomic Industrial Forum (JAIF), which can be accessed at the website of the magazine, Nuclear Engineering International. In illustration, the update of 3 May 2011 provided estimates of reactor core damage as follows: Unit 1, 55 percent; Unit 2, 35 percent; Unit 3, 30 percent.[4] The same update described the damage state of spent fuel in the pools of Units 1-4 as follows: Unit 1, unknown; Unit 2, unknown; Unit 3, damage suspected; Unit 4, some damage presumed based on the presence of radioactive material in water sampled from the pool.[5]

Many sources have described hydrogen explosions in the reactor buildings of Units 1, 3, and 4, causing severe damage to these buildings. Numerous photos and videos are available, showing these explosions and their aftermath. Also, many photos and videos show the measures taken by the licensee (TEPCO) and supporting agencies to add water to spent-fuel pools. Early on, these measures included dropping seawater from bags suspended from helicopters, and spraying water from police riot control vehicles and military fire trucks. Both approaches proved ineffective. Eventually, TEPCO brought a concrete pumping truck with a long boom to the site, and this proved effective in adding water to spent-fuel pools.

TEPCO implemented a variety of measures to add water to reactor vessels and reactor containments. When supplies of fresh water were depleted, TEPCO used seawater for this purpose. These activities occurred inside buildings or at ground level on the site. Thus, there is a comparatively sparse body of publicly-available photos and videos showing these activities.

Another current source of information about the Fukushima accident is a periodically updated review that appears on the website of the World Nuclear Association (WNA), a trade association. In illustration, the 19 May 2011 update described Japanese estimates of the atmospheric release of radioactive material from Fukushima, including the following isotopic releases: 130 PBq of Iodine-131; and 12 PBq of Cesium-137.[6] For comparison, the atmospheric release during the Chernobyl accident of 1986 has been estimated to include 1,300 PBq of Iodine-131 and 89 PBq of Cesium-137.[7]

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[4] More recent reports suggest that most of the core has been damaged at Unit 1, Unit 2, and Unit 3. See, for example: Normile, 2011.
[5] JAIF, 2011.
[6] WNA, 2011.
[7] Krass, 1991, Table 2. (These release estimates were by a group at Lawrence Livermore National Laboratory.)

Relevance of Fukushima knowledge to countries other than Japan

In the USA, the direct effects of the Fukushima accident are expected to be comparatively small. Thus, the accident will be viewed with particular attention to its implications for NPPs in this country. NRC has taken this view in establishing a Fukushima task force that is scheduled to conclude a short-term review in July 2011.[8] This will be followed by a longer-term review that has been described by the NRC Chairman as follows:[9]

> "The task force's longer-term review will begin as soon as we have sufficient information from Japan, and we plan for it to be completed in six months from the beginning of the evaluation. During this longer-term review, we expect to engage the public, licensees, and other key stakeholders in a way that the time constraints of the short-term review have not allowed."

In the European Union, the response to the Fukushima accident is to undertake "stress tests" of NPPs in member states. Parameters of these tests have recently been agreed, after vigorous debate. Some member states and the European Commission wanted each stress test of an NPP to include an assessment of the plant's robustness against malicious acts. Other member states and the nuclear industry opposed the consideration of malicious acts in stress tests. This difference in position has been resolved by a compromise whereby the effects of malicious acts will be considered in separate studies.[10]

III. Documents Relevant to this Report

As discussed in Section I, above, this report builds upon a previous report by the same author, completed in May 2006, that focused on pool storage of spent fuel at the Pilgrim and Vermont Yankee NPPs.[11] Hereafter, the May 2006 report is described as the Thompson 2006 report. The present report is consistent with the Thompson 2006 report, which is incorporated here by reference. Thus, the data, findings, and citations in the Thompson 2006 report are not repeated here.

Similarly, the present report incorporates by reference two other reports by this author that have direct relevance to the Pilgrim license extension proceeding. One report – designated hereafter as the Thompson 2007 report – was prepared in the context of a license extension proceeding for the Indian Point NPPs.[12] The second report – designated

[8] NRC, 2011a; NRC, 2011b.
[9] Jaczko, 2011.
[10] Phillips, 2011.
[11] Thompson, 2006.
[12] Thompson, 2007.

hereafter as the Thompson 2009 report – was prepared in the context of NRC rulemaking on Waste Confidence.[13]

The Thompson 2006 report cites a Generic Environmental Impact Statement (GEIS) for license renewal of NPPs.[14] Two offspring of that GEIS are particularly relevant here. One offspring – designated here as GEIS Supp 29 – is a supplement specific to the Pilgrim NPP.[15] The second offspring – designated here as GEIS Draft Rev 1 – is a draft update of the GEIS.[16]

Other documents referenced in the present report are cited normally.

IV. A Standard for Assessing New and Significant Information

NRC has provided a definition of information that is "new and significant" in the context of the Pilgrim license extension proceeding and the GEIS.[17] The Fukushima accident is a major event that is yielding a substantial body of information that meets NRC's definition. For each of the six issues addressed here, information from Fukushima has become available, or is likely to become available, that is both new and significant.

The Fukushima accident poses a major challenge to the nuclear industry and its regulators. After the Three Mile Island accident of 1979, industry and regulatory officials around the world assured citizens that the necessary lessons had been learned and implemented. When the Chernobyl accident followed in 1986, many in the industry and the regulatory bodies laid the blame on technology and safety culture that were said to be unique to the USSR and its satellites. Now, a severe accident is unfolding in Japan, which has an open society and an engineering capability that few countries can match. Moreover, the accident is occurring at NPPs whose design was developed in the USA. The Pilgrim plant has a similar design. Thus, it shares design deficiencies that have become evident at Fukushima, some of which are discussed in this report.

NRC and the Pilgrim licensee have both moral and practical reasons to consider information from the Fukushima accident with utmost seriousness, eschewing any temptation to view the accident as somehow uniquely Japanese. The accident has revealed, for all to see, basic limitations in the design of NPPs similar to the Pilgrim plant. NRC and the licensee have a moral responsibility to review that design from first principles, and to apply a similarly stringent review to plant modifications and other measures that allegedly compensate for limitations in the design.

From a practical perspective, if NRC and the licensee fail to exhibit scientific curiosity, analytic rigor, and objectivity in the Pilgrim license extension proceeding, they will lose

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[13] Thompson, 2009.
[14] NRC, 1996.
[15] NRC, 2007.
[16] NRC, 2009.
[17] NRC, 2008, page 46208.

public trust. That outcome would not serve the long-term interests of the nuclear industry.

Many people are already pre-disposed to mistrust the nuclear industry and its regulators. For example, a recent New York Times story described a widespread belief in Japan that its nuclear industry and regulators have repeatedly ignored or concealed dangers, and failed to respond to citizen petitions.[18] Some observers believe that a similar situation pertains in the USA. They would find support for that position in a recent Foreign Affairs essay by Victor Gilinsky, a former NRC Commissioner who is highly critical of NRC's approach to license extension.[19]

NRC and the Pilgrim licensee could view the Pilgrim license extension proceeding as an opportunity to demonstrate that Gilinsky's criticisms are unfounded.

V. NPP Design Weaknesses, and the Roles of SAMAs, SAMGs, and EDMGs

The Pilgrim NPP received a construction permit in 1968 and entered service in 1972. Its design reflects the status of commercial nuclear engineering in the 1960s. It is a member of the so-called Generation II cohort of NPPs. If granted a license extension of 20 years, it would operate until 2032, seven decades after its basic design was fixed.

The first systematic study of the radiological risk posed by an NPP was the Reactor Safety Study, published in 1975.[20] That study identified a variety of sequences of events that could lead to damage to a reactor core and a substantial release of radioactive material to the environment. The Reactor Safety Study initiated the art of probabilistic risk assessment (PRA) for NPPs. Since 1975, many PRA studies have been done, of which the most thorough and open was NRC's NUREG-1150 study.[21] Operating experience and critical reviews have shown limitations in the scope and accuracy of PRA findings.[22] Nevertheless, PRA studies have played a useful role in identifying and characterizing NPP design deficiencies that were not anticipated by the designers.

As discussed in Section VI.1, below, there have now been twelve core-damage accidents at NPPs. Of these accidents, five have both: (i) occurred at a Generation II plant; and (ii) involved substantial fuel melting. The first of these five accidents was at Three Mile Island Unit 2 in 1979. In addition, there have been many " accident precursor" events at Generation II plants. This body of experience, supplemented by PRA studies, has revealed a variety of design deficiencies in Generation II NPPs.

Some of the design deficiencies are fundamental. For example, many Generation II NPPs have pressure-suppression containments surrounding their reactors. The Pilgrim

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[18] Onishi and Fackler, 2011.
[19] Gilinsky, 2011.
[20] NRC, 1975.
[21] NRC, 1990.
[22] See, for example: Hirsch et al, 1989.

NPP and the plants involved in the Fukushima accident are in this group. Plant vendors took this approach to containment design in an effort to reduce construction costs.[23] However, pressure-suppression containments proved to have fundamental deficiencies, which are discussed further in Section VI.4, below.[24]

The basic designs of the Generation II plants are fixed, so fundamental deficiencies in their design cannot be corrected. However, particular components of a plant can be altered or replaced, new systems can be added, and operating procedures can be changed. If implemented appropriately, such modifications can reduce the probability and/or magnitude of a radioactive release. The Pilgrim NPP, like all Generation II plants in the USA, has undergone a variety of such modifications over the years. Other modifications could be made, as discussed in this report.

Some of the modifications that have been made are within the plant's design basis, which means that they are regulated in the same manner as the plant's original design features. Many modifications, however, are in the beyond-design-basis regulatory arena. This arena exists because, when plants such as Pilgrim were designed, core-damage accidents were assumed to be so unlikely that they did not require regulatory attention. Operating experience and PRAs have shown that this assumption was unsound.[25] As a result, Generation II plants are operated with the foreknowledge that they can experience events that they were not designed to accommodate. Thus, for example, the containment of the Pilgrim plant was not designed to retain radioactive material in the event of a core melt, yet the plant is allowed to operate with the foreknowledge that core melt is a foreseeable event. Given such foreknowledge, NRC has been obliged to extend the regulatory arena beyond the plant's design basis.

Regulatory weakness in the beyond-design-basis arena

In some parts of the beyond-design-basis regulatory arena, there are clear regulatory requirements. For example, licensees are required to take specific actions related to offsite emergency response. In other parts of the arena, however, requirements are unclear, secret, or absent. In important respects, licensee actions that could reduce accident risk are entirely voluntary. The lack of clear requirements and responsibilities, compounded by the debilitating effects of secrecy (as discussed in Section VI.3, below), makes it likely that: (i) opportunities to reduce accident risk are missed; (ii) accident risk is not properly understood; and (iii) licensee performance in an emergency will be sub-optimal. Indeed, during an NRC briefing in May 2011, NRC Commissioners and Staff members expressed confusion and concern about the state of regulation in the beyond-design-basis arena.[26]

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[23] Cochran, 2011.
[24] For background, see, for example: Hanauer/Hendrie, 1972.
[25] For background, see, for example: Okrent, 1981; Ford, 1982.
[26] NRC, 2011a.

Severe accident mitigation guidelines (SAMGs) and extensive damage mitigation guidelines (EDMGs) illustrate this area of regulatory weakness. SAMGs were introduced as a voluntary industry initiative in the 1990s, and are not covered by normal NRC inspections. They instruct operators in procedures to accommodate plant conditions beyond the design basis. EDMGs apply to secret measures that were introduced after the attacks on New York and Washington in September 2001, purportedly to address potential fires and explosions at NPPs. Deficiencies in these guidelines are discussed further in Sections VI.2 and VI.3, below.

After the Fukushima accident began, NRC realized that SAMGs and EDMGs are crucial to the ability of licensees to respond to NPP accidents. Accordingly, NRC is conducting urgent inspections to assess the implementation of SAMGs and EDMGs at the Pilgrim plant and other NPPs.[27] It is not clear, however, what objective criteria NRC is using to assess the implementation of these guidelines, or what authority NRC has to order improvements in the guidelines and their implementation.

Severe accident mitigation alternatives (SAMAs) are modifications in plant hardware and/or operating procedures that could reduce the probability and/or magnitude of an accidental release of radioactive material from an NPP. SAMAs are discussed further in Section VI.1, below. License extension proceedings devote a substantial amount of attention to SAMAs. Appendix G of GEIS Supp 29 sets forth an evaluation of SAMAs for the Pilgrim plant. Despite this attention, however, the NRC Staff asserts that a licensee is obliged to implement cost-effective SAMAs only if they address aging problems.[28] This assertion is equivalent to stating that SAMA analysis in the context of a license extension proceeding is largely an empty exercise. The legal validity of the Staff's assertion is a matter beyond the scope of this report. From the perspective of a technical professional, it would be pointless of NRC to engage in SAMA analysis if the findings have little practical force.

The path not taken: Robust design

SAMAs, SAMGs, and EDMGs are measures that seek to compensate for fundamental weaknesses in the design of NPPs. Another path has been open to the nuclear industry for several decades, but has not been taken. That path is to design NPPs that are much more robust and inherently safe. An appendix to this report discusses some work that has been done in this area, including preliminary design work by the NPP vendor ASEA-Atom on the PIUS concept. A plant designed to PIUS criteria could, for example, have ridden out the earthquake and tsunami conditions at Fukushima without difficulty. Those conditions would have been comfortably within the plant's design basis for nuclear safety.

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[27] NRC, 2011a; NRC, 2011b.
[28] NRC, 2006, Sections 4.4.5 and 4.4.6.

The discussion in the appendix shows that studies on designing NPPs for greater robustness have considered a threat spectrum that includes not only accident-initiating events considered in PRAs – operator error, earthquakes, etc. – but also acts of malice. For example, a plant designed to PIUS criteria would not only have ridden out the Fukushima earthquake and tsunami, but would also ride out an aerial attack using 1,000-pound bombs.

It is natural for designers of highly robust plants to consider a broad spectrum of threats, including malicious acts. The same basic design principles – such as passive mechanisms, large thermal mass, simplicity, and strength – are relevant across that spectrum. There is no substantial tradeoff between designing against malicious acts and designing against a range of other threats.

If all NPPs in the USA were highly robust, there would be substantial benefits in terms of homeland security. Those benefits would be further enhanced if other elements of the nation's critical infrastructure (e.g., chemical plants) were also designed to be robust and inherently safe. Benefits to homeland security would accrue as shown in Table V-1.

The Pilgrim NPP could not be modified so as to meet PIUS criteria. Meeting those criteria at the Pilgrim site would require decommissioning of the present plant and construction of a new plant. All that can be done to reduce the radiological risk posed by continued operation of the Pilgrim NPP is to attempt to compensate for its fundamental deficiencies through measures such as SAMAs. (See, for example, the measures discussed in Sections VI.4 and VI.6, below.) Neither the licensee nor NRC claims that such compensatory measures can eliminate the potential for a large radioactive release from Pilgrim. We know from the Fukushima accident and preceding accidents that such a release is possible.

Interestingly, there is one respect – discussed in Section VI.5, below – in which the original design of the Pilgrim NPP employed an inherent-safety mechanism that has since been discarded. The plant was originally designed to hold a comparatively small inventory of spent fuel in the pool adjacent to its reactor. With that small inventory, low-density, open-frame racks could be used in the pool. Loss of water from a pool equipped with such racks would lead to spontaneous ignition of spent fuel only in rare circumstances. Now, the pool is equipped with high-density, closed-frame racks. As a result, loss of water would lead to spontaneous ignition across a broad range of circumstances.

VI. Selected Issues Affected by New and Significant Information from Fukushima

VI.1 Issue #1: Probability of Reactor Core Damage and Radioactive Release, Accounting for Cumulative Direct Experience

The Fukushima accident adds to direct experience with severe fuel damage at an NPP that leads to a radioactive release. It is important to determine how the newly enlarged body of direct experience should affect the consideration of fuel-damage probability and radioactive-release probability in the Pilgrim license extension proceeding.

Background

The occurrence of a large, offsite radiological impact from operation of the Pilgrim NPP would involve a release to the environment of a substantial amount of radioactive material. That release would involve a two-part process. First, there would be severe damage to nuclear fuel, either in the reactor core or in a location where spent fuel is stored. At present, Pilgrim spent fuel is stored exclusively in an elevated pool adjacent to the reactor. Second, there would be an open pathway from the damaged fuel to the plant's environment (atmosphere, ocean, groundwater, etc.). This report focuses on pathways leading to the atmosphere. That focus does not imply that other release pathways would be insignificant.

Here (Section VI.1), the focus of discussion is on damage to fuel in the reactor core, accounting for cumulative direct experience. Section VI.5, below, discusses damage to fuel in the spent-fuel pool, accounting for Fukushima direct experience.

Estimates of the probability of core damage and an accompanying radioactive release play a role in the Pilgrim license extension proceeding. They find particular application in consideration of Severe Accident Mitigation Alternatives. SAMAs are modifications in plant hardware and/or operating procedures that could reduce the probability and/or magnitude of an accidental release of radioactive material from an NPP.

The licensee (and NRC) assess the benefit of a potential SAMA by estimating the present values, with and without the SAMA, of the monetized risk (assumed, in this instance, to be the monetized radiological impact multiplied by the probability) of a radioactive release, accumulated over the 20-year period of license extension.[29] The difference between these values, assuming that the with-SAMA value is lower, is the estimated benefit of the SAMA. The licensee compares that benefit with the projected cost of the SAMA.

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[29] NRC and this report provide a general definition of risk, as discussed earlier in this report, that extends beyond the simple multiplication of a consequence indicator and a probability indicator.

The methodology used by the licensee to assess the benefits and costs of potential SAMAs can be challenged in various respects. Here, however, we focus solely on the probability of a radioactive release. Using the licensee's methodology, the present value of the cumulative, monetized risk of a radioactive release is a linear function of the probability of that release.

The probability of severe core damage and an accompanying radioactive release can be estimated in two ways. First, the probability can be estimated using the techniques of probabilistic risk assessment. In a PRA study, analytic techniques such as fault trees are used to predict the occurrence of comparatively rare sequences of events that would lead to severe fuel damage and, potentially, a radioactive release. Second, the probability can be estimated from direct experience. The Fukushima accident has expanded the body of experience in this respect to the point that the data set of accidents involving severe fuel damage, although still sparse, can provide a reality check for PRA estimates.

Estimating core-damage probability using PRA techniques

As discussed in Section V, above, the most thorough and open PRA study conducted to date was NRC's NUREG-1150 study.[30] Figures VI-1 through VI-3 present some results from that study. Estimated core damage frequency (CDF) is shown for two NPPs. One is a Surry pressurized-water reactor (PWR) plant. The other is a Peach Bottom BWR plant, whose design is similar to that of the Pilgrim plant. The estimated conditional probability of reactor containment failure, given core damage, is shown in Figure VI-3.

The Pilgrim licensee has used PRA techniques to assess the potential for severe fuel damage and radioactive release. Drawing from that work, the Thompson 2006 report summarized (see Table 6-1 of that report) various licensee estimates, as of May 2006, of CDF and Early release frequency for the Pilgrim NPP.[31] As of May 2006, the licensee estimated the CDF from internal initiating events at 6.4E-06 per reactor-year (RY). (Previous licensee estimates were higher.) The licensee adjusted this estimate by a factor of 6 to account for external initiating events and uncertainty, yielding an overall CDF of 3.8E-05 per RY.

As discussed in GEIS Supp 29 (see page G-10 of that report), the licensee subsequently altered its SAMA analysis to use an adjustment factor of 5 to account for external initiating events but not uncertainty. With that adjustment factor the overall CDF would be 3.2E-05 per RY.

A conditional probability of Early release can be inferred from various licensee estimates of release frequency. The inference can use data shown in Tables 6-1 and 9-1 of the Thompson 2006 report. For the Pilgrim NPP, the latter table shows an overall CDF of

[30] NRC, 1990.

[31] An Early release at the Pilgrim NPP (see Table 6-3 of the Thompson 2006 report) would begin less than 24 hours after the core-damage accident begins.

1.3E-04 per RY and an Early release frequency of 4.2E-05 per RY. These numbers yield a conditional probability of Early release of 0.32.

To summarize, the licensee's current position regarding SAMA analysis is that the overall CDF at Pilgrim without SAMAs, not accounting for uncertainty, is 3.2E-05 per RY (1 event per 31,000 RY), and the conditional probability of Early release is 0.32 (32 percent). These numbers provide a baseline for assessing the benefits of SAMAs. To a first-order approximation, the benefit of a particular SAMA would scale linearly with baseline values of CDF and the conditional probability of release. Indeed, as indicated on page G-10 of GEIS Supp 29, the licensee has performed such linear scaling in accounting for the role of external initiating events.

Estimating core-damage probability from direct experience

Severe fuel damage at an NPP is often thought of as a rare event. Yet, a recent inventory lists twelve events involving severe damage to fuel in the reactor core of an NPP.[32] This inventory excludes similar events at non-power reactors. For example, it excludes the core fire and radioactive release experienced in 1957 by a reactor at the Windscale site in the UK. That reactor was used to produce plutonium and other materials for nuclear weapons.

Of the twelve core-damage accidents at NPPs, five have both: (i) occurred at a Generation II plant; and (ii) involved substantial fuel melting. These five events were at Three Mile Island (TMI) Unit 2 (a PWR plant in the USA) in 1979, Chernobyl Unit 4 (an RBMK plant in the USSR) in 1986, and Fukushima Daiichi Units 1 through 3 (BWR plants in Japan) in 2011.

These five events occurred in a worldwide fleet of commercial NPPs, of which 440 plants are currently operable. These plants and previous plants in the fleet had accrued 14,484 RY of operating experience as of 16 May 2011.[33] The five events provide a data set that is comparatively sparse and therefore does not provide a statistical basis for a high-confidence estimate of CDF. Nevertheless, this data set does provide a reality check for PRA estimates of CDF. Confidence in this reality check is enhanced by noting that the five events occurred in three different countries at three different types of NPP, involved differing initiating events, and happened on three distinct occasions over a period of 32 calendar years. This spread of accident characteristics is consistent with the diversity of circumstances that PRA analysis seeks to address.

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[32] Cochran, 2011.
[33] World Nuclear Association website, http://www.world-nuclear.org/, accessed on 16 May 2011.

Findings and implications

The occurrence of five core-damage events over a worldwide experience base of 14,500 RY can be translated to a CDF of 3.4E-04 per RY (1 event per 2,900 RY).[34] This value is an order of magnitude higher than the baseline CDF estimate of 3.2E-05 per RY (1 event per 31,000 RY) that the Pilgrim licensee developed using PRA techniques. One can reasonably find that the licensee has under-estimated the baseline CDF of the Pilgrim plant by an order of magnitude. Such a finding is supported by a technical literature describing the limitations of PRA techniques.[35] This finding is Conclusive, because there is general agreement that severe core damage has occurred at three NPPs at Fukushima.

Note from Figure VI-2 that the NUREG-1150 study, using Livermore seismic predictions, yielded a mean estimate of CDF at Peach Bottom of about 1.0E-04 per RY from earthquake initiation alone. This result is roughly consistent with a CDF estimate of 3.4E-04 per RY from direct experience worldwide.

In terms of the conditional probability of containment failure and radioactive release, direct experience tells us that the five NPP accidents discussed above yielded one small radioactive release (TMI), one large release (Chernobyl), and one apparently intermediate release (Fukushima). Future investigations will, presumably, refine estimates of the Fukushima release and will assign contributions to that release from the reactor cores of Units 1-3 and the spent-fuel pools of Units 1-4. For the time being, in the Pilgrim context, it may be appropriate to use the licensee's estimate of the conditional probability of an Early release at Pilgrim, namely 0.32.

In light of the above-stated finding, the licensee's SAMA analysis for Pilgrim should be re-done with a baseline CDF that is increased by an order of magnitude. To account for experience at Fukushima, the re-done SAMA analysis should encompass, among other SAMA options, measures to accommodate: (i) structural damage; and (ii) station blackout, loss of service water, and/or loss of fresh water supply, occurring for multiple days.

A CDF of 3.4E-04 per RY is equivalent to a cumulative core-damage probability of 0.7 percent over a 20-year period of license extension for one NPP. For the 104 NPPs in the USA, the cumulative core-damage probability over 20 years would be greater. Thus, in view of the evident impacts of the Fukushima accident, measures that can reduce CDF and/or the conditional probability of containment failure should be assessed and implemented with great seriousness. It follows that any accident-mitigation measure or

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[34] This simple estimate of CDF might be criticized because the three core-damage events at Fukushima had a common cause. However, there are some design differences between the three affected plants at Fukushima, and it appears that there were differences in the accident sequences at these plants. Also, multiple core-damage events with a common cause could occur in the future, potentially involving plants at single-unit sites.

[35] See, for example: Hirsch et al, 1989.

SAMA that is credited for the future licensed operation of the Pilgrim NPP should be incorporated in the plant's design basis. That implication holds across all six issues addressed in this report, and is designated as General Implication #1.

VI.2 Issue #2: Operators' Capability to Mitigate an Accident, and its Effect on the Conditional Probability of a Spent-Fuel-Pool Fire During a Reactor Accident

The capability of operators to mitigate an NPP accident could affect the probability of a radioactive release from the accident. For example, the operators' capability could affect the conditional probability of a spent-fuel-pool fire during a reactor accident. The Thompson 2006 report addressed that conditional probability in the context of the Pilgrim license extension proceeding.

The Fukushima accident adds to experience regarding operators' capability to mitigate an accident. It is important to learn from that experience in the Pilgrim context. Also relevant in this context are EDMGs that were recently placed in the public domain pursuant to the Fukushima accident.

Background

The earthquake and tsunami at Fukushima caused extensive damage at the site. As the resulting accident proceeded, hydrogen explosions produced further damage. Plant operators and other personnel were obliged to work in a highly disturbed environment where many items of equipment were non-functional and many parts of the affected plants were inaccessible. Operators encountered high radiation fields, high temperatures, smoke, debris, and steam. Supplies of electrical power and fresh water were interrupted for long periods.

These difficult working conditions have been described extensively in print and electronic media. Some citations to media coverage are provided here for illustration.[36] Future investigations may reveal inaccuracies in the media coverage. However, enough information has become available, and is confirmed by multiple sources, to support some initial findings about operators' capability to mitigate a severe accident. These findings extend to the Pilgrim plant.

A thorough, retrospective investigation of the Fukushima accident would systematically examine, among various other matters: (i) the extent to which the accident environment degraded operators' capability to manage or mitigate the accident; (ii) the effectiveness of measures taken by the operators to address that challenge; and (iii) the conditional probabilities that event sequences at the affected reactors and pools would have followed

[36] Bradsher and Tabuchi, 2011; Bradsher, 2011; Belson, 2011; Sanger and Broad, 2011; Onishi, Sanger, and Wald, 2011; Glanz and Broad, 2011; Tabuchi, Bradsher, and Wald, 2011.

a course such that the radioactive release was greater (or smaller) than was actually experienced.

Such an investigation has not been completed or even begun. Nevertheless, important lessons can be gained now by focusing on one aspect of accident mitigation at Fukushima – adding water to spent-fuel pools. TEPCO recognized the need for this addition at a comparatively early point of the accident. However, TEPCO's approaches to adding water to pools revealed a lack of preparation for this contingency.

Adding water to spent-fuel pools

Early on, TEPCO tried dropping seawater from bags suspended from helicopters, and spraying water from police riot control vehicles and military fire trucks. Both approaches proved ineffective. Eventually, TEPCO brought a concrete pumping truck with a long boom to the site, and this proved effective in spraying water into spent-fuel pools.

This experience is directly relevant to the Pilgrim plant. As at other plants in the USA, EDMGs at Pilgrim cover measures that seek to mitigate damage if the plant experiences an attack or an accident. The EDMGs were drawn up by the Nuclear Energy Institute (NEI), which is an industry association. They were secret until NRC recently placed them in the public domain.[37] NRC has made them a license condition for the Pilgrim plant.[38] The measures covered by these EDMGs at Pilgrim include measures for adding water to the spent-fuel pool.

Note that NRC placed the EDMGs into the public domain in response to the Fukushima accident. Thus, the newly-disclosed EDMGs add to the body of new and significant information that arises from the Fukushima accident.

In the newly-disclosed EDMGs, NEI calls for a capability to spray at least 200 gpm of water into the Pilgrim pool. This pool is high up in the reactor building. To accommodate this situation, NEI calls for the spray capability to include:[39]

> "Capability to lift/locate the monitor nozzle such that the spray can be externally directed into the spent fuel pool (e.g., from an adjacent building roof, fire truck extension ladder). The lifting capability (e.g., crane or fire truck with extension ladder) may be located off-site as long as the site has confidence (e.g., through an MOU) that it will be available for use on-site within the required timeframe (i.e., 2 hours or 5 hours). This may require a modification to the lifting device to allow the monitor nozzle to be affixed."

Presumably, the Pilgrim licensee has made an arrangement to bring a truck-mounted crane or a ladder fire truck to the site at short notice. This arrangement might work in

[37] NEI, 2006.
[38] Kim, 2007.
[39] NEI, 2006, page 13.

some situations. However, there are several factors that could render the arrangement unworkable, including:

- This arrangement can never be realistically tested.
- An event that initiates or co-initiates the accident (e.g., earthquake, hurricane, ice storm, blizzard, attack) could also render the truck unavailable.
- A radioactive release from a reactor accident could produce radiation fields that render the truck unavailable, or preclude its use on the site.
- There seems to be no provision for a radiation-resistant TV camera to guide nozzle positioning, or for shielding of the truck/spray operators.
- There seems to be no recognition that spraying water on exposed spent fuel could, in some circumstances, exacerbate the accident by feeding a zirconium-steam fire.

To some extent, NEI recognizes that its guidelines cannot guarantee that water can always be added to the pool. NEI says:[40]

> "It is understood that not all conceivable scenarios can be mitigated by sprays. The objective is for each site to work to identify means to spray the pool."

Findings and implications

The preceding discussion leads to the following findings. First, Fukushima experience shows clearly that the operators' capability to mitigate an accident at the Pilgrim NPP can be severely degraded in the accident environment. Second, examination of NEI's newly-disclosed EDMGs, focusing on the addition of water to spent-fuel pools as an illustrative example, reveals that the EDMGs are inadequate to mitigate the range of fuel-damage events that could occur at the Pilgrim plant.[41] That finding rests on a review of the EDMGs themselves, and on Fukushima experience. Third, in view of inadequacies in the EDMGs, it is clear that there is a substantial conditional probability of a spent-fuel-pool fire during a reactor accident at the Pilgrim plant. These are Conclusive findings.[42] Each of these findings supports General Implication #1. The third finding – regarding the conditional probability of a pool fire during a reactor accident – is reached again in Section VI.5, below, where it is based on direct experience of pool performance at Fukushima. Section VI.5 discusses the implications of this third finding.

The Thompson 2006 report (see Table 9-1 of that report) assumes that the conditional probability of a spent-fuel-pool fire at Pilgrim, given a reactor accident that leads to an Early release, is 50 percent.[43] That continues to be a reasonable assumption for the purposes of SAMA analysis.

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[40] NEI, 2006, page 14.
[41] For additional information on the limitations of EDMGs, see: UCS, 2011.
[42] In this instance there is some overlap between findings and implications, as illustrated in Table VI-1. Definitive wording of this report's findings and implications is provided in Section VII of the report.
[43] An underlying, reasonable assumption is that normal systems for makeup and cooling of spent-fuel-pool water would be lost during the progress of the reactor accident.

VI.3 Issue #3: Secrecy Regarding Accident Mitigating Measures

Since 2001, NRC has been highly secretive regarding measures for mitigating NPP accidents. NRC has extended this secrecy to technical analysis related to spent-fuel behavior following loss of water from a spent-fuel pool. It is important to examine information arising from the Fukushima accident, to determine how secrecy could affect the implementation of accident-mitigating measures at the Pilgrim plant.

Background

In 1992, two Harvard physicists with experience in nuclear-power issues published a paper about the 1986 Chernobyl accident, with the title, "Chernobyl: the inevitable results of secrecy". The paper's Abstract said:[44]

> "The Chernobyl accident was the inevitable outcome of a combination of bad design, bad management and bad communication practices in the Soviet nuclear industry. We review the causes of the accident, its impact on Soviet society, and its effects on the health of the population in the surrounding areas. It appears that the secrecy that was endemic in the USSR has had profound negative effects on both technological safety and public health."

These authors did not believe that secrecy was unique to the USSR. Indeed, they stated that it tends to occur in "all countries".[45]

The experience of Ivan Zhezerun illustrates the debilitating effects of secrecy. Zhezerun, a senior member of the USSR science establishment, began to write to Soviet officials in 1965 to notify them of a serious defect in the design of the RBMK reactor type – its positive void coefficient of reactivity. Chernobyl Unit 4 experienced an accident in 1986 that was due to this defect and the operators' and plant managers' lack of knowledge of the defect. Zhezerun's repeated warnings were ignored and attempts were made to silence him. Public disclosure of his concerns would have led to his imprisonment.[46]

The USA has a longstanding tradition of openness, which is generally believed to be a major pillar of the nation's prosperity and democracy. Contrary to that tradition, NRC has been highly secretive since the attacks on New York and Washington in September 2001. From a security perspective, secrecy on some subjects is appropriate. However, excessive secrecy can be counterproductive in terms of both safety and security, and has deleterious effects on society. This author has shown, in the Thompson 2007 report (see, for example, Section 8 of that report) and the Thompson 2009 report (see, for example, Section 9 of that report), that NRC's secrecy has been excessive and counterproductive.

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[44] Shlyakhter and Wilson, 1992, page 251.
[45] Shlyakhter and Wilson, 1992, final paragraph on page 251, continued to page 252.
[46] Medvedev, 1990, pp 258-259.

How secrecy can degrade accident mitigation capability

Mitigation of an NPP accident requires actions to be taken before and during an accident. One of the necessary conditions for effective conduct of those actions is that plant managers and operators have a thorough understanding of the phenomena that could occur during an accident. The Chernobyl accident illustrates the debilitating effect of suppressing information about such phenomena. At Chernobyl, plant managers and operators had been denied basic information about the accident propensity of the plant.

Similarly, mitigation measures to be taken during an accident – such as EDMGs – should be thoroughly understood by all persons who may be involved in the implementation of such measures. This requires that: (i) generic measures are thoroughly reviewed beforehand to assess their effectiveness; (ii) the findings of such assessments are circulated widely; and (iii) descriptions of the generic measures are widely circulated. If these conditions are not met, it is likely that ineffective measures will be adopted, and/or that measures will not be well implemented when needed. When the generic measures are applied at a particular site – such as Pilgrim – it may not be appropriate to publish details of their application. However, the details should be shared with all entities that may be involved in implementing the measures.

Lessons from the Fukushima accident

Lessons relevant to the Pilgrim license extension proceeding can be learned from events during the Fukushima accident, and from review of the EDMGs that have been disclosed pursuant to the Fukushima accident. Here, the focus, for illustrative purposes, is on lessons regarding measures related to the addition of water to spent-fuel pools.

TEPCO's response to the Fukushima accident revealed a poor understanding of the risk of a spent-fuel-pool fire. In illustration, TEPCO initially tried several ineffective measures for adding water to spent-fuel pools, as discussed in Section VI.2, above. Clearly, TEPCO had given little thought to this task before the accident. Also, a review of the newly-disclosed EDMGs reveals (see Section VI.2, above) an incomplete understanding of the phenomena that could be associated with a pool fire.

As discussed in the Thompson 2006 report, NRC documents published prior to 2001 revealed an incomplete understanding of phenomena that could be associated with a spent-fuel-pool fire. That weak state of knowledge contrasted with the more advanced state of knowledge about reactor accidents that NRC had acquired through studies such as NUREG-1150.[47]

Since September 2001, NRC has sponsored additional, secret studies – the so-called "Sandia studies" – that purportedly provide additional information about phenomena

[47] NRC, 1990.

associated with a pool fire.[48] It would be surprising if such secret studies rectified the pattern of deficiencies, which had persisted for two decades, in NRC's preceding, open studies. That skepticism is supported by severe inconsistencies in NRC descriptions of the secret studies. (See Thompson 2009 report, Section 5.2.) GEIS Draft Rev 1 discusses the Sandia studies in its Section E.3.7. That discussion does not acknowledge NRC inconsistency in characterizing the Sandia studies, and invites continuing skepticism as to whether these studies have corrected longstanding deficiencies in NRC's understanding of phenomena associated with a pool fire.

It should be noted that NRC's secret studies purportedly addressed engineering issues – such as convective and radiative heat transfer, and exothermic reactions of zirconium – that are highly pertinent to NPP accidents not initiated by malicious acts. Indeed, such issues will need to be examined in detail in any thorough, retrospective investigation of the Fukushima accident.

Thus, NRC secrecy about phenomena associated with pool fires has been counterproductive. It has retarded the development of technical understanding about pool-fire phenomena. As a result, NRC decision-making related to pool fires has not been well informed. Also, TEPCO has been deprived of information that could have helped it to mitigate the Fukushima accident.

In regard to mitigation measures to be taken during an accident – such as EDMGs – NRC's excessive secrecy has precluded a thoroughly understanding of these measures by all persons who may be involved in their implementation. Relevant measures have not been thoroughly reviewed beforehand to assess their effectiveness. Descriptions of the measures have not been widely circulated. Thus, TEPCO has been deprived of information that could have helped it to mitigate the Fukushima accident. Also, the EDMGs have deficiencies such as those discussed in Section VI.2, above. NRC has belatedly acknowledged these debilitating outcomes by recently disclosing the EDMGs. However, NRC has taken no step to systematically and openly assess the EDMGs in light of the best available knowledge about pool fires and other relevant matters.

Findings and implications

From the preceding discussion, this report reaches a Conclusive finding that NRC's excessive secrecy degrades the licensee's capability to mitigate an accident at the Pilgrim NPP. This finding supports General Implication #1. Also, this finding shows that: (i) NRC secrecy regarding the general characteristics of accident mitigation measures and the phenomena associated with spent-fuel-pool fires should cease; and (ii) NRC should sponsor open research on spent-fuel-pool fires and their mitigation.

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[48] NRC, 2008, page 46207.

VI.4 Issue #4: Hydrogen Control

Hydrogen explosions have caused severe damage to reactor buildings at the Fukushima Daiichi site. It is important to determine if these explosions could be replicated at the Pilgrim plant, and if prevention of hydrogen explosions has been adequately considered in the Pilgrim license extension proceeding.

Background

Generation of hydrogen during a reactor accident has long been recognized as a problem. It is an especially significant problem for a reactor surrounded by a low-volume, pressure-suppression containment – such as the BWR Mark 1 containment used at the Pilgrim plant and the NPPs involved in the Fukushima accident.

Internal correspondence within the US Atomic Energy Commission (AEC) in 1972 revealed AEC Staff concerns about pressure-suppression containments, in part related to their propensity to experience hydrogen explosion. One Staff member (Hanauer) said:[49]

> "I recommend that the AEC adopt a policy of discouraging further use of pressure-suppression containments, and that such designs not be accepted for construction permits filed after a date to be decided (say two years after the policy is adopted)."

In response, another Staff member (Hendrie) said that the idea of banning pressure-suppression containments was "an attractive one in some ways". He went on to say:[50]

> "However, the acceptance of pressure suppression containment concepts by all elements of the nuclear field, including Regulatory and the ACRS, is firmly embedded in the conventional wisdom. Reversal of this hallowed policy, particularly at this time, could well be the end of nuclear power. It would throw into question the continued operation of licensed plants, would make unlicensable the GE and Westinghouse ice condenser plants now in review, and would generally create more turmoil than I can stand thinking about."

The Hanauer/Hendrie correspondence occurred in the context of a design-basis accident, which would produce a comparatively small amount of hydrogen. In a beyond-design-basis accident, such as the one being experienced at Fukushima, the problem of hydrogen control is greatly exacerbated. Damage to the reactor buildings at Fukushima Daiichi Units 1, 3, and 4 provides clear evidence of the problem.

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[49] Hanauer/Hendrie, 1972.
[50] Hanauer/Hendrie, 1972.

Hendrie's fear of "turmoil" prevailed, and the use of pressure-suppression containments was allowed to continue. The Pilgrim licensee seeks to employ this deficient containment system at Pilgrim until 2032.

Hydrogen control measures have been introduced at NPPs in the USA that have pressure-suppression containments, in an effort to compensate for the basic design limitations of these containments. These measures include venting systems intended to disperse hydrogen into the atmosphere before it explodes.

Lessons from Fukushima about hydrogen control

When hydrogen explosions occurred at Fukushima, US officials said at first that this experience was not relevant to NPPs in the USA because the venting systems at US plants are more capable than the systems used in Japan. TEPCO later said that the affected plants at Fukushima had been equipped, years ago, with venting systems of the type used in the USA. Early in the accident, US officials also attributed hydrogen explosions at Fukushima to a delay by TEPCO officials in ordering the initiation of venting. Later, it emerged that aspects of the venting systems at Fukushima were inoperable, due to influences that are believed to include station blackout and earthquake damage.[51]

These events demonstrate the need for NRC and the US nuclear industry to undertake a measured, careful review of hydrogen control systems, rather than rushing to judgment. They also provide prima facie evidence that the venting systems at the Pilgrim plant and other NPPs in the USA have serious design deficiencies. Further evidence to that effect is correspondence five years ago in which engineers at the Monticello plant – a BWR plant with a Mark 1 containment – warned NRC that the hydrogen venting systems at Monticello and similar plants across the USA have design deficiencies.[52]

Findings and implications

The information set forth above allows a Provisional finding that hydrogen explosions experienced at Fukushima could be replicated at the Pilgrim plant, and that the potential for such explosions has not been adequately considered in the Pilgrim license extension proceeding. A considerable amount of further information related to this finding is likely to emerge from future investigations of the Fukushima accident and its lessons for NPPs in the USA.

An implication from this Provisional finding, and from the information that supports it, is that containment venting and other hydrogen control systems at the Pilgrim plant should be upgraded, and should use passive mechanisms as much as possible. A further

[51] Tabuchi, Bradsher, and Wald, 2011.
[52] Wald, 2011.

implication is that all hydrogen control measures at the Pilgrim plant, involving both hardware and operating procedures, should be incorporated in the plant's design basis.

VI.5 Issue #5: Probability of a Spent-Fuel-Pool Fire and Radioactive Release, Accounting for Fukushima Direct Experience

Section VI.2, above, discusses the potential for a spent-fuel-pool fire at the Pilgrim plant in terms of the fire's conditional probability during a reactor accident. That discussion focuses especially on operators' capability to add water to a spent-fuel pool during an accident. In that discussion, and here, it is assumed that loss of water would lead to a pool fire in the manner described in the Thompson 2006 report. The basis for that assumption is discussed below.

Section VI.1, above, draws from cumulative direct experience to discuss the probability of a reactor-core-damage accident and radioactive release at the Pilgrim plant.

Prior to the Fukushima accident, there was no direct experience with a spent-fuel-pool fire. During the accident, it appears that there has not been a full-scale fire of the type discussed in the Thompson 2006 report. However, it appears that there was fuel damage in at least one pool at Fukushima – Unit 4 – and there may have been an episode of steam-zirconium reaction in that pool.

It is important to understand what direct experience from Fukushima can tell us about the potential for a pool fire at the Pilgrim plant. However, much of the relevant information is not available at this time.

What we need to know

A thorough, retrospective investigation of the Fukushima accident would seek to determine, among various other matters: (i) damage to spent-fuel pool structures and pool support systems (e.g., cooling and makeup systems) from the earthquake, the tsunami, and hydrogen explosions; (ii) damage to spent fuel in the pools from the earthquake, hydrogen explosions, falling debris, and steam-zirconium or air-zirconium reactions; (iii) water levels and temperatures in the pools over time; (iv) radioactive releases from spent fuel via air or water pathways; (v) timing and extent of any steam-zirconium or air-zirconium reaction; and (vi) nature and effectiveness of mitigation measures.

What we know now

Absent such an investigation to date, we must use fragmentary information such as the following items. Early in the accident, press reports indicated that spent fuel in the Unit 3 pool may have been exposed to air, and the NRC Chairman asserted that for a period there was little or no water in the Unit 4 pool.[53] The information underlying that

[53] Bradsher and Tabuchi, 2011.

assertion has not been disclosed. There was a violent hydrogen explosion in the Unit 4 reactor building, leading many observers to conclude that a steam-zirconium reaction had occurred in the Unit 4 pool. Such a reaction could generate a copious amount of hydrogen. Recently, TEPCO has theorized that the hydrogen that exploded at Unit 4 had come from Unit 3 through a ventilation system. This theory rests, in part, on a lack of visual evidence of fuel damage in the Unit 4 pool.[54] Yet, water samples from the Unit 4 pool suggest that some fuel in that pool was damaged.[55]

To date, no evidence has emerged from Fukushima to suggest that a full-scale pool fire – as discussed in the Thompson 2006 report – will not occur if water is lost from the Pilgrim spent-fuel pool. Also, there has been no publication of technical literature, since the Thompson 2006 report was completed, to support such a suggestion. Thus, for the time being, the Thompson 2006 report describes the general state of knowledge about pool fires.

Given knowledge to date, direct experience from Fukushima does not yield a clear, quantitative estimate of the probability of a pool fire at the Pilgrim plant or elsewhere. The Fukushima accident does, however, provide direct experience of events that could be precursors of pool fires. At Fukushima, water may have been lost from pools through leaking, sloshing, and/or displacement by debris. Pool structures may have experienced earthquake damage. Cooling and makeup systems were inoperable for long periods. Ultimately, keeping spent fuel submerged relied entirely on jury-rigged systems for water addition.

Findings and implications

In view of the occurrence of an array of pool-fire precursor events at Fukushima, direct experience from Fukushima shows that there is a substantial conditional probability of a pool fire during a reactor accident at the Pilgrim NPP. That finding is Provisional.

As a separate matter, this report makes a Conclusive finding, set forth in Section VI.2, above, that there is a substantial conditional probability of a pool fire during a reactor accident at the Pilgrim NPP. That finding does not rest on direct experience. Instead, it rests on a judgment that jury-rigged systems may fail to add water to an affected pool in sufficient quantity to prevent a pool fire. At Fukushima, jury-rigged systems ultimately succeeded in covering the fuel with water.

Given these two, separate approaches to assessing the conditional probability of a pool fire, and the substantial probability of a reactor accident that is determined in Section VI.1, above, reducing the probability of a pool fire at the Pilgrim plant should be a high priority. Appropriate actions would include the consideration, in a re-done SAMA analysis, of a range of measures to prevent a pool fire. The measures to be considered

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[54] WNN, 2011.
[55] JAIF, 2011.

could include an emergency pool cooling and makeup system now being offered by Westinghouse.[56]

The most effective and reliable measure to prevent a pool fire at the Pilgrim plant would be to re-equip the Pilgrim pool with low-density, open-frame racks. In light of the findings in Section VI.1, above, together with conclusion C12 and supporting information in the Thompson 2006 report, SAMA methodology shows that the Pilgrim pool should be re-equipped with low-density, open-frame racks. Separate from SAMA analysis, prudent engineering principles also indicate that the Pilgrim pool should be re-equipped with low-density, open-frame racks. Finally, the finding set forth here supports General Implication #1.

VI.6 Issue #6: Filtered Venting of Reactor Containment

Section VI.4, above, describes the design limitations of reactor containments that rely on pressure suppression. The Pilgrim NPP and the NPPs involved in the Fukushima accident each have a low-volume, pressure-suppression containment – the BWR Mark 1 containment.

It has long been known that the BWR Mark 1 containment would be unable to retain the steam and hydrogen generated during a core-damage accident. Given the small volume of this containment, the pressure of steam and hydrogen would soon exceed the design pressure of the containment. Accordingly, the licensee would eventually be obliged to "vent" the containment – that is, to deliberately release its contents to the atmosphere.

During the Fukushima accident, TEPCO was obliged to vent the containments of the affected reactors. This venting produced two major, adverse outcomes. First, due to deficiencies in the design of the venting systems, as discussed in Section VI.4, above, destructive hydrogen explosions occurred. Second, a substantial amount of radioactive material was released with the vented steam and hydrogen, causing adverse radiological impacts at offsite locations.

Some of the radioactive material released to the atmosphere at Fukushima may have traveled through pathways other than the normal containment-venting pathways. Future investigations should identify the various release pathways and estimate the magnitude and composition of the release through each pathway. One factor to be addressed in such investigations would be the scrubbing of radioactive material passing through the suppression pools. This scrubbing may have reduced the radioactive release at Fukushima, but to a limited extent.

Given that deliberate venting of reactor containment is an expected feature of a core-damage accident, an option to reduce the radiological impact of this venting is to provide a filter in the vent pathway. The filter would be designed to retain a substantial fraction

[56] Westinghouse, 2011.

of the radioactive material released from the containment. This option is normal practice in some countries. For example, all BWR and PWR plants in Sweden and Germany have filtered venting systems.[57] Risk analysts have long advocated the addition of such systems to NPPs in the USA.[58]

If the Pilgrim NPP were equipped with an appropriately-designed system for filtered venting of its containment, the radiological impact of a core-damage accident at the plant could be substantially reduced. Such a system should, of course, be integrated with measures for hydrogen control. An appropriate design would use passive mechanisms – such as rupture disks – for its initial operation, but could also be shut off when the need for venting has passed.

Findings and implications

Based on experience at Fukushima, this report reaches a Provisional finding that filtered venting of the Pilgrim reactor containment could substantially reduce the atmospheric release of radioactive material from an accident at the Pilgrim NPP. Thus, filtered venting of containment should be considered in a re-done SAMA analysis for Pilgrim. Separate from SAMA analysis, prudent engineering principles indicate that the Pilgrim plant should be equipped with a filtered venting system that uses passive mechanisms. Also, any measures related to filtered venting should be consistent with General Implication #1.

VII. Conclusions

C1. The Fukushima accident has yielded information that is new and significant for six technical issues, selected for examination in this report, that are highly relevant to the Pilgrim license extension proceeding.

C2. Findings reached here on the six selected issues are either Conclusive or Provisional, as defined in Section I.

C3. Findings on the six issues, and the implications of these findings for the Pilgrim license extension proceeding, are illustrated in summary form in Table VI-1. Definitive statements of findings and implications appear in conclusions C4 through C9, below.

C4. On Issue #1, this report reaches a Conclusive finding based on cumulative direct experience of NPP accidents including the Fukushima accident. The finding is that the Pilgrim licensee under-estimates reactor core damage frequency by an order of magnitude. Thus, the licensee's SAMA analysis for Pilgrim should be re-done with a baseline CDF that is increased by an order of magnitude. In light of experience at Fukushima, the re-done SAMA analysis should encompass, among other SAMA options,

[57] NEA, 1998.
[58] See, for example: Beyea and von Hippel, 1982.

measures to accommodate: (i) structural damage; and (ii) station blackout, loss of service water, and/or loss of fresh water supply, occurring for multiple days. Also, in view of the high risk of a radioactive release at Pilgrim, any accident-mitigation measure or SAMA that is credited for the future licensed operation of the Pilgrim NPP should be incorporated in the plant's design basis. That implication – designated here as General Implication #1 – holds across all six issues addressed in this report.

C5. On Issue #2, this report reaches Conclusive findings based on operators' experience during the Fukushima accident and a review of the EDMGs – which were prepared by NEI – that were publicly disclosed pursuant to the Fukushima accident. The findings are as follows. First, the operators' capability to mitigate an accident at the Pilgrim NPP can be severely degraded in the accident environment. Second, NEI's newly-disclosed EDMGs are clearly inadequate to address the range of core-damage and spent-fuel-damage events that could occur at Pilgrim. Third, there is a substantial conditional probability of a spent-fuel-pool fire during a reactor accident at Pilgrim. Each of these findings supports General Implication #1. The third finding is set forth again in conclusion C8, where it is based on direct experience of pool performance at Fukushima. Conclusion C8 describes the implications of this third finding.

C6. On Issue #3, this report reaches a Conclusive finding based on operators' experience during the Fukushima accident and a review of the EDMGs that were publicly disclosed pursuant to the Fukushima accident. The finding is that NRC's excessive secrecy degrades the licensee's capability to mitigate an accident at the Pilgrim NPP. This finding supports General Implication #1. Also, this finding shows that: (i) NRC secrecy regarding the general characteristics of accident mitigation measures and the phenomena associated with spent-fuel-pool fires should cease; and (ii) NRC should sponsor open research on spent-fuel-pool fires and their mitigation.

C7. On issue #4, this report reaches a Provisional finding based on the occurrence of hydrogen explosions at Fukushima NPPs and on the reported experience of Fukushima operators with hydrogen control systems. The finding is that hydrogen explosions similar to those experienced at Fukushima could occur at the Pilgrim NPP. This finding shows that: (i) containment venting and other hydrogen control systems at Pilgrim should be substantially upgraded, and should use passive mechanisms; and (ii) all hydrogen control measures at Pilgrim should be incorporated in the plant's design basis. The latter implication is equivalent to General Implication #1 in regard to hydrogen control.

C8. On issue #5, this report reaches a Provisional finding based on direct experience at Fukushima regarding damage to spent-fuel pools and their support systems (for cooling, makeup, etc.). The finding is that there is a substantial conditional probability of a spent-fuel-pool fire during a reactor accident at Pilgrim. The same finding, reached through a different approach, is set forth in conclusion C5, above. This doubly-supported finding shows that measures to prevent a pool fire should be considered in a re-done SAMA analysis. In light of conclusion C4, above, and conclusion C12 (and supporting information) in the Thompson 2006 report, SAMA methodology shows that the Pilgrim pool should be re-equipped with low-density, open-frame racks. Separate from SAMA analysis, prudent engineering principles also indicate that the Pilgrim pool should be re-equipped with low-density, open-frame racks. Finally, the finding set forth here supports General Implication #1.

C9. On issue #6, this report reaches a Provisional finding based on the reported release of radioactive material to the atmosphere from NPPs at Fukushima. The finding is that filtered venting of the Pilgrim reactor containment could substantially reduce the atmospheric release of radioactive material from an accident at the Pilgrim NPP. This finding shows that filtered venting of containment should be considered in a re-done SAMA analysis for Pilgrim. Separate from SAMA analysis, prudent engineering principles indicate that the Pilgrim plant should be equipped with a filtered venting system that uses passive mechanisms. Also, any measures related to filtered venting should be consistent with General Implication #1.

C10. Findings in this report are consistent with findings in the Thompson 2006 report.

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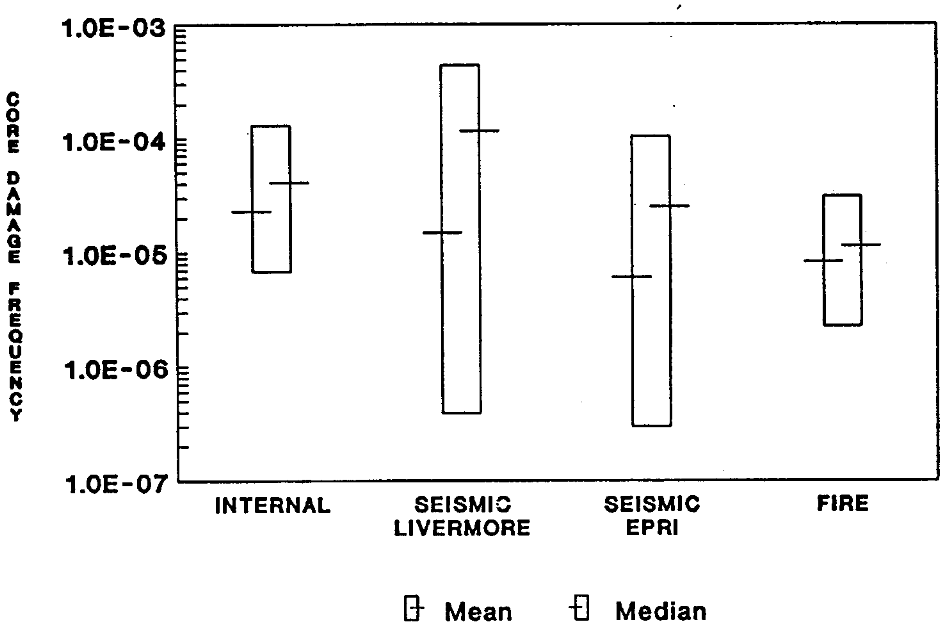
Table V-1
Selected Approaches to Protecting a Country's Critical Infrastructure From Attack by Sub-National Groups, and Some Strengths and Weaknesses of these Approaches

Approach	Strengths	Weaknesses
Offensive military operations internationally	• Could deter or prevent governments from supporting sub-national groups hostile to the Country	• Could promote growth of sub-national groups hostile to the Country, and build sympathy for these groups in some populations • Could be costly in terms of lives, money, etc.
International police cooperation within a legal framework	• Could identify and intercept potential attackers	• Implementation could be slow and/or incomplete • Requires ongoing international cooperation
Surveillance and control of the domestic population	• Could identify and intercept potential attackers	• Could destroy civil liberties, leading to political, social and economic decline
Secrecy about design and operation of infrastructure facilities	• Could prevent attackers from identifying points of vulnerability	• Could suppress a true understanding of risk • Could contribute to political, social and economic decline
Active defense of infrastructure facilities (by use of guards, guns, gates, etc.)	• Could stop attackers before they reach the target	• Requires ongoing expenditure & vigilance • May require military involvement
Robust and inherently-safe design of infrastructure facilities	• Could allow target to survive attack without damage, thereby enhancing protective deterrence • Could substitute for other protective approaches, avoiding their costs and adverse impacts • Could reduce risks from accidents & natural hazards	• Could involve higher capital costs

Table VI-1
Selected Issues Affected by New & Significant Information from Fukushima, Findings on those Issues, and Implications for the Pilgrim License Extension Proceeding: Illustrative Summary (See Section VII for definitive statements.)

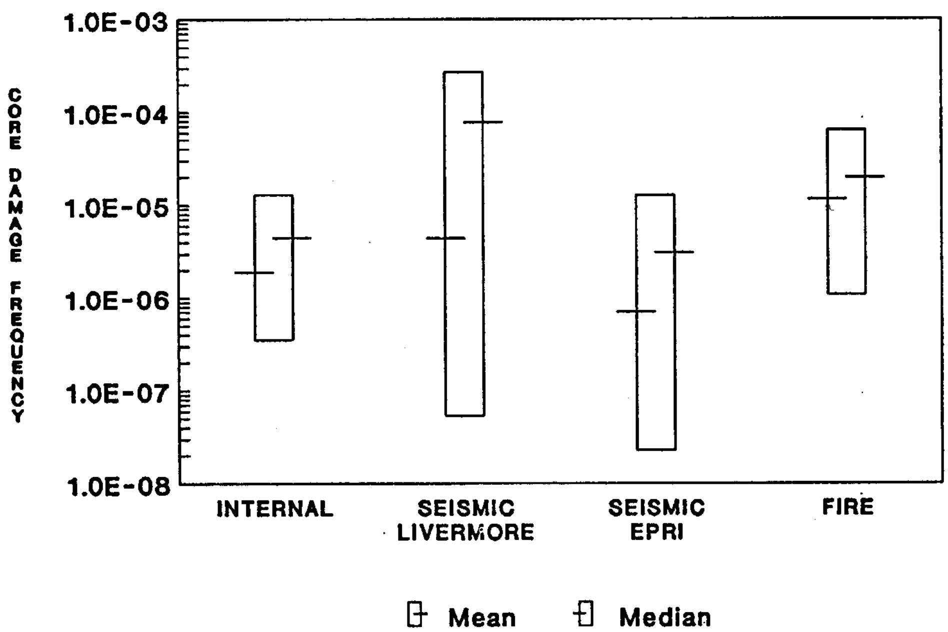
Information Issue & Finding	Implications for Pilgrim
Issue #1: Probability of reactor-core damage and radioactive release, from direct experience **Finding**: Cumulative experience shows that the Pilgrim licensee under-estimates CDF by an order of magnitude **Status of Finding**: Conclusive	• SAMA analysis should be re-done with increased baseline CDF; SAMA options to be considered should include measures to accommodate: (i) structural damage; & (ii) multi-day station blackout, loss of service water, and/or loss of fresh water supply • **General Implication #1:** Any mitigation measure or SAMA that is credited for licensed operation should be incorporated in the design basis
Issue #2: Operators' capability to mitigate an accident **Finding:** Experience shows that mitigation capability can be severely degraded in an accident environment **Status of Finding:** Conclusive	• EDMGs are inadequate • Conditional probability of a spent-fuel-pool fire is substantial (with implications as for Issue #5) • General Implication #1 as above
Issue #3: Secrecy regarding accident mitigation measures **Finding:** Experience shows that secrecy degrades accident mitigation capability **Status of Finding:** Conclusive	• Secrecy regarding accident mitigation measures and pool fires should cease • NRC should sponsor open research on spent-fuel-pool fires and their mitigation • General Implication #1 as above
Issue #4: Hydrogen control **Finding**: Hydrogen explosions similar to those experienced at Fukushima could occur at Pilgrim **Status of Finding**: Provisional	• Containment venting and other hydrogen control systems should be upgraded, and should use passive mechanisms • All hydrogen control measures should be incorporated in the design basis
Issue #5: Probability of a spent-fuel-pool fire and radioactive release, from direct experience **Finding:** Experience shows a substantial conditional probability of a pool fire during a reactor accident **Status of Finding:** Provisional	• Measures to prevent a pool fire should be considered in a re-done SAMA analysis • The Pilgrim pool should be re-equipped with low-density, open-frame racks • General Implication #1 as above
Issue #6: Filtered venting of containment **Finding:** Experience shows that filtered venting could substantially reduce radioactive release **Status of Finding:** Provisional	• Filtered venting should be considered in a re-done SAMA analysis • Pilgrim should be equipped with filtered venting that uses passive mechanisms • General Implication #1 as above

Figure VI-1
Core Damage Frequency for Accidents at a Surry PWR Nuclear Power Plant, as Estimated in the NRC Study NUREG-1150



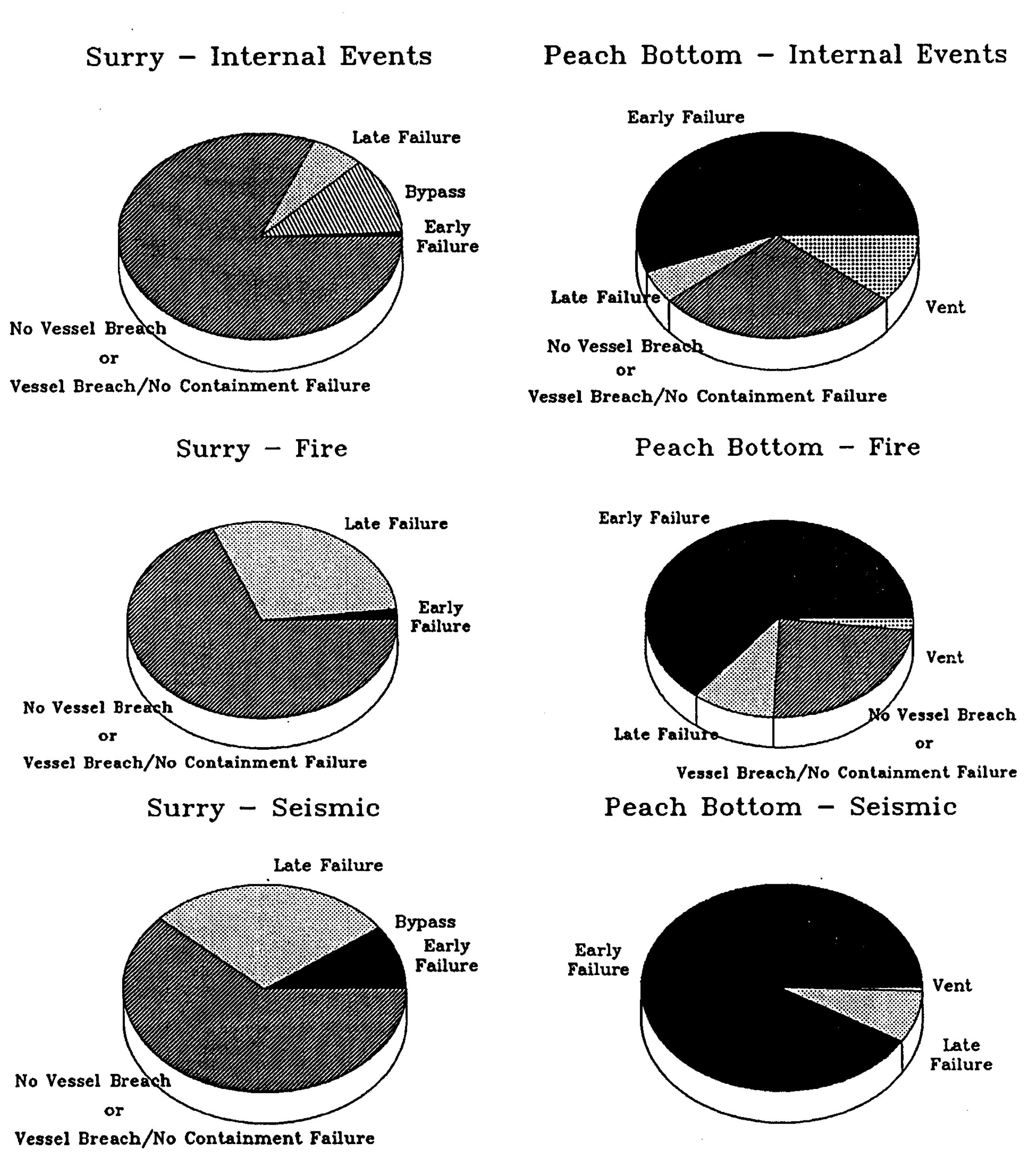
Notes:
(a) This figure is adapted from Figure 8.7 of: NRC, 1990.
(b) The bars range from the 5th percentile (lower bound) to the 95th percentile (upper bound) of the estimated core damage frequency (CDF). CDF values shown are per reactor-year (RY).
(c) Two estimates are shown for the CDF from earthquakes (seismic effects). One estimate derives from seismic predictions done at Lawrence Livermore National Laboratory (Livermore), the other from predictions done at the Electric Power Research Institute (EPRI).
(d) CDFs are not estimated for external initiating events other than earthquakes and fires.
(e) Malicious acts are not considered.

Figure VI-2
Core Damage Frequency for Accidents at a Peach Bottom BWR Nuclear Power Plant, as Estimated in the NRC Study NUREG-1150



Notes:
(a) This figure is adapted from Figure 8.8 of: NRC, 1990.
(b) The bars range from the 5th percentile (lower bound) to the 95th percentile (upper bound) of the estimated core damage frequency (CDF). CDF values shown are per reactor-year (RY).
(c) Two estimates are shown for the CDF from earthquakes (seismic effects). One estimate derives from seismic predictions done at Lawrence Livermore National Laboratory (Livermore), the other from predictions done at the Electric Power Research Institute (EPRI).
(d) CDFs are not estimated for external initiating events other than earthquakes and fires.
(e) Malicious acts are not considered.

Figure VI-3
Conditional Probability of Containment Failure Following a Core-Damage Accident at a Surry PWR or Peach Bottom BWR Nuclear Power Plant, as Estimated in the NRC Study NUREG-1150



Note:
This figure is adapted from Figure 9.5 of: NRC, 1990.

Appendix

Designing a Nuclear Power Plant to Pose a Comparatively Low Level of Risk of an Unplanned Release of Radioactive Material[59]

The most reliable option for reducing the risk of an unplanned release of radioactive material from a nuclear power plant would be to design the plant according to highly stringent criteria of safety and security. During the 1970s and 1980s, some plant vendors and other stakeholders sought to develop designs that could meet such criteria. One design approach was to provide a highly robust containment – which might be an underground cavity – to separate nuclear fuel from the environment. Another approach was to incorporate principles of "inherent" or "intrinsic" safety into the design. The two approaches could be complementary.

Underground siting

In the 1970s, there were several studies on constructing NPPs underground. Those studies are exemplified by a report published in 1972 under the auspices of the California Institute of Technology (Caltech).[60] The report identified a number of advantages of underground siting. Those advantages included highly-effective confinement of radioactive material in the event of a core-damage accident, isolation from falling objects such as aircraft, and protection against malicious acts. Based on experience with underground testing of nuclear weapons, the report concluded that an appropriately designed plant would provide essentially complete containment of the radioactive material liberated from a reactor core during a core-damage event.

The Caltech report described a preliminary design study for underground construction of an LWR power plant with a capacity of 1,000 MWe. The minimum depth of the underground cavities containing the plant components would be 150 to 200 feet. The estimated cost penalty for underground siting would be less than 10 percent of the total plant cost.

In an appendix, the Caltech report described four underground nuclear reactors that had been constructed and operated in Europe. Three of those reactors supplied steam to turbo-generators, above or below ground. The largest of those reactors and its above-ground turbo-generator made up the Chooz plant in France, which had a capacity of 270 MWe. In describing the European reactors, the report noted:[61]

> "The motivation for undergrounding the plant appears to be insurance of containment of accidentally released radioactivity and also physical protection from damage due to hostile military action."

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[59] A lengthier version of this discussion is provided in: Thompson, 2008.
[60] Watson et al, 1972.
[61] Watson et al, 1972, Appendix I.

Since the 1970s, underground siting of NPPs has been considered by various groups. For example, in 2002 a workshop was held under the auspices of the University of Illinois to discuss a proposed US-wide "supergrid". That grid would transmit electricity – via superconducting DC cables – and liquid hydrogen, which would provide cooling to the DC cables and be distributed as fuel. Much of the energy fed to the grid would be supplied by nuclear power plants, which could be constructed underground. Motives for placing those plants underground would include "reduced vulnerability to attack by nature, man or weather" and "real and perceived reduced public exposure to real or hypothetical accidents".[62]

The PIUS reactor

In the 1980s the reactor vendor ASEA-Atom developed a preliminary design for an "intrinsically safe" commercial reactor known as the Process Inherent Ultimate Safety (PIUS) reactor. An ASEA-Atom official described the company's motives for developing the reactor as follows:[63]

> "The basic designs of today's light water reactors evolved during the 1950s when there was much less emphasis on safety. Those basic designs held certain risks, and the control of those risks led to an increasing proliferation of add-on systems and equipment ending up in the present complex plant designs, the safety of which is nevertheless being questioned. Rather than to continue into this 'blind alley', it is now time to design a truly 'forgiving' light water reactor in which ultimate safety is embodied in the primary heat extraction process itself rather than achieved by add-on systems that have to be activated in emergencies. With such a design, system safety would be completely independent of operator actions and immune to malicious human intervention."

The central goal of the PIUS design was to preserve fuel integrity "under all conceivable conditions". That goal translated to a design specification of "complete protection against core melting or overheating in case of:

- any credible equipment failures;
- natural events, such as earthquakes and tornadoes;
- reasonably credible operator mistakes; and
- combinations of the above;

<u>and</u> against:

- inside sabotage by plant personnel, completely knowledgeable of reactor design (this can be considered an envelope covering all possible mistakes);

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[62] Overbye et al, 2002.
[63] Hannerz, 1983, pp 1-2.

• terrorist attacks in collaboration with insiders;
• military attack (e.g., by aircraft with 'off-the-shelf' non-nuclear weapons); and
• abandonment of the plant by the operating personnel".[64]

To meet those requirements, ASEA-Atom designed a light-water reactor – the PIUS reactor – with novel features. The reactor pressure vessel would contain sufficient water to cool the core for at least one week after reactor shut-down. Most of that water would contain dissolved boron, so that its entry into the core would inherently shut down the reactor. The borated water would not enter the core during normal operation, but would enter through inherent mechanisms during off-normal conditions. The reactor pressure vessel would be made of pre-stressed concrete with a thickness of 25 feet. That vessel could withstand an attack using 1,000-pound bombs. About two-thirds of the vessel would be below ground.

ASEA-Atom estimated that the construction cost of a four-unit PIUS station with a total capacity of 2,000 MWe would be about the same as the cost of a station equipped with two 1,000 MWe "conventional" light-water reactors. The PIUS station could be constructed more rapidly, which would offset its slightly lower thermal efficiency. Thus, the total generating cost would be about the same for the two stations. ASEA-Atom estimated (in 1983) that the first commercial PIUS plant could enter service in the early 1990s, if a market existed.[65] To date, no PIUS plant has been ordered.

PRIME reactors

In 1991, a study conducted at Oak Ridge National Laboratory examined various types of commercial nuclear reactor that were under development at the time.[66] Some types of reactor represented a comparatively small evolutionary step from existing reactors. Their safety systems tended to be simpler, and to rely more on passive mechanisms, than the safety systems of existing reactors. Other types of reactor were said to have PRIME characteristics. That acronym applied to designs with the features:

• Passive safety systems;
• Resilient safety systems;
• Inherent safety characteristics (no need for safety systems);
• Malevolence resistance; and
• Extended safety (remaining in a safe state for an extended period after an accident or attack).

The Oak Ridge study identified several types of reactor as being in the PRIME category. Those reactors, which were in various stages of development, were: the PIUS reactor; the ISER reactor being developed in Japan; the Advanced CANDU Project; modular, high-temperature, gas-cooled reactors being developed in the USA and Germany; and a

[64] Hannerz, 1983, page 3.
[65] Hannerz, 1983, pp 73-76.
[66] Forsberg and Reich, 1991.

molten-salt reactor being developed jointly by the USSR and the USA. The Oak Ridge study did not set forth a framework of indicators and criteria that could be used to assess the comparative merits of those reactors, or to determine if a reactor belonged in the PRIME category.

Design criteria for substantial reduction of risk

Table App-1 sets forth criteria for designing and siting a nuclear power plant that would pose a risk of unplanned release that is substantially lower than the risk posed by the Generation II plants that are now in use worldwide, and by the Generation III plants that vendors are currently offering. These criteria are similar to ASEA-Atom's design specification for the PIUS plant. Thus, there is evidence that the criteria set forth in Table App-1 are achievable. If ASEA-Atom's cost projections were accurate, there would be no overall cost premium for complying with such criteria.

Table App-1
Criteria for Design and Siting of an NPP that Would Pose an Unplanned-Release Risk Substantially Lower than is Posed by Generation II or III NPPs

Application of Criteria	Criteria
Safety performance of the plant during reactor operation (design-basis criteria)	<u>No significant damage of the reactor core or adjacent stored spent fuel in the event of</u>: • Loss of all electrical power (AC & DC), compressed air, other power sources, fresh water supply, and normal heat sinks for an extended period (e.g., 1 week); • Abandonment of the plant by operating personnel for an extended period (e.g., 1 week); • Takeover of the plant by hostile, knowledgeable persons who are equipped with specified explosive devices, for a specified period (e.g., 8 hours); • Military attack by specified means (e.g., 1,000-pound air-dropped bombs); • An extreme, specified earthquake; • Conceivable erroneous operator actions that could be accomplished in a specified period (e.g., 8 hours); or • Any combination of the above.
Safety performance of the plant during reactor refueling (design-basis criteria)	<u>A specified maximum release of radioactive material to the accessible environment in the event of</u>: • Loss of reactor coolant at a specified time after reactor shut-down, with replacement of the coolant by fluid (e.g., air, steam, or unborated water) creating the chemical and nuclear reactivity that would maximize the release of radioactive material, at a time when the plant's containment is most compromised; and • Any combination of the events specified above, in the context of reactor operation.
Site specification (radiological-impact criteria)	<u>In the event of the maximum release of radioactive material specified above, in the context of reactor refueling, radiological impacts would not exceed specified values regarding</u>: • Individual dose; • Population dose; and • Land areas in various usage categories that would be contaminated above specified levels.

Notes:
(a) The criteria in the first two rows of this table would apply to the reactor core and to spent fuel stored adjacent to the core. Separate criteria would apply to an independent facility for storing spent fuel, whether onsite or offsite.
(b) For a more detailed discussion, see: Thompson, 2008, Section 4.3.