COMMENTS ON THE FUKUSHIMA ACCIDENT

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Executive Summary

This report offers some comments on the Fukushima accident and the offsite responses to it. The Fukushima accident is unique in that it presents an opportunity to appreciably advance the state of knowledge of severe nuclear accidents. There has never been three similar nuclear power plants at the same site undergoing simultaneous core melt events caused by the same initiating event. This permits meaningful comparisons.

The main observations are:

(1) The offsite radiological health consequences from the Fukushima accident are and will be very small, are limited to a small area near the site, will only affect a portion of the general population in that small area, and will not affect future generations in a measurable way.

(2) The the non-radiological health consequences of long term evacuation and sheltering are serious and far outweigh the radiological health consequences. This observation calls for a modernization of the nuclear power plant emergency response process to better balance radiological and non-radiological consequences. Suggestions on how to modernize emergency planning are provided.

(3) Had this multiplant accident occurred at an average site in the United States, the offsite economic losses would likely be covered by the protection afforded by the Price-Anderson Act.

(4) Further analyses and plant inspections are needed to continue to develop severe accident computer codes, to refine severe accident management processes, and to minimize the possibility of turning a comparatively low consequence accident into a higher consequence accident.

(5) The three damaged plants at Fukushima were all station blackout sequences, but had different accident scenarios, largely because of the different timing and amounts of core cooling, often by sea water injection. Since the overall radiological consequences were very small, each of these somewhat different station blackout sequences itself had very limited offsite radiological consequences. This is supportive of the notion that these nuclear power plants acted like large, passive (no elec-

tricity or operator actions) filters during these different station blackout core damage scenarios.

(6) Station blackout sequences can be the result of internal fires or internal flooding or the result of large external natural events like flooding, tornadoes, earthquakes, and tsunamis. In theory, terrorist attacks on nuclear plants might precipitate station blackout conditions. So the lessons learned from Fukushima could be relevant to a broader spectrum of station blackout situations in Mark I and other nuclear plants. It is expected that other initiators of long term station blackout conditions in other nuclear plants would have their own initiator specific accident scenarios, but would still end up with low offsite radiological consequences because of the inherent passive source term reduction capabilities of water cooled reactors in general and Mark I plants in particular, provided catastrophic failure of the containment is avoided.

In order to address the concern that smaller consequence events like those at Fukushima might evolve into larger release events, this review of the Fukushima accidents concentrated on two related questions: (A) Were there some previously unidentified important phenomena that need to be considered, such as water hammer effects, that might lead to a catastrophic failure of the containment? and (B) Are there a set of accident conditions wherein plant personnel should shift most of their efforts from trying to limit core damage to trying to prevent catastrophic failures of the containment?

Two other subjects are covered by this report - the need for further efforts to protect spent fuel pools and the value of an additional drywell filtered vent system:

(1) With regard to protecting the spent fuel pools, this subject was examined by an earlier National Academy of Sciences Committee. The author provided reports and oral testimony to this earlier NAS Committee. The author draws upon this earlier NAS experience, improvements made by nuclear plant operators on protecting spent fuel pools in the interim time period, and experiences from Fukushima accident. It is concluded that the decisions reached by the earlier NAS committee are sound, the improvements that have since been made to protect spent fuel are valuable, and that no further changes are necessary.

(2) The benefits of installing another filter connected to the drywell have been evaluated. It is concluded that such a plant modification, in addition to being expensive, offers no significant radiological health effects benefit over what already exists at Mark I nuclear plants. Such a plant modification can not be made operational

quickly, whereas expanding the use of existing plant equipment to cope with severe accidents can be implemented quickly. As explained later, there may be situations which could turn Fukushima-like accidents into ones with more serious consequences. An additional filtered vent in the drywell would not be very useful in preventing or mitigating these more serious accidents. What is shown is that minor modifications to existing plant equipment could result in safety improvements greater than what might be accomplished by the addition of a new filter system in the drywell.

Most importantly, dwelling on unnecessary plant modifications could be a distraction. Even if there were only one additional fatality at Fukushima due to non-radiological causes, it would exceed the number of early fatalities from radiological sources. With a new report identifying over 1100 additional fatalities due to non-radiological health consequences at Fukushima because of over-evacuation, this health effect should be the primary focus of our attention. Additionally, preventing small consequence accidents like the one at Fukushima from turning into more serious accidents is far more important than adding another filter system to the drywell.

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1.0 Acknowledgements

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2.0 Introduction

The Fukushima accident is unique in that it presents an opportunity to appreciably advance the state of knowledge of severe nuclear accidents. There have never been three nuclear power plants at the same site undergoing simultaneous core melt events caused by the same initiating event. All three plants are boiling water reactors and all have Mark I containments. These plants are all based on a similar General Electric primary system design.

The accident sequences at these plants were both similar and different for a number of reasons. This permits meaningful comparisons: How were their responses similar and how were they different, and why? For example, if external water was injected in to the pressure vessel in one of these plants and not into the pressure vessel of another plant or injected much later, how did this affect the courses of the accidents at these two plants? The ability to make meaningful plant-to-plant comparisons is now greatly enhanced by advanced severe accident computer codes, such as Sandia's MELCOR code and EPRI's MAAP code, both of which have been applied to analyzing the Fukushima accident.

Analyses of the three Fukushima plants that experienced core melt sequences are particularly valuable in that they were all long term station blackout sequences, perhaps the most challenging of all severe nuclear accidents. Long term station blackout sequences can be the result of internal fires or internal flooding or the result of large external natural events like flooding, tornadoes, earthquakes, and tsunamis. In theory, terrorist attacks on nuclear plants might precipitate station long term blackout conditions. So the lessons learned from Fukushima could be relevant to a broad spectrum of long term station blackout situations.

Even though the accident sequences at the three Fukushima plants differed, the overall or cumulative offsite radiological health effects are very small. This implies that each plant had an accident sequence which resulted in very small offsite radiological health effects. As discussed later, the author has classified such small consequence accidents as leakage type accidents. Therefore the accidents at Fukushima represent three members of a class of leakage type (or small consequence) events.

In spite of different accident scenarios and somewhat different design features, all three damaged plants at Fukushima acted like large passive filters that significantly limited the release of radioactive material to the environment. While successful in limiting offsite radiological effects in a passive manner (no operator actions or electric power were needed), the Fukushima designs did not prevent highly destructive explosions. Preventing destructive explosions in a passive manner can be readily achieved at very low cost.

What also makes the Fukushima accident especially informative (1) is the close cooperation by the Japanese government and by TEPCO, the utility that operated these plants, with the many organizations that are analyzing these accidents.

This report, in Section 3, provides the reader with information on both risks and consequences. Risks are shown to be exceedingly small. As to consequences, Section 3 includes present and projected offsite radiological health consequences. Significant studies (2) have now been conducted, such as the study recently completed by the World Health Organization (WHO), to determine present and projected health effects caused by exposure to radioactive material released into the environment at Fukushima. Radiological health consequences are very small and highly localized. Those radiological health effects that have threshold limits, like early (or acute) fatalities, are thought to be zero. Additionally, some non-radiological health consequences are provided in Section 3, such as the effects of long term sheltering. Recent information (3) indicates that these non-radiological consequences are serious and far in excess of the radiological consequences. There are ways to reduce non-radiological consequences, principally through improved emergency responses and possibly through examining and applying improved criteria that establish long term re-occupation dose limits.

Lastly, Section 3 includes an estimate of the offsite economic consequences of contaminated land, had this accident happened in the United States. These economic consequences are then compared to the coverage afforded by the Price Anderson Act.

Section 4 includes a description of four postulated Mark I containment failure modes. These four postulated containment failure modes are then subjectively ranked by the author in terms of their potential offsite radiological consequences. This ranking of postulated containment failure modes extends from very limited releases of radioactive material to the environment from containment leakage type accident sequences to much more severe postulated accidents where there is a catastrophic failure of the containment. It is suggested that the Fukushima accidents be placed in the leakage category in this ranking scheme.

Section 5 examines a hypothetical situation which might cause a low consequence leakage type accident to evolve into a much more severe accident, a type D containment failure of the torus as described in Section 4. Analyses performed by Sandia Laboratories show that most of the CsI would end up in the suppression pool within a comparatively short time after a core damage sequence began. Therefore it is essential that the integrity of the torus be maintained so that cesium releases are kept to very small values. A review of the accident time line and supporting reports for the 1F2 plant was conducted to determine if there were possible phenomena occurring, such as water hammers, that might lead to type D containment failures with much larger consequences. Water hammers might create large forces which could lead to a tear in the torus. Practical solutions are suggested to prevent the evolution of leakage type accidents into more serious ones.

Section 5 also addresses what might be done to significantly reduce offsite non-radiological consequences.

Section 6 examines the proposal to modify Mark I containments with an additional drywell filtered vent system and concludes that this type of plant modification is unwarranted and impractical. More importantly, dwelling on such a plant modification may be a distraction from examining other more consequence-significant potential containment failure modes, from examining the need to remove the CsI from the suppression pool before there is torus leakage, and from examining the need to reduce non-radiological consequences.

Section 7 discusses the lessons learned from Fukushima about spent fuel pools. Section 8 compiles a number of recommended tasks. Section 9 contains Appendix A which provides an estimate of offsite economic losses for an average site in the United States and compares this to the coverage afforded by the Price-Anderson Act. Section 10 is a list of references and acronyms that were used in generating this report.

3.0 Risks and Consequences

Nuclear power risks are orders of magnitude smaller than the NRC's Quantitative Health Safety Goals. These NRC health safety goals were set at keeping nuclear power risk levels small compared to the everyday risks to which people are subjected. Numerically, these goals were to keep nuclear power early and latent fatality risks below one part in a thousand compared to normal background risk levels. Nuclear risks are extremely small because of both the low frequency of nuclear accidents that lead to releases of radioactive material into the environment and also because the amount of released radioactive material, and therefore the health consequences, is quite limited, far less than what was believed before. Because risks are so very small, the bulk of this report concentrates on consequences.

To keep discussions of consequences from nuclear accidents in perspective, the same large natural events that might cause an accident sequence to be initiated at a nuclear power plant would likely themselves cause far greater health and economic consequences than the affected plant(s). Fukushima is a case in point. The enormous earthquake and tsunamis that damaged the Fukushima plants had far greater economic and health effects than the meltdowns at Fukushima.

As shown in Table 1, the observed and projected health consequences, based on the World Health Organization's report, due to radiological exposure from the Fukushima accident are zero for threshold type consequences and inter-generational consequences (e.g., congenital defects) or are small increases in a selected part of the population (children under one year of age) and only from a selected location (the most contaminated area in the Fukushima Prefecture). Additionally, the WHO calculated consequences were based on conservative analyses. For Fukushima, the largest health effect is not be caused by radiation, but by the dislocation of people from their homes, neighborhoods, and family histories and for some, their placement in long term shelters. The fear of radiation is a larger health risk than the effects of the actual radiation itself.

Offsite economic losses, if such a release were to occur within the USA, would likely be covered by the Price-Anderson Act.

Table 1: Risks and Consequences

ITEM	Quantification	Can practical improvements be made?	Comment
A. Risks **(NRC Safety Goal)**	Early and latent fatality risks from a nuclear power plant should be less than 1/1000 of normal early fatality and latent fatality background risks.	No	Nuclear power risks already are orders of magnitude below the NRC's early and latent fatality safety goals (less than 1/1000 of background risks). Improvements are not necessary to meet safety goals and would only produce a tiny incremental benefit.
B. Radiological consequences			

ITEM	Quantification	Can practical improvements be made?	Comment
B.1 Early Fatalities - anywhere	~Zero (WHO)	No	There are no early fatalities if the average of the iodine + tellurium + cesium release is less than 5% of core inventory or if evacuation of innermost one mile begins at or before the start of the release. WHO reports that radiation doses were well below levels needed for deterministic effects (e.g. early fatalities and early injuries).
B.2 General population inside and outside of Japan	No observable increases in cancer rates above base-line rates anticipated. (WHO)	Yes	If cesium releases can be further reduced this would reduce projected long term health conse-quences. These projected consequences are already unob-servable from a statistical point of view and would still remain statis-tically unobservable with improvements.

ITEM	Quantification	Can practical improvements be made?	Comment
B.3 In the Fukushima Prefecture: Fetal development, outcome of pregnancies, spontaneous abortion, miscarriage, prenatal mortality, congenital defects, cognitive impairment	~ zero (WHO)	No	Estimated dose levels too low to cause these health effects. (WHO)
B.4 In the most contaminated location in the Fukushima Prefecture. Estimated increase in infants less than one year old.	See WHO numbers in a,b,c, and d, below.	yes	Consequences would be lower for people older than one year. Estimated cancers listed below are conservatively derived and therefore should be upper bounds.

ITEM	Quantification	Can practical improvements be made?	Comment
B.4 a. All solid cancers b.Breast cancer c.Leukaemia d.Thyroid cancer	a. ~4% in females b~ 6% in females c.~ 7% in males d. up to 70% in females over their lifetime	yes	a., b., and c. Reduce iodine and cesium releases. For d., reduce iodine releases. d.Present rate over lifetime is 0.75%, with increase due to radiological exposure, 0.75% + 0.50% = 1.25%
C. Non-radiological consequences			
Long term evacuation and/or sheltering of evacuees	Over 1100 fatalities reported. Far in excess of the radiological consequences.	Yes	Smaller cesium 137 releases, improved emergency responses, improved reoccupation levels. Investigate improved decontamination techniques.
Economic losses	Likely within coverage afforded by the Price-Anderson Act.	Yes	Reduce contaminated land by smaller cesium releases, improved reoccupation levels. Investigate improved decontamination techniques.

4.0 Postulated Containment Failure Modes

Four different postulated failure modes for Mark I nuclear plants are identified in Table 2. These failure modes are then subjectively radiologically ranked by the author in terms of potential health consequences. The least significant failure modes, from a health consequence point of view, are thought to be leakage type of containment failures. The author has placed the accidents at Fukushima in this classification. These rankings reflect consequences before recommended improvements are made which would further lower estimated consequences.

Table 2: Postulated Containment Failure Modes From Accidents

Postulated containment failure mode	**Comments**
A. Leakage-Containment overpressure, (Fukushima) Subjective ranking 1.0	Leakage type of nuclear accidents have very small source terms because the suppression pool captures almost all of the radionuclides that enter the pool, even with a closed wetwell hardened vent line. Additional source term reduction processes take place in the drywell by deposition processes and the reactor pressure vessel retains some of the CsI. These passive source term reduction processes are further backed up by use of the active drywell spray line. Spray droplets would "scrub" radioactive compounds out of the drywell gas space. Figure 1, reproduced from a Sandia MELCOR analysis of the 1F1 reactor at Fukushima (4), shows the limited amount of CsI in the drywell. Note that the amount of CsI calculated to be in the drywell includes both the airborne CsI and the CsI removed by deposition. Only the airborne fraction would be available for release though drywell leakage. Figure 1 is not based on the use of the active drywell spray system, but relies only on passive filtering processes (the SRVs would still be needed). In Mark I containments large source reductions is not dependent on operating the wetwell hardened vent system. The main purposes for opening the wetwell hardened vent line would be to prevent containment overpressure failure, to purge the containment of explosive gases, and to provide more pressure margin should there be a pressure spike if the reactor bottom head fails and corium falls into the water on the drywell floor. However, a timely opening of the wetwell hardened vent line might force airborne CsI in the drywell to flow into the suppression pool wherein it would be captured. If wetwell hardened vent line flow is delayed or if the drywell sprays are used before this flow begins, then there would be very little airborne CsI left in the drywell and venting would not be an important source term reduction process.

B. Leakage-Drywell Liner Failure, (Fukushima?) Subjective ranking 1.5 to 3.0	It is expected that only very limited amounts of CsI that might be released to the environment at the onset of a drywell liner failure. Source term reduction processes in the suppression pool would capture almost all of the radionuclides that entered pool via the SRV lines. Further CsI reduction would take place in the drywell due to deposition processes. The release of volatile CsI from the reactor core would always precede reactor vessel lower head failure, with possible liner melt through occurring after that. For example, the Sandia analysis of 1F3 (5) shows the onset of core melting around 42 hours after the station blackout sequence was initiated. By about 48 to 49 hours the Cs had relocated out of the heated reactor core with only about a few tenths of a percent of the initial cesium inventory available for release to the environment. Even as late as 96 hours in this Sandia analysis the reactor pressure vessel bottom had not been calculated to fail. This suggests that should there be later drywell liner failure, the radiological source term would be very small at the onset of drywell liner failure. The MELCOR analysis of 1F1 shown in Figure 1, below, extends out to 30 hours. During that time period, between about 13 to 15 hours, the bottom head of the reactor pressure vessel is calculated to fail. About 8-9% of the original cesium is sequestered in the reactor pressure vessel according to MELCOR. This sequestered cesium is not calculated to become airborne in the drywell even though calculated core melt progressions are sufficient to cause the reactor vessel bottom head to fail. However, core-concrete interactions produce significant quantities of hydrogen and carbon monoxide, both of which are explosive. These explosive gases would be added to the hydrogen generated by zirconium-steam interactions in the reactor vessel. Therefore the major concern about drywell liner failure could be explosions in the nearby reactor building. TEPCO is now using probes to investigate the condition of the drywell liners.

C. Containment becomes subatomspheric, implodes Subjective ranking 15	Mark I containments have significant pressure retention capabilities. Even with accident pressures around +100 psia or more at Fukushima, there were no catastrophic containment failures, just leakage. However, these containments can only withstand a negative pressure of around -2 psid. More negative subatmospheric conditions may lead to crumpling of the containment. Containment sprays in the drywell and torus gas space can both lower containment pressures and scrub radionuclides out of the containment atmosphere. Mark I plants use vacuum breakers to let in outside air should the containment pressure begin to go negative. With the wetwell vent closed, non-condensable gases would not be released to the environment. Actuating the containment sprays should not cause a subatmospheric problem because of these non-condensable gases. Containment pressures should gradually return to near atmospheric levels. Subatmospheric conditions should not occur as the vacuum breakers permit outside air to enter the drywell. Under long term station blackout conditions the containment would have to be vented to prevent overpressurization and to purge the combustible gases. Venting would also cause the release of most of the non-condensable gases. If venting is done gradually the containment should not become subatmospheric because water at saturation temperatures on the floor and on many other surfaces in the containment would vaporize, resulting in a slow depressurization. Once drywell pressures began to dip below atmospheric pressures, outside air should begin to flow into the drywell through the drywell-to-outside vacuum breakers. However, actuation of the containment sprays might depressurize the containment rapidly. Too high a spray rate might result in subatmospheric conditions that might begin to crumple the containment.

D.Tear at bottom of torus Subjective ranking 10	*1*If a tear develops in the bottom of the torus, pool water would *0*drain onto the floor below. If the tear was below the submergence level of the SRV line or below the multiple downcomers that are used during loss of coolant accidents, this would defeat the safety effectiveness of the suppression pool. This could create a direct pathway to the environment for both radionuclides and explosive gases. Less serious would be opening the torus boundary above the water line. A longer term issue that must be addressed is the clean-up of the CsI from the suppression pool. The integrity of the torus may degrade over time, especially if there is seawater in it. This could lead to pool water containing CsI leaking into the torus room.

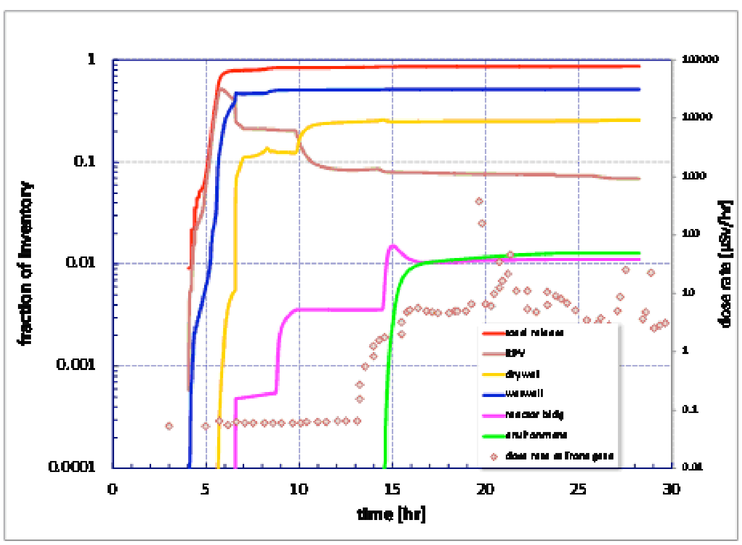
Containment failure modes A and B in Table 2 were given low subjective consequence rankings. In support of these rankings, consider Figure 1, below. This figure is a reproduction of an analysis performed by Sandia's MELCOR computer code for the Fukushima 1F1 nuclear plant. The top line represents the sum of all the spatial distributions of CsI over time plotted below this line, i.e., it is the total CsI fraction of the reactor core's initial CsI inventory. Directly below the top line is the CsI that ends up in the suppression pool. The important observation here is that the suppression pool is the dominant location where the CsI is trapped. Further, trapping the CsI in the pool occurs quite early in this accident sequence. Subsequent losses of containment integrity, should they occur, such as drywell liner failure or drywell leakage from overpressure conditions, are automatically limited in the sizes of their CsI source terms. This is because, by the time they occur, the bulk of the CsI is already in the suppression pool. The third line from the top is the amount of CsI in the drywell. However, not all of the CsI in the drywell is airborne. Some of it is deposited on surfaces or otherwise removed from entering the reactor building or the environment. Line four from the top is the CsI remaining in the reactor vessel. This CsI also is not available for release to the environment. Further down on this logarithmic scale is the CsI available to be released to the environment, about one percent of the initial inventory of CsI.

Figure 1 is based on a situation where the wetwell hardened vent isolation valves are closed. Even though the venting process did not occur, the majority of the CsI was still captured in the suppression pool and removed by deposition in the drywell. An additional MELCOR analysis of this same sequence, but with the wetwell hardened vent isolation valves open at some point, would likely show that some of the airborne CsI in the drywell would end up in the suppression pool. Further, the leakage path to the reactor building would have been closed off once venting of the drywell dropped pressures below the onset of leakage pressure. In this case, if the hardened vent isolation valves were open before about 8 hours, the small release to the environment from 1F1 at Fukushima would have been even smaller.

Other analyses have been performed by Sandia on the 1F3 accident sequence. Here core melt damage was much delayed compared to that at 1F1. Nonetheless, the amount of CsI available for release was also small.

Based on these Sandia analyses the airborne CsI source terms would already be small by the time the containment integrity is lost through either overpressure or drywell liner failure. Hence, the low subjective ranking.

Figure 1: Sandia Analysis of Locations of CsI versus Time, 1F1 Nuclear Plant



5.0 Reducing Health Consequences

5.1 Reducing radiological health consequences

5.1.1 Introduction

Section 4 identified four different potential containment failure modes for Mark I containments. In this section each of these four different containment failure modes is examined in greater depth and suggestions are offered to reduce the likelihood of these failure modes. Emphasis is placed on the use of existing plant equipment to accomplish these improvements.

There are benefits to improving the safety of nuclear power plants by using existing plant equipment, when possible. The plant operators should already be familiar with this equipment. The time to make such equipment operational is short because the equipment is already on site. This timeliness benefit also applies to its use during an accident since equipment at hand can be activated more quickly than bringing in safety equipment from offsite locations. Finally, expanding the use of existing plant equipment or improving operator responses provides safety benefits at minimal costs.

5.1.2 Type A failure: containment leakage because of overpressure

Figure 1 shows that the CsI source term released to the environment was very small because the CsI removal in the drywell by deposition and CsI removal in the suppression pool were very effective in capturing this compound. Source term reduction went on even though the wetwell hardened vent was closed for at least a day. This, then, means that the main purpose of the wetwell hardened vent is to prevent the containment from being overpressurized, leading to the creation of a leakage pathway to the reactor building and to prevent explosions. If the containment overpressure leakage pathway is prevented, the flow of explosive gases from the drywell into the reactor building along this pathway is also prevented. Accident recovery would have been much simpler without such explosions and plant personnel would have been better protected.

Venting harmlessly expels the explosive gases to the environment through an elevated release point and simultaneously lowers the drywell pressure which would eliminate the creation of a leakage pathway. The lower drywell pressure also provides an increased capability to handle a pressure spike should corium from a failing pressure vessel lower head fall into the water on the drywell floor. Lower containment pressures may permit core cooling with HPCI and RCIC to continue to operate instead of failing because of too high a backpressure.

The Fukushima wetwell hardened vents could not be opened in a timely manner because isolation valves were in the closed position and during a blackout condition there was no motive power (electricity or compressed air) to open these isolation valves. By the time these isolation valves were opened the containments were already overpressurized and leaking steam, hydrogen, and some CsI into the reactor building. Venting decreased the drywell pressure which then caused the leakage path to close. Opening the wetwell vents likely caused the explosions in the reactor buildings. This is postulated because as long as there was steam leakage into the reactor buildings there were several mechanisms to prevent hydrogen explosions. First, this leakage of steam, hydrogen and some CsI displaced oxygen in the reactor building. Second, the steam served to steam-inert the hydrogen/air mix. However, once the drywell-to-reactor building leakage stopped there was no new steam addition to inert the gas mix in the reactor building as the existing steam in the reactor building condensed on cooler surfaces. Without drywell leakage, oxygen would have flowed back into the reactor building making an explosive mixture. It is observed that there were two explosions in different reactor buildings, one at about 25 hours and the other at about 68 hours. Both explosions occurred about one hour after the venting process lowered the drywell pressure, causing the drywell flange area to reseal itself. This ended the flow of steam into the reactor building. With no new steam entering the reactor building, steam inerting of the hydrogen in the reactor building ended and this led to explosions.

There are several ways to overcome the inability to open the wetwell hardened vent line. One way is to have a dedicated power source to open these normally closed isolation valves in the wetwell hardened vent system. Some Mark I plants in the United States made this safety improvement years ago. Another approach is to change the position of the "normally closed" vent isolation valves to "normally open". Under this arrangement when the containment pressure rises beyond the failure pressure of the rupture discs downstream of these isolation valves, these rupture discs will burst and venting will begin. This "normally open" isolation valve arrangement is completely passive since it does not require any motive power or operator actions to initiate venting. Had the position of the isolation valves at Fukushima been changed to "normally open" there would not have been any explosions due to hydrogen from steam-zirconium reactions and even less CsI would have entered the environment. Such a simple valve position plant modification can be made very quickly at little to no cost.

Further improvements can be made to the wetwell hardened vent system. A source of motive power (electricity, compressed air) can be added or may even be available through the scope of equipment in the FLEX system. If such equipment were available it would still be advisable to place the vent isolation valves into a “normally open” position, and use the motive power to close/reopen these isolation valves when desired. With this arrangement of first passive operation, then active operation, a delay in providing motive power is not serious. As mentioned before, some Mark I plants already have dedicated power supplies to open and close the hardened vent isolation valves.

Some hardened vent designs, such as at Fukushima, have a further refinement. Here the two rupture discs are placed into two parallel vent lines, with one vent line smaller than the other. It is assumed that the rupture disc in the smaller vent line has a lower bursting pressure than the disc in the larger line. This arrangement would passively stage the flow down the vent line. The larger line, with the higher bursting pressure of the rupture disc within it, would only open if the venting provided by the smaller line proved to be insufficient. This parallel piping arrangement exists in a spool piece within the vent line that houses the two rupture discs. Elsewhere the vent line is a single pipe large enough to handle the flow if both rupture discs had burst open.

The rupture discs should have a bursting pressure higher than the calculated peak pressure from a loss-of-coolant accident, but below the pressure that would cause the containment to leak. Choosing an optimum rupture disc failure pressure between the above two pressure boundaries may require some trade-offs. Lower rupture disc bursting pressures may be beneficial to HPCI and RCIC so that they are less likely to fail on too high a containment back pressure. Lower rupture disc failure pressures provides an earlier ability tolerate pressure spikes that might occur upon failure of the reactor lower pressure vessel head. However, lower rupture disc failure pressures means that the time to reach saturation pressures in the suppression pool is reached sooner. Those areas in the suppression pool that are at or near the saturation point have decreased ability to capture the CsI injected into the pool when steam flows down the SRV line. Reaching the pool saturation temperature sooner may shorten the time when pool water hammer events become possible, as discussed later. If a dedicated source of isolation valve motive power is operable this could result in a series of hardened vent isolation valve openings and closings and this possibility could affect the determination of the optimum rupture disc failure pressure. In terms of improving the operation of the hardened vent isolation valves so that explosions like those at Fukushima would be avoided, the best arrangement seems to be to have these isolation valves in the normally open position and to have a source of motive power that would be able to open and close these valves during extended station blackout conditions.

In addition to having a passive capability to initiate hardened vent flow passively, the Mark I designs have a passive capability to end vent flow. This is because all pipes that might carry steam from a drywell location into the torus area are submerged in the pool water. Unless there is enough pressure in these submerged lines to clear the water out of the lines, there will not be any steam flow. For every two feet of submergence about one psi pressure difference is required to begin to have vent line flow. Thus, the drywell to wetwell vent lines, which have downcomers with a submergence of four feet, will not experience any flow through them if the differential pressure is less than +2 psid. So even if the hardened vent line's isolation valves are never returned to the closed position, the remaining gases inside the containment will be isolated from the outside environment whenever the containment pressure drops below the pressure needed to overcome the lines' submergence in the pool. Therefore the suppression pool acts passively to stop flow through the hardened vent line when driving pressure drops below that needed to overcome the hydrostatic pressure of the submergence.

5.1.3 Type B failure: containment leakage because of drywell liner failure

As discussed in Table 2, part B, by the time there might be a drywell liner failure, the amount of airborne CsI in the drywell should be very small, particularly if the drywell sprays have been actuated.

There are three possible benefits to using drywell containment sprays. These sprays scrub compounds of iodine and cesium out of the drywell gas space and put them into solution, likely in a pool of water on the drywell floor. Sprays can also reduce containment pressure, thereby making containment overpressure less likely. Finally, sprays can help build an inventory of water on the drywell floor. This layer of water may prevent drywell liner failure by freezing the flow of "corium" from a failed pressure vessel lower head and also may inhibit core-concrete interactions and the explosive gases such interactions can produce. At Fukushima there were ample supplies of water to fill the drywell up to the drywell-wetwell vent lines. This would require about 66,000 liters of water. The condensate storage tank at Unit 3 had roughly 1,650,000 liters of clean water and there were a few such storage tanks on site. The author does not know if these storage tanks survived the earthquake and tsunamis.

Whether or not a layer of water on the drywell floor is sufficient to prevent liner failure depends on the calculated corium temperature and pour rate from the reactor vessel lower head. Low-temperature slow-pour conditions are more likely to prevent drywell liner failure than high-temperature fast-pour rates. Different severe core damage computer models predict different temperature/pour rates. Further analyses need to be done to determine required water depth versus pour conditions to prevent drywell liner failure. The amount of water that needs to be sprayed into the drywell to create a layer of water on the drywell floor of a specified thickness may be estimated from ex-vessel quenching analyses (6). The time by which this layer of water should be in place can be determined from MELCOR or MAAP analyses which calculate the time it takes to reach reactor pressure vessel lower head failure. The fire protection system at these plants could be used to pump the drywell spray water.

It is not clear if any of the drywell liners failed at Fukushima. TEPCO is now using a probe to examine these liners and the results of this investigation will be useful. Some additional insights about drywell liner failure may be possible by comparing the calculated time for the lower head of the reactor pressure vessel to fail against measurements of drywell pressure over time. If the calculated reactor pressure vessel lower head failure time occurs before there is any indication of decrease in containment pressure due to leakage, then no liner failure would have occurred up until the time at which leakage begins.

If further investigation indicates that additional actions need to be taken to prevent drywell liner failure, then installing a curb in front of the liner could be useful. Additionally, it may be worthwhile to preemptively create a layer of water on the drywell floor when going back to power. The practicality of this preemptive layer of water has not been investigated.

The remaining issue is to avoid a drywell spray rate that might cause the containment pressure to drop below -2 psid. The acceptable spray rate depends on whether or not the drywell has been vented. As discussed in Table 2, Part C, as long as sufficient non-condensables are in the containment, there is no concern about becoming subatmospheric.

Note that the containment pressure would be elevated by the time the sprays were turned on, reducing the drywell spray system flow rate. This reduced spray rate may not be sufficient to prevent overpressurization if the wetwell hardened vent system is closed, particularly if the sprays are activated when the containment pressure is already high. The sufficiency of the drywell spray system to prevent containment overpressure as a function of the spray start time could be investigated by running the MELCOR or MAAP computer program.

Conversely, the drywell spray system might be initiated after the wetwell hardened vent line <u>was opened</u>. Under these conditions much of the non-condensable gases would have been vented, leaving the containment pressure near atmospheric conditions.After venting a portion of the gases, the drywell total pressure would mostly be that due to the partial pressure of steam. Initiating the drywell spray under these conditions puts in play a race between the depressurization caused by spray initiation and the inflow of outside non-condensable air through the outside vacuum breakers. If the spray depressurization rate exceeds the repressrization rate from the incoming air, an unacceptable subatmospheric condition may occur.

Once again, the MELCOR or MAAP codes could be used to calculate the maximum drywell spray rate to prevent unacceptable subatmospheric conditions during the already opened wetwell hardened vent system. See Section 5.1.5 for more information.

5.1.4 Type C failure: implosion of containment

Section 5.1.3 discussed the possibility of crumpling the containment if excessive subatmospheric conditions are reached after the containment has been vented and after the drywell sprays have been turned on. If sprays are turned on prior to venting excessive subatmospheric conditions would be avoided. If the containment has already been vented when the sprays are actuated, becoming excessively subatmospheric can be avoided by either limiting the spray rate or by delaying the initiation of the sprays until the drywell-to-outside air vacuum breakers have opened and restored a sufficient amount of non-condensable air to the containment, provided this does not interfere with preventing drywell liner failure. As a further precaution, it may also be possible to use other existing plant equipment to prevent unacceptable subatmospheric conditions. If the existing drywell vent is opened at the time of spray initiation, there should be air that backflows into the drywell once the containment begins to become subatmospheric. The use of this drywell vent would supplement the existing vacuum breakers that open to bring in air from outside to the drywell. This drywell vent would again be closed once the sprays are turned off.

It is noted that the Fukushima containments did not crumple after they were vented. This implies that the vacuum breakers that are designed to return outside air (noncondensables) to the containment performed adequately. Untested is how well these same vacuum breakers would work to prevent too low a containment pressure if the sprays had been actuated.

5.1.5 Type D failure: tear in bottom of torus

As discussed in Table 2 Section D, a tear in the bottom of the wetwell torus during a long term station blackout sequence might lead to significant offsite consequences because a large fraction of the CsI would have been in the pool water and because the bottoms of the SRV and downcomer vent lines would no longer be submerged in pool water. Without this submergence and with a tear in the torus boundary, a pathway from the reactor core to the environment might be created.

Creating a tear in the torus is not easily done. No reports have been issued that any torus at Fukushima had failed in spite of the very large earthquake and even flooding of the torus rooms at 1F2 and 1F4 from the tsunamis. It would take a very large force to create a tear in the torus boundary. However, water hammers have occurred in many different circumstances, especially where steam and unsaturated water come in contact and when steam, gases (air) and unsaturated water come into contact. Water hammers can create very large forces and therefore the author examined a number of Fukushima reports to see if there were any indications that a water hammer event might have occurred. Many water hammer events have occurred at both PWRs and BWRs and steps have been taken to minimize the frequency of their occurrence. There are some indications, but no conclusive proof, that the 1F2 plant may have experienced a series of increasing strength water hammers.

A few examples of past BWR water hammers have been selected for closer examination to determine if they might shed light on events that occurred at 1F2. During the original pressure suppression tests a water hammer event occurred (also called chugging) during small LOCA experiments. After expelling water out of the four foot deep submerged segment of the downcomer, stream from the drywell mock-up came in contact with colder pool water in the suppression pool, causing this steam to rapidly condense, creating a partial vacuum in the downcomer line. Water surged up into the downcomer line and fell back repeatedly in this experiment. Each cycle appears to have created shock waves and caused the vacuum breaker on downcomer line to open and close frequently and rapidly.

As another example, after a successful test of the HPCI system at an actual plant, the HPCI steam turbine was turned off. This led to a water hammer because the air in the HPCI turbine exhaust line was expelled when HPCI was started. When the test was concluded the remaining steam in the exhaust line condensed on the turbine exhaust line's cooler inner surface, creating a vacuum. Pool water rushed into the exhaust line causing a water hammer. These exhaust lines were later modified to include vacuum breakers to permit non-condensable gases to refill these exhaust lines and prevent further water hammers.

A more serious water hammer event may have occurred many years ago at a BWR in Europe. Only very limited information on this event is available to the author. It is recalled that large vibrations of the whole torus were reported to have occurred which ended when the reactor was scrammed.

Water hammer events in BWRs have led to design improvements such as greater use of vacuum breakers and passive devices to break up steam entering the pool into smaller bubbles.

When water hammers occurred before in nuclear power plants, they happened when these plants were in a normal operating mode. However, during severe accident conditions many changes are going on over time. Not all changes that can happen during an accident sequence are listed here, only those that might have some relevance to the creation of a potential water hammer situation.

First, equipment gets degraded. High temperatures caused by superheated steam and other hot gases may cause the main steam line to fail or the gaskets in safety relief valves to fail. Safety relief valves may fail open or closed because they are exposed to harsh environmental conditions and because they are actuated hundreds of times. Valve seizure (failure to reclose) was assumed to happen in an analysis by Sandia (7) when gas discharge temperatures exceeded ~1000K. The same Sandia analysis gave a 90% chance of failure to SRVs just due to the large numbers of times they are opened and closed in the course of an accident. On the SRV line there is a vacuum breaker. It should open and close each time the SRV opens and closes. Therefore it too would have a high probability of sticking open or sticking closed just because of the large number of times it would be actuated. Further, it may be possible for particles or metallic vapors, from a damaged reactor core and carried by the high temperature gases that travelled down the SRV line, to plate out on this SRV vacuum breaker, causing it to be sealed in the closed position.

In addition to the possible degradation of main steam lines, SRVs, and vacuum breakers on SRV lines, there are important changes in the suppression pool over time. A lot of the heat from the reactor core ends up in the suppression pool. It comes largely through the SRV discharges into the pool, but also comes from the turbine exhaust lines from the HPCI and RCIC systems. The pool temperature is not necessarily uniform. There may be vertical stratification with saturated water on top and subcooled water below. In the case of the 1F2 plant the torus room was flooded by the tsunamis up to a height about half way up the outside of the torus. Heat transfer through the steel torus itself could affect the temperature distribution in the pool.

Sandia National Laboratory (8) has calculated that there can be an angular temperature gradient across the pool water. Instead of modeling the suppression pool with a single node and therefore getting uniform calculated temperatures throughout the pool, a 16 volume representation of the suppression pool predicted a temperature gradient across the torus with a substantial time delay in reaching equilibrium conditions. So between vertical stratification and angular gradients, the pool is likely to have saturated conditions near the SRV, HPCI, and RCIC exit locations and unsaturated water in other locations. If the pool water was saturated in the vicinity of the bottom of the submerged SRV line, a bubble(s) of steam might migrate over to cooler, unsaturated sections of the suppression pool. Upon coming into contact with colder unsaturated pool water, the steam bubble would collapse instantaneously creating a vacuum in the pool water. This then might cause a water hammer as pool water rushes in to fill the vacuum caused by the rapid collapse of the steam bubble. Over time, more of the pool water will tend towards the saturated condition as more of the core's energy is transferred into the pool via the SRV line. It is possible that as the ability of the pool to absorb more energy from the SRV line decreases, the magnitude of the water hammers may increase.

With the above information about degrading SRVs, vacuum breakers, and steam lines, with the realization about non-uniform distributions of saturated and unsaturated water in the suppression pool, and the recognition that over time the average pool temperature would be rising, a review was conducted on the 1F2 plant using the time line information from INPO (9). A portion of the INPO timeline is presented in Table 3 and more information from Sandia is presented in Table 4. Postulated water hammer events are highlighted in red in these two tables.

Table 3: Portion of the INPO 1F2 Time Line

Clock Time in year 2011/ Elapsed time (hours)	Comments
March 11, 14:46/ 0.00	Plant trips on seismic event
March 14, 18:00/ 75.23	Operators are successful in opening an SRV and start to depressurize the reactor
March 14, 19:54/ 77.12	After refueling and starting a fire engine, seawater injection begins into the reactor via the fire protection system
March 14, 23:00/ 80.23	Based on increasing reactor pressure, operators suspect that there was not enough air left to open the selected SRV. The operators start to open other SRV switches in an attempt to depressurize the reactor.
March 15, 00:22/ 81.60	Operators continue cycling SRV control switches in an attempt to depressurize the reactor. Reactor pressure remains above 160 psig.
March 15, 06:14/ 87.47	A loud noise is heard in the area around the torus. Operators in the unit 1-2 MCR feel a shock - different than what they felt when the Unit 1 reactor building explosion occurred. While suppression chamber pressure drops to 0 psia *[actually, 0 psig]* indicating a potential instrument failure, drywell pressure remains high, indicating 105.9 psia, the reactor water level is 106 inches below TAF.

Table 4, below, adds information (10) from the Sandia's 1F2 Figure 10, which shows a significant divergence between the wetwell and drywell pressures starting around 80 hours. Up until about 80 hours the drywell and wetwell pressures track each other very closely. Then around 80 hours the wetwell pressure declines significantly while the drywell pressure climbs rapidly, up to a level (105.9 psia) which would have been sufficient to cause drywell leakage into the reactor building. Also reported in this Sandia analysis are two reactor pressure vessel repressurization events during this time of sustained high drywell pressure. Throughout the time period starting with the injection of seawater at 77.12 hours to the time when a shock is felt in the unit 1-2 MCR at 87.47 hours (about 10 hours and 20 minutes) the SRVs were activated hundreds of times and significant energy was transferred from the very hot reactor core to the suppression pool.

Table 4: Postulated Series of Increasingly Energetic Water Hammers at 1F2

Elapsed Time, hours	Actions/Postulated effects
77.12	Seawater injection starts.
77.5	First small repressurization in the reactor pressure vessel. Sandia has ruled out that this repressurization was due to seawater injection because the reactor pressure vessel pressure was near the shutoff head of the fire pump. The author postulates that the frequent opening and closing of the SRV line and the impact of hot gases flowing down the SRV caused the vacuum breaker on the SRV line to fail in the closed position. Without a functional SRV vacuum breaker, water hammers are possible every time an SRV opens, then closes. It is postulated that this RPV repressurization was the result of a small water hammer in the SRV line which caused some pool water to backflow past the SRV and into the reactor core where it was turned into steam causing a temporary repressurization.
?	Second repressurization in the reactor pressure vessel?

~80	Seawater injection continues, transferring more energy into the suppression pool and bringing the pool temperature closer to the saturation point. Then around 80 hours the wetwell pressure declined significantly while the drywell pressure climbed rapidly, up to a level (105.9 psia). These events raise the following question: what force might have been created that would be capable of simultaneously increasing drywell pressure while decreasing wetwell pressure? Recall that it was postulated that the vacuum breaker in SRV line was closed shut at 77.5 hours resulting in a water hammer that led to temporary pressure rises in the reactor vessel. By 80 hours, the pool water would have been even hotter. If that occurred, then a situation similar to the HPCI turbine exhaust line water hammer event described above might have happened, where a vacuum was created when there was steam condensation in a line without the ability to return non-condensable air to that line. If such a vacuum were created, a slug of pool water would have moved rapidly up the SRV line towards the reactor core. This slug of water would have been propelled upward by the high pressure in the wetwell gas space and the very low pressure in the SRV line. This slug of water would likely have pushed past the SRV. Some of this water could have impacted the main steam line. The main steam line itself would already have been in a weakened condition because of its exposure to very hot gases. Thus, it is postulated that the impact of this slug of pool water caused the pressure boundary of the main steam line to fail. This postulated failure of the main steam line would have led to the pressurization of the drywell and then to leakage into the reactor building. Finally, it is postulated that this same water hammer event created a hole in the torus boundary, similar in size to a penetration used for an instrument line.

77.12 to 87.47	Once the suppression pool became depressurized around 80 hours or so, the saturation point of the pool water dropped to a new temperature/pressure equilibrium point. This lower saturation temperature in the pool reduced the ability of the pool to absorb more energy making a more severe water hammer more likely. This set the stage for an even larger water hammer event than the one experienced at about 80 elapsed hours.
87.47	Seawater injection continues, transferring more energy into the suppression pool which now has a lower saturation temperature and much less margin to absorb steam from the SRV line. If the SRV were stuck open at this time because of the water hammer assumed to happen at around 80 hours, very hot gases and significant amounts of superheated steam could have travelled down the SRV line just at the time that the energy absorbtion capability of the pool would have been lowered because of the depressurization of the wetwell. Then at 87.47 hours a large noise was heard and a shock felt in MCR 1-2. This could have been the most severe water hammer up to that time or it might have been the explosion at Unit 4 which occurred within 20 minutes or so of the 87.47 hours reported from the MCR 1-2.
87.47 to ~ 96.0	Seawater injection continues until around 96 hours. The INPO time line ended at 87.47 hours. If further water hammers happened, they were not noted. Analyses performed by Sandia on 1F2 (Figure 8) indicates high seawater flow rates from around 90 to 94 hours. This implies a comparatively low reactor pressure vessel back pressure. A low pressure in the reactor pressure vessel may be the result of a stuck open SRV, the postulated hole in the main steam line, or both. No explanation is provided in the Sandia Figure 8 analysis as to why seawater injection stopped at about 96 hours. Once seawater flow stopped, the possibility for further water hammers would also have stopped.

5.1.6 Postulated water hammers

As discussed above, during the course of the 1F2 accident the SRV was opened and closed hundreds of times, particularly during the time of seawater injection, as indicated in Figure 8 of the Sandia analysis of 1F2. For each opening/closing cycle of an SRV there would be a corresponding opening/closing cycle of the vacuum breaker on the SRV line leading to the suppression pool. It is possible that because of these many SRV and vacuum breaker cycles, the vacuum breaker in the SRV line failed in the closed position. There are additional pathways to a failed closed SRV line vacuum breaker. For example, Sandia analyses of 1F1 have examined the possibility of main steam line and SRV failures due to a creep rupture mechanism brought about by the high temperature gases that flow towards the suppression pool. These same high temperature gases flow down the SRV line and might also cause the SRV vacuum breaker to fail in a closed position. If, indeed, the SRV vacuum breaker failed in the closed position this might precipitate a series of water hammers starting at 77.5 hours. Once this SRV vacuum breaker failed closed, the steam remaining in the SRV line after SRV closure would condense on the cooler SRV line's inner surface, causing a vacuum. The combination of the elevated pressure in the wetwell gas space and this vacuum in the SRV line would result in a slug of pool water being driven up the SRV line. This same surge of water up the SRV line might have caused pool water to back flow through the SRV. If water entered the reactor core this might explain the first small reactor repressurization at 77.5 hours. If a chugging type mechanism developed, as discussed before, it might explain a subsequent small reactor repressurization event. This kind of an event would be similar to the earlier water hammer event that occurred in the HPCI turbine exhaust line, discussed before.

During the time period between 77.12 hours and about 80 hours seawater was injected and the SRVs were frequently actuated. This would have transferred a lot of energy into the suppression pool, bringing more of the pool water closer to saturation temperatures. Data on pool temperatures versus time have not been provided. It is assumed that the higher pool temperature at about 80 hours caused a more severe water hammer than the one at 77.5 hours, sending shock waves throughout the whole torus. It is postulated that this more powerful water hammer at 80 hours was sufficient to create a small opening, perhaps a failed penetration, in the torus boundary above the water line. A small opening in the torus boundary above the water line might explain why the wetwell pressure decreased starting at 80 hours. In order to reconcile differences between Units 2 and 3 Sandia postulated a leak in the torus in Unit 2. A leak area with a 2 inch diameter best fit the data. If the calculated opening size is similar to the size of a penetration, perhaps that of the failed suppression chamber pressure monitoring instrument, that would be quite valuable information. TEPCO might consider attempting to send a probe to this penetration location and others of similar sizes to look for evidence of damage/failure.

It was postulated that the water hammer that may have led to creating a hole in the torus boundary also caused the main steam line to fail at 80 hours. The failure mechanism here is thought to be due to comparatively cold pool water slamming into the main steam line. Sandia has calculated that the main steam line would fail at around 84 hours because hot gases would have induced a creep rupture type of failure. Therefore, at 80 hours the main steam line would have been close to failure. If now struck by a slug of comparatively cold pool water, it seems plausible that the failure of the main steam line could occur somewhat sooner, at 80 hours. If there was some kind of a failure of the main steam line at 80 hours, that could explain the sudden steep rise in the drywell pressure, up to the point that leakage began to occur from the drywell into the reactor building. It is also possible that the force of a water hammer could have contributed to a failure of the SRV gasket which also could explain the drywell pressure rise.

The largest water hammer appears to be the event reported at 87.47 hours. It seems plausible that this would be the most severe water hammer in this series because the saturation temperature in the pool would have dropped when the wetwell pressure went down, starting at 80 hours. If the pool saturation temperature dropped because the wetwell overpressure ended, this would have caused a decrease in the ability for the pool to absorb more steam from the SRV line. To develop a idea of the importance of the decreasing overpressure in the torus, Figure 10 in the Sandia analysis of 1F2 was utilized. Just prior to 80 hours the pressure in the wetwell was around 60 psia. This corresponds to a saturation temperature of about 292 degrees F. and an enthalpy of 262 Btus per pound of water. The actual average pool temperature is likely to be well below 292 degrees, providing a capability to absorb more SRV line steam. However, at one atmosphere the saturation temperature is 212 degrees F. and the corresponding enthalpy is 180 Btus per pound. This large decrease in enthalpy, 82 BTUs per pound of water, implies that once the wetwell pressure started to decrease, the pool's ability to absorb more steam from the SRV line also sharply decreased.

A secondary effect of this late depressurization of the wetwell is that the pool's ability to trap soluble radioactive compounds decreases as temperatures approach the saturation point.

5.1.7 Discussion of Tables 3 and 4

Many previous probabilistic risk assessments (PRAs) analyzed station blackout sequences using data on the time to recover onsite or offsite power. The likelihood for an extended loss of offsite power decreases with elapsed time, reflective of the major efforts by utilities to restore power. However, a subset of the data is the power restoration time for extreme natural events that caused loss of offsite power. For example, extreme ice storms have caused the loss of offsite power for extended periods of time. Most of the risk from the loss of offsite power is thought to be in this subset. This special subset has to be further broken down into those external events that are also potentially capable of damaging emergency onsite power sources. This special subset would then be added to internal events like fires and internal flooding which have the potential to destroy the plant's ability to distribute electric power to safety equipment. Together these situations could give rise to a group of scenarios where normal offsite and onsite sources of electricity might be unavailable indefinitely.

Station blackout events can be characterized as situations where the normal path to the ultimate heat sink has been severed. In many station blackout cases electric power, either on site or off site or both, will be recovered before there is damage to the reactor core. However, very prolonged station blackouts are more challenging. In the case of nuclear plants with Mark I containments the water in the torus serves as a temporary heat sink, but has only a finite ability to absorb the energy discharged into it. If the energy discharged into the suppression pool becomes excessive, energetic reactions, such as water hammers in the pool, may occur. Should severe water hammers occur in the suppression pool, they may have the potential to cause a loss of integrity of the pool boundary. Even without water hammers the wisdom of adding more water to what remains of the reactor core has to be examined. At some point in a long term station blackout, adding more water to the reactor core may overburden the suppression pool as core cooling water is turned to steam which then flows down the SRV line into the pool water where there maybe incomplete condensation. This steam that is no longer fully condensed in the suppression pool might find its way back into the drywell via the wetwell-to- drywell vacuum breakers, eventually pressurizing the whole containment.

The different responses of the three damaged nuclear plants at Fukushima offer an exceptional opportunity to examine at what point severe accident management should shift from protecting the reactor core to protecting the torus boundary.

As a major lesson learned, it is important to know if, at some degree of core damage, the risks of injecting more water into the reactor vessel outweigh the benefits. Because of the potentially serious nature of water hammers in the suppression pool, further analyses using MELCOR or MAAP are warranted. As pointed out before, lessons might be gained by plant-to-plant comparisons. Computer analyses can compare the water level and pressure responses in the reactor vessel, SRV performance, drywell and wetwell pressure levels, and suppression pool temperature all as a function of when and how much emergency water is injected into the reactor vessel. If it is concluded that at some level of core damage further fire pump injection of water into the reactor vessel should stop, and possibly redirecting this fire pump water to help build a layer of water on the drywell floor, this would be a significant advance in severe nuclear accidents technology.

The Fukushima experience is very valuable. It seems that similar undamaged Mark I plants could take about three to four days of core cooling before limiting conditions in the pool were reached.

This issue of the pool's saturation temperature and the ability of the pool water to absorb steam energy from the SRV line has other implications. If the wetwell vent line is opened, the limiting water temperature for the pool becomes 212 degrees F. Although the failure pressure of the wetwell hardened vent line rupture discs should lie between being above the peak drywell pressure from a loss of coolant accident and being below the containment failure pressure, the optimum rupture disc failure pressure requires balancing different objectives. A lower failure rupture disc bursting pressure favors HPCI and containment spray rates. A higher rupture disc failure pressure favors maintaining higher saturation pool temperatures longer.

Optimization of the rupture disc failure pressure is therefore a rather complex analysis, well suited for the MELCOR or MAAP computer codes.Rupture disc bursting pressure may be a plant specific analysis. Not all Mark I plants are the same. There are differences in the mass of the pool water per thermal megawatt that might affect such optimizations. There are also limits on the amount of water that can be added to the pool before water heights reach that of the hardened vent line, however the hardened vent line should have been closed by this time.

Lastly, it may be useful to examine other severe accident management schemes, such as spraying water on the outside of the torus to turn the torus into a large radiator to lower pool water temperatures. This too could be accomplished with on site existing equipment.

5.2 Reducing non-radiological health risks

5.2.1 Introduction

Since the offsite radiological consequences due to a Fukushima type of accident are so low it is probable that the largest health risk is not directly radiological, but rather to the stresses introduced by evacuating people and then having some of these evacuees unable to return to their homes and some people placed into long term shelters. It is important to balance the radiological health benefit of evacuation against the detrimental health effects of overevacuation. This need for balance (11) was expressed years ago by the EPA:

"The decision to advise members of the public to take an action to protect themselves from radiation from a nuclear incident involves a complex judgement in which the risk avoided by the protective action must be weighed in the context of the risks involved in taking the action."

Data and analyses now becoming available from Fukushima indicate that there is a significant need to strike a better balance between radiological and non-radiological health consequences. On one hand the WHO radiological health consequences, listed in Table 1, are extremely low. On the other hand the non-radiological health consequences among evacuees are significant. In Japan the "Reconstruction Headquarters" has reported approximately 1100 disaster-related (premature) deaths among the evacuees due to psychosomatic effects (67%) and disruption of medical and social welfare facilities (18%) (12). What percentage of these evacuees are the result of the accident at Fukushima and what percentage of the evacuees are due to the earthquake and tsunami may be sorted out in a paper by Genn Saji, to be presented this summer at the ICONE 21. What is already known is that 160,000 people were evacuated because of the Fukushima accident of which, today, about 70,000 have not been allowed back to their homes. What is also known today is that because the non-radiological effects are significant it is timely and appropriate to rebalance these consequences to end up with a minimum total number of consequences.

In this section four actions are described which have the potential to strike a better balance between radiological and non-radiological health consequences so that overall health consequences are minimized.

5.2.2 Advanced emergency responses

The basis for developing advanced emergency plans is to understand the geographical distances over which different radiologically induced health effects might occur. Early fatalities require very high exposures and would be limited to one mile or less from the point of release. Early injuries, would be confined to two miles from the point of release. If the size of the radiological release is small, as was the case at Fukushima, early fatalities can not happen. Even if there were a very large release of radioactive material, early fatalities can still be avoided by evacuating the innermost one mile prior to or at the start of the release. Modern severe accident technology has shown that releases are too small to cause early fatalities. Further, these releases take quite a long time to start to enter the environment. Because of this, there should be ample time to evacuate the area within one mile of the damaged nuclear plant.

Lastly, there can be exposure to radiation from a nuclear power plant accident that could have long term effects such as latent fatality cancers. Although such long term effects would be too small to detect statistically, an advanced emergency plan can reduce radiation exposure that could lead to such long term effects.

With an understanding of these geographical distributions of health effects, emergency plans can now be developed using a combination of evacuation, sheltering, and selective relocation. To eliminate the early fatality risk and reduce long term health effects, a preemptive evacuation of the public within an inner one mile circle from a possible point of release should occur at the site emergency level, regardless of the size of the projected source term or wind direction. For most sites, this area is without a large public presence. If the site emergency continues for a number of hours or conditions worsen, evacuation in a circular area out to two miles should be initiated.

Evacuees could be brought to designated shelters or locations of their choice. Evacuation out to two miles would essentially eliminate the early injury risk and would further reduce radiation exposure that might contribute to long term health effects. Mass evacuation beyond two miles is not advised and could actually increase exposure to radiation in highly populated sites because it could result in a slow evacuation that would be less protective than sheltering. Beyond two miles downwind people should be advised to take shelter and continue to listen to emergency broadcasts. Sheltering could significantly reduce the latent fatality risk. While sheltering, people should take steps to reduce potential inhalation doses by closing off air inflow pathways.

Beyond two miles a process of selective relocation of sheltered people to group shelters should take place. It is expected that there would be specific locations close to the point of release where relocation of already sheltered people would be beneficial, such as where there were hot spot areas. The emergency response team would make the case-by-case decisions as to which people should be relocated from their individual shelters to group shelters, taking into account measured dose rates and the shielding characteristics of the structures in which people take shelter. Apartment buildings usually provide better sheltering whereas many trailers provide minimum sheltering and can be prone to allowing higher inhalation doses for their occupants.

This process of selective relocation would continue out to a distance, longer or shorter than the present ten mile EPZ radius, until a dose criterion is met. One possible dose criterion would be that exposures over a 24 hour period should not exceed a whole body dose equal to a single cranial multiple scan average dose of 5 rem, assuming the exposed person is in a normal activities protection mode. (Normal activities means that a person effectively acts as if he were unaware of the situation at the damaged plant, spending part of his day out of doors and part indoors.) People beyond the point where 24 hours of normal activities would not exceed the selected dose limit, would be kept informed through the media and emergency broadcasts. It would be their choice to stay in a normal activities mode or to evacuate.

This mixture of preemptive limited range evacuation, downwind sheltering, and selective relocation out to distances where doses fall below a dose limit criterion for people taking a normal activities response is consistent with fundamental emergency planning guidance to minimize the total of radiological plus non-radiological consequences.

5.2.3 Re-occupation criteria

Studies (13) have been conducted that indicate that the criteria for allowable radiation exposure during reoccupation of somewhat contaminated land could justifiably be set higher. If less stringent, but acceptable, reoccupation criteria were applied, fewer people would be dislocated from their homes following a nuclear accident that leads to a release of radioactive material into the environment. Part of this re-examination of reoccupation criteria should be a review of the scientific basis of the linear no threshold (LNT) and how this assumption affects the number of people who end up as displaced from their homes for long periods of time.

5.2.4 Improved decontamination methods

Lessons are being learned about land decontamination near Fukushima. It appears that much of the cesium deposited on land surfaces is confined to a comparatively thin top layer of soil, about half an inch thick. Removal of this layer of earth could be sufficient to justify having more people return to their homes. Progress in decontamination technology should be encouraged.

5.2.5 Smaller source terms

Even though the amount of cesium that was released from the Fukushima accident was quite limited, further reductions may be possible if the operation of the wetwell hardened vent is modified to place the isolation valves into a "normally open" position. This possible source term reduction should be verified by running the MELCOR or MAAP codes with these isolation valves "normally closed" (the Fukushima situation) and then run the codes again with these valves in the "normally open" position and then compare results.

5.3 Integrating prevention and mitigation - follow the cesium

A major lesson learned from Fukushima and from the Chernobyl accident is that the dominant isotope of concern is ^{137}Cs. At Fukushima the offsite health effects are very small and would have been even smaller if existing onsite equipment (e.g., the wetwell hardened vent isolation valves) were used more effectively and/or more extensively. Offsite <u>net</u> economic losses, if Fukushima had happened in the United States, would be expected to be small or zero if Price-Anderson payments were made. Explosions can be prevented with existing plant equipment with the modified wetwell hardened vent isolation valves placed in the "normally open" position. It is not radioactive iodine nor other radionuclides that are most important. ^{137}Cs dominates Mark I concerns because of its role in causing land contamination. Further, it is land contamination which can lead to the non-radiological health hazards of over-evacuating and long term sheltering. Going forward, the lessons learned from Fukushima should be the further development of severe accident management which should focus on preventing cesium from leaving the site, cleaning up cesium both on and off site as much as practical and minimizing non-radiological health consequences. Simply put, severe accident management should "follow the cesium".

6.0 Value of an Additional Drywell Filtered Vent System

6.1 Introduction

Proposals (14) have been made to modify Mark I plants by adding an additional drywell filtered vent system. In this section the benefits of such proposals are examined using the same format as in Table 2 where four different containment failure modes were examined.

6.2 Background

A number of filtered vent systems have been installed at nuclear plants in Europe (15). Many of the nuclear plants where these filtered vents were installed were pressurized water reactors. Pressurized water reactors do not have suppression pools or piping systems that direct radionuclides into the suppression pool during accident conditions, as the Mark I containment design does with its SRV lines and with the drywell-to-wetwell vent systems designed to handle LOCAs. Since the PWRs do not have the filtering capability of Mark I designs, the PWRs' use of additional filters has no bearing on applying similar filtering systems to Mark I plants. One European nuclear BWR plant, Lieberstadt, installed a filtered vent system in 1992 well before the present source term capability in MELCOR and MAAP was available. Present source term analyses show that releases of radioactive material under severe accident conditions, including station blackout, are far smaller than those predicted in 1992. The very small releases of radioactive material from the Fukushima accident are supportive of the analyses performed by today's advanced source term computer codes. Had today's source term technology been available in 1992 it is possible that a different decision might have been reached on the need for such a filter. The Lieberstadt design has a suppression pool arrangement which is likely very effective, as are the Mark I designs. In the Lieberstadt design the filtering system, two liquid-type vent tanks, is downstream of the suppression pool. Since it has been proven that the suppression pools are very effective in capturing CsI, the incremental value of these downstream filters is quite limited. The mass of water in these two vent tanks appears to be considerably less than the mass of water in the suppression pool.The Lieberstadt filtered vent design basically just adds more water mass to the suppression pool.

Drywell filter vent designs, like the one at Lieberstadt, are an add-on to the suppression pool and are not the same as the proposed filter designs to be added to the drywell. As such they, and other filtered vent systems used in Europe, are not relevant to determining if an additional drywell filtered vent system makes any sense.

There are aspects of proposed additional filtered vent systems in the drywell that are inferior to the present Mark I design. In the Mark I design CsI reduction processes and the prevention of explosions in the reactor building process take place in separate locations. In the Mark I design CsI reduction takes place upstream of the vent isolation valves and venting of explosive gases takes place downstream of the isolation valves. In some proposed drywell vent systems, both the filtration (CsI reduction) function and the control of explosive gases function take place together downstream of the isolation valves. This is a crucial difference. Failure to open the wetwell hardened vent isolation valves in a timely manner was the eventual cause of explosions at Fukushima. In these proposed drywell filtering systems, the same kind of failure to open isolation valves would cause the loss of both the CsI reduction and explosive gas removal functions. Stated differently, at Fukushima failure to open the wetwell vent isolation valves in a timely manner did not prevent the suppression pool from capturing most of the CsI and or the removal of CsI by deposition in the drywell.

In some filtered vent designs sand would be used as the filtering agent. However, the flow resistance in such a design would be significant, especially if the filtering medium was clogged with water from condensed steam. The greater the vent flow resistance, the larger the vent and its piping system would have to be to get drywell pressures down to low levels and to expel most of the explosive gases. The Mark I design is a well understood simple piping system and there is no concern that flow resistance would increase during the course of an accident.

An additional drywell filter vent system would use technology that is unproven under actual accident conditions whereas the Fukushima accident demonstrated that both the present filtering and the venting processes work very efficiently. Additionally, the functions of CsI filtration, removal of explosive gases from the drywell and containment pressure reduction can be supplemented with other existing plant Mark I equipment to provide more defense-in-depth. As an example of this, consider the combination of the containment sprays and the existing drywell vent system (not filtered). Careful use of the drywell spray system would supplement CsI removal accomplished by the suppression pool by removing airborne radioactive material from the drywell gas space. Drywell sprays would complement the wetwell hardened system's actions to lower drywell pressure. As the containment pressure nears atmospheric conditions because of the containment spray system action, the existing drywell vent can be opened and outside air would backflow through it, should the drywell begin to become subatmospheric. So, if further defense-in-depth for the hardened vent system is desirable, it can be achieved with existing plant equipment rather than by adding a new drywell filtering system.

If desirable, a more quantitative approach can be used to evaluate the benefits of proposed drywell filter systems. For example, two sets of MELCOR or MAAP calculations could be made for the well-studied Peach Bottom plant. One set of calculations would be based on the existing Peach Bottom design and the second set of calculations would be based on a design that had been modified to include a drywell with a filtered vent system. The difference between the consequence levels of these two sets of calculations would be a measure of the value added by having this proposed drywell filtered vent system. Half of this work is already done in that Peach Bottom without the drywell filtered vent has been extensively analyzed.

As such, devoting a lot of effort to this additional drywell filter would be a distraction. Limited time and resources should be spent on how to avoid more serious containment failures, how to rebalance the offsite emergency response so that large non-radiological fatalities are avoided, and how to re-orient severe accident management to focus on where the cesium is and how to prevent its release.

6.3 Drywell filtered vent systems and the four containment failure modes

Table 5 uses the same format as Table 2 to evaluate the benefits of an additional drywell filtered vent system. The benefits of an additional drywell filtered vent system appear to be very small.

Table 5: Evaluation of an Additional Drywell Filtered Vent System

Postulated containment failure mode	Comment
A. Leakage- Containment overpressure, (Fukushima)	With the very low radiological health consequences reported by WHO there is very little opportunity to improve upon this. Even if an additional filter system were added to the drywell, most of the CsI would still end up in the pool due to steam flow through the SRV lines. Some airborne cesium in the drywell would still be removed by deposition processes and some cesium would remain in the reactor vessel. Therefore there would be very little cesium to filter from the drywell, even if no changes were made in Mark I plants. As it is, the use of drywell sprays to lessen the possibility of liner failure would further reduce airborne cesium in the drywell and would lower drywell pressures. By converting the isolation valves in the wetwell hardened vent system to be normally open, drywell overpressure should be avoided passively as would reactor building explosions from the hydrogen released from zirconium-steam interactions that leaked into the reactor building when the drywell was overpressurized. There would be very little opportunity for an additional drywell filtered vent system to prevent containment overpressures because other existing systems would be doing this. An additional drywell filter offers no major improvement to this mode of containment failure.
B. Leakage- Drywell Liner Failure	Assuming that drywell liner failure actually occurred at Fukushima, this issue in other Mark I plants would be dealt with by using the drywell sprays, not an additional drywell filter.

C. Containment becomes subatom-spheric, implodes	Preventing subatmospheric conditions would be dealt with by controlling the drywell spray rate, possibly using the existing unfiltered drywell vent system to backflow air into the drywell and by controlling the opening and closing of the wetwell isolation valves. The use of these existing systems limits the value of an additional drywell filtered vent system.
D.Tear at bottom of torus	Water hammers in suppression pool is the main issue here. Prevention of water hammers can not be accomplished by the additional of a drywell filtered vent system.
Other issues 1.Non-radiological health consequences. 2. Spent fuel pool. 3. Offsite economic losses. 4. Re-orientation of severe accident management approach to focus on cesium control.	1. No effect. 2. No effect. 3. Very tiny effect if there is some reduction in the already limited amount of cesium released to the environment. 4.Very tiny effect if there is some reduction in the already limited amount of cesium released to the environment.

7.0 Spent Fuel Pool Considerations

7.1 Background information

Several years ago the National Academy of Sciences reviewed spent fuel pools at nuclear power plants. It had been postulated that the large inventory of ^{137}Cs in spent fuel pools could contaminate large land areas and cause a large number of latent fatalities if a spent fuel pool was drained and the zirconium cladding caught fire. It was proposed to remove the coolest 80% of the spent fuel and spread out the remaining 20% of the spent fuel so that it could be air cooled by natural convection if the pool water was drained. The 80% of the spent fuel that was to be removed from the pool would be placed into dry storage at the nuclear sites.

Even though only the hottest 20% of the spent fuel was to remain in the pool, this still represented a large source of CsI and therefore a high degree of confidence was needed to demonstrate that natural convection air cooling would be sufficient to assure adequate cooling. Several problems with air cooling became apparent upon closer examination. First, unless the location of the water drainage point was on the very bottom of the pool, the pool water level would first decrease down to the drainage point and then continue to decrease as the heat from the spent fuel caused the remaining water to evaporate. The spent fuel would be cooled by this evaporating water until the water level dropped to about the bottom 25% of the spent fuel element. Once that level was reached the rate of water evaporation would be insufficient to prevent the top of the exposed spent fuel from heating up. At that time there would still be some water below the 25% fuel height level and that would block any flow of air through the spent fuel. Analyses were not presented in the unclassified sessions of this committee that demonstrated that the remaining water that blocked air flow would be boiled away prior to the start of a zirconium fire.

It was further pointed out that any force large enough to damage a massive spent fuel pool would almost certainly also damage some of the spent fuel. No analysis was provided that demonstrated that damaged spent fuel could be adequately cooled by natural convection.

Attention turned to cooling the spent fuel with water. Analyses were presented that showed that very little water needs be sprayed on the spent fuel to keep it adequately cooled by evaporation. It was stated that an adequate amount of water needed for cooling the spent fuel, spread out over the large volume of the spent fuel pool, would be like having a mist throughout this volume. So the central issue was not providing large amounts of water but providing much smaller amounts of water, but reliably.

Additional heat transfer analyses showed that instead of moving this 80% of the spent fuel it would be beneficial to use the cooler elements as heat sinks for the hotter spent fuel elements, cooling them by radiative heat transfer. At the very least such radiative heat transfer would delay any onset of a zirconium fire by several hours giving the plant personnel additional time to add water to the pool via a fire hose arrangement. A potential example, subject to verification of adequate air cooling, using the cooler spent fuel to advantage appears in Figure 2 of reference 17. Such an arrangement would significantly reduce the amount of spent fuel that would have to go into dry storage by about 75%. However, although surrounding the hotter spent fuel with cooler spent fuel is advantageous, it still leaves open the questions raised before about air flow blockage by water at the bottom of the spent fuel pool and how to cool damaged fuel. A water spray is still needed.

Once it is concluded that it is necessary to provide some kind of a reliable emergency way to spray water into the pool, this would end the rationale for removing any of the spent fuel on an accelerated basis. The same spray water would be able to cool all the spent fuel.

In the years since the first NAS committee met on this subject nuclear power plants have put in place systems to provide emergency water to the spent fuel.

7.2 Lessons from Fukushima on spent fuel pools

In spite of a magnitude 9 earthquake, aftershocks, tsunamis, and explosions in reactor buildings, none of the spent fuel in the spent fuel pools was damaged. At Fukushima the fire protection systems survived the earthquake and tsunamis and were used to inject water into the reactors. The availability of these fire protection systems adds to the reliability of providing the limited amount of water needed to protect the spent fuel.

Debris from the Fukushima explosions did land in the spent fuel pools. It was pointed out by Dr. Alvarez, et al, (16) that debris falling into the spent fuel pool could defeat the open lattice air cooling approach to spent fuel protection.

The conclusions reached by the earlier NAS committee has withstood the passage of time and, if anything, there is even less reason to modify or supplement the spent fuel pool improvements that have already been put in place.

8.0 Suggested Tasks to be Completed

A number of actions might be considered to extract more severe accident information from the Fukushima accident, such as:

A. Develop emergency response plans that balance radiological consequences with non-radiological consequences as discussed in Section 5.3.

B. Re-orient the approach to severe accident management to focus on cesium control.

C. Make plant-to-plant comparisons. Look for evidence of other water hammer events in 1F1 and 1F3. How did the timing and rates of water injection amongst the plants differ and how did this affect the accident progressions?

Expand the MELCOR and MAAP analyses to calculate suppression pool temperature/saturation temperature versus time for all three plants.

D. Use MELCOR or MAAP analyses to calculate optimum wetwell hardened vent rupture disc failure pressures and accident management strategies. This may have to be a plant specific analysis in that there are, among the Mark I plants, different pool water mass to thermal power ratios and different drywell gas space to wetwell gas space ratios, both of which affect the pool temperature transients in the power plants and therefore the potential onset of water hammer events. Additionally, if motive power is assumed to be available to open and close the isolation valves, use these computer codes to determine if a series of hardened vent openings and closings provides a better severe accident management profile.

E. Recalculate Figure One but with the wetwell hardened vent opened at the optimum rupture disc failure pressure and determine how venting further reduces the CsI source term. Separate the CsI in the drywell into that which is airborne and that which has been removed by deposition.

F. Use MELCOR and MAAP analyses to calculate optimum use of the drywell spray system in conjunction with the use of the wetwell hardened vent analysis so that containment does not become subatmospheric. Use core-concrete interactions and drywell liner failure analyses to determine how much spray water is necessary to create a thick enough layer of water on the drywell floor and the time by which this water has to be in place. Account for the water on the drywell floor produced by the accident itself. Determine if there are any problems in using the existing drywell vent to backflow outside air to be a supplemental way to avoid excessive subatmospheric conditions. These computer codes can also indicate whether opening this unfiltered vent line leads to unacceptable releases of radioactive material to the environment.

G. TEPCO has examined portions of the torus using robots and did not find any indication of a point of leakage. This inspection is expected to continue. If possible, TEPCO should examine the main steam lines and SRV gaskets to see if their pressure boundaries were opened up. Similarly, if possible, examine the operability of the vacuum breakers on SRV lines. The vacuum breakers used to return outside air to the drywell to prevent subatmospheric conditions should be examined.

H. Create a list of essential instruments and monitors needed for a long term station blackout condition. TEPCO might check to see how these essential instruments and monitors acted during the Fukushima accident. These essential instruments and monitors should be capable of operating under station blackout conditions for four days.

I. Determine if the scope of onsite equipment, including FLEX equipment, is sufficient to meet all severe accident needs, such as power to essential monitors, motive power for opening and closing vent isolation valves, motive power for the containment sprays, lighting, communications, etc.

J. Further develop severe accident guidance based on the above tasks. Examine the implications of long term loss of the ultimate heat sink on other accident sequences, such as the TW sequence.

K. Look for novel ways to keep torus pool water temperature from becoming too high, such as spraying the outside of the torus with fire water. Look for novel ways to create a feed and bleed circuit to help keep suppression pool water levels from becoming too high. The feed part could be accomplished by adding water to the reactor core, the bleed part might be accomplished by draining some of the torus water, provided that this drained contaminated torus water can be placed in a secure storage tank.

9.0 Appendix A: Estimate of Offsite Economic Losses

Estimates (17) of offsite losses (in billions of dollars) have been made for five nuclear power plants as a function of the number of megacuries of ^{137}Cs released. For an average site, a release of 35 megacuries of ^{137}Cs, the estimated economic losses were calculated to be $385 billion dollars and for a release of 3.5 megacuries of ^{137}Cs, the estimated economic losses were calculated to be $91 billion dollars. A third point can be added to the above figures, for zero megacuries of ^{137}Cs released there would be zero offsite economic losses.

Using the above figures it is possible to estimate the number of megacuries of ^{137}Cs that, on average, would cause about $12.6 billion dollars worth of economic losses. The $12.6 billion dollar figure is the automatic coverage afforded by the Price-Anderson Act. Interpolating between a zero cesium 137 release and a 3.5 megacurie cesium 137 release, a release of about 0.45 megacuries of ^{137}Cs would, on average, be expected to cause economic losses equal to $12.6 billion.

Based on TEPCO measurements taken on April 1, 2011 the amount of ^{137}Cs that was measured was 1.1 PBq on land; another 0.9 PBq went into the sea for a total of 2.0 PBq. A more recent measurement of the cesium release was taken on May 24, 2012. At this later date it is estimated that the ^{137}Cs release on land was about 10 PBq and another 3.6 PBq at sea for a total of 13.6 PBq of ^{137}Cs. Using the conversion factor of 1.0 megacurie = 37 PBq, a release of 13.6 PBq equals 0.37 megacuries, somewhat below the 0.45 megacuries needed to have offsite losses equal to the Price-Anderson coverage. A certain amount of ^{134}Cs was also released at Fukushima. This isotope of cesium has a two year half life compared to the 30.2 year half life of ^{137}Cs and therefore its contribution to long term land contamination is much less. The short term economic impact of ^{134}Cs was not accounted for in this estimate.

Based on these estimates, if the three reactor meltdowns at Fukushima happened in the United States at an average population site it seems likely that the offsite losses would fall within the Price-Anderson Act coverage.

10.0 References and Acronyms

10.1 References

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10.2 Acronyms

BWR	Boiling Water Reactor
CsI	Cesium iodide
EPRI	Electric Power Research Institute
HPCI	High Pressure Coolant Injection
ICONE	International Conference on Nuclear Energy
INPO	Institute of Nuclear Power Operation
LOCA	Loss of Coolant Accident
MCR	Main Control Room
NAS	National Academy of Sciences
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PBq	Peta Becquerel
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
SRV	Safety Relief Valve
TAF	Top of Active Fuel
TEPCO	Tokyo Electric Power Company
WHO	World Health Organization