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Sent: Friday, December 07, 2007 10:17 AM
To: Prato, Robert \RES
Subject: A Summary of AEC / NRC Risk and Consequence Reports
Attachments: AEC NRC Studies.doc

Bob

As an effort to become familiar with prior studies, I put together a short summary of everything that was published in the past. I thought you may find it useful as a quick reference. Feel free to forward it as you please.

Jim Vail

C/17

A Summary of AEC / NRC Risk and Consequence Reports

In the first decade of nuclear power the reactors were low power and of experimental designs. These low power reactors were more tolerant of what today would be accident initiating events. As newer designs approaching 500 MWt were developed, the AEC began serious studies of accidents and their consequences. Over the following 40 years the AEC and later the NRC would produce 5 significant reports that examined the broad spectrum of reactor risk and consequence. Each study would build on the prior study and add newer research and experience to sharpen the models of nuclear accidents.

WASH-740 *Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants*, 1957

An important technical input to establishing the indemnity provisions of the Price-Anderson Act was the report WASH-740 which was prepared by Brookhaven National Laboratory and published by the AEC. Using what would prove to be extremely pessimistic assumptions including a core meltdown with the release of fifty percent of the core fission products to the atmosphere, the worst case consequences of a 500 MWt reactor accident were estimated to be 3,400 early fatalities, 43,000 acute injuries, and 7 billion (1957) dollars.

There was a consensus among those involved in the WASH-740 study that the likelihood of a meltdown accident was low, but quantitative probability estimates could not be supported given the lack of operating plant experience. Similarly, the likelihood of containment failure (or bypass) given a meltdown accident was not quantified (or quantifiable, at the time). However, until 1966, the containment building was treated as an independent barrier, which should remain intact even if the core melted, thereby preventing any large release of radionuclides to the atmosphere. It was recognized that failure of the containment building and melting of the core could occur--for example, as a consequence of gross rupture of the reactor pressure vessel--but such events were not considered credible. Containment failure was not expected to occur just because the core melted.

WASH-1250, *The Reactor Safety Study of Nuclear Power Reactors (Light Water-Cooled) and Related Facilities,* 1973.

Senator John Pastore requested a comprehensive assessment of reactor safety. The AEC's first response to this request was the WASH-1250 report, which was published in final form in July 1973. WASH-1250 provided factual information regarding the conservatisms applied in the design of nuclear power plants. It did not, however, address the likelihood or potential consequences of *beyond-design-basis,* that is, failures beyond those postulated under the single failure criteria.

Conservative offsite doses from design-basis accident analyses from WASH-1250

	Two Hour Exclusion Boundary (3200 feet or 975 meters)		Duration of Accident Low Population Zone (4 miles or 6.4 km)	
Accident	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
Loss of Coolant	155	3	81	3
Control Rod	<1	<1	<1	<1

Ejection				
Fuel Handling	2	2	<1	<1
Steam Line Break	16	1	3	1
10 CFR 100 Dose Guideline	300	25	300	25

WASH-1400, (NUREG-75/014), *Reactor Safety Study*, 1975.

As indicated above, the radionuclide releases from fuel assumed in conservative design-basis LOCA analyses could only be realized if significant core melting occurred. Consequently, for a severe accident in which containment remained functional, the resulting offsite doses would be comparable to those conservatively calculated in the Safety Analysis Report for design-basis LOCAs. Yet the possibility remains of severe accidents in which containment is either bypassed or breached as a result of severe accident phenomena. Depending on the mechanism, location, and timing of containment failure, and the meteorological conditions, offsite doses could be substantially (100 times) worse than conservatively calculated for the design-basis LOCA. That is, the accidents with the greatest potential public consequences are uncontained severe accidents.

In this light, several questions had to be addressed in order to respond to Senator Pastore's request for a comprehensive assessment of reactor safety. What accidents could result in significant core damage and containment breach or bypass? How likely are these accidents? What would be their health and economic consequences? These are fundamental questions that WASH-1250 did not address. Such questions are addressed in probabilistic risk assessments, but, at the time, relevant probabilistic estimates were quite limited in scope and/or highly subjective.

In the summer of 1972 the AEC initiated a major probabilistic study, the Reactor Safety Study. Professor Norman C. Rasmussen of the Massachusetts Institute of Technology served (half-time) as the study director. Saul Levine of the AEC served as full-time staff director of the AEC employees that performed the study with the aid of many contractors and consultants.

The team attempted to make a realistic estimate of the potential effects of light water reactor accidents on the public health and safety. One BWR, Peach Bottom Unit 2, and one PWR, Surry Unit 1, were analyzed in detail to estimate the likelihood and consequences of potential accidents.

Its stated purpose was to quantify the risks to the general public from commercial NPP operation. This logically required identification, quantification, and phenomenological analysis of a very considerable range of low-frequency, relatively high-consequence scenarios that had not previously been considered in much detail. The introduction here of the notion of "scenario" is significant; as noted above, many design assessments simply look at system reliability (success probability), given a design basis challenge. The review of nuclear plant license applications did essentially this, culminating in findings that specific complements of safety systems were single- failureproof for selected design basis events. Going well beyond this, WASH-1400 modeled scenarios leading to large radiological releases from each of two types of commercial NPPs. It considered highly complex scenarios involving success and failure of many and diverse systems within a given scenario, as well as operator actions and phenomenological events. These kinds of considerations were not typical of classical reliability evaluations. In fact to address public risk, WASH-1400 needed to evaluate and classify many scenarios whose phenomenology placed them well outside the envelope of scenarios normally analyzed in any detail.

The team adapted methods previously used by the Department of Defense and NASA to predict the effect of failures of small components in large, complex systems. The overall methodology, which is still utilized, is called probabilistic risk assessment (PRA).

The team first identified events that could potentially lead to core damage. Event trees were then used to delineate possible sequences of successes or failures of systems provided to prevent core meltdown and/or the release of radionuclides. Fault trees were used to estimate the probabilities of system failures from available data on the reliability of system components. Using these techniques, thousands of possible core melt accident sequences were assessed for their occurrence probabilities. The public health and economic consequences of the identified severe accidents were estimated using computational models that were developed as part of the overall effort.

The Reactor Safety Study indicated that risks to the public from potential U.S. nuclear power plant accidents were small compared to other risks encountered in a complex technological society. Other sources of risk that were compared in the study included fires, explosions, toxic chemical releases, dam failures, airplane crashes, earthquakes, tornadoes, and hurricanes.

It was assumed that there are 100 power reactors and that they all have risks equal to the average risks for Surry and Peach Bottom. There is no evidence to support this assumption; however, the other 98 reactors would have to be orders of magnitude worse than Surry and Peach Bottom for the general conclusions to be rendered invalid. While the risks from nuclear power appear to be very low, the Reactor Safety Study did indicate that core melt accidents were more likely than previously thought (approximately 5x 10^{-4} per reactor year for Surry and Peach Bottom), and that light water reactor risks are mainly attributable to core melt accidents. The Reactor Safety Study also demonstrated the wide variety of accident sequences (initiators and ensuing equipment failures and/or operator errors) that have the potential to cause core melt. In particular, the report indicated that, for the plants analyzed, accidents initiated by transients or small LOCAs were more likely to cause core melt than the traditional design-basis LOCAs.

The risk to the surrounding population over the 40 years following a reactor accident is presented in the following table:

		Consequences	
Chance per Reactor-year	Early Fatalities	Early Illness	Latent Cancer Fatalities (per year)
One in 20,000	<1.0	<1.0	<1.0
One in 1,000,000	<1.0	300	170
One in 10,000,000	110	3,000	460
Normal Incidence	-	-	17,000

These results represented more than an order of magnitude reduction of the consequences estimated by the earlier WASH-740 report.

WASH-1400 was arguably the first large-scale analysis of a large, complex facility to claim to have comprehensively identified the risk-significant scenarios at the plants analyzed. Today, most practitioners and some others have grown accustomed to that claim, but at the time, it was received skeptically. Some skepticism still remains today. In fact, it is extremely challenging to identify comprehensively all significant scenarios. Methods have improved in some areas since the time of WASH-1400, but many of the areas considered controversial then remain areas of concern today. Completeness was one issue. Quantification, and especially quantification of uncertainties, was also controversial then and remains so today. Despite the early controversies surrounding WASH-1400, subsequent developments have confirmed many of the essential insights of the study, established the essential value of the approach taken, and pointed the way to methodological improvements.

In addition to providing some quantitative perspective on severe accident risks, WASH-1400 provided other results whose significance has helped to drive the increased application of PRA in the commercial nuclear arena. It showed, for example, that some of the more frequent, less severe IEs (e.g., "transients") lead to severe accidents at higher expected frequencies than do some of the less frequent, more severe IEs (e.g., very large pipe breaks). It led to the beginning of the understanding of the level of design detail that must be considered in PRA if the scenario set is to support useful findings (e.g., consideration of support systems and environmental conditions). Following the severe core damage event at Three Mile Island in 1979, application of these insights gained momentum within the nuclear safety community, leading eventually to a PRA-informed re-examination of the allocation of licensee and regulatory (U.S. Nuclear Regulatory Commission) safety resources. In the 1980s, this process led to some significant adjustments to safety priorities at NPPs; in the 1990s and beyond, regulation itself is being changed to refocus attention on areas of plant safety where that attention is more worthwhile.

NUREG / CR-2239 *"Technical Guidance for Siting Criteria Development"* (1982)

Following the TMI accident, NRC contracted Sandia National Laboratory to develop a technical guidance report for siting future reactors. Guidance was requested regarding (1) criteria for population density and distribution surrounding future sites, and (2) standoff distances of plants from offsite hazards.

The 92 plant scope of study was so large that rather than model the release from each plant separately, 5 types of accidents would be imposed on each plant. The accidents or "siting source term events" would be derived from the previous Reactor Safety Study (WASH-1400) and each SST event would be assumed identical regardless of the reactor size or plant design. Although the absolute numerical results may be questionable due to the arbitrary source terms, the relative impact of population density, weather, and evacuation times would be apparent for every site in the US.

SST1 – Core damage and loss of all containment after 1.5 hours. (Later found not credible.)

SST2 – Core damage and containment isolation bypass path after 3 hours.

SST3 - Core damage and 1% per day leakage of containment after 1 hour.

SST4 – Modest core damage and degraded containment systems.

SST5 – Limited core damage and full functioning of containment.

The outcomes were computed as mean value results. The results were also fitted to smooth probability curves that were extended out to 1 in 1000 probability weather conditions that would, basically, pipeline the radioactive plume to the nearest large city and then dump it all within the city by means of an isolated rainstorm that would increase consequences by a factor of about ten. The mean value is representative of the conservative expectations and is used here.

The results for most of the 92 reactor plants were similar due to a low population density. Using the SST1 model with a population density of 50 persons per square mile:

Early fatalities	= 47 to 140
Latent cancers	= 730 to 860

High population density sites had higher consequences, although not proportionally higher. This was a result of computing latent cancers based on the aggregate population dose rather than the individual dose. Thus, most of the latent cancers were in very large distant populations that had received very small individual doses.

Using the SST1 model with a NYC population density (Indian Point reactor - 42 million persons):

Early fatalities with summary evacuation	= 831
No evacuation	= 3580
Best evacuation	= 176
Latent cancers	= 8110 (0.06% increase over normal incidence)

For the more realistic release represented by SST2 events, the mean values from typical plants were:

	Early Fatalities	Latent Cancers
Peach Bottom	0	140
Surry	0	95
Sequoia	0	95
Grand Gulf	0	60
LaSalle	0	200
Indian Point	0.08	590 (0.004% increase over normal incidence)

As rules of thumb:
The typical radioactive release difference between SST1 events and SST2 events is 10 times.
The typical radioactive release difference between SST2 events and SST3 events is 100 times.
Rain will increase effects of the radioactive release by a factor of 10 times locally.

The insights gained from the NUREG/CR-2239 guidance would shape reactor siting decisions right up to today.

NUREG-1150, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, 1990

NUREG-1150 documents the results of an extensive NRC-sponsored PRA. The study examined five plants of varying designs to give an understanding of risk for these particular plants. Selected insights regarding classes of plants were also obtained in the study, and these were further developed by the plant licensees through the Individual Plant Examination program. The improved PRA methodology used in the NUREG- 1150 study significantly enhanced the understanding of risk at nuclear power plants, and can be considered as a replacement for the Reactor Safety Study.

The five nuclear power plants analyzed in NUREG-1150 are:

- Unit 1 of the Surry Power Station, a Westinghouse-designed three-loop reactor in a sub atmospheric containment building, located near Williamsburg, Virginia;

- Unit 1 of the Zion Nuclear Power Plant, a Westinghouse-designed four-loop reactor in a large, dry containment building, located near Chicago, Illinois;

- Unit 1 of the Sequoyah Nuclear Power Plant, a Westinghouse-designed four-loop reactor in an ice condenser containment building, located near Chattanooga, Tennessee;

- Unit 2 of the Peach Bottom Atomic Power Station, a General Electric designed BWR-4 reactor in a Mark I containment building, located near Lancaster, Pennsylvania; and

- Unit 1 of the Grand Gulf Nuclear Station, a General Electric-designed BWR-6 reactor in a Mark III containment building, located near Vicksburg, Mississippi.

The internal-event core damage frequency distributions from NUREG-1150 are shown below. The table shows the 90% uncertainty ranges along with the mean and median values.

Plant Name	Internal Core Damage Frequency Ranges per Year			
	Mean	Median	5th Percentile	95th Percentile
Surry	4.0E-5	2.3E-5	6.8E-6	1.3E-4
Peach Bottom	4.5E-6	1.9E-6	3.5E-7	1.3E-5
Grand Gulf	4.0E-6	1.2E-6	1.7E-7	1.2E-5
Sequoyah	5.7E-5	3.7E-5	1.2E-5	1.8E-4
Zion (retired)	3.4E-4	2.4E-4	1.1E-4	8.4E-4

The above table reflects core damage frequencies that are relatively low. Except for a particular sequence involving component cooling water at Zion (plant changes were subsequently made to address this sequence resulting in a mean risk of 6E-5/yr), there are no serious vulnerabilities that yield unusually high risk.

The frequency of core damage initiated by external events has been analyzed for two of the plants in NUREG-1150, Surry and Peach Bottom. The analysis examined a broad range of external events (e.g., lightning, aircraft impact, tornadoes, and volcanic activity). Most of these events were assessed to be insignificant contributors by means of bounding analyses. However, seismic events and fires were found to be potentially major contributors and thus were analyzed in detail.

The external-event core damage frequency distributions for Surry and Peach Bottom from NUREG-1150 are shown below. The table shows the 90% uncertainty ranges along with the mean and median values.

Plant Name	External Core Damage Frequency Ranges (Fire & Seismic)			
	Mean	Median	5th Percentile	95th Percentile
Surry	2.6E-5	1.4E-5	8.4E-7	1.4E-4
Peach Bottom	2.3E-5	1.3E-5	1.1E-6	6.4E-5

The various accident sequences that contribute to the core damage frequency from internal initiators can be grouped by common factors into categories. NUREG-1150 uses the accident categories depicted in the table below: station blackout, anticipated transients without scram, other transients, reactor coolant pump seal LOCAs, interfacing system LOCAs, and other LOCAs. The selection of such categories is not unique, but merely a convenient way to group the results.

Plant Name	Internal Initiators					
	SBO	ATWS	TRANS	RCP Seal	SG/IF Sys	LOCA
Surry	2.2E-5	1.6E-6	2.0E-6	-	3.4E-6	6.0E-6
Peach Bottom	2.2E-6	1.9E-6	1.4E-7	-	-	2.6E-7
Grand Gulf	3.9E-6	1.1E-7	-	-	-	-
Sequoyah	5.0E-6	1.9E-6	2.6E-6	-	2.4E-6	3.6E-5
Zion	9.3E-6	-	1.4E-5	3.1E-4	-	-

NUREG-1150 also computed consequences of core damage events and fitted them to risk curves in a manner similar to WASH-1400.

Cumulative latent cancer fatalities consequences and their accident frequency from NUREG 1150 are:

Chance per Reactor-year	One in 100,000	One in 1,000,000	One in 10,000,000	*50 Mi Pop*
Peach Bottom	0	2000	10000	706,500
Surry	0	2000	9000	1,805,500
Sequoia	60	5000	10000	942,000
Grand Gulf	0	300	2000	235,500
Zion	0	8000	30000	5,338,000

With an average 50 mile radius population of 1.800,000 the normal incidence of cancer fatalities would be 670,000 over that population's lifetime.