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PROPOSED RULE (NUREG-0956) (46)

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Chris Ryder
UNRC
Washington, DC 20555

Dear Mr. Ryder:

Please accept this letter including enclosures as my comments on NUREG 0956 Draft Report for Comment.

The report reasses the technical bases for estimating source terms. A dangerous bias in the report assaults this commenter's intelligence. The report is heavily biased toward reducing the existing design basis and present source term. Reducing the existing source term will justify minimizing safety thru neglecting needed backfits, changing EPZ from 10 miles to 2 miles or less, and any other cost cutting which the industry wants.

Reducing the source term is similar to a steamship running out of fuel burning its lifeboats for fuel. The growth of the nuclear industry can be linked to a steamship that run out of fuel. Growth in the nuclear industry has reached zero for new powerplant orders and the nuclear industry is looking for new financial incentives. Allowing the industry to "burn" safety to provide financial incentives is very like a steamship burning its lifeboats for fuel.

Eventually both will meet an accident without any resources to rescue them.

Sincerely,

Marvin I. Lewis, R.P.E

Marvin I. Lewis 1/3/86

D310
add: M. Silberberg 1130SS
D Ross 1130SS
Wm Olmstead 9604MNBB

JAN 13 1986 pd

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Ross, Olmstead, RIDS, Silberberg,

INTRODUCTION I

The NUREG 0956 depends heavily on data generated from the TMI 2 accident. This TMI data provides the basis for much of the source term developed in NUREG 0956.

However the TMI data has shown little dependability or reliability. Much of the TMI instrumentation did not work, and the extent of the accident still defies definition. A synopsis of the TMI accident follows to provide the reader with some background.

What really happened at TMI on March 28, 1979

A reactor core, which had only operated for 3 months, melted. For 5 years, the extent of the damage was not known to the public. Originally GPU, the owner/operator, reported 1% of fuel might have melted. Over 5 years later, samples and remote cameras showed 70% damage with extensive melting.

In the days after the accident, GPU & NRC announced that there was little danger to the public due to radiation release. GPU & NRC still maintain that there was little danger to the public--despite a seven-fold increase in cancers in areas surrounding TMI. (Aamodt Study)

Obviously, the pronouncements and announcements of the NRC and GPU do not fit with the actual facts of the accident at TMI2. Usually, the distance between what GPU & NRC stated happened and what actually happened would be a mere annoyance. Discrepancies continue to be litigated. (NRC & CLI 85-18 12/18/85)

However, nuclear power plants will be designed and "backfitted" using the "source term." The "source term" is the amount of radiation that escapes during an accident such as that which occurred at TMI2. If the industry relies on very little radiation getting out in a major accident, the industry can then justify less expenditures for safety. Less expenditures for safety will mean that there will be fewer backfits for safety and fewer safeguards designed into future and operating plants.

Indications of large radiation release the Aamodts of Coatsville, Pa. have intervened at TMI before the NRC.

Recently Mrs. Aamodt stated before a congressional committee that GPU reported data from a radiation survey which had not been done. GPU stated that off-site radiation had not reached levels which would cause harm. These statements related to GPU's radiation surveys. At least one of the surveys, upon whch GPU based its assurances, had just not been done!

Other surveys remain unreported and out of the public's view. Still other questions just linger. The NRC questioned the adequacy of the offsite TLD array for dose assessment. Sensitivity of array geometry to dose assessment still lingers as an unknown. (B&W #622 Item 12)

PA. NATIONAL GUARD SURVEY

The PA. National Guard were manning helicopters with NBC (nuclear-biological-chemical) NCO's (non-commissioned officers). The Guard was performing a radiation survey using RADIAC meters. Personnel carried personal dosimeters also. The helicopter survey was performed over and around the TMI site during the accident.

Many RADIAC meters were reading high levels of radiation. Upon reporting this high level of radiation, the N.G. NBC NCO's were told that the high readings were due to instrument malfunction. One NCO threw his personal dosimeter down on a table, and told the reporting authority, "Well, read this." The personal dosimeters had also read very high levels of radiation. To this day, the data from the actual RADIAC readings in the National Guard helicopter survey remain out of the public's view.

Since the RADIAC readings from the National Guard survey were high, the readings would contradict the small "source term" that the NRC and GPU have assumed for the TMI accident.

The concealment of these very high RADIAC readings shows that the small source term used by the NRC for the TMI accident has little factual basis.

The source term in the NUREG 0956 is also based partially on the NRC estimate of the TMI accident source term. Since the concealment of very high readings produced an erroneously small TMI source term, the source term developed in NUREG 0956 is also erroneously small.

Therefore, design based on the source term in NUREG 0956 provides inadequate safety.

THRU THE EYES OF MY CAMERA

Bob Forsyth, Middletown area resident and Civil Defense worker wrote a detailed brochure on his experiences during the TMI accident on a pamphlet entitled, "Thrugh the Eyes of My Camera." He reports several touch down points for the plume at 50 mp/hr. He also reports readings up to .4 rem/hr or 400 mr/hr. which he personally measured and observed.

Considering these very high readings, the NRC's estimate of 85 millirem to the "most exposed individual" in the plume exposure pathway appears very suspect. At 400 millirems per hour, an individual, on Aspen St. at 3:15 p.m., 30 Mar. 79, would receive 85 mrem in only 14 minutes. 85 mrem is the NRC estimate for "most exposed."

Mr. Forsyth has reported ground level readings far in excess of any reported by any other agency. His report remains unchallenged.

Nonetheless, the NRC has not used the NG or Mr. Forsyth's readings and measurements in developing a source term in NUREG 0956. Because NUREG 0956 ignores important data it provides inadequate assurance of safety.

PA. DEPT. OF HEALTH

Recently, the PA. Department of Health released a study which attempted to rebut the Aamodt study showing a 7 fold increase in cancer rate in 5 and 10 mile radius of TMI. Several academicians expert in health statistics took exception to the methodology of the Dept. of Health study, and the principal author admitted an "error." (Harrisburg Patriot 10/6/85) The error was that the population figures were inflated with many people who lived outside the 5 and 10 mile study areas. The inflated population reduced the observed cancer rate to expected levels.

Conversely, a study conducted by Marjorie and Norman Aamodt of Coatesville showed a 7 fold increase in cancer in areas exposed to the plume from the Mar. 29, 79 TMI accident. The Aamodt study has successfully withstood challenge, unlike the Health Department Study.

"NO SIGNIFICANT HAZARD"

A 7 fold increase in cancer in the area of TMI makes NRC and industry pronouncements of "no significant hazard" meaningless. At a minimum, the NRC and GPU should attempt to find the cause of the increased cancer rate. A 7 fold increase in cancer rate is a significant hazard to reasonable people. Pronouncements of " no significant hazard" make little impact until the cause of the increased cancer rate is found and shown to lack causal relationship to the TMI reactor.

Conversely, the increased cancer rate may very well have a direct relationship to the TMI accident. The relationship can stem from more radiation exposure than previously reported due to lack of reporting high radiation readings or due to very low radiation levels being more damaging than previously assumed. (Radiation & Human Health Gofman p. 134)

CAUSAL RELATIONSHIPS

There are clusters of high cancer rates in the TMI area. These cancer rates occur in the areas close to a radiation accident. The public perceives cancer rates as a legitimate concern, and the public perceives a causal connection between the accident and cancer in the area. Until an analysis uncouples the TMI accident from the area cancer rates, the public will continue to perceive a causal connection.

Thomas Cochran of the Union of Concerned Scientists speaking to the NRC Commissioners on 11/19/85 explained the public's perception:

"But you (NRC) are going to get a lot of citizen concern over this (TMI cancer rates) because the citizens still believe it is TMI related but the (Dept. of Health) analyses doesn't really get uncoupled from whether there may be something there or not."

Stated another way, the public perceives that the NRC or GPU assumes that there is no significant danger and relies only that data which supports the "no significant hazard" criteria. The same assumptions and weighting produce the source term in NUREG 096. The source term in NUREG 0956 depends upon data which supports a reduction in source term and ignores data which show large radiation releases during the TMI accident.

ELIZABETHTOWN SCHOOL DISTRICT

Some unofficial but very damning data has come out of the TMI area. The NRC and GPU have given this data little weight. An example of hard data given little weight by the NRC and GPU is the Elizabethtown Area School District (600 E. High St., Elizabethtown, Pa. 17022) Cancer Survey. A 7 fold increase in cancer rate over the expected rate has assaulted the Dept. of Health statisticians without effect.

LACK OF ATTENTION TO DATA

This lack of attention to increased cancer rates promotes the suspicion in the minds of the public that the increased cancer relates to the TMI accident and that the lack of attention continues an ongoing coverup. Many areas of concern remain years later. TMI Cleanup Programmatic Environmental Impact Statement discussed some of these concerns.

However the PEIS did not address these concerns adequately. For instance, worker radiation exposure has been six times the total exposure predicted in the PEIS to complete cleanup, but the cleanup continues. The PEIS ignores radiation exposure from the disposition of wastes once DOE accepts ownership or responsibility. Nonetheless, the public radiation exposure goes on despite the transfer of ownership. The NRC is handling the TMI accident by neglecting many continuing exposures. This negligence continues into the development of a source term. One example: when the utility declares an emergency is over, the utility does not have to "count" any further releases into a source term, but the releases continue.

NUREG 0956 needs to specify the cut off or last release which must be counted into the source term. Subsequent releases from the damaged reactor years after the accident increase the source term. The NRC should abolish an artificial cut off point beyond which releases caused by an accident need not be included in the source term. Instead, regulation must require that all releases and exposures related to an accident add into the source term without consideration of the passage of time.

CONTROVERSY ABOUT IODINE RELEASES

The NRC calculated a very small release of radioactive iodine from the TMI release. EPA has reported a release of 27 Ci of radioactive iodine. Japanese authors have stated releases or radioactive iodine as high as 5 figures. (An Examination of Pathways and Source Term of Released Radioactive Iodine in the Early phase of the TMI Accident, S. Kume, H. Koide, T. Seo Kyoto Univ. Japan Nov. 9, 84). Various explanations abound why the release of iodine was so small compared to the original core inventory during the accident. Due to the limited early surveys and due to the contradictions discussed previously, a large amount of radioactive iodine may have escaped without detection. Before the NRC uses a source term based on a small release of radioactive iodine, the NRC should make every effort to substantiate its use of a small iodine release during the TMI accident.

The NRC is doing very little to substantiate a small iodine release thru actual data. This substantiation would be very easy to do as follows: Only 15% of the radioactive ^{129}I and 2% of the radioactive ^{131}I of the core inventory were accounted for in the TMI-2 reactor building. (GEND 042 "TMI-2 Reactor Building Source Term Measurements: Surface & Basement Water & Sediment." p. 76) The remaining 85% of ^{129}I and 98% ^{131}I must have either escaped or remain in the reactor core. That which does not remain in the reactor core or in the reactor building must have escaped.

Very careful analysis of the iodine remaining in the degraded core and buildings would give an excellent estimate of that iodine which was released.

$$I_{\text{released}} = I_{\text{originally in core}} - I_{\substack{\text{remaining}\\ \text{in reactor}\\ \text{building or}\\ \text{on site.}}} - I_{\substack{\text{remaining}\\ \text{in degraded}\\ \text{core.}}}$$

Since I_{129} is chemically similar to I_{131}, the longer lived iodine will still be present in sufficient quantity to make these determinations. NRC can then safely assume that the shorter-lived isotape acted the same as its longer lived sister.

However, instead of the NRC analyzing the degraded core iodine inventory very carefully, Mr. Wm. Travers of the NRC has testified that GPU has requested exemptions to careful analysis and accounting of the degraded core materials. (TMI 2 Citizens Advisory Panel Oct. 85 meeting; also letter Travers to Standerfer 11/12/85 and letter Snyder to Standerfer 10/17/85). These exemptions requested by GPU will destroy any chance of verifying the magnitude of the iodine release from the TMI 2 accident. The iodine contribution to the source term will depend on unsubstantiated assumptions.

The unsubstantiatedly small iodine release in NUREG 0956 requires verification thru careful analysis of the degraded core before acceptance of any change in present source term.

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The response here concerning liberation of iodine is most telling.

"The change in vapor composition in the containment due to hydrogen burning is accounted for, insofar as reduction in H_1 concentration and increase in H_2O is concerned. Other chemical changes in the vapor are not accounted for, nor in vaporization of aerosol particles and potential changes in their composition computed, although there is a possibility of liberation of iodine in this process. This represents a shortcoming in the analyses and the possible impact on the mass fractions has not been assessed."

The NUREG 0956 source term ignores the thermodynamic radiochemistry of the vapor composition in the containment. For instance, large amounts of hydrogen enter the containment but do not enter into consideration in NUREG 0956. This added hydrogen affects the failure mode analysis and is discussed below.

Secondly this "shortcoming" also extends to the treatment of CsI. The form of I2, I or CsI is dependent on the H:0 ratio (BMI 2104 VII p. 38). After a detonation with H_2 from sources ignored in BMI 2104 and NUREG 0956, the H:0 ratio will be very high in H_2. The H_2 will provide an excellent mix to produce volatile species. The volatile I species can then easily exit any opening in containment including unplanned atmospheric dumps thru damaged or degraded once thru steam generator tubing.

All this adds up to a very inadequate treatment of iodine in NUREG 0956.

ACTION OF HYDROGEN ON IODINE

The actual accident can provide hydrogen. Also hydrogen from on site storage can enter the primary thru a let down line from the make up tank. Both forms of hydrogen are chemically identical. All the hydrogen will try to reduce CsI to free I within thermodynamic restraints.

Added to these hydrogen considerations, recent NRC studies show that CsI can form free I in high radiation. (Inside NRC 11/11/85; Nucleonics Week 11/7/85)

Most likely, a large amount of free I will escape in a major accident. Justification remains that a major accident will release large amounts of free iodine and the small release of radioactive I at TMI seems either a fluke or an outright error hiding a large release.

CONTINUING CONTROVERSY ABOUT I DATA.

Although the nuclear industry and the NRC have clung to a belief in a small radioactive iodine release from the TMI accident, GPU and the NRC based their 15 Ci of radio iodines on very limited and flawed monitoring. The iodine measurements depended on wind direction, location of instruments, and reliability of workers and volunteers. As stated in Rogovin Report (Vol. 1), the 15 Ci which were later revised several times derived from calculations and assumptions.

In contrast iodine measurements require many samplings and then only provide an estimate. In GEND 042 EG&G TMI 2 bdng. source term. Page 28 last paragraph, differences of iodine concentrations in different areas range over a factor of 10.

Now here is the nub of the problem of believing NRC and GPU estimates of iodine releases. The iodine concentrations measured within the auxiliary and containment buildings vary over a factor of 10 from location to location. These samples can easily be rechecked and kept in archives for later reverification. Still these samples vary over a factor of 10 from location to location.

Conversely GPU and the NRC insist that only 27 Ci of radioactive iodine escaped in the accident. The NRC and GPU insist that the figure is a maximum, insist that the assumptions and calculations are exact, and that the data suffices despite inadequate, failed and absent instrumentation.

Again facts and figures emphasize that a large radio iodine release is most likely in a major accident and that a small radio iodine release developed from improper reliance on inaccurate instrumentation, mythical surveys, and pure invention.

PLUTONIUM, AMERICIUM AND CERIUM:

The actinides present a special problem at TMI for several reasons and affect the source term. Plutonium, americium, and cerium aerosols did not significantly contribute to aerosol releases at TMI or in the NUREG 0956 source term. A sample of aerosol analyzed by Inhalation Toxicology Research Institute, Lovelace Biomedical and Environmental Research Institute in Albrequerque, N.M. (Characterization of an aerosol sample from TMI reactor auxiliary building" Kanapilly G.M., et al. 1981) contradicts many assumptions leading to a minimal contribution of actinides to the NUREG 0956 source term. The auxiliary building sample taken years after the accident shows PA., A.M., auxiliary building air.

Since plutonium and americium exists in the auxiliary building air, these isotopes could have existed from the auxiliary building before, during or after the accident as an aerosol.

This presents a serious challenge to the accuracy of the NUREG 0956 source term--which is in part based on the assumed releases during the TMI accident. Also, the completeness of the monitoring of nuclides actually released during the accident lacked any mention of plutonium and americium releases. Further, the residents of the TMI area remain ignorant of any insult to their health which arises from the presence of plutonium and americium.

TMI Personnel supplied the aerosol sample well after the accident. The actinide activity released during the accident thru the auxiliary building air should exceed the actinide activity found on this analyzed sample taken well after the accident by a large margin. The NUREG 0956 source term needs a corection desparately to include plutonium and americium releases. The actinide activity in an auxiliary building filter sampling taken well after the accident challenges many assumptions concerning release path, aerosol activity and mobility, and isolation assumed both by the accident investigators and NUREG 0956 for actinides.

The finding of actinides on an auxiliary building air sample demands a comprehensive investigation answering many questions including:

1. When did actinides first enter the auxiliary building atmosphere: before, after or during the TMI accident?

2. What was the total load of actinides in the auxiliary building? Total released to outside air? or water?

3. What harm did these actinides do? To whom?

Many release paths exist without catastrophic containment failure.

NUREG 0956 reports failure modes. Failure modes directly affect the amount of radioactive releases. Failure modes involve containment integrity. If the integrity of the containment fails, the containment allows radioaction to escape into the air and water.

HYDROGEN CAN CAUSE CONTAINMENT FAILURE.

One mechanism of containment failure involves hydrogen. Hydrogen either ignites or detonates increasing the pressure to cause the containment to fail. NUREG 0956 and the American Physical Society APS study (page S97) investigated the failure mechanism caused by hydrogen only generated by the reaction of zirconium and steam during an accident.

Other sources of hydrogen do exist in a nuclear reactor both during operation and in an accident. Some of these other sources are small. Effective corrosion control and PH balance minimizes the amount of hydrogen generated by electro chemical or galvanic potentials.

However, hydrogen, injected into the make up tank, migrated to areas of concern during the TMI accident. Hydrogen was injected into the make up tank at TMI to provide false lease rate readings according to the testimony of a technician. (Order CLI-85-18 12/18/85 page 2) This raises grave concern about improper procedures.

First hydrogen in make up tanks have produced explosions in piping "upstream from the make-up tank." (NUREG/BR-0051 Vol. 4 #3 Mar. 82 page 9) Some large hydrogen explosions have caused damage in unexpected areas. (IE Info Notice 84-80 Nov 8, 84. Page 1 Ranco Seco I) Also there are many uses for hydrogen at TMI: hydrogen surge system, corrosion control and hydrogen side seal oil pumps. These uses require large tankage of hydrogen to be stored at TMI. Reserve tankages have connections to the make up tank. The make up tank connects with the "let down line." During the accident, the "let down line" was not isolated. (APS Study II. A.7 last sentence 3rd) This means that during the TMI accident, hydrogen introduced into the make up tank had a "unisolated" route into the primary coolant system and out thru the PORV to the containment. GPU/NRC ignored whether any hydrogen followed this route, increased the hydrogen concentration in containment or could cause containment failure due to increased detonation pressure in future accidents.

Personnel heard two "bumps" or "thumps" at the time that instruments recorded a pressure rise to 27 psig. The instruments could not record a pressure rise of detonation or ignition accurately as the reaction time of the recorder was too slow. The 27 psig. reading is probably much lower than the actual instaneous pressure.

Also the report of 2 "bumps" suggests that hydrogen continues to enter the containment after the first indication and that the "bumps" were detonations or localized ignitions. In either case, the potential damage would be greater than that assumed in NUREG 0956 from hydrogen.

The hydrogen effects in NUREG 0956 and BMI 2104 needs revision in light of this new source of hydrogen.

Further, hydrogen introduced into the make up tank could travel thru a let down line into the collant and thru the relief valves into the containment in quantity and over long periods of time. Hydrogen could build up and detonate or ignite many times in the months proceeding the accident. The lack of attention given to the "thumps" during the accident suggest that the operators were injured or just used to hearing these "thumps." These "thumps" and lack of attention to them suggest that these "thumps" had occurred enough to warrant little or no attention.

Therefore the hydrogen could have entered the containment, caused "thumps" and endangered the containment integrity for months before the accident.

READY & ABLE TO MELT THRU RPV WALL

The source term defines the amount of radiation which escapes during a major accident. The NRC has viewed the TMI accident as "the" major accident of record in a commercial nuclear power plant.

However, some very fortuitous circumstances arose during the TMI accident which minimized the amount of radio active releases. One of these circumstances involves a "B pump transient" at 174 minutes into the accident. Subsequently and consequently, the B pump transient caused a non-coolable geometry to shatter extensively. ("Thermal Hydraulic Features of the TMI Accident" INEL/EG&E B Tolman et al Conclusions and Summary) The shattering of the non-coolable geometry avoided what scoping calculations indicate, "scoping calculations indicate that if the core material were to rapidly flow downward onto the reactor vessel, melt thru of the vessel wall would occur within several minutes." (IDCOR Technical Summary Report "Nuclear Power Plant Response to Severe Accident" Nov. 84)

Obviously, the TMI accident had progressed to a thermal-hydraulic stage adequate to melt thru the reactor pressure wall. A fortuitous albeit unexplained, B-pump transient avoided the melt thru. Unexplained B pump transients or other unexplained circumstances cannot provide assurance to avoid vessel wall failure in future accidents. The conservative approach requires assuming vessel wall failure in a major commercial power plant accident.

The NUREG 0956 source term assumes that 75% of the fuel must melt before the entire core falls into bottom of reactor vessel. From the circumstances of the TMI non-coolable geometry, a much smaller, but very thermally hot, core section could fall into the bottom of the reactor vessel. If the non-coolable geometry survives the drop, the non-coolable geometry would eventually melt thru the bottom head. This scenario produces extensive complications:

1. There would be corium-concrete interactions.

2. Fuel remaining in reactor would continue to heat and melt causing difficulties not investigated in NUREG 0956. One of these difficulties is high pressure ejection of molten fuel (BMI-2104 VII p. 28) This scenario provides a supportable basis for including H pressure ejection of molten fuel.)

3. Fuel remaining in reactor would pump heat, hydrogen, and pressure into the containment.

4. Any attempt to inject coolant into reactor at this time would increase pressure in the containment endangering the integrity of the containment.

5. Recriticality becomes an immediate possibility. The core can preserve enough integrity to go critical if boration of all coolant, including building sprays are inadequate.

6. Under these circumstances and considering hydrogen detonations mentioned earlier, pressures in the containment could easily exceed the containment pressure capability. (NUREG 0956 C.1-7)

7. NUREG 0956 source term does not include the above scenario. Therefore the meltdown model in BMI-2104 as used in NUREG 0956 is deficient. Any change in source term requires an adequate melt down model unlike the deficient melt down models in BMI 2104 and NUREG 0956.

EXISTING PENETRATIONS OF RPV WALL.

Many complications arise when considering RPV wall failure. The reactor pressure wall is not a continuous unbroken sheet of steel. There are many penetrations in the lower plenum which receives the molten core, 70 air and instrument lines penetrate the steel. How these existing penetrations would react to a molten core needs definition. Similar penetrations dot the entire reactor. All the penetrations will degrade over time and when exposed to hydrogen ignitions or detonations.

The many RPV wall penetrations provided ways for the RPV wall to fail during an accident. Very little consideration of the above complications have entered into the analysis in BMI 2104 or NUREG 0956 source term.

Reference: Burns & Roe drawing General Arrangement - Reactor and Control Building Area Section "A-A" TMI-2.

DISCUSSION OF BMI-2104

The source term in NUREG 0956 depends heavily on the data and analysis in BMI 2104. The best summary of the BMI 2104 data occurs as a response to a comment on aerosols on P 31, Vol. VII.

"This represents a shortcoming in these analyses and the possible impact on the released mass fractions has not been assessed."

This response summarizes much, even most, of in BMI-2104. Also the shortcomings trend in the direction of estimating source term on the low side. The result predicts a source term which is much lower than any reasonable approach achieves. As stated on Page 2-3 Vol. I, "The intent of this work to produce . . . best estimates of source term."

The question becomes "best estimates" for whom. Clearly, BMI-2104 and NUREG 0956 assume "best estimates" refer to smallest estimate. This bias shows in the 14 member Invited Peer Review Group members. 9 work for the nuclear industry. 5 work in academia, but academia depends on grants and contracts from the nuclear industry. Independent experts or environmental groups do not appear as invited peers. Although "observers" attended peer review meetings, the weight given the "observers" comments depended heavily on the basis of Batelle and NRC.

Throughout the responses, comment after comment receives temporizing or rationalizing responses such as:

"The influence of this decay heat is ignored...." P. 6 Vol. VII.

"Chemical forms are not known... Paucity of experimental data and complications of thermodynamics... P. 8

"This was crudely modeled... P. 6

"Experimental programs are in place which <u>will</u> provide...

"There is considerable uncertainty P. 9

"It was assumed (not analyzed) that plugging was not significant... P. 10

"Resuspension... not fully understood... P. 12

"Deposition is not well understood...

"More throughout theoretical approach would be preferred in SPARC code...

"Foaming unlikely in absence of surfactants...

(not true--surfactants arise from furmanite and other materials used in RCS.)

"We do not have a means of quantifying the degree of such further impairment of fission product scrubbing... P. 16

The admissions of inadequacies and deficiencies go on and on. The answers to comments admit that the report is inadequate and deficient. Many of the inadequacies and deficiencies spotlighted in Vol. VII have plagued reactor design engineers for decades.

Allowing reactors to be designed to a mythically small source term, will reduce safety. Further any logical approach prohibits reducing the source term now or in the foreseeable future. The TMI accident has provided little dependable data. Much of the data which existed at TMI has succumbed to expediency. NRC granted GPU exemptions to allow transportation of wastes off the island without extensive accounting and physical data which might have provided some of the needed parameters to define the source term. Presently, GPU is attacking the degraded core with drills and tamping tools. One barrel of core fell into the reactor while employees tamped the wastes tightly into the barrel. This mishap endangered irreplaceable physical information applicable to the source term. (Harrisburg Patriot 12/18/85; 12/19/85 & 12/20/85)

Considering that the procedures for peer review of BMI-2104 are biased and that the physical evidence applicable to the source term has been destroyed at TMI, release of NUREG 0596 fails any test or appearance of fairness or concern for health and safety of the public.

References:

Letters	
Snyder NRC to Standerfer GPUNC	10/17/85
Lewis to Snyder NRC	10/24/85
Lewis to Travers	Undated
Travers NRC to Standerfer	11/12/85
Travers to Lewis	12/23/85

Lewis to Travers undated reply to 12/23 letter.

INCLUSION OF ACRS COMMENTS ON SOURCE TERM

The ACRS has commented on NUREG 0956. Many of their comments point to the many deficiencies in the source term.

The ACRS concludes, "the report can best be characterized as a status report for a task well begun but far from conclusion."

I agree. I also agree with the ACRS that some documents are not readily available. Some documents are near impossible to find.

Reference: NRC-ACRS Comments on NUREG 0956.
Letter Ward to Palidine 12/12/85

THE INTEGRITY OF THE CONTAINMENT

The NRC has based much of the assurance of safety on the integrity of the containment. Integrity of the containment refers to isolation of radiation from the air and water outside of the containment building. Containment design attempts to stop radiation releases even if the reactor pressure vessel wall fails.

However, experience shows that radiation will escape containment as presently designed and operated. Some of the experiences include "the June 9, 85 Davis Besse event demonstrated that the PRA analyses were wrong. Davis Besse had a loss of all feed water that involved the failure of 14 separate pieces of equipment. (See NUREG 1154)" Luckily none of the 14 failures affected containment integrity this time.

Six years before, Ed Wallace GPU Licensing Manager wrote in his notes, on the TMI accident, "NRC stated water hammer frequently ocurs on trips at other plants both BWR/PWR. The water hammer bothers Lopinski that this event would always lead to atmospheric dump which without SG tube leaks would have offsite releases." (B & W Exhibit 624 Page W 36962) (Also B&W Exhibit 719 W. Zewe primary to secondary leak before isolating.)

Of course, the OTSG tube leaks constituted a large item of litigation at the TMI #1 restart hearings. All the ingredients, water hammer and OTSG tube leaks, existed at TMI for offsite releases. Luckily the reactor pressure vessel remained intact despite fuel temperatures which could have breached the RPV wall. The reason that the RPV wall remained intact at TMI is still partially unexplained as the fuel temperature and geometry was sufficent to lead to failure. An unexplained 'B' pump transient caused a non-coolable geometry to shatter, which shattering lead to a coolable geometry.

The coolable geometry allowed the fuel to cool to temperature which preserved the RPV wall.

Had the RPV failed, the Lopinski concern, "atmospheric dumps," would have released large amounts of radiation from the TMI accident.

The containment building isolates radiation from escaping to the outside environment. The ability to contain radiation depends on valves, tubing and equipment to work as designed. Again and again, equipment has failed to work as designed.

<u>Reference</u>: [1] <u>"Thermal Hydraulic Features of the TMI Accident"</u>
B. Tolman et al INEL Idaho Falls A.C.S. May 85.

QUALITY CONTROL AFFECTS SOURCE TERM:

One concern of source term prediction requires that the analyst is privy to enough information about the severity and probability of the design basis accident to predict accurately. Unhappily, the information about the accident scenarios is highly flawed. The NRC and the nuclear industry boast excellent quality control and assume that if one engineered safety function fails, the backup for that failure will work. (NRC single failure criterion)

The TMI accident, Salem anticipated accident without scram, thru the recent Davis Besse failure of all feedwater demonstrate that the quality control in the nuclear industry has failed miserably. The TMI accidents had a history of major problems before startup. The "Loss of Feedwater Flow Leading to the Accident of March 28, 1979" dated Sept. 1, 79 (B&W Exhibit 343) reported many deficiencies known long before start up: Caustic and acidic resin regeneration process fluids were drained into main control panel for polishers. (Page A4) Polisher modifications--"some were redundant and some were counter productive." (Page B6) "This electrical alteration in itself would render all 8 beds incapable of coping with either loss of air or control power."Page B7

"gross lack of system knowledge, attention or both"

"lack of total circuit comprehension"

"no record of positioner calibration" Page B8

"system response to failure modes was not checked" Page B8

"The presence of conflicting real and circumstantial evidence currently prohibits the establishment of an overall cause/effect relationship." Page B14.

The list of contradictions in the TMI accident goes on and on.

Test wells on TMI showed high tritium levels. TMI Restart Hearing Board questions on tritium in the TMi test wells rested without definitive answers. These high tritium levels indicated poor management practices.

Horrendous maintenance shippages failed to raise any action from NRC staff or GPU. TMI Action contended that maintenance shippages of over one year constituted management incompetence and a danger to the public. Eventually, the TMI #1 Restart Board required only that GPU perform a few TMI related actions previous to restart. The Maintenance practices, which contributed to the TMI accident, continue.

All this reinforces the conclusion that lack of adequate maintenance in the nuclear industry presents a continuing danger to the safety of the public.

The NRC Staff, the nuclear industry, BMI 2104, and NUREG 0956 all assume an adequate maintenance program. The "single failure criterion" makes adequate quality control design, and maintenance a policy. The facts and the reality shout that the maintenance, quality control, and design are lacking.

The source term needs reworking to show the great likelihood of a major accident with above design basis consequences. The inadequate maintenance, quality control and design in contrast to

NRC policy and regulation also raise the possibility of an accident more severe than the design basis accident.

MY PAST CONTRIBUTION TO SAFETY

I have written hundreds of comments on many actions of the Nuclear Regulatory Commission. Regulations contain several of my sugestions. I have also recommended that the NRC appoint antinuclear activists to positions such as commissioner or member of the Advisory Committee on Reactor Safeguards. Some of the names, that I suggested, were Judith Johnsaud, Ph.D., Chauncey Kepford, Ph.D. and Wm. Lochstet, Ph.D. Any of these good people would have aided the NRC to perform its regulatory duty a bit better.

In view of the highly deficient way that the present source term has issued, I now suggest at least a partial cure. I propose that the NRC needs a comittee of anti nuclear activists and interested citizens. The new committee will try to protect the health and safety of the public. Without ties to vested nuclear interests and having its own interests tied to the health and safety of the public, the members will try their best to protect the public as legislated by the Atomic Energy Act.

I am also proposing that I serve on this new committee or help by providing applicants for membership. I cannot serve full time, but I can serve about 20 hours/week. I would really like to hear from the NRC about my recommendations.

-Misc Notice (NUREG-0956) (47)

MARTHA DRAKE
230 FAIRVIEW
PETOSKEY, MICH. 49770
(616) 347-4645 OR 2596

'86 JAN 10 P12:33

Jan. 5, 1986

Docketing and Service Branch
Secretary of the Commission
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

RE: NUREG-0956

Dear Sirs:

I would like to object to your lowering the source term standards for radiation.

The American Physical Society has reviewed the calculations and concluded that the research cannot yet be regarded as adequate. This is such a serious matter that I feel the research should be credible to all parties, that it should be published in a accepted journal and be submitted to peer review and peer acceptance before standards are changed based on it. Once the radiation is allowed into the environment it cannot be reclaimed. We cannot do this to our children's future.

The new codes do not model external events such as earth quakes and sabotage.

I understand new areas of uncertainty have been identified since the study was finished.

I hope that you will reconsider and submit the conclusions to more scrutiny and peer review.

Thank you,

Martha Drake

Martha Drake

DS10
add: M. Silberberg, 1130SS
D Ross 1130SS
Wm Olmstead, 9604 MNBB

pd

1/0

DOCKET NUMBER
PROPOSED RULE PR-Misc Notice
(NUREG-0956) (48)

GERALD A. DRAKE, M. D.
511 WAUKAZOO AVE.
PETOSKEY -:- MICHIGAN
TELEPHONE DI 7-8300

'86 JAN 10 P12:33

Jan. 5, 1986

Docketing and Service Branch
Secretary of the Commission
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

RE: NUREG-0956

Dear Sirs:

I wish to express the following objections to NRC plans to lower standards for nuclear plant operation based on recent studies using new computer codes:

1. An American Physical Society panel has concluded that the research is not adequate.
2. Computers are not immune to the biases of the programmers nor operators. For example a project done by a couple of pastors using computers concluded that the Bible is devine. An old engineering rule says, "what goes in must come out".
3. In such complex studies computer models cannot be expected to properly include or weigh all the uncertainties. Human error goes into everytning we design, build and operate including computers.

The conservatism of WASH-1400 should be preserved.

Yours truly,

Gerald A. Drake, M.D.

DS10
add: M. Silberberg, 1130SS
D. Ross, 1130SS
Wm. Olmstead, 9604MNBB

pd

1/0

PROPOSED RULE PR-~~Misc~~ Notice (49)
(NUREG-0956)

DOCKETED

'86 JAN 10 P12:32

DOCKETING & SERVICE BRANCH

To NRC - Secretary of the Commission
From H.V. Lynde Jr.
Subject NUREG-0956

I have recently become aware of a proposal to change the estimate of source terms via NUREG-0956. Unfortunately, I have not been able to obtain a copy of NUREG-0956. I expected to find a copy in a major library close to me (located in Nashua, N.H. - N.H.'s second largest city) but the library was unable to help me.

Therefore, I request a copy of NUREG-0956 and an extension of time to comment. My address is:

Harold V. Lynde Jr.
Mercury Lane
Pelham, N.H. 03076

It is my understanding that the effect of NUREG-0956 would be to reduce estimates contained in WASH-1400. This is going in the wrong direction because of the uncertainties of WASH-1400, other estimates more severe than WASH-1400 and the occurence of TMI.

Also, it is my understanding that NUREG-0956 has been found lacking by the American Physical Society and has not had adequate nor rigorous peer review.

Under the present circumstances surrounding NUREG-0956 the NRC would not be fulfilling its designated role unless it allowed for a rigorous examination of the contents and the basis of NUREG-0956 and allowed sufficient time for comment.

DS10 Add: M Silberberg, 1130SS
D Ross 1130SS
Wm Olmstead, 9604MNBB.

JAN 13 1986 pd

1/0

PROPOSED RULE PR-Misc Notice (52)
(NUREG-0956)

DOCKETED USNRC

'86 JAN 10 P12:34

EPRI

Electric Power
Research Institute

January 6, 1985.

Dr. Denwood F. Ross
U.S. Nuclear Regulatory Commission
Willste Building
Mail Stop 1130SS
7915 Eastern Avenue
Silver Springs, MD 20912

Dear Dr. Ross:

The opportunity to review draft NUREG-0956 is appreciated. It is clear that a significant effort went into its preparation. It is a characteristic of reviews of this type that negative aspects receive more emphasis than the positive ones. The first part of this letter will cover general comments. An addendum will cover more specific comments which are referenced to NUREG-0956.

While the draft document attempts to establish the technical basis for source term, we are concerned by the absence of overall judgment about the improvements in understanding since the issuance of WASH-1400. While there are statements about improvements, the implication of those improvements are not apparent nor are they integrated for the regulator. We feel some statements like those in the conclusion of the OECD/NEA Newsletter Report (Vol. 3, No. 2, Fall, 1985, p. 11 on Regulatory Implications) are warranted. Without such visible interpretation of the extensive results obtained, including those at TMI, the draft report becomes a scientific quagmire for the non-specialists who have to use the information. The specialists share an obligation to make their results visible for use, lest the scientific quagmire continue endlessly with attendant continuing ultraconservatism uses that penalizes the plant operator beyond the point of reason.

General Comments

(1) The title of NUREG-0956 mentions the words Technical Bases for Source Term. A reading of the report, however, shows that the authors have:

DS10
Add: M. Silberberg, 1130SS
D. Ross, 1130SS
Wm. Olmstead, 9604MNBB

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JAN 13 1986 pd

(a) considered mainly the technical work sponsored by NRC;

(b) in general, ignored the technical work sponsored by EPRI;

(c) ignored the technical development and application work performed by the IDCOR Program, Stone and Webster and others. On the other hand one German Risk Study and Sizewell-B source term estimates are mentioned.

Thus, the implication is that only the technical work sponsored by NRC (and largely performed by the National Laboratories) can form the technical bases for reassessment of the source term. Such an implication is unfortunate. In fact, the industry sponsored work has been directed for several years, towards resolution of outstanding technical issues and in furthering the knowledge base for making accurate estimates of the source term.

(2) NUREG-0956 argues that the BMI-2104 code suite is an archival reference methodology for source term assessment. This claim troubles us since the "BMI-2104 code suite "is not a well defined article, has not been properly reviewed in its latest form, is in the process of change and is inadequate to analyze some types of reactors. It is not clear what purpose will be served by designating it as "archival". To elaborate:

(a) Several versions of the BMI-2104 suite of codes appear to be in circulation and use.

(b) The peer review referred to on page 5-1 of NUREG-0956 was conducted about two years ago. Since that time, BCL has indicated that numerous changes have been made in the codes, so that what is now proposed to be archived is not what was reviewed. A new review of a well described and stable set of codes is needed if they are to become of permanent value.

(c) We are told that the available version of MARCH, one of the BMI-2104 codes, is not suited for analysis of BWRs. MARCH was originally a PWR code which has been modified for use on BWRs. Apparently the modification does not represent

BWRs effectively.

(d) The BMI-2104 code suite does not represent the state-of-the-art in 1985 and, in spite of the statement on page 3-41 of NUREG-0956, probably did not do so in 1983-4. The coupling between codes in the suite is still partly manual or by tape reading rather than by step-by-step output-input coupling, as it should be, where feedback from one code to another is required. Important phenomena, such as in-vessel recirculation, structural heating by deposited fission products and revaporization of deposited fission products are not adequately modeled.

It is not explained what purpose will be served by making this code suite "archival". Any information or object, regardless of its quality, utility or value, can be declared to be archival. In this case, it appears (page 8-6) that the intention is to make the code suite not only archival, but to use it as the reference code suite for estimation of LWR source terms. In its present condition and with its present scope, this code suite is not suitable for use as a reference suite without extensive modification.

(3) NUREG-0956 makes extensive use of the results of the QUEST study, SANDIA 84-0410, "Uncertainty in Radionuclide Release Under Specific LWR Accident Conditions". That study has been widely criticized as exaggerating the range of significant uncertainties, primarily because the ranges of parameter values used were not weighted with a probability distribution (NUREG-0956, page 3-29). As a result, NUREG-0956 concludes that the "uncertainty in the source term is broad (span is on the order of 100). . ." for the Surry TMLB' case. It also concludes that "the source term uncertainty range is a factor 1000 for releases proceeding through the suppression pool" (Grand Gulf). Uncertainty estimates made without probability weighting are misleading.

(4) There are instances of a lack of traceability. For example, the release fractions shown on Table 4.11 on page 4-23 are not consistent with those appearing in the source material, i.e., BMI-2104, Volume 2. Especially noteworthy is the discrepancy in the La release, which is lower by a factor of 25. It is <u>very important</u> to establish the principle of

traceability between source documents and summary documents and to adhere to that principle. Otherwise, credibility of the summary document can be called into question.

(5) Appendix D presents the details of what was done to re-evaluate the public risk estimate for the Surry PWR for the purposes of the NUREG-0956 report. It involved a significant amount of data synthesis to generate source term results for source term bins not directly computed in BMI-2104 or QUEST and for refractory fission product groups also not directly calculated in BMI-2104 or QUEST. Insufficient detail is given in the appendix in these cases however to permit a step by step review of the process used to construct these results which, surprisingly, are listed to two significant figures in Table D.3. This deficiency in documentation is a very troubling aspect of this very important appendix. Further specific comments are given later in this letter.

(6) Mention is made of the Sandia work on the thermal decomposition of CsI by hydrogen combustion. We feel that these experiments may be flawed with regard to prototypicality. When a review was recently conducted on the CsI radiation decomposition experiments the review committee in our opinion reached the conclusion that the experimented technique was flawed as well as the interpretation of the experiments.

(7) By way of summarizing the general observations given above this section addresses directly the conclusions and recommendations presented in Section 8.

Conclusion 1. The BMI-2104 suite of computer codes represents a major advance in technology and can be used to replace the Reactor Safety Study methods.

We do not believe that the BMI-2104 suite of computer codes are sufficiently well developed to provide "best-estimate results".

Conclusion 2. Principal omissions and oversimplifications in the Reactor Safety Study methods have been corrected in the new source term codes. Fission product chemistry, retention in the reactor coolant system, and mechanistic aerosol behavior are now accounted for, at least in an approximate manner.

It has not been established to our satisfaction that the BMI-2104 suite of codes adequately model the in-reactor vessel accident progression phenomena.

Conclusion 3. Remaining areas of uncertainty have been identified in the new source term analytical procedures and indicate areas of research that should be pursued. Uncertainties persist in some of the areas where major advances have already been made.

We agree.

Conclusion 4. The new analytical procedures have been extensively reviewed, including a review by a special study group of the American Physical Society, and all phases of the source term reassessment effort have been documented.

We do not agree with Conclusion 4. The 14 scientist review was carried out some two years ago. The code suite has been changed since then and should be reviewed again. The depth of the critical comments by the American Physical Society concerning the code suite is also much understated.

Conclusion 5. The analytical procedure is complex and involves several scientific disciplines. Successful application of the analytical procedure requires a thorough understanding of the problem to be solved, including the plant characteristics, the accident sequence description, and the purpose of the analysis. A quality assurance procedure is also required.

We agree.

Conclusion 6. New source terms have been calculated for selected accident sequences for five reference plants that represent major reactor and containment types in operation in the United States. These selected sequences have provided a sufficient test of the capabilities of the computer codes.

When the NRC has completed the code improvements on the BMI-2104 code suite the calculations should be repeated.

Conclusion 7. For most accident sequences, the largest single factor affecting source terms is containment behavior. A delay of several hours in containment failure will reduce source terms significantly.

We agree.

Conclusion 8. Source terms were found to depend strongly on plant design and construction details, thus making development of useful generic source terms difficult.

We agree.

Conclusion 9. New source terms for many accident sequences were found to be lower than those in the Reactor Safety Study, but some were larger. The reductions were found mainly because containment integrity was maintained and natural processes reduced airborne concentrations of fission products. The larger source terms resulted from early containment failure, which is still predicted in some cases, and the improved description of ex-vessel processes, which leads to larger release estimates. Therefore generalizations are inappropriate.

There is considerable truth in the statements made in Conclusion 9. However, we wish to emphasize that the BWR Mark I and Mark II source terms are undergoing further intensive study.

Conclusion 10. A comparative risk appraisal for the Surry plant using the Reactor Safety Study accident frequencies, source terms based on BMI-2104 results, and a preliminary re-evaluation of the behavior of the containment shows a reduction in estimated risk compared with the Reactor Safety Study. The reduction results about equally from new source terms and new evaluations of containment behavior.

No comment.

Conclusion 11. For the other plants, further analyses need to be made before any conclusions can be drawn about changes in estimated risk. The fact that source terms for some accident sequences are not lower than those in the Reactor Safety Study suggest that significant reductions in estimated risk may not be found in all cases.

We are not yet prepared to accept the higher source terms, for example, for the Mark I. We feel further analyses are necessary.

Conclusion 12. Research programs that address the remaining major areas of uncertainty in the source term technology are currently in place and being pursued by the NRC.

The statement is parochial in that it ignores research done by EPRI and the rest of the world.

Will the NRC research indeed be finished by the end of calendar year 1987 as Figure 7.1 implies. We have seen other milestone charts which seem to extend further (i.e., MELPROG and SCADAP development).

Conclusion 13. A major conclusion of the American Physical Society study group confirms the NRC staff position that source term research must be continued in order to complete the regulatory actions being considered.

We agree, if the program is intense, and not extended out for many years in the future.

Recommendation 1. The new source term analytical methods should be used to re-evaluate regulatory practices that are based on the Reactor Safety Study methods. Insights from new analyses should be applied to reconsider the use of TID-14844 assumptions. Improvements are so significant that utilization of the new methods is warranted while additional confirmatory research is being completed.

We agree in general. However, the challenge to the methodology is greater in evaluating some regulatory practices and less in other cases. Therefore the evaluation of regulatory practices should be undertaken with an appreciation of some of the limitations of the methodology.

Recommendation 2. A particular version of the new codes called the Source Term Code Package will be maintained as a reference code and is the recommended analytical tool for NRC analyses of accidents severe enough to result in complete core melting. Additional technical insights can be obtained for all accident conditions with the RC's detailed mechanistic codes and their experimental data bases.

Reservations concerning the readiness of the code suite to assume the role mentioned above have already been expressed.

Recommendation 3. The Source Term Code Package was designed to provide best-estimate results (i.e., without intentional bias). In any regulatory application, careful consideration must be given to the purpose of the evaluation, to the desired margins, and to the uncertainty levels. Close coupling between the research effort and the regulatory effort will be required in assessing uncertainties and evaluating technical issues.

We agree.

We feel the very issuance of the draft report is a significant accomplishment to provide a basis for discussion of scientific work. We are uncomfortable with the lack of synthesis of results in hand that could provide a basis for early and meaningful regulatory action. We feel that even with the concerns about specifics as indicated above and in the enclosure, monumental progress has been made since WASH-1400 was issued. Such progress has been achieved through NRC, industry and overseas efforts. We would be more comfortable with the draft report if there were more of an attempt to integrate the information in the context of both near-term and long-term regulatory action. We would be even more comfortable with the draft report if the relative likelihood of various phenomena were considered along with attendant uncertainties in mechanistic phenomena. For example, the likelihood of various high efficiency events is probably much lower than that of their low efficiency counterpart.

We will be happy to discuss these thoughts with you in more detail.

Sincerely,

R. C. Vogel

Richard C. Vogel
Sr. Scientific Advisor
Safety Technology Department

RCV/11

cc: J. J. Taylor
W. B. Loewenstein
Tony Buhl
Miles Leverett
M. Silberberg

Addendum

This addendum contains a number of specific and sometimes detailed comments on NUREG-0956.

Page 1-2, line 28

Others besides the NRC staff have been engaged in risk assessments of specific nuclear power plants over the past few years.

Page 1-3, line 42

None of the source term R&D work supported by EPRI during the last few years is acknowledged in the subsection on research and technical issues. Actually since the section addresses applied analysis rather than research support it probably should carry a different title.

Page 2-11

Lack of reference to IDCOR, specifically the results as published in Chapter 10 of the IDCOR Technical Summary Report, is conspicuous. If the German Risk Study and The Sizewell-B Source Term Estimates are discussed, the IDCOR source term estimates should be noted as well.

Page 3-9

The list of principal uncertainties in the MARCH code is much shorter than it should be. Some models in MARCH code, e.g., the mode of vessel failure can not be justified.

Page 3-9, third full paragraph

The statement is made that it is "just as easy to imagine enhanced oxidation caused by the exposure of fresh zircaloy and wetting as it is to imagine retarded oxidation as the result of channel blockage". No supporting evidence for this statement is provided. The argument given in its support also ignores the observed decrease of Zircaloy oxidation (due to steam starvation) in the German fuel melt experiments and in the PBF. It is not supported by any of the BMI-2104 documentation. It is also not supported by ORNL/TM-8842 (the Status of Validation Report). Indeed, if one were to consider the significant decrease in surface-to-volume ratio that results when the fuel becomes molten, one would be hard-pressed to make such a statement.

Page 3-9

The statement regarding the unreliability of MARCH-2 predictions (page 3-9) for assessing containment loadings and performance is significant.

Page 3-10

The discussion on MERGE codes mentions the absence of natural convection models, however, it claims that MERGE calculated temperatures were close to those obtained from more detailed calculations which include natural convection models. The detailed calculations mentioned do not model circulation between core and upper plenum and therefore can not serve as checks on MERGE modeling.

A statement is made to the effect that the calculations performed by EPRI contractors are inadequate to resolve the natural convection issue. However, no arguments to support this statement are advanced.

Page 3-12

The statements about aerosolization and revolatilization are based on perceptions which are not supported by recent results of analyses with EPRI sponsored code CORMLT or by the NRC sponsored code MELPROG. The estimated values reported for temperatures and revolatilization, etc. will most probably need updating.

Page 3-14

The statement to the effect that a good data base for corium concrete interaction does not exist is most appropriate. The statement about CORCON-1 code giving satisfactory results is probably not appropriate, since CORCON-1 (Sandia version) did not reproduce the data acquired at BETA facility. Secondly the artificially high melt temperatures predicted by CORCON-1 will lead to artificially high release of the fission products during core concrete interaction.

Page 3-17 to 3-19

The discussion of the CORSOR code seems adequate. However, with respect to Te forming compounds with core metals, believe the recent work of Alexander at Battelle indicates that dissolution rather than telluride formation is the principal reason for Te retention by Zircaloy cladding.

Pages 3-21 and 3-22

The TRAP-MELT computer program calculates the fission product retention in the primary system. Its calculational boundaries are the core and primary system up to the containment. Like all other aerosol codes of the bin type (formerly log-normal), thermal hydraulic parameters are required input from some other code. In the case of the BMI-2104 study, these inputs were obtained from the MERGE code. NUREG-0956 adequately lists TRAP-MELT's strengths and deficiencies. Most of the correlations for aerosol deposition, agglomeration and settling are standard state-of-the-art. One significant deficiency is, however, glossed over, namely that it is not capable of accommodating recirculating flows. To adjust the fundamental equation set for recirculating flows requires a virtual rewrite of the code; the flow equations have to be rewritten for two-dimensional flow rather than the singly-dimensioned once through flow that is currently modeled. Primary system fission product retention depends principally on the input flows and, therefore, dependent on the user.

Pages 3-22, 3-23 and 3-24

This section provides a very brief description of the VANESA code, which figures so very prominently in source term assessments. The description should be strengthened and an independent formal review should be performed. The report rightfully states that validation of the code is lacking due to dearth of applicable data. What is also lacking in the case of VANESA code is the code exercising by individuals other than the developers of the code.

The list of uncertainties in VANESA modeling mentioned on page 3-24 should be expanded. The temperature estimates are the most important. It is worth noting here that the BETA tests the temperature of the melt dropped rapidly (within $\approx$ 3 minutes) to solidus temperature. This is in spite of large amount of electrical power input into the melt.

Page 3-24

NUREG-0956 states that NAUA is not extensively validated because "there is not a large data base requirement for validation" sic. NAUA has, of course, been partially validated through separate effects experiments at KfK. The report should point this out, and go on to mention that two large scale experiments (DEMONA and LACE) are well underway expressly to provide the needed data base. Recently favorable comparisons have been reported between the NAUA predictions and the measured data in DEMONA.

Also does the BMI-2104 version of NAUA really treat homogeneous nucleation of water droplets, as stated? (No other version of NAUA that we are familiar with does.)

Page 3-25

NUREG-0956 states that "(1) in some BWR sequences, the relatively high drywell volume can reach temperatures that are high enough to revaporize volatile fission products, and (2) hydrogen burning can create temperatures high enough to revaporize materials that are airborne as aerosols in containment". Both these "facts" could be challenged. We would consider them to be at most hypotheses, yet to be proved experimentally. The first assumes that fission product are not retained or bound to the surface of concrete in a way that hinders revolatilization, and the second ignores the thermal capacity of water bound up in the aerosol droplet that may "moderate" the thermal history the fission product aerosol sees (undoubtedly some (all) of the water may evaporate, creating smaller particles, but the fate of the aerosol particle "core" is still speculative).

Pages 3-27 and 3-28

Although it is stated on page 3-27 that "it is believed that the present set of codes provides a credible basis for defining the major influence on source terms", it is further noted on page 3-28 that improved modeling and additional research is needed on the simultaneous coupling between the transport and deposition of fission products (as heat sources) and the thermal hydraulics of the reactor coolant system. It has become apparent, through the IDCOR work that when such a coupling is made, the consequences of some accidents change dramatically, relative to the predictions made in BMI-2104. This is particularly true for Mark I BWRs in which revaporization of CsI is calculated to result in significant relocation out of the reactor coolant system, either into the suppression pool or into the secondary containment building. It appears, then, that assertion #2 on page 3-27 is premature.

Pages 3-28 to 3-33 (QUEST - General)

The results of the QUEST uncertainty study performed by Sandia Laboratory is briefly described in this section. As known from previous readings of the full report, the study is flawed since it assumes large variations of individual parameters with no probability distributions for the variations. The resulting variations in source terms are three orders of magnitude, which defeats the purpose of a best-estimate analysis of the type attempted with the BMI-2104 code suite.

Pages 3-27 to 3-28 (Code Validation Review - General)

This subsection simply quotes the conclusions as given in the ORNL report ORNL/TM-8842. It may be of interest that EPRI supported R&D programs are addressing most of the areas identified in the report as needing additional work.

Page 3-29

It is stated that the uncertainty in the source term is broad and is very dependent on time. The "broadness" referred to arises because of the methodology used to carry out the uncertainty analysis.

Outputs from one code which are, in turn, inputs to other codes were varied over what were judged to be "reasonable" ranges. The problem with this approach is that, in reality, the relationships are too complex to be treated with separate stand-alone codes. Couplings and feedbacks between thermal hydraulics, material relocations and fission product transport and deposition are often very strong. This observation was recognized by IDCOR, which then proceeded to integrate the various required models into a single code package. Once this was done, limited uncertainty analyses were carried out, which showed considerably narrower ranges than observed in SAND84-0410. The principal reason for this is that the parameter variations chosen were such that care was taken to not exercise the models in "nonphysical" regimes. Considerable work has also been done by EPRI, which largely bears out this observation. The EPRI work also shows that source term results depend more strongly on changes in accident progression than on the types of variations chosen in the QUEST study.

The thrust of this comment is similar to that for page 3-27 and 3-28.

Page 3-32

In the results of the Surry TMLB' study, it is mentioned (page 3-32) that resuspension is possible. This, we believe, not to be the case, as shown by several experiments.

In one of the NRC/IDCOR issue resolution meetings, it was agreed that resuspension is not likely to significantly augment fission product release. Why is the issue raised again here?

Page 3-32

A statement is made about suppression pool disruption in conjunction with a high pressure vessel failure, giving rise to a high source term. It is difficult to imagine such a scenario since wetwells are designed to withstand the blowdown from a large-break LOCA. Even if such could occur, the volatile fission product species such as CsI and CsOH, would deposit in-vessel (the vessel would remain at high pressure only if the ADS valves remained closed). The source term would then be dominated by the ex-vessel release. It is suggested that reference to this type of event be eliminated.

Pages 3-33 to 3-37

It is a good idea to point out where major technical advances have been made. In doing so, however, it is imperative to point out where supporting data exists or is currently being developed. In that regard, various EPRI-sponsored and other experimental programs should be pointed out as follows:

a. The chemical forms of cesium, iodine, tellurium, etc., in the primary system can be established through the STEP experiments at TREAT and from the FP-2 test at LOFT.

b. The data base for in-vessel melt progression, hydrogen generation and control rod behavior will also be enhanced by the STEP experiments and the LOFT-FP2 test results.

c. The EPRI-sponsored pool scrubbing experiments at BCL will support improved mechanistic treatment of this phenomenon.

Page 3-33, line 33

The abundance of different iodine forms is determined by the predicted physical and chemical conditions as well as the thermodynamic properties.

Page 3-35, line 13

Is the conclusion from NUREG-0772 regarding the degree of organic iodide formation no longer valid? And if so, why?

Page 3-35, line 15

Alexander's work at BCL indicates Te may be retained in Zircaloy via dissolution rather than by compound formation.

Page 3-38 to 3-40

The major areas of uncertainty have been and are being addressed by various industry efforts. These should be pointed out as follows:

a. Natural circulation in the reactor vessel is modeled in the EPRI-developed CORMLT code and in the IDCOR-developed MAAP code. EPRI has an experimental program ongoing in this area.

b. Core-melt progression and hydrogen generation effects can be determined from the STEP tests and from the LOFT-FP2 experiment. Improved information on in-vessel fission product release from fuel is also resulting from these programs.

c. Revaporization of fission products in the reactor coolant system is modeled in MAAP. It is also being measured in an EPRI-sponsored experimental program at ANL.

d. Aerosol generation from core-concrete interactions is also modeled in MAAP, EPRI has experimental work underway at ANL to improve the data base.

e. The pool scrubbing work at BCL, mentioned on page 3-40, is sponsored by EPRI.

Page 3-39, line 15

Another important effect of natural circulation within the reactor vessel and coolant system during a severe accident would be to delay the onset of core melting compared to predictions made in which that process is not considered.

Page 3-40, line 9

The MAAP code has this capability. Perhaps some useful insights could be provided here through reference to recent MAAP results.

Page 3-41, line 45

The subsection and summary make no mention of any capability for calculating fission product/aerosol transport in secondary structures (auxiliary buildings, reactor buildings, etc.) which can be part of the pathway to the environment in many accident sequences. Will updated source term calculations with the Source Term Code Package include such analyses and if so, how?

Pages 4-6 and 4-7, Tables 4.3 and 4.4

The order of events listed for the TC sequence calculations is not the same as in BMI-2104. In the actual analyses, containment failure was predicted to preceed loss of makeup for both the Peach Bottom and Grand Gulf plants.

Pages 4-19 and 4-20, Figures 4.4 and 4.5

It is suggested that it be noted directly on these figures that cesium iodide and cesium hydroxide are assumed to follow the same retention curve.

Page 4-25

It is stated that the small sump below the vessel in Peach Bottom could retain corium in a confined configuration. This is not actually the case for Peach Bottom. In fact, the sump is about 6 inches deep and could only confine a very small fraction of the core inventory. In the event of a vessel breach, it would be expected that a large fraction of molten corium would quickly enter the drywell once melting through the flimsy door at floor level.

Page 4-32, line 27

It is not clear from the discussion starting on this line why the Sandia calculations and the Stone & Webster calculations depicted in Figure 4.13 do not agree, particularly for large size openings. Probably the basis used for the two calculations are too different to permit a meaningful comparison.

Pages 4-35 and 4-36

Figure 4.14 doesn't make the point (and it should) that aerosols leaked after start of concrete attack will contain a much larger mass fraction of inert aerosols than those leaked prior to concrete attack.

Page 4-35 (first paragraph)

Aerosol stratification, observed at DEMONA, is likely to be important in real accidents. Early containment failure, with high gas flow rates and direct paths for aerosol leakage to the environment, would likely only occur from the upper part of the RCB where stratification effects would likely reduce aerosol concentration, thereby partially mitigating the consequences.

Page 4-40 and 4-42, Table 4.13

Many release fraction values given in this table are different from corresponding values in BMI-2104 Volume V (Surry) or Volume II (Peach Bottom). For some sequences and some fission product groups in all the sequences, the release fractions were not calculated in BMI-2104. This lack of traceability between the two pieces of work requires clarification in the present report.

Page 5-5

The list of 18 issues between NRC and IDCOR is presented with no indication of which of the issues have been resolved, i.e., it reads as though the entire list was open. Actually many of them are closed.

Page 5-12

Line 13 may leave the impression that these IDCOR/NRC meetings are not open to the public. That is incorrect; they are open.

Page 5-13

The areas of agreement between IDCOR and NRC are much more extensive and on much more important issues than those listed here. Possibly this page was written before the two recent (March and April) IDCOR/NRC meetings at which much agreement was reached. We suggest that Zoltan Rosztoczy be asked to rewrite page 13. He is the NRC person who chairs and reports the results of the IDCOR/NRC technical exchange meetings now.

Pages 6-1 and 6-7

Chapter 6, Comparative Risk Appraisal for Surry and Risk Insights for Peach Bottom, is focused almost entirely on Surry. The risk appraisal is really not up to date because it uses the Reactor Safety Study accident frequencies. The discussion is basically a presentation of the conclusions of the work reported in Appendix D of the report. These conclusions indicate that as a result of applying BMI-2104 procedures the Surry risk of early fatalities is about a factor of 10 below the WASH-1400 estimate while the risk of latent fatalities is about a factor of 4 lower. These should be contrasted with our EPRI NP-4096 updated risk results which indicate zero risk of early fatalities and a risk of latent fatalities which is more than a factor of 20 lower than the WASH-1400 estimate. In plotting the CCDF curves in

NUREG-0956 the authors elected to extend the lower end of the frequency scale to 10^{-10} yr^{-1}. This is a departure from past practice (the WASH-1400 cutoff was at 10^{-9} yr^{-1}) and seems to yield little additional information of value, particularly with respect to the re-evaluated risk profile.

Page 6-1

End of second paragraph acknowledges that the Peach Bottom results are being recalculated, but on pages 4-16, 4-23 and 4-40 quotes results of earlier, out-of-date calculations. What purpose is served by quoting such results? Publication of such results can only lead to later confusion and conveyance of false impressions which may last long after the correct results have been published.

Page 6-7 (ninth from last line)

Apparently a phrase has been omitted in the parentheses (with the exception of bypass of the reactor building).

Page 6-7

Since Peach Bottom is being re-analyzed, Section 6.2 (page 6-7) should be omitted.

Pages 7-4, 7-5, 7-6, 7-7 (Section 7.3, Applicability to the Regulatory Process

This section draws a fairly sharp line between the conservative estimates of the source term and risk presented in the Reactor Safety Study, and the best-estimate or realistic results of IDCOR and, presumably SNL and BCL. It is stated that "single-valued best-estimate predictions-----cannot be used to ascertain compliance with regulations that require an upper bound of potential consequences of accidents", such as paragraph 11 of IOCFR100. On the other hand it is stated that "source term methodology" (i.e., best-estimate) appears suitable for assessing the need for protective action in response to an accident.

It would be desirable to have the report make it clear that source term methodology is suitable for application to severe accidents generally, i.e., to determination of the overall safety of a plant. Otherwise, some may feel that conservative estimates should be applied to such determinations. This would be highly counter-productive.

Pages 7-1 to 7-6

The present NRC research program for natural circulation consists of analysis only. Reference is made to the natural circulation experiments performed by Westinghouse under EPRI sponsorship. They should be employed for code validation.

Page 7-6

In the Core Melt Progression and Hydrogen Generation no mention is made of the CORMLT code sponsored by EPRI or of the TREAT (STEP) test results. There does not appear to be a mention of validation of methods using the existing experiments. In this respect, the data from TMI-2 examination program is perhaps the most valuable and it should be used. Also analysis should include natural convection modeling and coupling of thermal hydraulics and aerosol transport.

Page 7-8

The report states that NRC has instructed BCL to construct an integrated code package out of the individual codes used in BMI-2104 and "all code options are being fixed as they were used in BMI-2104 and most of the code interfaces are being directly linked".

This is a disturbing development since it is strongly suspected that the BMI-2104 codes still contain inadequacies and need a thorough airing. Additionally, some non-BCL improvements on the BCL codes should be considered if the projected code package is to be taken seriously. Incidentally, the relationship between this code package and several other packages of somewhat similar kind being assembled under NRC sponsorship is not clear. Offhand, it appears that there is a lot of duplication and overlap involved among them. This point should be clarified at some point in NUREG-0956.

Pages 7-6 and 7-7

In this section on In-Vessel Fission Product Release from Fuel and Aerosol Generation, in addition to the STEP work noted, EPRI also supports the RAFT code work which incorporates aerosol formation models. Another relevant program not identified in this section is the Marviken experimental test series.

Pages 7-7 and 7-8

In the section on Retention and Revaporization of Fission Products in the Reactor Coolant System the EPRI supported development of the RAFT code is relevant since it also contains models for fission product chemistry effects on deposition and revaporization.

Page 7-8

In the section on Fission Product Release and Aerosol Generation from Core Concrete Interaction there appears to be an attempt to obtain data on core-concrete interaction process and the resultant aerosol release. This is commendable, however, use of data from German BETA tests does not appear to be playing a prominent role in the validation of CORCON and VANESA codes. This data already exists and should be used.

Direct heating and aerosol formation due to high pressure discharge are mentioned. Natural circulation induced system failures are important and we feel that it is highly unlikely that a high pressure melt discharge will occur in the PWR high pressure severe accident scenarios.

Page 7-9

In the section on Containment Pressure Loads the mention has been made of the results obtained from the large-scale hydrogen combustion tests performed at Nevada under EPRI, NRC and International sponsorship. The Sandia hydrogen program tends to experiment with extreme conditions which are highly unlikely to occur in any scenario.

The direct heating research does not mention the results of tests performed by ANL under EPRI sponsorship.

Page 7-13, Figure 7.2

It is suggested that this figure be supplemented with a table showing, for each code, its required inputs and its outputs.

Page 8-3

Conclusion 6 says "These selected sequences have provided a sufficient test of the capabilities of the computer codes". Unfortunately, merely showing that a code can be made to give an answer is not enough. We are not aware that any appreciable amount of validation of BMI-2104 code suite has been performed by analyzing experiments such as PBF, LOFT or the TMI-2 accident.

Page 8-12

Tests on concrete containments are covered in a short paragraph on this page which tells what NRC plans to do, but gives no recognition at all to the extensive large-scale work already done under EPRI sponsorship at the Portland Cement Association. To our minds, the EPRI work pretty well settles the principal question about concrete containments (leak before break). It should certainly be mentioned in NUREG-0956.

Page A-9, Figure A-5

This figure does not show the pressurizer, which is a major piece of PWR equipment. Same comment applies to Figure A-3.

Page A-10, Figure A-6

This figure contains three errors:

a. The steam separators and dryers are located inside the reactor, not outside as implied.

b. The high pressure steam extraction lines apparently terminate in the feedwater heater, with no indication of the disposition made of the condensate.

c. The jet pump support plates are not shown. As drawn, most of the reactor coolant would by-pass the core.

Pages B-3 and B-21, Appendix B

This appendix is described as a summary of a SNL report on "Containment Event Analysis and Estimation of Source Term Frequencies". It contains 22 detailed (and almost illegible) tables of calculated results and about 14 pages of text. Considerable emphasis is placed on distinctions among the "optimistic", "central" and "pessimistic" estimates (with two subcases under each of the last two estimates). While it is proper to reflect the fairly large uncertainties which attend some source term estimates, the impression left by Appendix B is that of inability to come to a best-estimate. The tables in Section 4 of the report are crisper in this respect but Appendix B tends to blur the impression left by Section 4. It is questionable whether Appendix B serves a useful function in its present form.

Page D-6, Appendix D, line 1

The four characteristics chosen for the binning of source terms appear consistent with the BMI-2104 analyses for Surry. Actually, several of the bins represent analytical uncertainties as much as scenario alternatives.

Page D-8, Appendix D, line 11-19

The discussion of the details of spray drop size effects is not very helpful and seems irrelevant in view of the approximations introduced in the analysis by the source term synthesis strategy already described.

Page D-8, Appendix D, line 30

Since the BMI-2104 analysis of $S_2D\ \gamma$ assumed no spray operation after containment failure it is confusing how it could be used to estimate a case where sprays continue operating.

Page D-8, Appendix D, line 45

It should be noted that this source term may be significantly over estimated.

Page D-9, Appendix D, line 2

This appears to be another case where effect of spray operation is included somehow but the details of the process are left undisclosed.

Page D-11, Appendix D, Table D.3

It is suggested that the zero core release fractions given in the table for bins 13 and 14 be replaced with the term negligible to more accurately reflect the expected results.

Page D-13, Appendix D, line 13

EPRI supported analyses for Surry as reported in EPRI NP-4096 also made this determination for the outcome of the S_2C accident sequence.

Page D-23, Appendix D, line 1-5

The EPRI supported source term update work for Surry predicted significantly lower risks for early and latent fatalities than indicated here.

Pages D-25 and D-26, Appendix D

The comment on Peach Bottom risk is premature except that it highlights the need to consider the reactor building as an attenuating volume in source term calculations.

We very much appreciate the opportunity to comment on NUREG-0956. If there are questions concerning our comments we would be happy to discuss them.

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OECD NUCLEAR ENERGY AGENCY / 38 BD SUCHET / 75016 PARIS / FRANCE

Note to the reader: *Since the creation of the* ***NEA Newsletter*** *in December 1983, one special and four regular issues have been published. Given the encouraging response, publication of the* ***Newsletter*** *will be continued on a regular twice-yearly basis. In an effort to improve presentation and make the* ***Newsletter*** *more attractive to readers, a new format which includes the ordering of sequence by volume and number has been adopted with this issue.*

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Contents

Source terms: Evaluating new information

D.F. Torgerson

One of the most intensive areas of nuclear safety research over the past 10 years has concerned "source terms". In general, "source terms" characterise the potential release of radioactivity following a severe reactor accident. A severe reactor accident is an accident in which the core of a reactor is sufficiently damaged (due to loss of cooling) so that fission products are released from the fuel matrix. Such releases include those associated with fuel melting and with pressure vessel (the vessel that contains the core) melt-through. It is important to note that source terms may be defined differently depending upon the end use of the information. For example, to an analyst assessing the performance of a reactor containment building, the source term describes the radioactive material released from the reactor core to the containment building. To an air cleaning specialist, the source term describes the challenge to a filtration system. However, from a nuclear safety point of view, the most widely used definition is that the source term is the quantity, timing, and characteristics of the release of radioactivity to the environment following a postulated severe accident.

Source terms are used by regulators for such activities as emergency planning, risk assessment, setting research priorities, evaluation of potential backfits, and the resolution of safety issues. Obviously, any changes to source terms could have significant impact on these regulatory activities, and on the utilities operating nuclear power plants. Over the past few years, there has been considerable progress in the development of our knowledge of source terms. This article summarises the current situation, and indicates how the NEA's Committee on the Safety of Nuclear Installations (CSNI) is evaluating the new information.

Barriers to release of radioactivity

Although severe accidents have low probabilities of occurring, the calculated consequences of such accidents could be high if source term values associated with severe accidents are large. For this reason, it is useful to review the principal barriers to the release of radioactivity, assuming that the various engineered safety systems are not operating. The first barrier is the fuel, which is a dense uranium oxide (UO_2) matrix in which most of the fission products are fixed. The UO_2 matrix is surrounded by a metal sheath, such as Zircaloy or stainless steel. If the fuel becomes sufficiently hot, the sheath fails and relatively small amounts of fission products are released into the primary coolant circuit – *i.e.*, the "gap inventory" of fission products that are between the sheath and fuel matrix. If the temperature continues to rise, for example to melting temperatures, then more fission products would be released from the fuel matrix.

The next barrier is the primary circuit, which contains the core and the water coolant. Fission products may be depleted in the primary circuit by a number of mechanisms, depending on the particular accident sequence. In very severe accidents, the core may melt through the primary vessel into the containment building, and additional radioactivity would be released from the fuel, due to the interaction between the melted core and the concrete basemat.

The containment building surrounds the reactor and is designed to withstand high pressures and temperatures. Within the containment building, natural processes occur that deplete radioactivity from the gas phase. In addition, there are other removal processes due to the operation of devices such as water sprays and fan coolers. As we shall see later, a key ingredient contributing to improved source term values is the progress that has been made in understanding performance of the containment building during a severe accident.

The 1975 WASH-1400 Reactor Safety Study

The modern history of the source term begins with the WASH-1400 Reactor Safety Study, commissioned by the US Atomic Energy Commission, and published in 1975. The study classified the source term values into "release categories" that were associated with various levels of damage to nuclear power plants.

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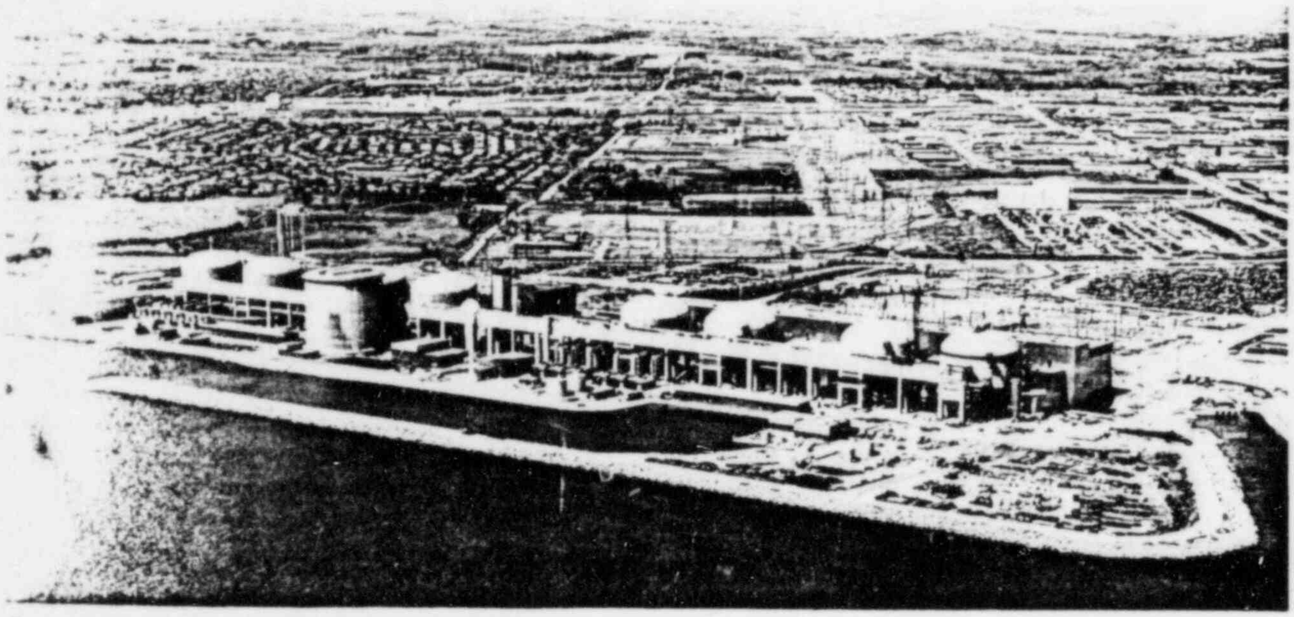


The Pickering "A" and "B" Generating Stations in Canada
Credit: Ontario Hydro

At the time of WASH-1400 it was recognised that there was little knowledge of some of the important phenomena associated with source term technology. In some cases, the important phenomena had to be neglected (such as the effectiveness of some engineered safety systems), and in other cases, simplified models had to be used that contained conservative (*i.e.*, pessimistic) assumptions concerning the behaviour of radioactive material. In particular, such phenomena as fission product retention in the primary coolant circuit, steam condensation in containment buildings, aerosol behaviour in containment buildings, fission product release pathways, and some important aspects of the chemistry of volatile fission products were largely neglected. The result was that many of the source terms predicted in WASH-1400 were highly pessimistic with respect to the quantity and timing of the release of radioactivity from nuclear power plants. However, WASH-1400 was the only comprehensive description of reactor accidents available to regulators, and the information in WASH-1400 is still used today in several areas of regulation.

New information

Since WASH-1400, there has been considerable research activity that has led to a better understanding of the source term phenomena that were originally neglected. As a result, a number of new studies have recently been prepared to reassess source term technology. These include reports prepared by the American Nuclear Society, the American Physical Society, the Industry Degraded Core Rulemaking Program, and work performed by, and for the US Nuclear Regulatory Commission. Various other studies have been done by such organisations as the Stone and Webster Engineering Corporation, the Electric Power Research Institute, and the New York Power Authority. More specialised reports have also been prepared, based on work in several NEA Member countries, including Canada, France, the Federal Republic of Germany, Italy, Japan, Sweden, the United Kingdom, and the United States. Finally, a vast pool of relevant technical information has become available from research programmes in various countries.

At the November, 1984 meeting of the NEA's Committee on the Safety of Nuclear Installations it was decided to organise a Special Task Force on Source Terms to review the studies as well as ongoing work. Since most of the studies were available by early 1985, the Task Force began its work in February and completed the review in October, 1985. The Task Force consisted of an international group of scientists and engineers who are experts in the various technical areas of importance for evaluating source terms. A few examples of the Task Force's findings will serve to illustrate the impact of the new information on source terms.

Source terms

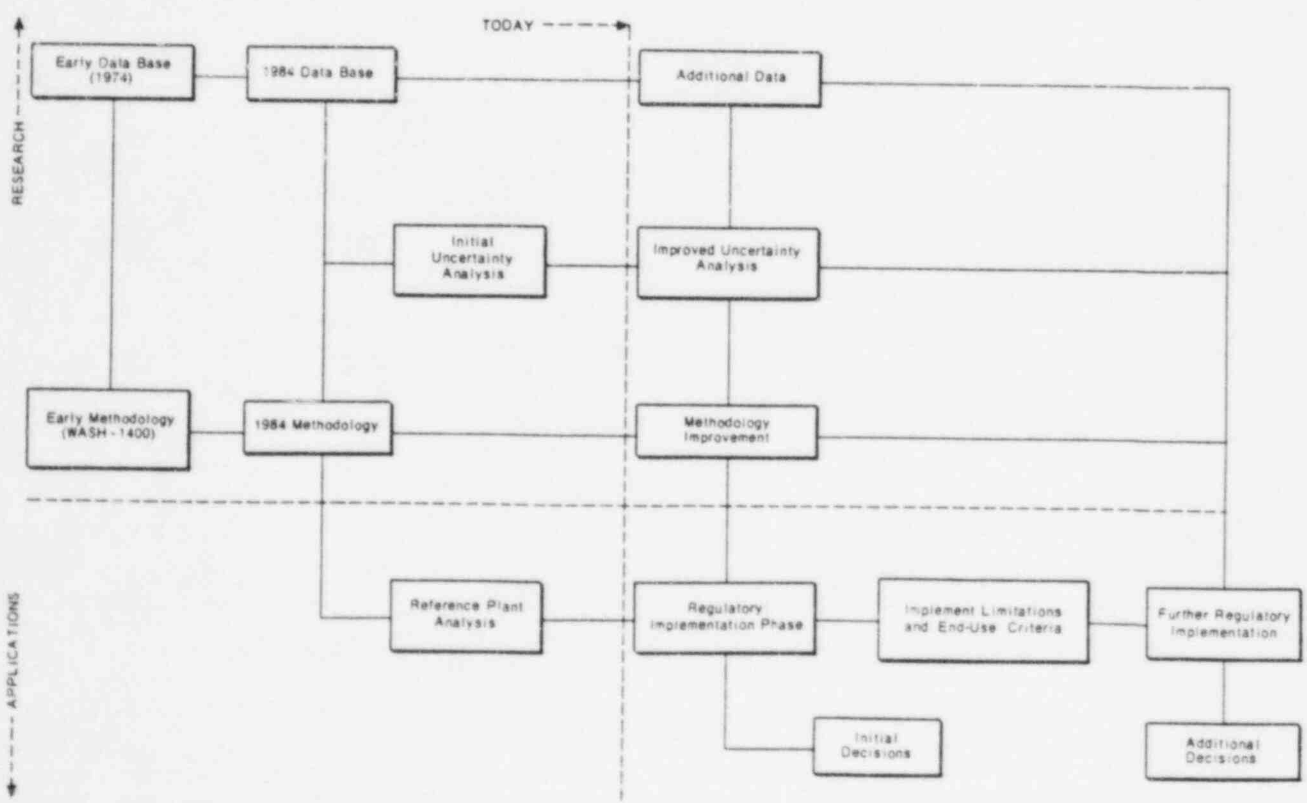
One of the main developments since the 1975 WASH-1400 report is that most source terms for nuclear power plants can now be calculated on a mechanistic basis. That is, the important mechanisms have been identified and source term values can be based on technical information, not on pessimistic assumptions. As a result of this, many of the old source term values have been found to be over-estimated, sometimes by large factors.

Another major factor resulting in reduced source terms is a better understanding of the performance of nuclear power plants. For example, some containment buildings have been found to be 2-4 times stronger than their design pressure. During a severe accident, these containment buildings would fail, if at all, after longer times. This is important since recent information on fission product/aerosol behaviour in containment buildings during a severe accident indicates that most of the radioactivity would not be in the gas phase at the longer times, and therefore would not be available for release.

There have also been significant advances in the understanding of fission product chemistry during reactor accidents. For example, at the time of the WASH-1400 study, it was known that radioiodine would likely react with cesium (like iodine, cesium is a fission product formed in fuel, but is about 10 times more abundant than iodine) to form the low-volatile, water-soluble salt, CsI. However, since there was a poor data base for characterising this reaction, it was assumed that iodine would form volatile iodine (I_2). This led to very large source terms for iodine release in some postulated accidents. However, the importance of CsI formation became strongly evident during the Three Mile Island accident in 1979. Although a large fraction of the core inventory of iodine was released from the reactor core, the iodine source term was only a very small fraction of what had been expected if the iodine had formed I_2. Today, there is sufficient information to characterise CsI formation during reactor accidents and in many accidents much lower iodine source terms are justified.

The dependence of fission product transport on thermalhydraulics (the flow of heat and mass) in reactors is another area where there have been notable advances. Most of the important in-vessel thermalhydraulic phenomena are now recognised, and the current methodology is probably adequate to predict fission product retention in the primary coolant circuit. Also, the improved understanding of aerosol transport in containment buildings is allowing analysts to couple aerosol physics with containment thermalhydraulics.

Figure 1. **RELATIONSHIP BETWEEN SOURCE TERM RESEARCH, UNCERTAINTIES AND IMPLEMENTATION**



The above are just a few examples of new information and how the new information is affecting source term values. At the same time, the new information has resulted in the definition of phenomena that need further characterisation. Examples are the need to improve the models for core-slump behaviour, for core concrete interactions, and for some aspects of hydrogen combustion. These areas are all being addressed in international research programs, and it is highly unlikely that any important phenomena are being neglected.

Remaining uncertainties

Although source terms for many accident sequences have been over-estimated in the past, it may prove difficult to arrive quickly at specific source term reduction factors that are universally acceptable. The main reason for this is that different code sets may result in different source terms, depending on the specific plant and accident sequence being analysed. These differences can be traced to differences in the mathematical models representing the physical phenomena, differences in the physical properties of materials, possible omission of important phenomena, specifications of the accident sequence and plant geometry, and numerical approximations.

One way of characterising uncertainties is to recognise that the importance of a particular uncertainty depends on the timing and mode of containment failure. As discussed previously, stronger containments reduce the probability of early containment failure, and aerosol depletion processes reduce the impact of late containment failure. Thus, for these stronger containment buildings, uncertainties associated with early and late containment failure would have less impact than the uncertainties associated with intermediate times. The reduction of these remaining uncertainties will undoubtedly receive the highest priority in future research programs.

Regulatory implications

The remaining uncertainties notwithstanding, all the current studies indicate that the source terms for many postulated accidents can be reduced from WASH-1400 values. The implications of the current work for regulatory activities are summarised in Figure 1, which shows the historical, current and future relationships between source term research and eventual applications. The early 1974 data base has now been replaced by the 1984 data base, which includes all the information available at the time of preparation of the recent studies.

While there is always a temptation to delay implementation of new information until the last "i" is dotted and "t" is crossed, most source term experts feel that efforts should now begin to use the new information in regulatory applications. The vertical dashed line in Figure 1 is the current situation. The additional data, improved uncertainty analysis and improved methodology arising from research activities should now lead to a regulatory implementation phase. The remaining research to be done should be part of an interactive process, whereby comparison of the application requirements (as developed by regulatory agencies) with the remaining source term uncertainties will determine the need for future research or code improvements. Such an approach is feasible today owing to the outstanding progress that has been made in source term technology ■