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Regroupement pour la surveillance du nucléaire

Canadian Coalition for Nuclear Responsibility

REACTOR SAFETY:

TWO CRITICAL PAPERS

Ontario Legislature Investigates Nuclear Safety

Gordon Edwards

Nuclear Fuel Melting as seen by The Rasmussen Report (WASH-1400) with some Additional Comments Re The CANDU Reactor

compiled by Gordon Edwards

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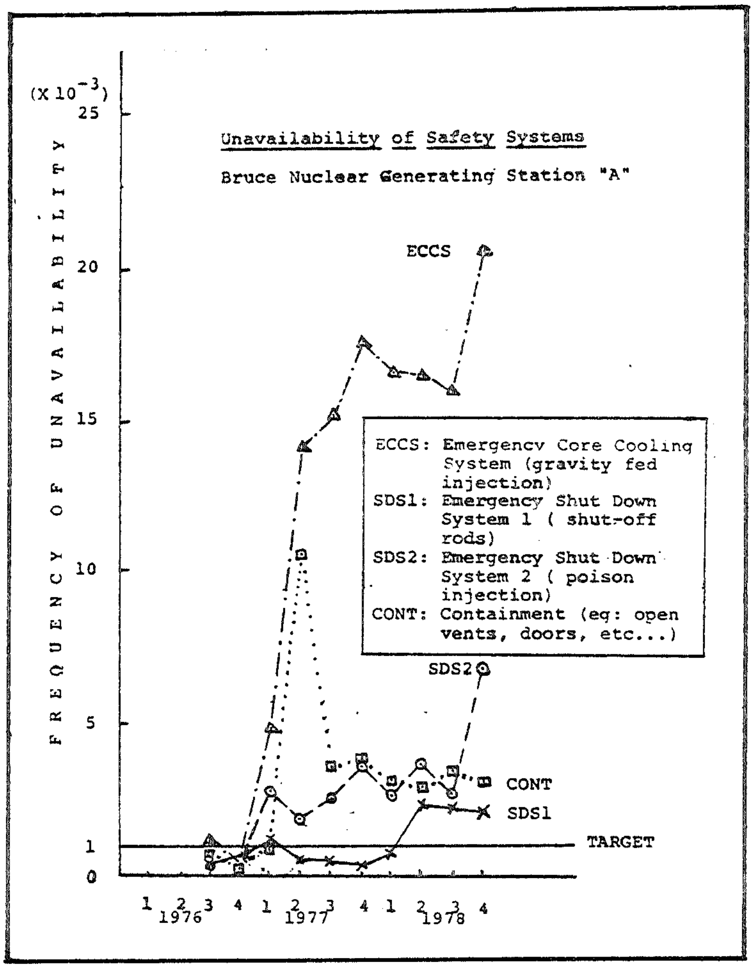
Unavailability of Safety Systems

Bruce Nuclear Generating Station "A"



The graph represents the *unavailability* of the safety systems at the Bruce Nuclear Generating Station, "A", for the time period 3rd quarter 1976 through to 4th quarter 1978. Each point represents the frequency that a specific system was not operable within the corresponding time interval. All values, as indicated on the vertical axis, must be multiplied by the factor 10^{-3} (ie: .001 or 1/1000).

This number corresponds to the AECB licencing criteria which stipulates that all safety systems must be available 99.9% of the time - this is the "target" as indicated in the figure. Over the course of one year, this implies that each system cannot be unavailable for more than *7.5 hours*. The Emergency Core Cooling System (ECCS), for example, was not operable for 39 hours (ie: total time allowed for unavailability / number of time periods X frequency of unavailability = 7.5/4 X 21) during the fourth quarter 1978; this is *21 times above the target*. It is important to note that the target level was exceeded by all systems over the 3 year time span.

The graph appeared in a letter from T. J. Malloy, Manager, Power Reactor Division of the AECB to L. W. Woodhead, Director, Nuclear Generation Division of Ontario Hydro, June 15, 1979 (SCOHA Exhibit E-71).

March 19, 1980

Gordon Edwards

ONTARIO LEGISLATURE INVESTIGATES NUCLEAR SAFETY

In the wake of the Three Mile Island accident, the Ontario Legislature has become involved in matters of nuclear safety.

In June there was a full Parliamentary debate on a resolution to restrain Ontario Hydro from operating the small N.P.D. reactor north of Ottawa until serious allegations concerning the safety of the plant had been heard by the Federal Appeals Court. The resolution was defeated by a narrow margin (two votes).

In addition, the Legislature's Select Committee on Ontario Hydro Affairs held 13 weeks of hearings on CANDU safety during 1979, culminating in an Interim Report to the Legislature in December. Copies of the transcripts and the Interim Report have been deposited in the Legislative Libraries of every province in Canada. The safety hearings are expected to conclude in March 1980.

The Interim Report: Recommendations

The Committee's Interim Report, entitled The Safety of Ontario's Nuclear Reactors, contains a number of constructive recommendations (20 in all). If implemented, they will go a long way towards clarifying the nature of the risks associated with CANDU reactors:

* For the first time in the 40-year history of nuclear power in Canada, an attempt will be made "to analyze the likelihood and consequences of a catastrophic accident in a CANDU reactor". (Recommendation VII)
* All pertinent information on nuclear safety will be made publicly available, on a routine basis, in as accurate and comprehensible a form as possible. (Recommendations I, II, VI, XIX)
* An independent forum will be established wherein controversies related to the biological effects of low-level ionizing radiation and the adequacy of existing exposure standards can be publicly scrutinized and acted upon. (Recommendation III)

* The procedures and criteria by which nuclear reactors are licensed for operation will be made much more explicit. (Recommendations VIII, X, XI, XII, XIII)
* Increased attention will be focused on human error, which is acknowledged to be the weakest link in the chain of cause and effect that may lead to major reactor accidents. (Recommendations IV, V, XIV)
* The effectiveness of the Atomic Energy Control Board will be strengthened, largely as a result of the greater degree of public accountability which will result from the other proposed changes. (Recommendations IX, XV, XVI, XVII, XVIII, XX)

There are all welcome developments, long overdue.

Unfortunately, the text of the Report is not as constructive as its recommendations. There one encounters a number of inconsistencies, confusions, and oversights of a rather fundamental nature. In addition, the superficial tone of political expediency adopted in the Report does not adequately reflect the seriousness of the issue or the quality of the testimony heard to date. Perhaps the Final Report will avoid some of these pitfalls.

The Safety Record: Good and Bad

One of the accomplishments of the Select Committee is the liberation of about four tons of hitherto confidential safety documents from Ontario Hydro, AECL and AECB. At the insistence of the Committee, these documents have been made available for public perusal.

For the first time, interested citizens have an opportunity to glimpse the strange and murky world of significant event reports, forced outages, safety system unavailability, and regulatory squabbles. The result is impressive and not altogether reassuring, as the documents show that:

> "accidents, mistakes, and malfunctions do occur in nuclear plants: equipment fails; instrumentation gives improper readings; operators and maintainers make errors and fail to follow instructions; designs are inadequate; events that are considered 'incredible' happen. It is a sobering reality, but one we must accept, that no matter how careful we are, we must expect accidents; we must anticipate the unexpected." (Interim Report, p.33)

Despite the disturbing regularity with which things seem to go wrong in Ontario's nuclear power reactors, the Committee was consoled by the fact that the public has never been exposed to highly dangerous amounts of radioactivity as a result of these untoward incidents. Indeed, "the incidents themselves do give some assurance that CANDU reactors are properly designed and operated. When incidents happen, the reactor does shut down safely, and there is enough redundancy that in the incidents that have happened, the public has not been at risk." (pp.33-34). Nowhere in the Report are the Chalk River accidents of 1952 and 1958 mentioned -- not even briefly!

Radioactive Pollution Not a Major Concern

In the absence of serious accidents, the worst the public has to fear from the operation of CANDU reactors is exposure to routine radioactive emissions, which are usually quite low. Even when serious spills occur, such as the 28,000 curies of tritium that leaked into the water system near Pickering in February of 1979, the Committee found little cause for concern. After all, "the irradiated water was well diluted by the time it reached the municipal water supply". In the Report, no mention is made of the fact that the Pickering Town Council was not informed of the spill in question until three months after the fact.

Nor is there any reference to the testimony of Dr. Edward Radford, Chairman of the prestigious B.E.I.R. Committee of the U.S. National Academy of Sciences. Dr. Radford told the Committee in July that tritium appears to be considerably more hazardous than previously thought, and that sharp "pulses" of tritium may be particularly harmful to the developing foetus if a pregnant woman drinks the tritium-contaminated water at a critical stage in foetal development. The Select Committee likewise makes no mention of the ongoing concern expressed by the federal Department of the Environment to the Porter Commission that tritium levels in the Great Lakes are measurably increasing and may become a problem by the turn of the century.

> "The Committee was impressed, however, by the continuing attention that must be paid to questions of radiation health effects... Studies that appear to show increased danger at low levels need continuous review... The basic Canadian regulatory standards come from a body [ICRP] reputed to be dominated by those the critics call the nuclear establishment... the Committee believes that there is an additional need for a body that can focus on Ontario problems, that can do so openly and with participation from the public, and that can relate directly to the federal body [AECB]." (pp.23-24)

Evidence related to another chronic pollution problem associated with CANDU reactors (the buildup of radioactive carbon-14) will be heard by the Committee early in 1980.

<u>Catastrophe Risk "Acceptable"</u>

Although nuclear reactors are relatively non-polluting in normal operation, the Report makes it plain that "catastrophic accidents" are an inescapable possibility:

> "It is not right to say that a catastrophic accident is impossible... The worst possible accident could involve the spread of radioactive poisons over large areas, killing thousands immediately, killing others through increasing susceptibility to cancer, risking genetic defects that could affect future generations, and, possibly contaminating for further habitation or cultivation, large land areas." (pp.9-10)

The reader is left to speculate whether these "large land areas" might include downtown Toronto, about 20 miles from Pickering.

Reassuringly, however, "the Committee found that the chance of a very serious accident occurring in any single reactor is extremely small and, on the basis of the information considered to date, that the reactors are therefore 'acceptably safe'. In so doing, the Committee recognized fully that it was making a political/societal judgment that the benefits derived from its nuclear reactors are worth the risks incurred by the people of Ontario." (p.32). Thus the Committee has decided on <u>political grounds</u> that the probability of a serious reactor accident -- whatever it may turn out to be in precise scientific terms -- is acceptably small.

Of course, the Committee members could have reached a more cautious conclusion. They could have stated that there is insufficient evidence to determine whether or not the catastrophe risk posed by CANDU reactors is acceptable or not. This statement would not necessitate shutting down operating reactors, as such a move would be very expensive, and there is as yet no compelling evidence that they constitute an unacceptable risk. However, such a statement would clearly signal the need for further, more detailed studies before any expansion of the nuclear industry is allowed to occur. It would lend a sense of urgency to Recommendation VII.

Instead, the Committee members chose to express an outright political judgment in favour of nuclear power, by saying that the chance of a serious accident is "acceptably small". They were careful to hedge their bets by adding: "This basic conclusion is, of course, subject to change following further Committee investigation before the final report is completed." (p.2).

What makes their affirmation of faith so remarkable is the enormous degree of uncertainty that surrounds the subject of probabilities as applied to reactor accidents. The Select Committee has chosen to base its most important conclusion on the most controversial and least reliable concept in the entire field of reactor safety analysis.

Rasmussen Probabilities for Meltdowns

In the early '70's, Dr. Norman Rasmussen of M.I.T. spent three years and four million dollars trying to determine the probability of a complete core meltdown in an American light water reactor. But despite all his efforts, he couldn't even pass the exam. Shortly before the Three Mile Island accident, the U.S. Nuclear Regulatory Commission publicly repudiated the Rasmussen probability estimates by withdrawing any "explicit or implicit support" for them. The N.R.C. announced that it had "no confidence" in Rasmussen's numbers.

The N.R.C. action was prompted by the findings of a special scientific review of Rasmussen's work carried out by the Lewis Committee. The review, completed in September 1978, showed that there are very fundamental questions about the validity of Rasmussen's calculations. The Lewis Committee found a host of problems "ranging from lack of data on which to base input distributions to the invention and use of wrong statistical methods". At one point, referring to a particularly important probability calculation by Rasmussen, the Lewis Committee observed that "the arbitrariness of this [Rasmussen's] procedure boggles the mind".

In view of the Lewis Committee findings, the N.R.C. found it prudent to warn against the use of such questionable probability figures as a basis for deciding public policy.

Lack of Canadian Studies

As it happens, Ontario Hydro and Atomic Energy of Canada Limited both leaned very heavily on Rasmussen's Report during the Porter Commission Hearings (Ontario Royal Commission on Electric Power Planning). They wanted to "show" that the probability of a major accident in a CANDU reactor is acceptably small. Because they had not done their own detailed studies, they expressed confidence in Rasmussen's calculations. In so doing, they revealed that they do not know, in any technical sense, what the probability of a meltdown in a CANDU reactor is.

It is not clear how, in view of these circumstances, the Committee could have possibly "found" that the chance of a meltdown in a CANDU reactor is either acceptable or unacceptable.

The Select Committee knows that Hydro still hasn't done its own meltdown studies. That's why the Interim Report recommends that AECB should get busy and do them. (After all, why should Ontario pay for the research when Ottawa may be persuaded to pick up the bill? Taxpayers from other provinces surely won't object; they never have in the past!)

In view of Hydro's track record, the Committee would have been wiser to reserve judgment until the evidence is in.

Probability Estimates for a CANDU Meltdown

Following the Three Mile Island Accident, Ontario Hydro spokesmen tried to allay public apprehensions by citing probability figures for meltdowns which had already been thoroughly discredited. Indeed, the Hydro number quoted on radio and TV was about 50 times more reassuring than Rasmussen's number (one in a million compared with one intwenty thousand per reactor per year).

Dr. Arthur Porter, Chairman of the Royal Commission on Electric Power Planning, felt compelled to go on radio himself to state that Hydro's probability estimate was about a hundred times too low. Having carefully listened to 15 days of detailed cross-examination on CANDU safety, and having studied the relevant background documents, the Porter Commission concluded 18 months ago that one in ten thousand per reactor per year was a "more realistic" estimate for the probability of a meltdown in a CANDU (A Race Against Time, p.78).

Although one-in-ten-thousand sounds quite small, it is actually uncomfortably large. With a thousand reactors operating worldwide some time soon, it means one meltdown per decade, on the average. And as the Royal Commission pointed out in their report A Race Against Time, this estimate could be out by a factor of five either way. If the true probability is five times worse, then even with 50 reactors operating at some time in the future here in Canada, one might expect a meltdown about once every 40 years.

In the Toronto area, 12 reactors are already committed -- eight at Pickering and four at Darlington. Each reactor is expected to have a lifetime of 30 years. Using the Porter Commission probability estimate, the chances that at least one of these reactors will suffer a core meltdown during its lifetime can be easily calculated:

(1/10,000) x (12 reactors) x (30 years) = 1/27 (approx.)

This is hardly a low probability. It is one-third higher than the probability of rolling a "twelve" with two dice (which is one in thirty-six). And if the Porter Commission estimate happens to be low by a factor of five -- which is a distinct possibility -- then the probability of a nuclear disaster from one of these plants near Toronto is about 1/6 -- which is the probability of rolling a "one" with a single die!

An Unanswered Question

Does the Select Committee regard this probability estimate of one in six as "acceptably small"? The Interim Report would suggest that it does. Perhaps in its Final Report the Committee will give some indication of just how large the probability of a serious accident would have to be before it would become "unacceptably large". Or perhaps the Committee members have concluded that the risk is worth taking, no matter what the probability may be. If so, they should say so. Not everyone would agree.

The Ostrich Syndrome

There is a disturbing possibility that the Select Committee is simply refusing to face facts by adopting the traditional ostrich-like posture. The Interim Report maintains a stony silence on mundane matters like evacuation plans, emergency medical facilities for radiation victims, insurance mechanisms to compensate for billions of dollars in offsite property damage, and suchlike details. One gets the impression that the Committee members have persuaded themselves that no serious accidents will ever actually happen at a CANDU nuclear power station.

One wonders how the Committee managed to arrive at such a conclusion, if indeed it did. In November, just before the Interim Report was published, a prominent U.S. pro-nuclear scientist named Alvin Weinberg was telling a United Nations Technology Conference in Montréal that by the year 2020, we could expect an average of one meltdown every two years, worldwide.

There is evidence, however, that the Select Committee is gambling that it won't ever happen, and is using scientific double-talk to avoid the issue. The Interim Report states:

> "A key element in determining what is 'reasonable' is the technical concept of risk. In the scientific community, risk is a precise term, defined mathematically as probability times consequence..." (p.10)

(Isn't science wonderful? We do not know the probability accurately, we have not measured the consequences, but nevertheless risk is a "precise term" equal to the product of these enigmatic quantities!)

The Interim Report gives no consideration to the potential consequences of a meltdown accident. The Committee does not even discuss the most obvious question of all: how do you clean up the mess after a CANDU reactor has melted down, and how much will it cost? In a minority report, the four N.D.P. Committee members call attention to this glaring inconsistency:

> "In Committee Hearings, we were told many times that RISK= PROBABILITY x CONSEQUENCE. Without any serious evaluation of the CONSEQUENCE of a catastrophic accident, we do not feel that a proper evaluation of RISK has yet been undertaken by Hydro or the federal or provincial governments or indeed by this Committee." (p.63)

Consequences of a Meltdown

In the event of a core meltdown, U.S. studies have shown that "the molten mass would be expected to eat its way through the concrete floor into the ground beneath" (Executive Summary, Rasmussen Report). However, Ontario Hydro spokesmen told the Select Committee and the Porter Commission that the "meltthrough" phenomenon would not occur in the event of a meltdown at a Pickering reactor. When challenged to prove this remarkable claim, which is in direct contradiction to all published studies on the subject, Ontario Hydro declined to do so. The Select Committee did not pursue the subject nor did it see fit to mention the dispute in the Interim Report.

The cleanup costs following the Three Mile Island accident were originally estimated at $400 million, but the Kemeny Commission Report indicated that those costs will probably exceed $1 billion. No one has yet estimated the cost of decommissioning a power reactor whose core has melted down.

If Ontario Hydro is right and the molten core does not penetrate the concrete floor, whom will they send into the reactor building to clean up the mess? Indeed, how can they clean up the mess? How will they package the congealed core materials, and where will they send them? How much radioactivity will have to be released into the atmosphere or into the Lake to make it safe enough for the workers inside? How many workers will be required for the job, what kind of pay will they require, and what levels of radiation will they be exposed to?

On the other hand, if Hydro is wrong and the molten core does penetrate the concrete floor within 20 hours or so after core melting begins, how much of that enormous radioactive inventory will find its way into Lake Ontario? What kind of steam explosions might occur when the molten core comes in contact with soil moisture? How will Hydro deal with the residual molten blob, which the Rasmussen Report says will reach a maximum diameter of about 50 feet and which will remain partially molten for about a year?

On all these matters the Interim Report is silent. The Report does not even inquire as to how many days the vacuum building would be able to contain the radioactive fission products without pumping them out into the atmosphere or dumping them into the Lake. Nor does the Report comment on the economic repercussions in Ontario resulting from the prolonged forced shutdown of eight reactors on the Pickering site, representing more than 4000 Megawatts of electrical power. (Actually, a major accident would probably necessitate the shutdown of all reactors in Ontario, thereby bringing about the kind of massive and prolonged energy shortages that nuclear power is supposed to prevent.)

In view of these oversights, it is difficult to avoid the conclusion that the Select Committee does not believe that such serious accidents are really possible.

Magical Materials Don't Melt

There is a key passage in the Report (pp.17-20) -- a kind of "worst case" scenario -- which may explain why the Committee has such faith that catastrophic accidents won't happen. Basically, it is assumed that a complete loss of cooling of the reactor takes place: this entails a loss of regular cooling (including the primary heat transport system and the moderator cooling system) accompanied by a loss of emergency cooling. According to expert U.S. testimony, such an accident would surely lead to a complete core meltdown. However, the Select Committee Report makes no mention of fuel melting whatsoever in this passage (though it does discuss the melting of the zirconium sheath which encases the fuel rods).

The passage in question may reflect a fundamental misunderstanding of the rudiments of reactor safety on the part of the Select Committee members. If they do not believe that fuel melting will occur, even with no cooling whatsoever, then of course they can all relax and stop worrying about catastrophic reactor accidents resulting from loss of cooling. The only indication that they are aware of the problem of fuel melting at all is on page 14 where they mention in passing that "temperatures can, in some circumstances, build up to a level close to, or even slightly beyond, the fuel melting point".

Testifying at the Cluff Lake Board of Inquiry in Saskatchewan, Norman Rasmussen (Head of Nuclear Engineering at M.I.T.) was much more blunt:

Rasmussen:	It's assumed that this accident lost the regular cooling.
Questioner:	Now, if the emergency core cooling system were not there I presume you'd have massive melting?
Rasmussen:	Indeed you would... That's one way to melt the core and it's one of the ways we looked at in our study.
	(Cluff Lake Transcript, vol.85, pp.8686-91)

Although Rasmussen's probability calculations have been disputed, his facts about fuel melting have never been disputed, because they are based on straightforward engineering calculations.

Testifying at the Royal Commission on Electric Power Planning, Mr. Liberty Pease (Manager of the AECL's Safety Analysis Division) confirmed Dr. Rasmussen's analysis:

Pease:	Under those circumstances, since one has heat production and since one doesn't have heat removal, the temperature must rise. It will continue to rise until the system melts.

Questioner: Mr. Pease, would that amount of heat production be sufficient... to cause massive melting of the core if there were no emergency cooling available?

Pease: Well, yes... If you produce heat and don't take it away, the material must melt unless it is a magical material.

(Porter Commission Transcripts, vol.136, pp.17357-58)

The heat referred to by Dr. Rasmussen and Mr. Pease is the so-called "decay heat", the result of the intense radioactivity of the nuclear fuel. Unlike the nuclear chain reaction, this decay heat cannot be shut off. Indeed, it persists for centuries, but at gradually diminishing rates. As Dr. Rasmussen told the Cluff Lake Board of Inquiry:

> "Right after shutdown the heating rate of the reactor is about 7% of the full power rate, and then as the radioactivity decays away it decreases; in a few hours it's less than 1% [of full power heat] and it keeps going down... Even one percent is a lot of heat, and if you don't remove that heat it's surely enough to overheat the fuel and melt it, and therein lies the problem that the reactor safety analyst must face -- to assure himself that the heat produced by these decaying fission products is reduced, is removed for periods of weeks or months... Every core melt in our study is assumed to melt its way through the bottom of the reactor." (Cluff Lake Transcript, vol.85, p.8686-87)

Moderator as a Heat Sink

One witness, Dr. Terry Rogers of Carleton University, told the Committee that the moderator cooling system would probably prevent fuel melting in a CANDU even if all other cooling systems fail. However, the Select Committee's "worst case" scenario assumes that the moderator cooling system is completely inoperative along with the main heat transport system and the emergency cooling system. The Interim Report cites no evidence whatsoever to indicate that fuel melting won't occur with all these cooling systems inoperative. It seems to be a gratuitous assumption which contradicts the basic laws of physics. As Mr. Pease has said, "If you produce heat and don't take it away, the material must melt unless it is a magical material".

By the way, despite Dr. Rogers' opinions on the subject, even if the moderator cooling system is operational and the other cooling systems aren't, fuel melting may still occur. Dr. Rogers' analysis (a 1978 study commissioned by AECB which indicates that the fuel won't melt) was hardly conclusive on this score:

> "The pressure tube [containing the fuel] will heat up and at about 1000°C it will sag...The fuel bundle slumps because it overheats... The zircaloy [surrounding the fuel] would eventually melt... The fuel reaches a maximum [calculated] temperature around 2600°C, which is about 200°C below the melting point...
>
> "Once you get up to about 1000°C, there can be a very rapid [chemical] reaction between the zircaloy and the steam. This generates heat... This [reaction] was ignored in [my] analysis... If you had just the right steam flow it [the chemical reaction] would result in about a 200°C increase in temperature... It would get close to the melting point [of the fuel], right... If you take that further to the peak, it would be slightly above the melting temperature... Certainly from my analysis I couldn't say definitely that the UO_2 fuel won't melt...
>
> "[Even without considering the zircaloy-steam reaction] there is enough uncertainty in my analyses that you could say, 'You are predicting 2650°' or whatever it is -- it might be 2750°C -- 'which is getting close to the melting point [of the fuel]'. In other words, there are always uncertainties in any analyses..." (Dr. Rogers' Testimony, Select Committee Transcripts, July 23)

The melting point of UO_2 fuel is 2800°C (about 5000°F). Since Dr. Rogers ignores the extra heat produced both by zircaloy combustion and by hydrogen combustion, his conclusions are not as firm as one might hope.

Other Confusions

Not only does the Select Committee Report ignore fuel melting as a possibility in its "worst case" scenario, it also reveals a number of basic confusions:

* It is not necessary to assume, as the Report does, that the initial pipe break "is the worst kind and in the worst location" (p.18). A small pipe break is just as bad as a large pipe break if emergency cooling is totally unavailable, because the core is left uncooled, "causing the fuel bundles to fall apart and the pressure tubes to split and break" (p.19).

COMMENTS ON THE DIAGRAMS

FUEL BUNDLES AND FUEL SHEATHS

The accompanying illustration of a CANDU fuel bundle shows how the zircaloy sheaths completely contain the uranium oxide fuel pellets (visible only through the "cutaway" portion). Accordingly, radioactive fission products cannot escape from the fuel bundle unless this protective sheath is damaged (e.g. by overheating).

Prolonged overheating can melt the sheath at about 1200°C, but the sheath will rupture before it melts because of other factors (softening of the metal and "ballooning" caused by gas pressures).

As soon as the sheath is damaged, all of the radioactive fission gases (krypton-85, argon-41, and up to 2% of the iodine-131) are released from the fuel bundle into the piping system. However, most of the radioactive inventory (including strontium-90, caesium-137, ruthenium-103, technetium-99, plutonium-239, and the bulk of the iodine-131) cannot escape unless the ceramic pellets overheat and melt at a temperature of about 1800°C.

Without fuel melting, a catastrophic release of radioactive materials from a nuclear power plant (causing massive radiation doses and long-term offsite contamination) is simply not possible.

CALANDRIA TUBES AND PRESSURE TUBES

The core of a CANDU reactor is a large cylindrical vessel lying on its side, penetrated by hundreds of hollow "calandria tubes" which run horizontally from end to end. This vessel is filled with heavy water "moderator" (outside the calandria tubes).

Inside each calandria tube is a smaller, concentric "pressure tube" which is made of zircaloy. Inside each pressure tube there are eight or twelve fuel bundles (depending on the design of the reactor). The fuel bundles generate heat. Coolant flows through the pressure tubes to transfer heat away from the core (to the boiler).

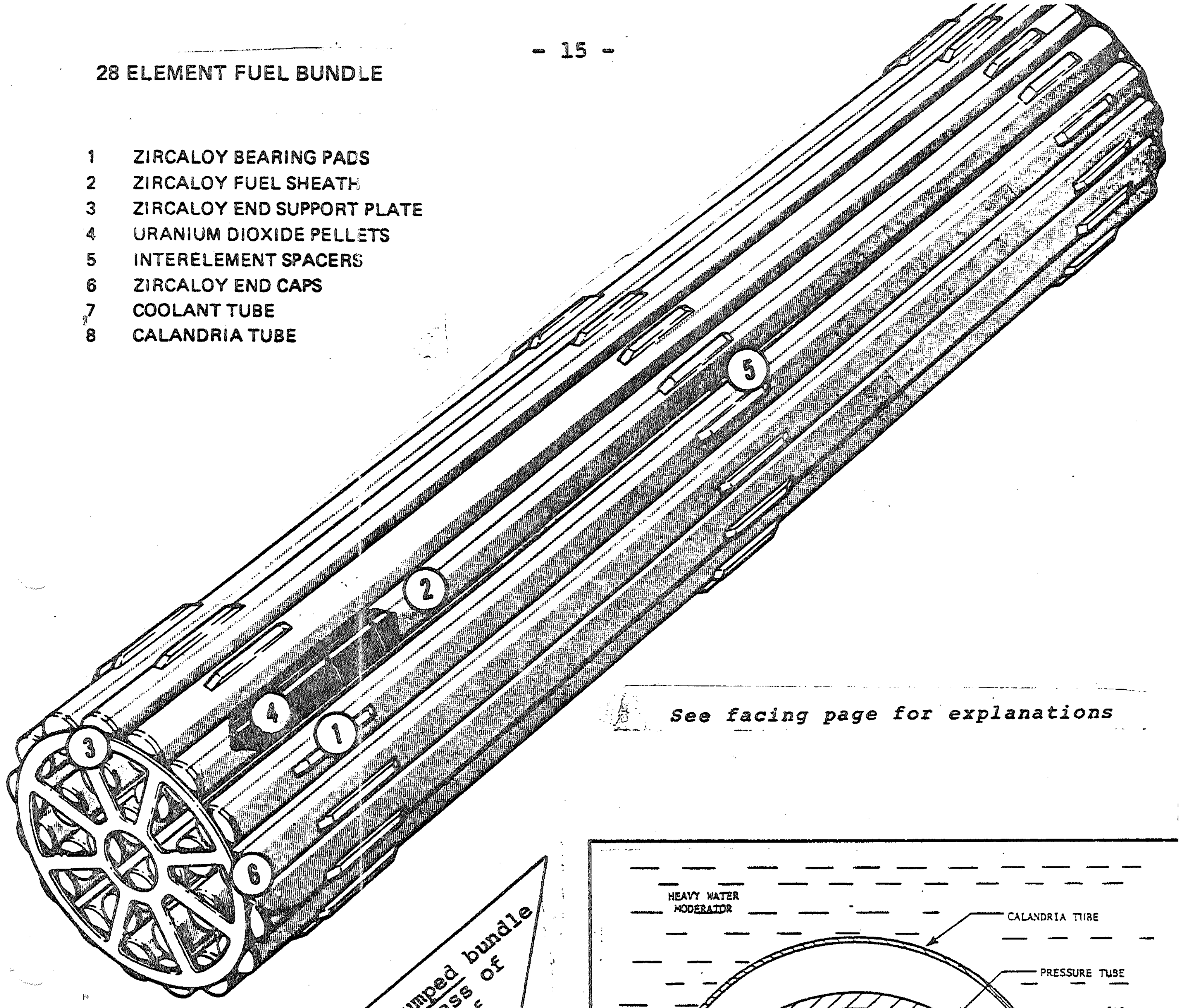
If the regular coolant is lost because of a pipe break, and if no emergency cooling is available, then the fuel "dries out" and begins to overheat. The fission reaction can be shut off immediately, but the "decay heat" produced by the intense radioactivity of the fission products cannot be shut off. This heat will cause the zircaloy pressure tubes and fuel bundles to sag and slump at about 1000°C, as shown in the cross-sectional diagram which is inset on the opposite page.

In his testimony to the Select Committee, Dr. Rogers argued that the decay heat can be transferred to the moderator as soon as the slumped pressure tube comes in contact with the calandria tubes, as shown, without exceeding the fuel melting temperature. However, he ignores the fact that zircaloy will burn very vigorously in a steam atmosphere at about 1000°C, adding more heat to the system and possibly driving the fuel temperature beyond its melting point.

When zirconium reacts with steam, more than three times as much heat is produced as when an equivalent amount of sodium reacts with water. Moreover, hydrogen gas is produced as a byproduct of the reaction, which is also capable of burning or exploding, thereby adding even more heat to the system. Not very reassuring.

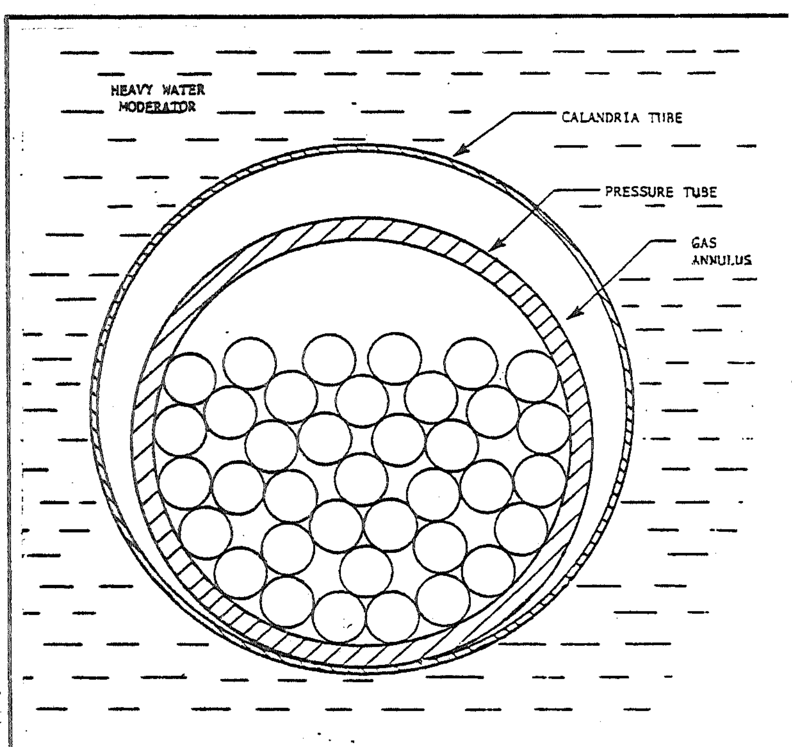
28 ELEMENT FUEL BUNDLE

1 ZIRCALOY BEARING PADS
2 ZIRCALOY FUEL SHEATH
3 ZIRCALOY END SUPPORT PLATE
4 URANIUM DIOXIDE PELLETS
5 INTERELEMENT SPACERS
6 ZIRCALOY END CAPS
7 COOLANT TUBE
8 CALANDRIA TUBE



See facing page for explanations

In reality, the slumped bundle would be just a mass of rubble made up of single fuel pellets.



SAGGED PRESSURE TUBE-SLUMPED BUNDLE CONFIGURATION

Inset: Loss of Geometrical Integrity Caused by Overheating

Taken from SCOHA Exhibit E-52
submitted by J. T. Rogers, July 23, 1979.

* It is not necessary for the "pressure relief valves on the calandria to open, permitting radioactive steam to flash into the reactor building" (p.19). In the circumstances described, radioactive steam will automatically and immediately exit into the reactor building through the same hole in the piping system that caused the original loss of coolant.

* It is not necessary that "the containment system fails to function properly" in order for radioactive gases to be released to the environment. The overheated and damaged fuel will generate large volumes of non-condensible gases that will require purging to the atmosphere (even with a vacuum building).

Dual Failures and Triple Failures

Perhaps as a result of these and other basic confusions, the Select Committee has arrived at the dubious conclusion (p.19) that a catastrophic accident in a CANDU reactor would have to involve, "at the least, a 'triple failure' accident" (involving the simultaneous failure of three independent systems, such as regular cooling, emergency cooling, and containment). This conclusion is unambiguously stated on page 42 of the Interim Report:

"1. A single failure accident should be a very infrequent event (say once every three years) and its consequences should fall within the safe limits set for releases during normal operation.

2. A dual failure accident should be expected to occur very rarely (about once in every 3000 reactor years) and its consequences should be well below dangerous radiation levels.

3. A triple failure, or catastrophic accident, must be expected to be a highly improbable event."

According to current AECB licensing criteria, the public must be protected in the event of any hypothetical dual failure accident. Triple failure accidents are beyond the scope of the licensing criteria.

The difficulty with this neat dissection of possibilities is that it may not correspond to reality. A dual failure which is sufficiently severe may automatically trigger a triple failure, so that the two categories overlap. For example, a loss of coolant coupled with a loss of emergency coolant, precipitating a meltdown, would almost certainly breach the containment, thereby turning a dual failure into a triple failure.

The probability of a small pipe break is between 1/10 and 1/100 (according to the Point Lepreau Safety Report) and the probability that emergency cooling will be unavailable is also between 1/10 and 1/100 (according to Ontario Hydro operating records). The combined probability of this dual failure accident, which could be catastrophic, is between 1/100 and 1/10,000 per reactor per year -- somewhat higher than the Porter Commission's probability estimate for a CANDU meltdown. The average of these two extremes is 1/200. With 20 reactors operating, the average probability of such an accident is about 1/10 (10%) per year.

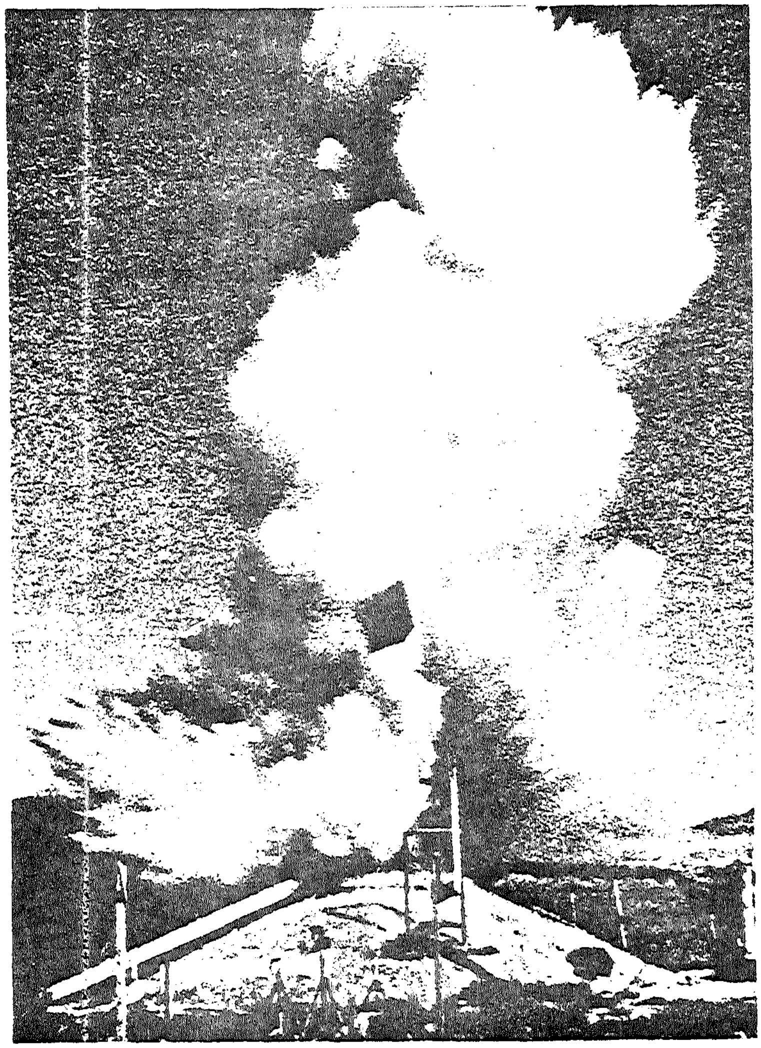
Destructive Power Surges

Another type of dual failure accident to worry about (at Pickering) is a loss-of-regulation (that is, a sudden power surge) combined with a failure-to-scram (due to unavailability of the fast shutdown system). According to Dr. Richard Webb, author of The Accident Hazards of Nuclear Power Plants (University of Massachusetts Press, 1976), such an accident in a CANDU reactor would definitely cause a core meltdown. The violence associated with such an accident is illustrated by the accompanying photograph, taken from Dr. Webb's book. (next page)

Dr. Webb writes, "It cannot be demonstrated theoretically that the containment of a CANDU would not be breached in this event" (p.152). He is referring to the containment shell bursting at a weak point due to overpressurization, or being penetrated by an energetic missile, thereby establishing a pathway for the atmospheric release of huge quantities of radioactive materials. In a footnote, he attributes his information to a conversation with W.G. Morison, chief design engineer for Pickering "A" Generating Station, Dec. 1975.

In AECL document PP-32 by Brooks and Renshaw, entitled "Safety in CANDU Power Reactors", this type of accident (loss of regulation with failure to scram) is also briefly alluded to: "Operation at excessive power levels [i.e. loss of regulation] could lead to melting of the UO_2 fuel and/or failure of the fuel

This photograph, taken from The Accident Hazards of Nuclear Power Plants by Richard E. Webb, University of Massachusetts Press, 1976 (p.68), illustrates the violence associated with an unterminated power excursion in a nuclear reactor.



A destructive nuclear runaway (power excursion) test conducted in 1954 of a small nuclear reactor—a 1/500 scale boiling water reactor. This test posed no serious hazard to the public, because little radioactivity was involved and the reactor was located in an Idaho desert. The core was small and operated a short time at a low power level prior to the test; so there was little buildup of fission products (no plutonium was used). Also, the fuel was metallic, which minimizes fission product release due to lower melting temperature, and only a fraction of the fuel melted (no afterheat was involved).

ABOUT THE CANDU

"An autocatalytic power excursion accident [loss of regulation] would definitely occur during a loss-of-coolant accident, if a reactor SCRAM fails to occur, which would cause a core meltdown... It cannot be demonstrated theoretically that the containment would not be breached in this event... The consequences of such an accident in CANDU could be disastrous..." (p.152) sheaths and/or overpressure rupture of the heat transport system" if not promptly terminated [i.e. failure to scram]. It is later pointed out that in such an event, "the containment might not be able to cope with the energy release", and in certain circumstances even scramming the reactor might not prevent these severe consequences from occurring. Such an accident could be a truly catastrophic event.

At Pickering "A", there were six loss-of-regulation accidents within four years (A Race Against Time, p.79). Fortunately, the shut-off rods functioned as designed in each case, being driven vertically into the core of the reactor with great force, shutting the reaction off within two or three seconds and preventing a catastrophic failure. However, at both Pickering "A" and Bruce "A", AECB documents reveal that the fast shutdown systems have been frequently unavailable -- much more often than they are supposed to be. If these critical safety systems happen to be unavailable during a loss-of-regulation accident, we may have to kiss Pickering good-bye, and possibly parts of Toronto as well.

Back in 1952, the small NRX reactor at Chalk River underwent a violent power excursion (i.e. loss of regulation plus failure to scram) that destroyed the core of the reactor, causing some fuel melting in the process. Unaccountably, the shut-off rods failed to fully descend into the core. A series of hydrogen gas explosions (or steam explosions) hurled the four-ton gasholder dome four feet through the air where it jammed in the superstructure. Thousands of curies of fission products were released to the atmosphere, and a million gallons of radioactively contaminated water had to be pumped out of the basement and disposed of in shallow trenches. Young Jimmy Carter, then a nuclear engineer in the U.S. Navy, was among those who participated in the NRX cleanup. This bit of history shows that such accidents do occur, improbable as they are. Because NRX is a pygmy compared with Pickering, the result of the 1952 accident was not a catastrophe. At Pickering, such an accident would almost surely be a catastrophe.

Since there is only one fast shutdown system in each of the Pickering "A" reactors, a loss of regulation with failure to scram is yet another example of a hypothetical dual failure accident which would trigger a triple failure either by penetrating the containment shell at a weak point and/or by precipitating a meltdown. It is therefore apparent that the Pickering "A" reactors are not operating in conformity with the AECB dual failure licensing criterion.

The probability of a loss-of-regulation is about 1/3 (AECB target, already exceeded), and the probability of unavailability of the scram system is about 1/1000 (AECB target, already exceeded). The combined probability of such an accident at Pickering is therefore about 1/3000 per reactor per year (optimistic!) -- considerably larger than the Porter probability estimate for a meltdown. The chance that such an accident will occur at one of the four Pickering reactors during their useful lifetime of 30 years is about 1/3000 x 4 x 30 = 1/25. Not very reassuring.

Additional Scram Systems

In the Bruce reactors and in subsequent CANDU designs, unlike the Pickering "A" reactors, there are two fully independent fast shutdown systems. This makes the loss-of-regulation-plus-failure-to-scram scenario truly a triple failure accident at Bruce. It also sharply reduces the probability of such an accident occurring. That is small consolation to people living near Pickering.

Ironically enough, the Select Committee singles out the second fast shutdown system at Bruce for special criticism (p.48), yet fails to point out that the Pickering reactors -- sited in a more densely populated region than any other reactors in the world -- do not even have a second fast shutdown system. (The moderator dump at Pickering is too slow to qualify as a scram system; that is one reason it has been eliminated from subsequent CANDU designs.)

Nuclear critics have charged that Pickering "A" is operating in violation of its license because Ontario Hydro cannot demonstrate that the public will be protected in the event of the dual failure accidents described in the previous sections. During the hearings, it was suggested to the Select Committee that Pickering "A" be operated at reduced power (to provide an extra margin of safety against a destructive power surge) until it has been retrofitted with an additional scram system. However, the Committee did not see fit to record the suggestion in their Interim Report.

The Case of the NPD Reactor

Many of these seemingly abstract considerations are well illustrated by the case of the small, 22 MW Nuclear Power Demonstration (NPD) reactor situated at Rolphton, Ontario. In April 1979, local citizens, acting through the Renfrew County Citizens for Nuclear Responsibility (RCCNR), petitioned the AECB to hold public hearings on the safety of the plant. They alleged that the plant was operating in violation of its license because of admitted inadequacies in the emergency coolant injection system. (Analysis had revealed that any size of pipe break in any location on the "inlet" side of the reactor core would result in at least half the fuel failing, thereby releasing radioactive gases and iodine, contrary to assurances given in the NPD licensing documents.)

The AECB refused to grant the citizens' petition because modifications were being made to improve the safety systems of the NPD plant. Among other things, the calandria spray cooling system would be upgraded to compensate for the deficiencies in the emergency coolant injection system. These modifications had been going on for three years, during which time the plant was allowed to operate at full power. The Renfrew citizens still wanted a public hearing, so they lodged a complaint against the AECB with the Federal Court of Appeals.

At this point the Select Committee recommended to the Ontario Legislature that the NPD plant, which was then shut down for "routine maintenance", should remain shut down until the case had a chance to be heard in court. Premier Davis promised a full Parliamentary debate on the matter, but NPD was started up again the night before the debate was scheduled to take place!

Fuel Melting at NPD

Meanwhile, AECB released documents (dating from 1976) that revealed that fuel melting would occur in the NPD reactor if there were a loss of coolant coupled with a loss of emergency coolant: "The melting of the core was presumed to occur with complete incapacitation of the ECCS [emergency core cooling system] since no account was taken of the effect of the calandria spray system". (History of ECCS at NPD, by D.J. Martin of AECB, May 30, 1979) As it turns out, it was wise that no account was taken of the calandria spray system: subsequent analysis showed that a loss-of-coolant accident would have unavoidably flooded the pumps needed to drive the calandria sprays, rendering them useless.

Strangely enough, even though the core was presumed to melt, it was not presumed to melt down (let alone melt through)! Instead, AECL calculated radiation doses three kilometers away from the plant on the assumption that the containment would not be impaired in any way, that 97% of the radioactive iodine would stick to the metal surfaces inside the plant and not escape, and that the filters in the NPD stack would function at top efficiency. The result of this analysis was that an infant's thyroid might receive a maximum dose of 10,000 rads, and that 100 delayed deaths due to cancer might result from such an incident. (An infant receiving such a thyroid dose would no longer possess a thyroid gland, according to medical evidence.)

More recently, it has been pointed out that the 97% retention factor is probably too high, and that the filters would not work as well as planned in an emergency, because they would be damp from the steam.

It has also been discovered that the NPD emergency cooling system was manually shut off, while the reactor was operating, for more than a hundred days within the last year. If a small pipe break had occurred during that time, theory might have been translated into practice, and we might know for sure whether a meltdown would melt through.

In September, the Federal Appeals Court refused to hear the RCCNR case, suggesting that AECB was under no obligation to answer to the public for its decisions or to obey its own regulations.

Without going into much detail, the Select Committee's Interim Report concludes rather superficially that the NPD plant "was allowed to continue to operate because it met its original licensing criteria: the risk was equivalent to one expected death in 100 years, safer than a coal plant" (p.47). The Report does not comment on the fact that this highly theoretical calculation is based on the doubtful assumption that the NPD core can somehow manage to melt without actually melting down or melting through.

However, the Report does point out that "the time may not be far off when this plant will no longer be 'acceptably safe'. At that time Hydro and the AECB should have... no hesitation in closing the plant" (p.38).

Conclusion

There is much food for thought in the Interim Report, and a very healthy emphasis on the need for openness and public participation in matters of nuclear policy, especially those pertaining to safety. In addition, the transcripts are a gold mine of information.

There are indeed a great many oversights in the Report, only some of them touched on in the transcripts. In part, this is because the summer hearings were oriented exclusively to the cooling systems of the CANDU reactor. There were no hearings specifically on the electrical systems or the containment systems. There were no hearings on sabotage, or on external events (such as earthquakes, lightning strikes, airplane crashes, hurricanes, etc.).

For instance, the Report does not mention that Pickering is still using the same kind of flammable urethane sealant that contributed to the Brown's Ferry fire of 1975, or that a lightning strike at the NPD plant in 1979 brought the plant perilously close to a loss-of-cooling (as opposed to a loss-of-coolant) situation. Five agonizing minutes passed before power was restored to the NPD pumps; another few minutes and the fuel would have overheated.

Nevertheless, despite its shortcomings, the Select Committee has definitely broken new ground. For the first time in Canada, elected representatives have grappled with the technical problems surrounding reactor safety. They have made it much easier for other Canadian citizens to begin to learn the truth about nuclear hazards with some depth of understanding. They have dispelled the myth, once and for all, that reactor safety should be left to the experts alone.

Further Reading

1. The Safety of Ontario's Nuclear Reactors, Interim Report, Select Committee on Ontario Hydro Affairs, Ontario Legislature, Queen's Park, Toronto (December 1979).
2. A Race Against Time, Interim Report on Nuclear Power, Royal Commission on Electric Power Planning, ONtario Government Publications, 800 Bay St., Toronto (September 1978).
3. The Accident Hazards of Nuclear Power Plants, Richard E. Webb, University of Massachusetts Press, Amherst, Mass. (1976).
4. Documents on Nuclear Safety, Gordon Edwards, Canadian Coalition for Nuclear Responsibility, 2010 MacKay St., Montréal (December 1978).
5. Summary Argument to the Porter Commission, Ralph Torrie and Gordon Edwards, Canadian Coalition for Nuclear Responsibility, 2010 MacKay St., Montréal (April 1978). Contains sections on "CANDU Safety Standards" and "Fuel Melting Accidents".
6. "The Harrisburg Nuclear Accident Could Happen Here", CCNR Press Release and Background, Canadian Coalition for Nuclear Responsibility, 2010 MacKay St., Montréal (April 1979, 14 pages).
7. "Nuclear Fuel Melting as Seen by the Rasmussen Report (Wash-1400) with some additional comments re the CANDU reactor", Gordon Edwards, Canadian Coalition for Nuclear Responsibility, 2010 MacKay St., Montréal (July 22, 1979).

NOTE: APPROXIMATE VERSUS EXACT PROBABILITIES

In this article, the probability "P" that a mishap will occur when "n" reactors operate for a period of 30 years is calculated as follows: $P = 30nf$ (where "f" is the probability that such a mishap will occur in any one reactor in any one year). Please note that this is only an approximation to the correct formula: $P = 1 - (1 - f)^{30n}$. As applied in the article, the approximations are all very good, as you may check.

As a rule, when 30nf is much smaller than 1, the approximation is good. As 30nf approaches 1, the approximation gets increasingly worse. When $30nf = 1$, it is absurd. For example, if the probability of a small pipe break is $f = 1/30$ per reactor per year, it would be foolish to think that, over a 30-year life, the probability of a small pipe break in a single reactor is $P = 30nf = 30(1/30) = 1$ (which would indicate complete certainty). The correct answer is $P = 1 - (1 - 1/30)^{30} = 0.64 = 64\%$.

NUCLEAR FUEL MELTING

AS SEEN BY

THE RASMUSSEN REPORT (WASH-1400)

WITH SOME

ADDITIONAL COMMENTS RE THE CANDU REACTOR

Compiled by Dr. Gordon Edwards, Chairman, Canadian Coalition for Nuclear Responsibility

(This document was filed with the Select Committee as Exhibit E-60-C on July 25, 1979. All exhibit numbers cited herein, however, refer to the earlier hearings of the Ontario Royal Commission on Electrical Power Planning (RCEPP), commonly known as the Porter Commission.)

March, 1980

Introduction:

Despite its shortcomings, the Rasmussen Report provides valuable insight into the question of catastrophic reactor accidents.

To begin with, the report makes it clear that catastrophic accidents cannot occur unless a substantial portion of the nuclear fuel melts. This is because most of the fiercely radioactive fission products which are chemically and biologically active (including 98% of the iodine-131) are locked up in the ceramic fuel and cannot be released except by prolonged overheating and eventual melting of the fuel. Without fuel melting, you cannot have thousands of people hospitalized with radiation sickness, nor can you have long-lived radioactive contamination of soil and buildings (leading to the $14 billion in property damage discussed in the report).

Secondly, the report makes it clear that once a substantial portion of the fuel melts, it cannot be resolidified by any conventional cooling methods. Drowning the molten mass with water would only serve to form a solid crust, beneath which the melt would continue to generate heat faster than it can be removed. Since the melting point of the fuel (5000°F) is considerably higher than the melting point of any of the structural materials in the reactor building (including steel and concrete), it follows that the molten core will inevitably eat its way through the steel containment vessel and the concrete floor of the reactor building into the ground beneath.

Thirdly, the report indicates that without adequate cooling, the core of a reactor will melt spontaneously even after it has been shut down for days or weeks. The so-called "decay heat", resulting solely from the radioactivity of the fission products (which cannot be shut off) is sufficient to melt the fuel even after it has been removed from the reactor (e.g. in a spent fuel bay) unless there is sufficient cooling available. This no doubt is why Sir Brian Flowers (author of the U.K. Royal Commission Report on Nuclear Power and the Environment, 1976) concluded that if nuclear power had been developed before the time of the second world war, large parts of Europe would still be uninhabitable today (simply because of conventional warfare: bombs, sabotage, etc.). It is an important fact that a nuclear reactor cannot be shut down and left in a safe condition (REQUIRING NO MAINTENANCE) under emergency conditions. The possibility of a meltdown in a damaged reactor is ever-present.

Fourthly, the report indicates that there are many ways in which the containment shell can be damaged, releasing a deadly cloud of radioactivity into the atmosphere. Under meltdown conditions, the radioactivity released would be very intense and would cause severe illness and death within days among the surrounding population -- unlike the serious but non-catastrophic releases which occurred during the Three Mile Island accident. In such a circumstance, evacuation may be permanent, as the levels of radioactive contamination would make it extremely difficult, expensive, and dangerous to decontaminate. The report calculates that, in the worst scenario that it considers, about 45,000 people would require hospitalization with radiation sickness, of which about 3,300 would die. In one of the appendices, it is pointed out that those who survive this ordeal may require such radical medical treatment as bone marrow transplants.

In Canada, the nuclear authorities steadfastly refuse to talk about fuel melting accidents -- which are the only kind of reactor accidents with catastrophic potential. The Interim Report of the Select Committee on Ontario Hydro Affairs (December 1979) doesn't even mention fuel melting, except in one sentence of the report. This reflects an astonishing degree of dishonesty in dealing with the issue of reactor safety in Canada. For a discussion of possible fuel melting accidents in CANDU reactors, see the CCNR Summary Argument to the Porter Commission (available from CCNR for $2.00).

Gordon Edwards

I. EXCERPTS FROM PAGES 6 & 7 OF THE EXECUTIVE SUMMARY OF WASH-1400

2.6 HOW CAN RADIOACTIVITY BE RELEASED?

The only way that potentially large amounts of radioactivity could be released is by melting the fuel in the reactor core. The fuel that is removed from a reactor after use and stored at the plant site also contains considerable amounts of radioactivity. However, accidental releases from such used fuel were found to be quite unlikely and small compared to potential releases of radioactivity from the fuel in the reactor core.

The study has examined a very large number of potential paths by which potential radioactive releases might occur and has identified those that determine the risks. This involved defining the ways in which the fuel in the core could melt and the ways in which systems to control the release of radioactivity could fail.

2.7 HOW MIGHT A CORE MELT ACCIDENT OCCUR?

It is significant that in some 200 reactor-years of commercial operation of reactors of the type considered in the report there have been no fuel melting accidents. To melt the fuel requires a failure in the cooling system or the occurrence of a heat imbalance that would allow the fuel to heat up to its melting point, about 5,000°F.

To those unfamiliar with the characteristics of reactors, it might seem that all that is required to prevent fuel from overheating is a system to promptly stop, or shut down, the fission process at the first sign of trouble. Although reactors have such systems, they alone are not enough since the radioactive decay of fission fragments in the fuel continues to generate heat (called decay heat) that must be removed even after the fission process stops. Thus, redundant decay heat removal systems are also provided in reactors. In addition, emergency core cooling systems (ECCS) are provided to cope with a series of potential but unlikely accidents, caused by ruptures in, and loss of coolant from, the normal cooling system.

The Reactor Safety Study has defined two broad types of situations that might potentially lead to a melting of the reactor core: the loss-of-coolant accident (LOCA) and transients. In the event of a potential loss of coolant, the normal cooling water would be lost from the cooling systems and core melting would be prevented by the use of the emergency core cooling system (ECCS). However, melting could occur in a loss of coolant if the ECCS were to fail to operate.

The term "transient" refers to any one of a number of conditions which could occur in a plant and would require the reactor to be shut down. Following shutdown, the decay heat removal systems would operate to keep the core from overheating. Certain failures in either the shutdown or the decay heat removal systems also have the potential to cause melting of the core.

2.8 WHAT FEATURES ARE PROVIDED IN REACTORS TO COPE WITH A CORE MELT ACCIDENT?

An essentially leaktight containment building is provided to prevent the initial dispersion of the airborne radioactivity into the environment. Although the containment would fail in time if the core were to melt, until that time, the radioactivity released from the fuel would be deposited by natural processes on the surfaces inside the containment. In addition, plants are provided with systems to contain and trap the radioactivity released within the containment building.

Even though the containment building would be expected to remain intact for some time following a core melt, eventually the molten mass would be expected to eat its way through the concrete floor into the ground below. Following this, much of the radioactive material would be trapped in the soil; however, a small amount would escape to the surface and be released. Almost all of the nongaseous radioactivity would be trapped in the soil.

It is possible to postulate core melt accidents in which the containment building would fail by overpressurization or by missiles created by the accident. Such accidents are less likely but could release a larger amount of airborne radioactivity and have more serious consequences.

2.9 HOW MIGHT THE LOSS-OF-COOLANT ACCIDENT LEAD TO A CORE MELT?

Loss of coolant accidents are postulated to result from failures in the normal reactor cooling water system, and plants are designed to cope with such failures. The water in the reactor cooling systems is at a very high pressure (between 50 to 100 times the pressure in a car tire) and if a rupture were to occur in the pipes, pumps, valves, or vessels that contain it, then a "blowout" would happen. In this case some of the water would flash to steam and blow out of the hole. This could be serious since the fuel could melt if additional cooling were not supplied in a rather short time.

The study has examined a large number of potential sequences of events following LOCAs of various sizes. In almost all of the cases, the LOCA must be followed by failures in the emergency core cooling system for the core to melt.

2.10 HOW MIGHT A REACTOR TRANSIENT LEAD TO A CORE MELT?

The term "reactor transient" refers to a number of events that require the reactor to be shut down. These range from normal shutdown for such things as refueling to such unplanned but expected events as loss of power to the plant from the utility transmission lines. The reactor is designed to cope with unplanned transients by automatically shutting down. Following shutdown, cooling systems would be operated to remove the heat produced by the radioactivity in the fuel. There are several different cooling systems capable of removing this heat, but if they all should fail, the heat being produced would be sufficient to eventually boil away all the cooling water and melt the core.

In addition to the above pathway to core melt, it is also possible to postulate core melt resulting from the failure of the reactor shutdown systems following a transient event. In this case it would be possible for the amounts of heat generated to be such that the available cooling systems might not cope with it and core melt could result.

2.11 HOW LIKELY IS A CORE MELT ACCIDENT?

The Reactor Safety Study carefully examined the various paths leading to core melt. Using methods developed in recent years for estimating the likelihood of such accidents, a probability of occurrence was determined for each core melt accident identified. These probabilities were combined to obtain the total probability of melting the core. The value obtained was about one in 20,000 per reactor per year. With 100 reactors operating, as is anticipated for the U.S. by about 1980, this means that the chance for one such accident is one in 200 per year.

NOTE

Both AECL and Ontario Hydro, in their submissions to the Porter Commission, cited the Rasmussen Report (WASH-1400) as part of their 'proof' that nuclear risks are acceptable. See Exhibit 158 (pp.14-15) & Exhibit 2 (2.1-43/44).

In Exhibit 28, G. A. Pon, Vice President of Power Projects, said of WASH-1400: "Although the study was prepared in the US assessing the risks associated with their light water nuclear power plants, the findings should not be significantly different for the CANDU reactor."(p.5)

In RCEPP-R1061, p.8629, Dr. Rasmussen says of CANDU meltdown possibilities: "although the Canadian design philosophy differs in some of its approaches ... it achieves, in my judgment, about the same safety level as far as I can tell."

WORST CASE CONSEQUENCES IN WASH-1400:

45,000 nuclear sickness cases (hospital)
3,300 prompt deaths
45,000 fatal cancers (over 50 years)
250,000 non-fatal cancers (over 50 years)
190 defective children per year
$14 billion in property damage

These consequences are not the worst that can be imagined, however.

II. EXCERPTS FROM THE MAIN REPORT RE FUEL MELTING (WASH-1400)

Taken from pp. 25-29.

The identification of all significant sources of radioactivity, the fact that a gross release of radioactivity can occur only if fuel melts, knowledge of the factors that affect heat balances in the fuel, and the fact that mechanisms that could lead to heat imbalances have been scrutinized for many years, all provide a high degree of confidence that those accidents of significance to risk have been identified.

3.3 LOSS OF COOLANT ACCIDENTS

A LOCA would result whenever the reactor coolant system (RCS) experiences a break or opening large enough so that the coolant inventory in the system could not be maintained by the normally operating makeup system. Nuclear plants include many engineered safety features (ESFs) that are provided to mitigate the consequences of such an event.

The specific LOCA initiating events analyzed in this study are:

a. Large pipe breaks (6" to approximately 3 feet equivalent diameter).

b. Small to intermediate pipe breaks (2" to 6" equivalent diameters).

c. Small pipe breaks (1/2" to 2" equivalent diameter).

d. Large disruptive reactor vessel ruptures.

e. Gross steam generator ruptures.

f. Ruptures between systems that interface with the RCS.

The basic purpose of the ESFs is the same for both PWR and BWR plants. However, the nature and functions of ESFs differ somewhat between PWRs and BWRs because of the differences in the plant designs. A number of the ESFs are included in a group termed the emergency core cooling system (ECCS) whose function is to provide adequate cooling of the reactor core in the event of a LOCA. Other ESFs provide rapid reactor shutdown and reduce the containment radioactivity and pressure levels that result from escape of the reactor coolant from the RCS.

In early power reactors the power level was about one tenth that of today's large reactors. It was thought that core melting in those low power reactors would not lead to melt-through of the containment. Further, since the decay heat was low enough to be readily transferred through the steel containment walls to the outside atmosphere, it could not overpressurize and fail the containment. Thus, if a LOCA were to occur, and even if the core were to melt, the low leakage containments that were provided would have permitted the release of only a small amount of radioactivity.[2]

However, as reactors grew larger, several new considerations became apparent.

The decay heat levels were now so high that the heat could not be dissipated through the containment walls. Further, in the event of accidents, concrete shielding was required around the outside of the containment to prevent overexposure of persons in the vicinity of the plant. Finally, it became likely that a molten core could melt through the thick concrete containment base into the ground. Thus, new sets of requirements came into being.

Emergency core cooling systems were needed to prevent core melting which could, in turn, cause failure of all barriers to the release of radioactivity. Systems were needed to transfer the core decay heat from the containment to the outside environment in order to prevent the heat from producing internal pressures high enough to rupture the containment. Finally, systems were needed to remove radioactivity from the containment atmosphere in order to reduce the amount that could leak from the containment into the environment.

The major goal behind these changes was to attempt to provide ESFs designed so that the failure of any single barrier would not be likely to cause the failure of any of the other barriers. For example, if the RCS were to rupture, ECC systems were installed to prevent the fuel from melting and thereby protect the integrity of the containment. Other features were added to aid this positive objective. For example, additional piping restraints and protective shields were required to lessen the likelihood of ESF damage that could result from pipe whip following a large break in the RCS. Knowledge that large natural forces such as earthquakes and tornadoes could cause multiple failures led to design requirements that attempted to reduce the likelihood of dependent failures from such causes. Appendix IX provides more detailed descriptions of the above and many more of the safety features in current nuclear plants.

MORE ABOUT FUEL MELTING ....

Prior studies have indicated that a core meltdown in a large reactor would likely lead to failure of the containment (Ref. 1,2). Thus, a commonly held opinion regarding core melting is that such an event would result in a very serious accident with large public consequences. This is evidently one of the reasons that major safety efforts have been devoted to the prevention of core meltdown and little attention has been directed toward the examination of the potential relationships between core melting and containment integrity.

At two key stages in the course of a potential core meltdown there would be conditions which would have the potential to result in a steam explosion that could rupture the reactor vessel and/or the containment.[1] These conditions may occur when molten fuel would fall from the core region into water at the bottom of the reactor vessel or when it would melt through the bottom of the reactor vessel and fall into water in the bottom of the containment. It is predicted (see Appendix VIII) that if such an explosion were to occur in the reactor vessel, it may be energetic enough to change the course of the accident. For reactors enclosed in relatively large volume containments, it is considered improbable that a steam explosion outside the reactor vessel would rupture the containment. If a steam explosion were to occur within the reactor vessel, it is considered possible that both large and small containments could be penetrated by a large missile. Such occurrences might release substantial amounts of radioactivity to the environment. However, these modes of containment failure are predicted to have low probabilities of occurrence.

3.5 ACCIDENTS INVOLVING THE SPENT FUEL STORAGE POOL

In section 3.2 the spent fuel storage pool (SFSP) is identified as having a significant radioactivity inventory, second in amount to the reactor core. Further, the decay heat levels in freshly unloaded fuel assemblies that may be stored in the pool may be sufficiently high to cause fuel melting if the water is completely drained from the SFSP. Because the maximum amount of fuel stored in the pool immediately after refueling is smaller than that in the core and because it has had time (72 hours minimum) for radioactive decay, it is a less intense heat source than a reactor core (about one-sixth) and therefore melt-through of the bottom structure of the pool would occur at a much lower rate and, in fact, may not occur at all. On the average, fuel in the pool will have undergone about 125 days of decay, and it is questionable that such fuel would melt. However, to assure that the risk would not be underestimated, it has been assumed that even this fuel would melt.

Although the pool is not within a containment building, filters in the SFSP building ventilation system and natural deposition of radioactivity within the building both aid in reducing the amount of radioactivity that might be released to the environment in the event of a spent fuel accident.

The analyses of accidents that could potentially lead to loss of fuel cooling in the SFSP are discussed in section 5 of Appendix I. The most probable ways in which such accidents could occur have been determined to be the loss of the pool cooling system or the perforation of the bottom of the pool. The latter could occur, for example, by dropping a shipping cask in the pool or on the top edge of the pool. Both this type of accident and the loss of cooling capability, are of low likelihood.

"Right after shutdown the heating rate of the reactor is about 7% of the full power rate, and then as the radioactivity decays away it decreases; in a few hours it's less than 1% and it keeps going down.... Even one percent is a lot of heat, and if you don't remove that heat it's surely enough to overheat the fuel and melt it, and therein lies the problem that the reactor safety analyst must face -- to assure himself that the heat produced by these decaying fission products is reduced, is removed for periods of weeks or months.... Every core melt in our study is assumed to melt its way through the bottom of the reactor." Norman Rasmussen, testifying to the Cluff Lake Board of Inquiry, RCEPP-R1061, p.8686.

III. EXCERPTS FROM APPENDIX VIII OF WASH-1400 ON CORE MELTDOWNS

A1.1.2.3.1 Core-Meltdown Behavior. The behavior of a core during a meltdown accident is uncertain. No cores have been melted. Experiments involving more than a cupful of molten UO_2 are still in the planning stages. Some properties of molten UO_2 are known: melting point, boiling point, and heat of fusion. However, little is known about the viscosity, internal thermal convection, surface tension, or the metallurgical effects of various diluents. Nevertheless, considerable insight has been developed by people working in the field about the possible course of core meltdown.

Because the core power tends to peak (about 2.5 times average) at the center of the core and because the cladding-steam reaction increases the heatup rate of the hotter regions, core melting starts at the center of the core. Because of the power peaking and the presence of water in the bottom of the core, the core temperatures a foot away from the melted region are frequently calculated to be more than 1000 F below the melting point of the fuel. In these relatively cool regions, the UO_2 would remain solid although the cladding could be melted. Because the fuel rods in the core are relatively closely packed, there is not room for solid fuel pellets to fall out of the core nor for gross distortion of the solid portions of the core. In this situation, it is believed a region of solid rubble would form under the molten fuel, and the molten fuel would tend to be retained in the core. However, since the rubble continues to generate heat, it will eventually melt, and the increasingly larger molten region will move downward. If the pool moves downward fast enough, it will intercept the water that is boiling out of the bottom of the core. When this happens either (1) steam explosions will occur or (2) the boiloff rate and, therefore, the cladding-steam reaction rate will increase. When the molten region grows to include 50 to 80 percent of the core, it becomes questionable whether or not the molten region can be retained inside the core. At this time, the molten pool will be 3 to 4 ft thick, and will presumably be held up by a layer of rubble. When large fractions of the core are molten, the core-support plates and shrouds are also exposed to high thermal loadings. Failure of these major structural members would release the molten pool and either (1) the rest of the water boils out of the pressure vessel or (2) a steam explosion results.

CONTAINMENT VESSEL MELTTHROUGH

The processes by which the molten core interacts with the concrete floor of the containment building are very complex and not fully understood. In the absence of definitive experimental information it is only possible to estimate approximately the time required for penetration of the containment floor by the molten core. When the molten fuel (together with molten zirconium, zirconium oxide, steel, iron oxide, etc.) falls onto the concrete, vaporization of free water below the surface will cause spalling of the concrete and result in a very rapid penetration rate of the melt into the concrete. Based on the vaporization of the free water, initial spalling rates are calculated to be 15-30 ft/hr. As the concrete heats up it will give up its water of hydration at about 900 F; then at 1400-1600 F the limestone will decompose into CO_2 and CaO. It is expected that the water vapor and carbon dioxide would escape from the melt and be released to the containment atmosphere, but the calcium and silicon oxides would stay with the melt. As the products of concrete decomposition are absorbed and/or dissolved by the melt, the melt temperature will decrease until constituents of the mixture begin to precipitate. It is estimated that by the time the melt has penetrated through 1-1/2 feet of concrete, UO_2 would begin to precipitate from the multicomponent mixture. From this point on the progress of the melt through the concrete is controlled by the rate of decay-heat generation, i.e., the quantity of concrete penetrated is directly proportional to the energy available to decompose the concrete and raise the temperature of the products to the temperature of the melt.

After the initial rapid penetration of concrete by spalling, the melt is expected to remain relatively viscous, decomposing and dissolving concrete at a rate compatible with decay-heat generation. Because the silica and calcia that are introduced at the lower surface are less dense than the body of the melt, convection currents that should be established will prevent the center of the melt from reaching very high temperatures. Also, the carbon dioxide and water vapor released by the concrete, and/or their further decomposition products, will provide additional agitation as they rise through the melt.

The upper surface of the melt is likely to be covered with a solid crust and to be radiating heat to the remains of the pressure vessel and to the walls of the reactor cavity. If there is water in the cavity it will be vaporized by the melt, but even the continued addition of water would not avert containment meltthrough since the geometry of the molten mass is not favorable for effective cooling. If the structures above the melt reach elevated temperatures, they could fall into the melt.

That at least part of the mass of fuel, structural, and other material remain molten is required by heat-transfer considerations. If this mass should solidify at some point during the accident, then the only mechanism for dissipating decay heat would be conduction. For conduction to transfer the decay heat out of the largely low conductivity mass of material involved, temperatures at the center of the mass would have to exceed boiling points of some and melting points of all the constituents. Hence the conclusion that at least some of the mass will remain in a fluid state for considerable time, with convection within the melt tending to maintain the melt temperature near the effective melting point of the mixture. It is recognized that as the size of the melt increases and the heat-generation rate per unit volume decreases due to dilution, solidification must eventually occur. Solidification is not, however, expected to take place prior to the penetration of the containment foundation mat.

In estimating the time required to penetrate the 10-ft-thick foundation mat three different configurations were considered for the quantity of concrete decomposed by the molten core: (1) a 15-ft-diameter, 10-ft-high cylinder, (2) a 10-ft-radius hemisphere, and (3) a 32-ft-diameter, 10-ft-high cylinder. The first case assumed that the melt progressed downward through the concrete faster than it did horizontally; the second case assumed equal rates of progress of the melt downward and horizontally. The third case assumed that the molten core materials spread within the confines of the reactor cavity and then attacked the concrete at equal rates in the downward and horizontal directions. Because the carbon dioxide and water vapor produced by the decomposition of concrete would tend to sparge the melt of fission products, the decay heat utilized for this analysis corresponded to 60 percent of that at the time of LOCA. This implies complete loss of the volatile fission products and fractional removal of some of the less volatile species from the melt. Assuming all the decay heat goes into the concrete, the times required to penetrate the concrete and bring the entire melt to 4000 F will be 7, 9, and 36 hours for the three cases considered. These times are based on contact of the concrete by the molten core at 2-3 hr after the start of the accident. Making an allowance for heat losses from the upper surface of the melt, the best estimate for the time required to penetrate the containment foundation mat is 18 hours.

More rapid meltthrough of the concrete than calculated could occur if the spalled pieces of concrete were floated to the surface of the melt without undergoing dissolution and being elevated to the temperature of the melt.

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After the molten material has penetrated the concrete floor, the melt front will proceed into the underlying gravel and possibly into the earth. The ultimate extent to which the molten zone can grow depends upon the heat removal processes at the upper and lower surfaces and the chemical and physical processes within the melt. Estimates have been made of the ultimate extent of the growth of this region. Assuming that heat removal at the surface is limited to conduction, the maximum radius of molten sphere has been calculated to be 30 ft and 50 ft for growth into media of limestone and dry sand, respectively (Ref. 14). The analyses of Jansen and Stepnewski (Ref. 12) for basaltic concrete indicated a maximum radius of 38 ft for a molten hemisphere. Since the ground underneath containment is well below the level of the water table, conduction heat transfer at the surface of the melt should be augmented by steam generation and convection. It is therefore likely that the melt will not proceed more than 10-50 ft below the bottom of the containment building. Greater depths could only be achieved if the core material were able to melt through the underlying material without mixing and being diluted by the products of decomposition. Although it has been predicted that small pellets of solid UO_2 could travel some distance before being dissolved into the molten products of the medium being penetrated (Ref. 14), good mixing should occur between molten UO_2 and the products of decomposition of concrete or soil in the configurations expected in the meltdown accident.

During a major nuclear accident involving significant fuel melting, tremendous pressures can be generated which will test the integrity of the containment systems.

This is particularly true in the event of violent steam explosions or hydrogen detonations, such as those which occurred during the 1952 accident at the small NRX reactor located in Chalk River. The explosions inside the NRX reactor were sufficient to hurl a four-ton gasholder dome four feet through the air, where it lodged in the superstructure.

The following excerpts from Appendix VIII-B of the Rasmussen Report (pages VIII-77 to VIII-79) describe briefly what is known about the kind of steam explosions which can occur when molten metal comes in contact with water. A paragraph on non-condensible gases is also included for completeness.

Physical Explosions Resulting from Contact of Molten Materials and Water

B1. REVIEW OF LITERATURE

B1.1 INTRODUCTION

When molten metal comes into contact with a quenching fluid, a violent explosion can occur. This so-called "vapor or physical explosion" is a well-known, but little understood, phenomenon. A vapor explosion is usually characterized by an initiating event that leads to fragmentation and by the sudden conversion of thermal energy to mechanical energy due to very rapid heat transfer accompanied by a subsequent pressure wave. The violence, or magnitude, of such an explosion depends upon the quantity and rate of energy release. Numerous incidents have been reported in the literature (Refs. 1-12). Such explosions have occurred in the steel (Refs. 1-4), aluminum (Refs. 5,6), copper smelting (Ref. 5), paper (Refs. 7,8), and nuclear industries (Refs. 9,10).

The mechanism that triggers or initiates the explosion is not known; however, two basic facts have been established. First, the causative mechanism is not due to chemical reaction (Ref. 13), and second, fragmentation of the sample material is usually involved. Both experimental results and analyses (Ref. 14) have shown that the heat-transfer rates required to release the observed energy from a smooth metal sample are several orders of magnitude higher than the maximum rates that can be obtained in laboratory studies. Thus it has been concluded by numerous investigators that fragmentation of the metal to generate large surface areas is required to obtain the observed explosion violence.

B1.2 REPRESENTATIVE INCIDENTS INVOLVING STEAM EXPLOSIONS

Explosive incidents periodically occurring in the paper and metal industries have been reported. Several such incidents have been summarized (Ref. 11) and are cited to demonstrate the magnitude of the destructive forces present and the physical circumstances leading to the incidents.

B1.2.1 METAL INDUSTRY

Explosive accidents are infrequent in the metal industry but when they do occur, the destruction is severe.

B1.2.1.1 Mallory-Sharon Incident (Ref. 35).

In 1954, a titanium arc-melting furnace, which was water-cooled, exploded at a plant in Ohio. Nine injuries included four fatalities and property damage was $30,000. The explosion was believed to result from water entering the melting crucible.

B1.2.1.2 Reynolds Aluminum Incident (Ref. 35).

In 1958, an aluminum-water explosion occurred in Illinois involving some 46 injuries, 6 fatalities and approximately $1,000,000 in property damage. The explosion "rocked a 25 mile" area. Wet scrap metal was being loaded into a furnace when the explosion was triggered.

B1.2.1.3 Quebec Foundry Incident (Ref. 4).

The accident occurred in a foundry building approximately 18 million cubic-foot volume. One hundred pounds of molten steel fell into a shallow trough containing about 78 gallons of water. The resulting explosion injured mill personnel (one fatally) and caused $150,000 damage to the foundry building including cracking a 20-inch concrete floor, breaking 6000 panes of glass, and structural damage to the walls and ceilings. Damage was also incurred by another structure separated some 75 yards from the foundry building. This accident is one of the better documented incidents.

B1.2.1.4 Western Foundries Incident (Ref. 36).

In 1966, while 3000 pounds of molten steel was being poured from an electric furnace into a tile-lined ladle, a cable broke and the hot steel dropped into a water filled pit. The result was a violent explosion that injured three workers and tore a 600-square-foot hole in the roof of a building of some 12,000-square-foot floor area. The explosion was heard some 3 miles from the foundry.

B1.2.1.5 Armco Steel Incident (Refs. 37,38).

In 1967, an explosion occurred when molten steel fell on "damp" ground. A ladle containing some 30 tons of molten steel had been elevated some 40 feet when the ladle fell. Injuries sustained by some 30 workers included 6 fatalities. Evidently, sufficient moisture was present in the porous ground to trigger small-scale explosions that showered molten steel over a wide area. Although the injuries were attributed primarily to burns, an explosion accompanied the incident.

B1.2.1.6 East German Slag Incident (Ref. 2).

Appearing in 1959, an East German article discusses a number of slag-water explosions that have occurred in German open-hearth steel mills. Two accidents were discussed in which explosions resulted from spraying water on molten slag in open slag pits. One of the explosions resulted in a fatality and a number of other injuries. Severe structural damage was also noted. The second explosion was less severe. Both explosions were attributed to excess water on the slag passing down into the cracks to the hot molten material below. A third instance resulting in an explosion occurred when a slag pot was placed on a slag bed that had been previously sprayed with water. An explosion occurred, killing one man. The explosion was attributed to the heavy slag pot causing cracks in the surface of the hot slag bed and excess water on the surface getting into these cracks. Other explosions briefly described include rainwater leaking through an unsealed roof over a slag bed, resulting in an explosion, and two instances of explosions resulting when molten slag was poured into dump cars that had small amounts of water in the bottom.

B1.2.1.7 British Slag Incident (Ref. 3).

In 1964, an explosion occurred in a British steel mill when a ladle being used to tap a blast furnace was sprayed with lime water and returned to service. When it was next used, the ladle exploded when it was about three-fourths full of slag (12 to 14 tons). Damage to the structure and injuries to personnel were reported.

B1.2.2 PAPER INDUSTRY

The paper industry experiences explosions similar to those in the metal industry more frequently but they are less destructive. Explosions occur when paper smelt (mostly fused sodium carbonate with a few percent of sodium sulfide, sodium chloride, and minor ingredients) is quenched in large containers of "green liquor" (Refs. 7,8). Also, explosions frequently occur when boiler tubes in waste-heat boilers fueled by "black liquor" fail, and water is injected into hot molten smelt and black liquor. These explosions occur with considerable destruction to the furnace and plant facilities.

B1.2.3 NUCLEAR REACTOR INDUSTRY

Explosive vapor formation when hot, molten core materials have come in contact with water have also been observed in nuclear reactors.

B1.2.3.1 Canadian NRX Reactor (Ref. 35).

In 1952, at Chalk River, Ontario, during a low-power experiment, a nuclear excursion was experienced. Although the duration of the incident was less than 62 seconds, the damage was sufficient to result in contamination of the facility. The reaction between uranium and steam (or water) was the principal cause of damage.

B1.2.3.2 Borax I Reactor (Ref. 35).

In 1954, at the National Reactor Testing station in Idaho, the Borax I reactor was deliberately subjected to a potentially damaging power excursion in reactor safety studies: A power excursion lasting approximately 30 milliseconds produced a peak power of 19,000 megawatts with a total energy release of 135 megawatt-seconds. The power excursion melted most of the fuel elements. The reactor tank (1/2-inch steel) was ruptured by the pressure (probably in excess of 10,000 psi) resulting from the reaction between the molten metal and the water. The sound of the explosion at the control station (1/2 mile away) was comparable to that from 1-2 pounds of 40 percent dynamite.

B1.2.3.3 SPERT 1-D Reactor.

During the final test of the destructive test program with the SPERT 1-D core, damaging pressure generation was observed. Pressure transducers recorded the generation of a pressure pulse larger than 3000 psi which caused the destruction of the core. The pressure pulse occurred some 15 milliseconds after initiation of the power excursion. The power excursion rapidly overheated the fuel plates; the increased temperature melted the metal and the cladding of the fuel plates. After the transient, much of the fuel that had been molten was found dispersed in the coolant.

B1.2.3.4 SL-1 Reactor.

In January, 1961, a nuclear excursion occurred in the SL-1 reactor in Idaho. The total energy released in the excursion was approximately 130 Mw-sec (Ref. 51). Of this, 50 Mw-sec was produced in the outer fuel elements in the core. This portion of the energy was slowly transferred to the water coolant over 2 sec period, and no melting (uranium-aluminum alloy) of the outer fuel elements occurred. About 50-60 Mw-sec of the total energy release was promptly released by 12 heavily damaged inner fuel elements to the water coolant in less than 30 msec. This prompt energy release resulted in rapid steam formation in the core which accelerated the water above the core and produced a water hammer that hit the pressure vessel lid. The vessel, weighing about 30,000 lbs with its internals, sheared its connecting piping and was lifted approximately 9 feet into the air by the momentum transferred from the water hammer. Calculations of the mechanical deformation of the vessel indicate that about 12 percent of the prompt energy release or 4.7 percent of the total nuclear release was converted into mechanical energy (Ref. 52).

In each instance, under differing circumstances, a hot molten material fell, dropped, or spewed into a mass of cooler liquid and destructive pressure generation resulted. The complex mechanisms triggering this type of reaction are not completely understood.

Internal steam pressure can be greatly reduced by using a dousing system to condense the steam. It is much more difficult to reduce the pressure from non-condensible gases which are formed in great quantities during a meltdown. This passage is from page VIII-117.

Noncondensable Gases

There are two major sources of noncondensable-gas generation in a reactor accident in which core meltdown occurs. Metal-water reactions in the pressure vessel will produce hydrogen gas (H_2) shortly before and during the meltdown process. Later in the accident, carbon dioxide (CO_2) will be generated as the molten core material causes thermal decomposition of limestone aggregate in the concrete base mat of the containment structure. The production of these gases presents two potential threats to containment integrity. Both gases will cause a buildup in internal gas pressure in the system. Hydrogen generation can also lead to combustible mixtures with the oxygen already present in the containment atmosphere. Ignition of the hydrogen-oxygen mixtures can produce an exothermic chemical reaction which, depending upon conditions, might develop into a detonation. The introduction of additional thermal energy into the containment atmosphere will cause a rise in pressure, perhaps coupled with a shock-wave loading on the containment walls if the detonation occurs. The important question in all of these situations is whether, or under what conditions, would they likely result in containment failure by overpressurization. The hydrogen-generation problem is examined first for each type of water-reactor system; an analysis of the carbon dioxide generation problem follows.

The kind of meltdown accidents envisaged in WASH-1400 require a much more extensive evacuation plan than any that is currently envisaged in Canada, as indicated in this very brief excerpt from Appendix VI.of WASH-1400.

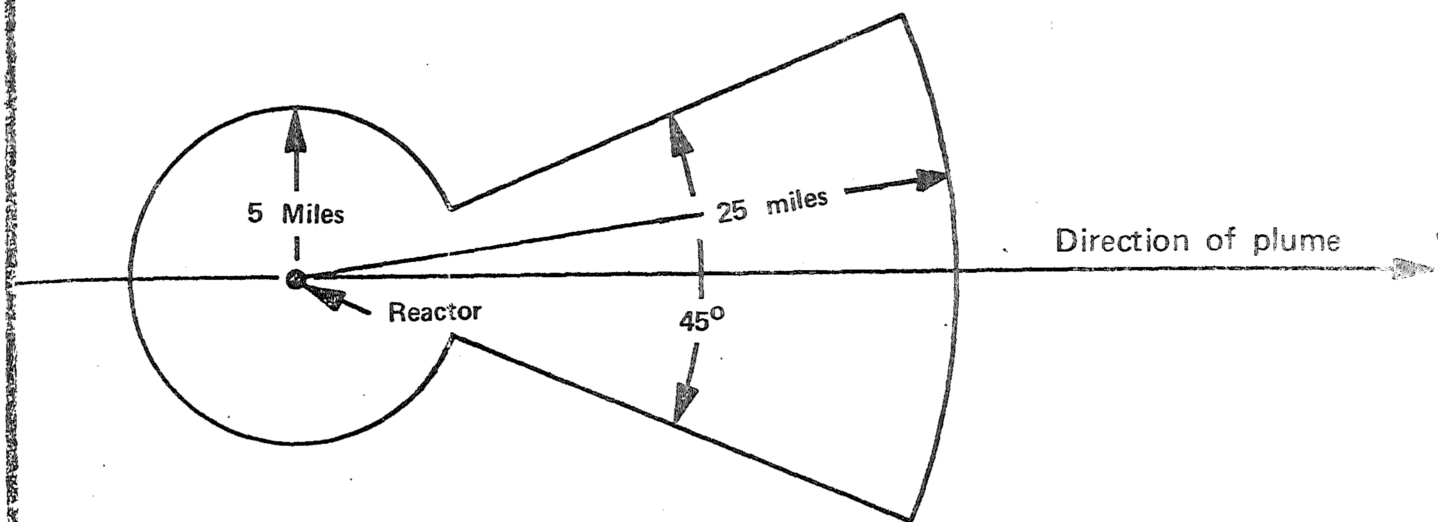

FIGURE VI 11-2 Evacuation area used for cost calculations.

In the evacuation model incorporated into the consequence calculations, the evacuation area is postulated to be shaped like a keyhole centered on the prevailing wind direction at the time of the release. The dimensions of the area are chosen to be 5 and 25 miles and 45° (see Fig. VI 11-2) for the following reasons. The evacuation would be carried out to mitigate the early exposure to individuals; the early exposure from the passing cloud would contribute little to the population dose. Since the resources of the local authorities -- all that would be available immediately after the accident -- are limited, it would be desirable to minimize the evacuation area and the number of evacuees. On the other hand, the goal would be to evacuate anyone who might receive a significant dose. The values 25 miles and 45° represent a compromise. In addition to this sector, it was judged prudent to evacuate all people within a 5-mile radius of the reactor. The evacuation costs are calculated on the basis of the number of people living in this evacuation area.

In order to calculate doses to individuals within the evacuation area, people are postulated to move radially away from the reactor at a specified effective evacuation speed until the cloud reaches them and then to move in a circumferential direction. For example, if an effective evacuation speed of 1 mph is assumed, people located between 2 to 3 miles from the reactor are assumed to be 7 to 8 miles away from the reactor 5 hours after the warning.

Fuel melting accidents release more than 200 different radioactive substances, of which the following 54 are considered the most dangerous. CANDU safety analyses routinely restrict themselves to only a small handful of these: iodine, the inert gases krypton and xenon, and, in some cases, cesium -- all released before melting actually begins.

TABLE VI 3-1 INITIAL ACTIVITY OF RADIONUCLIDES IN THE NUCLEAR REACTOR CORE AT THE TIME OF THE HYPOTHETICAL ACCIDENT

No.	Radionuclide	Radioactive Inventory Source (curies x 10^8)	Half-Life (days)
1	Cobalt-58	0.0078	71.0
2	Cobalt-60	0.0029	1,920
3	Krypton-85	0.0056	3,950
4	Krypton-85m	0.24	0.183
5	Krypton-87	0.47	0.0528
6	Krypton-88	0.68	0.117
7	Rubidium-86	0.00026	18.7
8	Strontium-89	0.94	52.1
9	Strontium-90	0.037	11,030
10	Strontium-91	1.1	0.403
11	Yttrium-90	0.039	2.67
12	Yttrium-91	1.2	59.0
13	Zirconium-95	1.5	65.2
14	Zirconium-97	1.5	0.71
15	Niobium-95	1.5	35.0
16	Molybdenum-99	1.6	2.8
17	Technetium-99m	1.4	0.25
18	Ruthenium-103	1.1	39.5
19	Ruthenium-105	0.72	0.185
20	Ruthenium-106	0.25	366
21	Rhodium-105	0.49	1.50
22	Tellurium-127	0.059	0.391
23	Tellurium-127m	0.011	109
24	Tellurium-129	0.31	0.048
25	Tellurium-129m	0.053	0.340
26	Tellurium-131m	0.13	1.25
27	Tellurium-132	1.2	3.25
28	Antimony-127	0.061	3.88
29	Antimony-129	0.33	0.179
30	Iodine-131	0.85	8.05
31	Iodine-132	1.2	0.0958
32	Iodine-133	1.7	0.875
33	Iodine-134	1.9	0.0366
34	Iodine-135	1.5	0.280
35	Xenon-133	1.7	5.28
36	Xenon-135	0.34	0.384
37	Cesium-134	0.075	750
38	Cesium-136	0.030	13.0
39	Cesium-137	0.047	11,000
40	Barium-140	1.6	12.8
41	Lanthanum-140	1.6	1.67
42	Cerium-141	1.5	32.3
43	Cerium-143	1.3	1.38
44	Cerium-144	0.85	284
45	Praseodymium-143	1.3	13.7
46	Neodymium-147	0.60	11.1
47	Neptunium-239	16.4	2.35
48	Plutonium-238	0.00057	32,500
49	Plutonium-239	0.00021	8.9×10^6
50	Plutonium-240	0.00021	2.4×10^6
51	Plutonium-241	0.034	5,350
52	Americium-241	0.000017	1.5×10^5
53	Curium-242	0.0050	163
54	Curium-244	0.00023	6,630

EXCERPT FROM WASH-1400, APPENDIX VII, PAGES 6&7.

1.2 MELTDOWN RELEASE COMPONENT

The conditions pertaining to this release period begin with rapid boiloff of the water coolant which uncovers the reactor core. Steam, flowing up through the heating core, initiates $Zr-H_2O$ reaction and this accelerates the rate of temperature rise. The cladding begins to melt within one minute and in a few more minutes fuel-melting temperatures are approached in the hotter regions. The process spreads throughout the core and within 30 minutes to 2 hours (Ref. 1) nearly the whole core is molten at temperatures ranging from roughly 2000 to 3000 C. During the later stages of this process molten core material can run through or melt through the grid plate and fall into the bottom of the pressure vessel. If a steam explosion does not occur when residual water is contacted in the lower portion of the pressure vessel, partial quenching and temporary solidification of portions of the molten mass can take place. However, the internal heat generation causes remelting and the inevitable downward migration continues until the pressure vessel fails, probably by meltthrough. Pressure vessel failure is expected to require about 1 hour after most of the core has melted (Ref. 1). Prior to this the high internal temperatures have caused melting of some of the pressure vessel steel and interior structural components. The molten iron is not miscible with the core material (oxide phase) although partial conversion to iron oxides could produce some dissolution in and dilution of the core material. Nevertheless some fission products (i.e., the noble metals) would tend to distribute to the metallic iron phase.

Initial fuel melting is expected to occur in only the center regions of the rods on almost a pellet by pellet scale. Thus the melting fuel will offer a relatively high surface area for release of fission products. As the melting front moves outward, the melting of the individual pellets may continue, but it is also conceivable that larger sections of fuel may collapse into the molten mass. If this fuel melts within the mass rather than at the edge, then fission product release could be inhibited by the time required for transport to a free surface. On the other hand, gaseous fission products, present as bubbles in the UO_2 could rise quickly to the surface of the molten mass and escape. It appears that most of the fission product release that does occur will take place early in the melting period at each core location. Then as the melted fuel mixes with the rest of the molten mass and the mass increases in size, fission product release rates will become much slower. The melting of structural steel in the pressure vessel during this later period is expected to produce a layer of molten iron above the molten core material which would offer a further barrier to fission product release. Other factors which can inhibit release during melt down in the pressure vessel include t' possibility of crust formation at the melt surface and partial quenching when melt runs or falls into water that may be left in the bottom of the vessel.

The atmosphere in the core region and pressure vessel during meltdown is expected to be a steam-hydrogen mixture with small concentrations of fission product and core material vapors and aerosols. This may be classified a nonoxidizing atmosphere for most fission products, and it, of course, results from partial consumption of steam by metal-water reaction, yielding H_2 in the core region. Although the metal-water reaction that does occur is steam supply limited, some steam flow passes through cooler portions of the core region without complete reaction. It is estimated that during the meltdown phas only about half the Zircaloy is reacted and other metal-water reaction produces only about 50 percent more H_2. Thus total metal-water reaction is only the equivalent of 75 percent $Zr-H_2O$ reaction.

Thermal analyses of core meltdown provide only generalized data on core temperature profiles, geometry changes, and melt behavior versus time. This, combined with the uncertainties which exist in fission product properties at very high temperatures, argue against construction of a highly mechanistic model to calculate fission product release during the meltdown phase. Therefore, in this work, fission product release is treated as being simply proportional to the fraction of core melted.