DEVELOPMENT OF CONTAINMENT PERFORMANCE CRITERIA
FOR LIGHT WATER POWER REACTORS

W. Kerr

Introduction

The first nuclear power plant containment in the United State, constructed at Schnectady, N.Y. in the early 1950s, set a precedent that has been followed by all light water commercial power reactors (LWPRs) constructed in the U S. All have included some type of containment structure. It appears that the West Milton site was originally selected for a fast breeder reactor, and was approved by the Atomic Energy Commission's Reactor Hazards Evaluation Committee for that purpose provided the entire plant was enclosed in a sphere capable of containing any reactor accident produced by a reactivity-induced energy release. (1)

However plans for the site changed and the design of the first containment, a 225 foot diameter sphere fabricated of one inch thick steel, was based on plans for a sodium cooled prototype of a propulsion reactor, the SIR Mark I. The design of the containment was dictated by the requirement that it withstand the pressure which would result from the burning of the sodium which made up the primary coolant of the sodium cooled prototype.

With the decision to build the Shippingport reactor power plant, near Pittsburgh, and to build several water-cooled propulsion prototypes on the West Milton site, came a concurrent decision to include a containment as part of each plant. The double-ended instantaneous pipe break (DEPB) in the primary system was chosen arbitrrially on an entirely non-mechanistic basis only to establish the design pressure and temperature to be used in the design of these new containments. It was also used to check the suitability of the containment, originally designed for the SIR, after it had been decided that, instead of the SIR, a water cooled destroyer prototype, the D1G would be installed. This approach to determining containment design criteria, i.e. use of the DEPB to establish the design pressure and the design temperature, was also accepted for all of the early, relatively low power LWPRs that were constructed and operated in the US.

In the early 1960s as LWPRs proposed for licensing became larger, and as the core power density of the designs increased, those responsible for reactor licensing became concerned about the capability of the containments to contain fission products in case of a postulated accident in which all or a significant fraction of the core became molten because of the loss of capability to remove decay heat from the core.

In the course of its consideration of the construction permit for the Zion reactor power plant, with reactors significantly larger than any previously licensed, the Advisory Committee on Reactor Safeguards (ACRS) raised questions concerning the possibility of cooling a severely damaged core. The accident hypothesized was a loss of coolant accident, followed by shutdown of the fission process,followed by failure of the systems designed for decay heat removal. Under these circumstances severe core distortion or core melting may occur. The question was raised as to whether, if a source of water becomes available after severe damage has occurred, cooling sufficient to arrest further significant core damage, and to avoid penetration of the containment by a molten core, can be assured.

In a letter of October 1966 to the then chairman of the Atomic Energy Commission, Dr. Glen Seaborg, the ACRS made the following recommendations:

> "A vigorous research program should be initiated promptly on the potential modes of interaction between sizeable masses of molten mixtures of fuel, clad, and other materials with water and steam, particularly with respect to steam explosions, hydrogen generation, and possible explosive atmospheres. Work should be directed toward understanding the mechanisms of heat transfer connected with such molten masses of material, the kinds of layers formed at cooled surfaces, the nature and consequences of any boiling of the fuel, and the manner and forms in which fission products escape from bulk molten fuel mixtures. Further studies should be initiated by industry to develop nuclear reactor design concepts with additional inherent safety features or new safeguards to deal with low-probability failure of the emergency core cooling system." (2)

The Commission subsequently appointed a task force, chaired by W. K. Ergen of Oak Ridge National Laboratory, with members from industry and other national laboratories, to study these issues and to make recommendations. A major question was whether there should be a requirement for a licensee to provide some system to contain, for at least some specified interval, if not indefinitely, a molten core inside an intact containment. In 1967 the task force reported its conclusions that the state of knowledge at that time did not permit the design of systems that would guarantee, with high confidence, that melt through of containment would not occur, given a severely damaged core, even if a source of coolant was available. (3)

In light of this conclusion, the task force recommended that emergency core cooling systems (ECCS) of high reliability be required for the new plants being considered for licensing. In effect this was a recommendation that the ECCS reliability be sufficient that the residual risk of penetration of containment by a molten core could be ignored.

The AEC accepted this recommendation, and the staff began the task of preparing specifications for an appropriate ECCS. Performance criteria for the required systems were eventually adopted in a rule-making which produced the requirements now found in 10CFR50.46 and the accompanying Appendix K.(4)

The task force also recommended a program of research to permit the collection of information needed to design systems to deal with a molten core. However that part of the recommendation was not implemented at the time.

Containment performance criteria, specified in 10CFR100, Reactor Site Criteria (5), do not take specific account of the challenge to containment that would be produced by a molten core. The only concession to possible severe core damage is the specification that the leak rate of the containment be demonstrated to be not more than a value determined by a specified calculated off-site absorbed dose to an individual. The absorbed dose to be associated with a particular containment design is to be calculated with the assumption that at the beginning of the accident a fission product inventory consisting of all the noble gases and a major fraction of the volatile fission products is immediately available for leakage from the containment.

However the containment environment--which can have a significant influence on the leakage rate--to be used for the leakage calculation, is that calculated to be produced by a so-called design-basis loss of coolant accident (LOCA). (Of course, if the ECCS has been properly designed, and operates correctly, no significant core damage will be produced by a LOCA). Thus in calculating the postulated leakage, no consideration is given to the environment that would be associated with a molten core that has escaped the pressure vessel, and may challenge containment integrity. It is hard not to conclude that for the first two decades of power reactor design and operation in the US, most of the emphasis was on prevention of severe reactor accidents, with little attention being given to mitigation.

The publication of the Reactor Safety Study (6) in 1975 provided evidence that the probability of severe accidents could be sufficiently large that they should not be ignored in reactor system design. However it was not until the TMI 2 accident in 1979 that serious attention was given by those in the nuclear industry to the predictions of that report. And despite the fact that large quantities of manpower and money have been spent on severe accident research since that time, design and performance criteria that are currently being used to judge the suitability of containment systems proposed for newly designed LWPR plants are based largely on those developed before severe accidents were taken seriously.

Reactor Safety Study Predictions of Containment Performance

The Reactor Safety Study referred to above marked the first serious effort to predict containment system performance under severe accident conditions. Two types were treated, a large, dry, subatmospheric containment, and a pressure suppression system, the GE, Mark I containment. A number of failure modes were postulated for each containment, and the fission product releases were categorized partly on the basis of the method of containment system failure. Failure modes postulated for the large dry included early failure caused by an in-vessel steam explosion, failure caused by over-pressurization due to hydrogen burning and steam formation, failure to close off containment penetrations (failure to isolate), and base mat melt through. Failures hypothesized for the pressure suppression containment included an in-vessel steam explosion, as well as over pressurization caused by failure to remove decay heat. Failure to isolate was also identified as a possible failure mode. The report also includes a statement that "...the vapor suppression system that has some capability for removal of radioactivity is largely ineffective in a number of the core melt cases". In those cases of core melt referred to, the principal mechanism for removal of radioactive material is assumed to be natural deposition on the surfaces inside the containment and the reactor building. At the time of the study that produced this report there was a paucity of information about many of the phenomena contributing to the postulated containment failures, and in many cases the description of the sequence of events producing containment failure was little more than an educated guess.

In addition to containment failures caused by the consequences of a severe accident, the study also hypothesized some risk-significant and previously unidentified event sequences that could produce early by-pass of the containments, i.e. fission products could escape, in large quantities, very soon after core melt, without the failure of the containment structure. Notable among these is the failure of valves designed to isolate the high pressure primary coolant system from low pressure auxiliary systems. Some hypothesized failures were found to provide a direct path from the primary system, through low pressure piping which penetrates containment, to an auxiliary building outside containment. In this case, containment integrity is destroyed, not by the after-effects of a severe core-melt accident, but by a previously unidentified mechanism which, in some circumstances, can not only breach the containment boundary, but can also result in unavailability of the ECCS, leading to severe core damage with an unavailable containment system. (This sequence is frequently referred to as an Interfacing Systems LOCA.) Recognition of this weakness in some of the containment systems in operating reactors, led to changes in both equipment and operation. As a result, the risk from this type of sequence is calculated to have been decreased significantly.

Other mechanisms which degrade or make the containment function unavailable include valves left open inadvertently, or failure to properly close a penetration originally installed for some purpose, but later abandoned. Information available from US experience shows a significant number of cases in which containments have failed a leak rate test before corrective measures were taken. A recent report of "international experience" concludes that there is about a ten per cent probability of having a leak equivalent to a 0.4 sq cm opening in an operating containment. (7)

Severe Accident Research

Shortly after the TMI 2 accident both industry and the NRC began research programs to develop information about the phenomena expected to be encountered during the course of a severe accident. (8) A significant amount of this research dealt with the hypothesized interaction with the containment structure and containment systems of the molten mass produced during core melt and vessel penetration. Of overriding importance in predicting the course of the interactions that may occur is the state of the material that is released from the pressure vessel into the containment. Among parameters important to determining further behavior are such things as the amount of cladding material that has been oxidized, the amount of alloying of uranium with zirconium that may have occurred, the amount of non-fuel material that has become molten, the temperature of the mixture as it enters containment, and the rate at which the molten mass enters containment. The amount of molten material in the containment at any time is also of considerable significance.

The research programs involved a combination of analytical modeling and small scale experiment, with the experiments aimed primarily at validating computer codes that would be used to calculate the behavior of risk-important power plant systems during severe accidents. In the Reactor Safety Study it was assumed that the molten material that resulted from melting of the core was retained in the vessel until some predetermined state was reached, and then all of the material entered the containment together. No serious effort was made to model in any detail the progression of core melt. Partly because of this precedent, partly because it was considered to be a conservative approach, and partly because of lack of data, much of the subsequent modeling of severe accident progression has retained this approach.

There was no penetration of the vessel during the course of the TMI 2 accident, the first US power reactor accident in which significant melting of the core was observed. This was accounted for in early studies of the accident by assuming that little or no melting of the core material had occurred. Even though later information indicated that perhaps as much as 50 to 60 percent of the core became molten, and that as much as 20 per cent of the original core may have moved, as molten material, to the bottom of the vessel, (9) these data have been largely ignored in the development of the analytical models for treating severe accident progression.

It can be argued that it is impossible to model core melt progression in detail for the variety of possible accident scenarios that might occur. However, if this is hypothesized, it is important to decide what information about the state of the molten corium entering the containment is needed in order to predict containment performance in sufficient detail and with adequate confidence, and to concentrate on obtaining this information.

Development of Containment Performance Criteria

Once it is recognized that the criteria used in the design of contemporary containments are obsolete, it becomes clear that the behavior of containment systems in a severe accident environment requires further consideration if the containment is to be considered as a significant contributor to defense in depth. Many current PRAs of operating plants, in which containments system performance is modelled for sequences in which a significant fraction of the core becomes molten, and enters the containment, produce risk predictions which are interpreted to meet the NRC's Safety Goals. Even if one accepts these results and their interpretation as valid, (It is generally agreed that major uncertainties exist) there is still a question as to whether containment performance criteria should be specified. A major contributor to uncertainty is the modelling of the performance of containment systems when challenged by the presence, in the containment, of significant quantities of molten core material. A first question might be stated as, "If the risk predicted for a plant without containment is less than that established as acceptable under the criteria that grow out of the Safety Goals, should containment be required?".

The early formulation of the safety philosophy used in power plant design identified containment as a final "echelon"in an overall approach identified as "defense in depth". It was justified as a compensation for the uncertainty in the predicted reliability of those systems and procedures that were depended upon to make the likelihood and the consequences of a severe accident acceptably low. Typically three echelons were described. The first two were measures for prevention of core damage. Only the third, the containment, included mitigation of the effects of core damage. There were inconsistencies in containment system design, in that little account was taken of the environment produced inside the containment by the presence of a large mass of molten material. However one of the purposes of the containment was certainly to mitigate the consequences of severe core damage.

There are those who conclude that with the increased understanding of reactors systems that has been developed, and with the analytical methods that have been formulated and used, conventional containment can be eliminated in favor of a reactor system designs which results in very low predicted core melt probability. Others argue that the uncertainty in the values that are calculated is such that containment should be required. If containment is to be required, some containment criteria appear to be needed if for no other reason than to specify the extent to which a designer can depend on prevention as an alternative to mitigation in establishing an acceptable design while taking account of uncertainties.

Among the forms of criteria that have been suggested are what we shall call design criteria and performance criteria. The two are not necessarily mutually exclusive, but might be described as follows:

> Design criteria -- These might specify such characteristics as free volume of the containment, design pressure and temperature, material of construction (e.g. the use of basalt rather than limestone aggregate for concrete in that part of containment likely to be in contact with the molten corium), approval of or prohibition of pressure suppression, and methods for dealing with hydrogen generation.
>
> Performance criteria -- These might, e.g, specify the conditional probability that, given some accident or set of accident sequences, the probability that a release of more than some fraction of the core fission product inventory present at the onset of the accident be less than some selected value.

The ACRS, in a report to the NRC dated May 13, 1987, which discusses an implementation plan for the Commission's Safety Goal Policy, makes the following recommendation:

> We recommend that as a minimum the containment performance objective should be such that there is less than one chance in ten for a large release for the entire family of core melt scenarios." (10)

The report does not define a large release, but does say that the ACRS believes that a large release, if its definition is to be consistent with the Safety Goal Policy, should be one that produces a significantly larger dose than 25 rem whole body to an individual at the plant boundary. Although somewhat ambiguous, and probably overly restrictive, the ACRS recommendation is an example of one possible approach to defining containment performance criteria.

At one time the NRC Staff undertook the task of developing containment performance criteria.(11) The task was never completed. For a time emphasis was being given to an investigation of the expected performance of the Mark I pressure-suppression type containment in a severe accident situation. The priority given to the Mark I containment was due in part to an interpretations of the results in the draft NUREG-1150 which led to a conclusion that, given a molten core in the containment, there was about a 90 per cent probability of containment failure. (12)

<u>Treatment of Containment Performance in Later PRAs</u>

Not long after the TMI 2 accident, there were estimates by members of the NRC staff that the Zion Plant posed an unusually high risk to the public, primarily because of its location near a densely populated area.(13) In response to this concern, the operators of the plant performed a full-scope PRA. (14) In the course of the analysis, an accident sequence was postulated which involved a previously unanalyzed mechanism for release of molten core material into the containment. The analysis predicted that under some circumstances this sequence could lead to early containment failure.

The sequence analyzed, Direct Containment Heating (DCH), involved the hypothesis that core melt occurred with the vessel at high pressure. It was further assumed that the molten corium would rapidly melt through the bottom of the vessel at one or more of the penetrations of the instrument tubes through which fission chambers are inserted into the vessel. Vessel pressure would result in the expulsion of the molten material as a jet, which would break up into small droplets. The large surface area of these droplets would produce rapid transfer of heat to the containment atmosphere. In addition, unoxidized metal in the molten material would be oxidized, producing additional heat input to the containment.

Calculations, using this set of assumptions, but based on a paucity of data, predict a pressure build up large enough to fail the containment within a few minutes of the start of melt expulsion from the vessel. Studies of other large dry containments have led to concerns that this may be a generic problem involving a variety of plants. For example, on the basis of the results reported in the draft Reactor Risk Reference Document, NUREG 1150,(12) the NRC staff identified DCH as one of the principal uncertainties in the risk prediction for power plants that use large dry containments. It is also postulated that DCH might occur in the course of some sequences identified in the analyses of BWRs. (13?)

There is no general agreement on the likelihood of this sequence and its consequences. Key determinants in the consequences of the sequence are the temperature of the molten core material, and the fraction of the core which is molten and at the bottom of the vessel when the sequence reaches the stage at which melt through is postulated. Because of the serious consequences accompanying an early failure of containment, and because it is hypothesized that this sequence will produce early failure, the NRC staff concluded that the issue deserved further investigation. A major research program (both experimental and analytical) was undertaken to study some of the phenomena that have been judged to be important to a more accurate prediction of the effects of such a sequence (so-called direct containment heating or DCH). It is worth noting that those European countries with operating reactors do not seem to have the same concern about DCH as does the NRC staff. For example a recent risk study performed in the FRG concludes that the probability of high pressure core melt is of the order of one in ten million per year. (Ref)

In a report dated May 13, 1987, addressed to the Chairman of the Commission, the ACRS made the following comments on a part of the NRC research program that is related to containment performance:

"We observe that estimates of accident progression at key points in the core melt sequence depend on the prediction, using inadequately based computer codes, of such parameters as melt temperature and time required for vessel melt through. There appear to be significant uncertainties in the predictions of a number of these key parameters that tend to be masked by the codes. Since vessel penetration, core-concrete interactions, and the concurrent release of fission products, for example, are all very sensitive to melt temperature, we urge that efforts, including both experiments and independent calculations, be made to provide some independent and more transparent assessment of the behavior of key parameters. Comparison with another code embodying the same underlying assumptions is not sufficient." (10)

The Mark I containment

The concerns about some of the predictions of the performance of Mark I containments in postulated accidents that involve melting of a major fraction of the reactor core are illustrative of the problems of making the transition from the early approach to containment design, with its emphasis on design basis accidents, to today's regulatory climate which recognizes that severe accidents have a predicted probability sufficiently high that they should not be ignored in setting performance criteria for containment systems. It must be added that although there is general agreement that account must be take of severe accidents, there is still no consensus in the US as to how it should be done.

The GE Mark I pressure suppression containment was designed with the specific characteristics of the BWR in mind. The large inventory of water that is part of the containment system will not only produce condensation of the steam released by the blow down expected to follow a large LOCA, but also provides for capture, inside containment, of radioactive materials in the water and steam released from the primary system during the course of transients which involve loss of heat sink at full load, and the consequent opening of safety/relief valves to prevent a large pressure surge that would otherwise occur. These are transients which occur during the normal operation of the BWR, and thus the containment systems play an important role in normal operation as well as providing a barrier against fission product release in the case of a severe accident. The suppression pool can also serve as a temporary source of emergency cooling, although eventually the heat input to the pool must be removed if over-pressurization of the containment is not to occur. The use of pressure suppression also permits the containment performance required to cope with design basis accidents to be achieved with a significantly smaller containment volume than would be required without pressure suppression. For example the volume of Mark I containments in the US is typically in the range of 280,000 cubic feet--7900 cubic meters--(Peach Bottom) with a design pressure of about 62 psi, as compared to the large dry containment at, say, Zion which has a

free volume of 2.6 million cubic feet--73,600 cubic meters--and a design pressure of
47 psi. (15?)

This design has been demonstrated to meet the requirements of containment system performance that existed when the plants were licensed. And indeed, with an inerted containment atmosphere, it meets existing requirements. However it is clear that for postulated severe accidents of the type that produce significant quantities of non-condensable gases, the pressure suppression capability of the Mark I may not cope adequately with the resulting pressure increase inside containment. On the other hand, the large volume available in the large dry containment is just as effective against non-condensables as it is against steam. However, if an accident sequence occurs which produces severe core damage, the suppression pool might provide a filter for fission products that would otherwise be released outside containment in the event of either deliberate containment venting or of containment rupture.

We are faced with the fact that since the publication of The Reactor Safety Study in 1975 it has been known that Mark I containments may be especially vulnerable, in severe accident situations, to over pressure caused by ignition of hydrogen, to over pressure produced by non-condensable gasses, or to melt-through of molten core material. Currently the hydrogen problem is treated by inerting the containment. However during the study that produced the draft version of NUREG 1150 it was concluded that there is a high (50-90%) probability, given that a significant fraction of a molten core enters the containment dry well in a time interval of a few minutes, that the Mark I containment will be breached by melting of the dry well metal liner. This melt-through would be expected to occur in a matter of a few hours after breach of the pressure vessel.(15?) This result is in contrast to that of the Industry Degraded Core study which predicts only about a 10% probability of failure, given severe core damage. (16)

A number of proposals have been made in an effort to decrease the vulnerability of the Mark I to over pressurization. The one that has probably received the most consideration is venting the containment before the pressure reaches a value at which containment failure is likely to occur. In the ATWS accident sequence usually postulated for the BWR it is expected that with natural circulation established, the reactor power will reach a level of about 30% of full power.(15?) It is estimated that a closed containment will reach a pressure likely to result in rupture after about 30 minutes of operation at this power level. It is postulated that a vent in the wet well would permit pressure relief, and thereby at least delay more serious damage. Under the ATWS sequence, damage to the core at the time of venting is thought not to be serious, and the filtering available from the suppression pool should permit pressure relief directly to the atmosphere. However there is some concern that if venting is used for this purpose, the operation of ECCS pumps, which might be needed at some later point in the recovery process, may be defeated. In addition the vent sizes currently being proposed for operating Mark I containments are typically not large enough to carry the steam generated by operation at the postulated power level.

There is also no clear cut basis for determining at what point venting should occur, since the decision to vent requires that one make a decision about what is likely to occur during the course of a process that is likely to have been unanticipated.

As a result of the Chernobyl accident, and the subsequent discussion of the performance of U S containments in the event of severe accidents, the possible vulnerability of the Mark I has received special attention. The NRC staff is currently giving priority to the issue, and indications are that a staff position on Mark I containment performance will be released in the near future. It appears that the process of deciding what, if any, changes are to be required in existing systems or procedures will give considerable weight to the results of PRAs that predict plant risk. Thus, for example, if the calculated likelihood of core melt is sufficiently low, the performance of the containment systems, given a severe accident, will be of less concern. The draft version of NUREG-1150 predicts a core melt probability for the Peach Bottom reactor of slightly less than one in 100,000 per year.

A consideration that makes the setting of containment system performance criteria especially difficult is that if the criteria are such that they are not met by some containment systems of currently operating reactors, it is likely that pressure would be generated either to make changes to those systems or to somehow limit the operation of reactors that have these systems. Thus the setting of criteria may be constrained to take account not only of what are the best criteria, but also of what criteria will not impose an unacceptable penalty on operating reactors.

REFERENCES

1. "A History of Containment Design Requirements", W. E. Cooper, Private Communication.

2. Letter Report, D. Okrent to G. Seaborg, October 12, 1966.

3. "Emergency Core Cooling, Report of the Advisory Task Force on Power Reactor Emergency Cooling"; TID-24226; W. K. Ergen; USAEC, 1967

4. Code of Federal Regulations, Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities".

5. Code of Federal Regulations, Title 10, Part 100, "Reactor Site Criteria".

6. "Reactor Safety Study, An Assessment of Risks in U. S. Commercial Nuclear Power Plants", WASH-1400 (NUREG 75/014), October, 1975.

7. "Severe Accident Risk Reduction by Containment Leakage Testing During Operation"; B. De Boeck; Fourth Workshop on Containment Integrity; Arlington, Va.; June, 1988.

8. "Nuclear Power Plant Severe Accident Research Plan"; G. P. Marino, NUREG-0900, Rev. 1, 1986.

9. Ref to report on amt of core melted

10. ACRS Letter Report, W. Kerr to L. W. Zech, Jr., May 13, 1987.

11. "Plan for Developing a Containment Performance Design Objective for Safety Goals", Office of Nuclear Reactor Regulation, October 2, 1985.

12. "Reactor Risk Reference Document"; NUREG-1150 (Draft); Office of Nuclear Regulatory Research; February, 1987.

13. Denton pronouncement

14. "Zion Probabilistic Risk Analysis"; Performed for Commonwealth Edison under supervision of Pickard, Lowe and Garrick; 1982.

15. "Containment Performance Working Group Report"; NUREG-1037;

Office of Nuclear Reactor Regulation; May, 1985.

16. IDCOR

17. "Survey of the State of the Art in Mitigation Systems"; NUREG/CR-3908; J. N. Castle, et al; R & D Associates; January, 1988.

wk 26-IV-93