September 29, 2000

MEMORANDUM TO: Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

FROM: Ashok C. Thadani, Director **/RA/** original signed by M.V. Federline
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER RIL-0005, "COMPLETION OF RESEARCH TO ADDRESS DIRECT CONTAINMENT HEATING ISSUE FOR ALL PRESSURIZED-WATER REACTORS"

This research information letter provides the results and findings of extensive research to resolve the potentially risk significant issue related to early containment failure in pressurized-water reactors (PWRs) by direct containment heating (DCH). DCH was a longstanding concern in the NRC's consideration of severe accidents that has been resolved by the syntheses of experimental and analytical research using a probabilistic framework. This melding of deterministic analysis methods under a probabilistic framework has been demonstrated now, by repeated applications, to be a proven success path for resolving complex technical issues. By this letter we summarize in a single document the research results and provide references to the testing and analysis performed over the past decade. This research information letter can be considered by licensees as part of their efforts to update their probabilistic risk assessments (PRAs) or in conjunction with their efforts to analyze risk informed initiatives.

<u>Executive Summary</u>

DCH refers to the phenomenology whereby, under certain reactor accident scenarios, molten core debris is ejected under high pressure from the reactor vessel into the containment atmosphere. The subsequent rapid heating of the containment atmosphere, in conjunction with possible hydrogen combustion, can lead to early containment failure. DCH was first identified in the Zion Probabilistic Safety Study (Ref. 1) of 1981. Subsequent evaluations by the NRC in connection with NUREG-1150 (along with industry assessments conducted in the individual plant examinations [IPEs]) identified DCH as one of the important contributors to early containment failure for PWRs.

The large uncertainty attached to DCH as a dominant early containment failure mechanism prompted the NRC to acknowledge same in Generic Letter 88-20, which provided guidance on the performance of an IPE. Citing the need for additional research, the generic letter advised utilities that no major modifications would be needed until research was completed and the NRC reached generic conclusions on containment performance. The NRC further concluded that IPEs need not extensively address DCH due to the then general lack of understanding. NRC activities, focused under the Severe Accident Research Plan (NUREG-1365), then commenced, addressing a broad range of tasks including development of a severe accident scaling methodology, integral and separate effects testing, and phenomenological modeling.

The first investigations addressed the Zion and Surry designs (both were NUREG-1150 plants). These studies concluded that DCH posed no tangible risk for these designs. The methodology developed and applied to these designs was then applied to each operating plant, taking into account plant specific features. The plants were grouped into three classes; 1) Westinghouse plants with large dry and subatmospheric containments, 2) Babcock & Wilcox (B&W) and Combustion Engineering (CE) plants, and 3) Westinghouse (W) plants with ice condenser containments. For all PWR plants, except the ice condenser plants, the threat from DCH was shown to be acceptably small, and in some cases we conclude that utilities were unnecessarily conservative in their IPE evaluation. Therefore, no follow up regulatory action is needed for those designs and their inherent safety has been confirmed, even for the more severe of challenges.

Ice condenser plants, however, present a more complex picture due to their greater vulnerability to a variety of phenomena. DCH, per se, only represents a moderate threat, resulting in a conditional containment failure probability of approximately 0.1 for one plant, McGuire. All ice condenser plants, though, are vulnerable to failure from hydrogen combustion during station blackout scenarios since their hydrogen control systems (i.e., igniters) would not be operable during those blackout sequences. This vulnerability to hydrogen for an important class of sequences is being addressed as part of our efforts to risk inform 10 CFR 50.44. Under the framework developed for Option 3 (SECY-00-0198), if a class of sequences represents a significant fraction (e.g., more than 10 percent) of the quantitative guidelines for large early release frequency ($<10^{-5}$/yr) then additional measures would be considered for implementation. Thus, it would be necessary to show that station blackout type sequences have a frequency $<10^{-6}$/yr, or develop performance-based measures that can be used, in order to preclude consideration of alternate power to igniters. Additional efforts are being considered to determine the backfit potential for this risk significant failure mode.

Background and Discussion

The consequences of a core melt accident and failure of the reactor vessel at higher pressure received increased attention following the Three Mile Island-2 accident. The first explicit consideration of high-pressure core debris dispersal was addressed in the Zion Probabilistic Safety Study. Subsequent evaluations of core damage events conducted as part of the NUREG-1150 study and individual plant PRA also identified the high-pressure melt ejection (HPME) of molten corium from a failed reactor vessel as a potentially important failure mode.

Early experimental studies supported by the Electric Power Research Institute (EPRI) and the NRC, as well as those performed in the United Kingdom, while important to observing fundamental characteristics associated with HPME, were difficult to apply directly to a nuclear power plant (Ref. 2).

As part of the NRC's Revised Severe Accident Research Plan (Ref. 3), the staff outlined key issues and tasks that needed to be addressed and performed to resolve the issue of early containment failure from DCH. The first key issue dealt with scaling of severe accident phenomena so that fundamental issues such as DCH could be defined and combined with other assessments to provide an integral picture of the phenomena in question. The activity to develop a Severe Accident Scaling Methodology (Ref. 4) represented a major contribution to the resolution of DCH by providing a rational approach to scaling, guiding the testing at different scales and the development of analytical formulations enabling integral tests to be correlated and extrapolated to nuclear power plants.

The NRC experimental program which ensued from the scaling rationale represented a major effort involving the close coordination of three research organizations (Sandia National Laboratories, Argonne National Laboratory and Purdue University) performing integral and separate effects tests. Counterpart experiments conducted at Sandia and Argonne at different scales (1/40, 1/10 and 1/6 linear scale) provided an extensive database confirming repeatability of fundamental behavior. Separate effects testing at Purdue similarly confirmed fundamental behavior while providing insights on detailed melt transport phenomena. The vital role played by the experimental program stemmed in large part on the rigor with which different facilities were similarly designed and scaled and most importantly faithfully simulated the important design characteristics of nuclear plants which influence DCH. Table 1 lists the research reports describing the NRC's experimental program and its results.

Table 1 Research Reports Describing NRC's Experimental Program and It's Results

1	NUREG/CR-5809, "An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution," N. Zuber, et al., 1991.
2	SAND 91-1173, "Experiments to Investigate the Effects of Water in the Cavity on Direct Containment Heating in Surtsey Test Facility – the WC-1 and WC-2 Tests," M.D. Allen et al., Sandia National Laboratories, March 1992.
3	NUREG/CR-6044, "Experiments to Investigate Direct Containment Heating Phenomena with Scaled Models of the Zion Nuclear Power Plant in the SURTSEY Test Facility," M.D. Allen et al., (SNL), SAND 93-1049, Mary 1994.
4	NUREG/CR-6152, "Experiments to Investigate Direct Containment Heating Phenomena with Scaled Models of the Surry Nuclear Power Plant," T.K. Blanchat et at., (SNL), SAND 93-2519, June 1994.
5	NUREG/CR-6168, "Direct Containment Heating Integral Effects Tests at 1/40 Scale in Zion Nuclear Power Plant Geometry," J.L. Binder et al., (ANL), ANL-94/18, September 1994.
6	NUREG/CR-6469, "Experiments to Investigate Direct Containment Heating Phenomena with Scaled Models of the Calvert Cliffs Nuclear Power Plant," T.K. Blanchat et al., (SNL), SAND 96-2289, February 1997.
7	NUREG/CR-6267, "Air-Water Simulation of Phenomena of Corium Dispersion in Direct Containment Heating," M. Ishii et al., (Purdue University), PU NE-93/1, October 1994.
8	NUREG/CR-6510, Vols. 1 & 2, "Corium Dispersion in Direct Containment Heating," M. Ishii et al., (Purdue University) PU NE-96/2&3, September 1999.
9	NUREG/CR-5746, "Direct Containment Heating Experiments at Low Reactor Coolant System Pressure in the Surtsey Test Facility," T.K. Blanchat et al., (SNL), SAND 99-1634, July 1999.

As the results of DCH experiments became available, there were parallel efforts to develop models for capturing the dominant physical phenomena and for extrapolation to reactor scale. Also, as part of the issue resolution effort, a generic methodology was applied to integrate aspects of the problem under a probabilistic framework, for example, ROAAM (Risk Oriented Accident Analysis Methodology).

Under the probabilistic framework, initial conditions of the core melt (melt mass, fraction of zirconium oxidized) were quantified as independent probability density functions. These functions were then combined with a causal relationship which represents model uncertainty in predicting containment pressurization. This distribution function of containment pressurization was then combined with a containment fragility curve to obtain a probability distribution function of containment failure.

In addition to the above process, the DCH issue resolution effort also developed a method for factoring in the likelihood of unintentional depressurization of the reactor coolant system by creep rupture of the hot leg. Such a failure, which was demonstrated to be credible by testing and analysis conducted under a joint EPRI/NRC(Ref. 5) research program, would lead to a lower probability of HPME and thus DCH and would lower the weighted frequency of containment failure (by reducing the contribution from high-pressure sequences). The analysis of DCH loads and the evaluation of containment failure probabilities was performed on a plant specific basis for all PWRs, and thoroughly peer reviewed, as described in the reports listed in Table 2.

Table 2 Research Reports Describing the Evaluation of DCH Containment Failure Probabilities

1	NUREG/CR-6075, “The Probability of Containment Failure by Direct Containment Heating in Zion,” M. Pilch, H. Yan, T.G. Theofanous, SAND 93-1535, December 1994.
2	NUREG/CR-6075 Supplement 1, “The Probability of Containment Failure by Direct Containment Heating in Zion,” M. Pilch, et al., SAND-93-1535, December 1994.
3	NUREG/CR-6109, “The Probability of Containment Failure by Direct Containment Heating in Surry,” M. Pilch, et al., SAND 93-2078, May 1995.
4	NUREG/CR-6338, “Resolution of the Direct Containment Heating Issue for All Westinghouse Plants With Large Dry Containments or Subatmospheric Containments,” M. Pilch, M.D. Allen, E.W. Klamerus, SAND 95-2381, February 1996.
5	NUREG/CR-6475, “Resolution of the Direct Containment Heating Issue for Combustion Engineering Plants and Babcock and Wilcox Plants,” M. Pilch, et al., SAND 97-0667, November 1998.
6	NUREG/CR-6427, “Assessment of the Direct Containment Heating Issue for Plants With Ice Condenser Containments, M. Pilch, K.D. Bergeron, J.J. Gregory, SAND 99-2253, April 2000.

The fundamental behavior, repeatedly demonstrated in the DCH tests, that was key to resolving the issue was the phenomena by which intermediate subcompartments in the containment trap most of the debris dispersed from the reactor cavity. It was further demonstrated that DCH was also limited by incoherence in the steam blowdown and melt entrainment process; the molten core debris is exposed to only a small fraction of the steam blowdown. Thus there is no mechanism to transfer the bulk of the sensible energy of the melt to the containment atmosphere.

Using these insights as the basis for simplified equilibrium models (and a more complex transient code model), it was demonstrated that the conditional containment failure probability for most large dry and subatmospheric plants was negligible ($<10^{-3}$) and, in fact, for many designs, there was no intersection of the distribution of containment loading with the fragility curve. For some designs there was a somewhat greater vulnerability due to their geometrical layout which led to less efficient trapping of debris in subcompartments, and for some of those plants (CE-like designs similar to that of Calvert Cliffs), it was useful to also consider the probability of unintentional depressurization of the reactor system by creep failure of the hot leg prior to vessel failure. Some of the CE and B&W plants were more vulnerable due to higher hydrogen concentrations in the containment. Even for large dry designs more susceptible to DCH, the majority within that group had conditional containment failure probabilities between 0.01 and 0.1, given a high-pressure core melt. Since the probability that vessel breach occurs at high or intermediate pressure is estimated at ~0.1 (for accidents starting at high pressure) due to unintentional depressurization, then it can be concluded that the probability of containment failure given core melt is further reduced to an acceptably low level. The two most susceptible designs, Calvert Cliffs and Maine Yankee, had conditional failure probabilities (given a high-pressure core melt) greater than 0.1 (0.15 and 0.21 respectively). Thus after consideration of unintentional depressurization, the containment failure probability (given accidents starting at high pressure) is ~0.02. Further refinement of these estimates may be achieved by integrating the DCH evaluation into the plant specific PRA.

The ice condenser plants were evaluated as a separate class of W designs due to their smaller volume and lower design pressure. Because of their design characteristics, the DCH evaluation for ice condensers involved a more detailed study to address all early failure modes for representative station blackout and non-blackout scenarios. The general conclusion of the study was that the ice condenser plants are substantially more vulnerable to early containment failure than large dry containments and that this vulnerability is not confined to DCH but includes other phenomena such as hydrogen combustion. In fact, early containment failure was dominated by non-DCH hydrogen combustion events rather than by DCH; ice condenser igniter systems are not operable during station blackout events. A plant-specific evaluation of the containment event trees (from the IPEs) showed that all plants, except McGuire, have an early containment failure probability below 10 percent (0.4 percent to 6.0 percent) for full power internal events. The McGuire early containment failure probability was only slightly higher, at approximately 14 percent, because of the relatively higher station blackout probability at McGuire. Even though the early failure probabilities for ice condenser plants were estimated to be higher than for large dry PWR containments, the containment failure probabilities remain consistent with a general objective of a conditional containment failure probability of 10 percent. The apparent vulnerability of the designs to station blackout sequences is being addressed currently under NRC activities to risk inform 10 CFR Part 50 and the first application, 50.44, which deals with hydrogen control requirements.

Conclusions and Insights

With the completion of our study of DCH the NRC has resolved a longstanding severe accident safety issue and in the process developed certain general insights and confirmed others with respect to the resolution of complex technical issues.

The first among those insights was the clear importance of well designed and scaled experiments (at different scales) which are sufficiently integral and prototypic of actual reactor designs to capture the important physical behavior. Such testing, while expensive, is vital to address complex interrelated phenomena. The acknowledged success of the DCH experimental program was due to the expertise of the researchers at Sandia, Argonne and Purdue and to the guidance of the independent design review committee.

The DCH research program spawned important research to address the general issue of reactor coolant system (RCS) failure during high pressure severe accident sequences that has since been used in other applications, namely assessment of the risks posed by steam generator tube flaws. It was through research to address unintentional depressurization of the RCS, in part for the purpose of mitigating DCH, that the NRC further refined its modeling of natural circulation flows, important for predicting the environment seen by steam generator tubes. Also, the issue of depressurization of the RCS provides some insights on accident management. The general conclusion that plants are not vulnerable to DCH suggests that active measures, implemented through emergency procedures, to depressurize a PWR are not necessary to mitigate DCH. For a few designs, however, depressurization was seen to be beneficial for DCH and would, in any event, be beneficial with respect to steam generator tube integrity.

Resolution of complex problems significantly benefits from applying consistent integrated analysis for specific accident sequences. By this approach one avoids the compounding of conservatisms that inevitably occurs when the problem is tackled as a sum of its parts. Further, the use of a probabilistic framework to address the uncertainties in boundary conditions also limits the extent to which unnecessary conservatism is introduced. Future efforts should build upon the synthesis of probabilistic and deterministic analytical methods.

References:

1. Zion Probabilistic Safety Study, Consolidated Edison Company (CECo), Chicago, IL 1981.
2. High Pressure Melt Ejection (HPME) and Direct Containment Heating (DCH), State-of-the-Art Report, NEA/CSNI/R(96) 25, OCDE/GD(96)194, December 1996.
3. NUREG-1365,"Revised Severe Accident Research Program Plan," August 1989.
4. NUREG/CR-5809, "An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution," November 1991.
5. Electric Power Research Institute, EPRI TR-1028-15, "Natural Circulation Experiments for PWR High-Pressure Accidents," August 1993.

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