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MEMORANDUM FOR: Dennis J. Dougherty, Chief, Technical Assistance Contracts Branch

FROM: Denwood F. Ross, Acting Director, Division of Project Management

SUBJECT: RS-NRR-80-101
"ADVANCED REACTOR ACCIDENT DELINEATION AND ASSESSMENT" (SECY-80-67)

I am forwarding to you a revised Statement of Work for the subject technical assistance (TA) contract. The revised statement now deals exclusively with TA work associated with severe accident mitigation for LWRs, with the focus on the Zion/Indian Point plants (See draft 3 of TMI-2 Action Plan, Section II.B.6). This revision has been coordinated with Mr. William Menczer of your staff. Three points should be made concerning this revision.

First, it meets the two objections raised by the Commission in their letter to William Dircks of March 10, 1980, namely, by directing the work wholely to LWRs (as opposed to the initial direction to fast breeder reactors as well as to water reactors) and placing appropriate emphasis on the licensing nature of the work (as opposed to what appeared to the Commission to be a "research" orientation in the initial wording).

Second, even though the end product in terms of types of reactors is somewhat different, the scope of the work is similar to the initial scope because the subject matter is generic and applicable to a variety of reactor types. The original Statement of Work had three Tasks (Task IA, Task IB, and Task II). Task IA (Heavy Water Reactors) has been dropped in the revision but was never intended to be a major portion of the total effort. Task IB (Light Water Reactors) is expanded somewhat and directed specifically to problems associated with the present Zion/Indian Point action. In the revision this task (original Task IB) becomes Tasks II and III. The original Task II (Fast Breeder Reactors) is very similar to the revised Task I but directed to core melt problems associated with Zion/Indian Point as opposed to the original direction to Fast Breeder Reactors. Basically, the content of the effort remains the same; the specific direction to the Zion/Indian Point Action is what is different. This re-emphasis is apparent in the "Background" statement.

Third, the duration of this contract has been changed to 18 months (from 36 months) because the Zion/Indian Point Action program conducted by the Commission is expected to continue for about 18 months. Because of

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the nature of the contract program we view the accomplishments to be about half those initially anticipated in the 36 months contract. Specific Zion/Indian Point milestones for this contract will have to, of course, be developed (the overall Z/IP plan should be completed shortly).

I wish to emphasize the urgency of implementing this contract. We view this work to be an integral part of the Z/IP effort within NRR. The Commission is committed to completing the study of features to reduce the probability or mitigate the consequences of severe accidents for the Zion Units 1 and 2 and the Indian Point Units 2 and 3 on a very tight schedule. Interim actions imposed on Z/IP are just that; the implementation of long-term permanent actions are key for continued operation of these units. We anticipate a continuation of the fine cooperation your office has given us in expediting this matter, in particular, through the efforts of your staff member, Mr. William Menczer. If there is any way we can be of assistance, do not hesitate calling me or Dr. James Meyer of my staff on X-27976.

Original Signed By
D. F. Ross

Denwood F. Ross, Acting Director
Division of Project Management

cc: William Menczer

DIST:
Central File
ARB Rdg. File
TSpeis
JMeyer

AMarchese
DRoss
DEisenhut/JOlshinski
PRiehm

OFFICE	ARB/DPM	ARB/DPM	Act.Dir/DPM		
SURNAME	JMeyer:dlr	TSpeis	DRoss		
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PART I - STATEMENT OF WORK

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A. Background

As a result of the TMI-2 accident, the NRR staff has recognized the need to re-examine the emergency preparedness plans and capabilities of all nuclear power plants. Although programs are underway that will evaluate all nuclear plant sites, two sites have been singled out for additional evaluation at this time. The sites are those for the Zion Station, Units 1 and 2, in northern Illinois and the Indian Point Station, Units 2 and 3, in New York. These sites are being evaluated because they represent the four operating reactors which are located in areas of unusually high population density and therefore are believed to present a disproportionately high contribution to the total societal risk from reactor accidents. This evaluation will determine if additional design and procedural preventive or mitigative measures are warranted in order to reduce the probability of occurrence or to reduce the consequences of an accident more severe than the current design bases at these sites.

In the event of such a severe accident, releases of radioactivity to the public may be conveyed through the air, through the ground water supply, or by both paths. Postulated radioactivity releases might occur rapidly due to a rupture of containment, or such postulated releases may occur at a later time due to a slow development of containment overpressure or as a result of a core melt-through which could eventually lead to liquid pathway releases. A rapid release of a given amount of radioactivity could result in high public consequences if there was insufficient time to implement protective measures such as evacuation.

Several rapid atmospheric release paths being evaluated include: steam explosions in the reactor vessel or the containment building which rupture the containment; hydrogen explosions; open vents in the containment at the time of an event; and Event V considerations. Releases at various other time frames are also being evaluated and include: loss of power events; loss of heat removal events; and slow overpressurization. In addition, slow liquid pathway releases will be evaluated.

Recent studies (see for example, NUREG-0440 "Liquid Pathway Generic Study") indicate that the probability of a steam explosion rupturing containment during such a postulated accident is relatively small in comparison to other release mechanisms and therefore, in this evaluation, priority will be given to the study of the other release mechanisms. The potential hydrogen explosion release path, open containment release path, and Event V check valve failure path will be examined further, and, if found necessary, design and procedural preventive and mitigative measures will be required in order to reduce the probability of occurrence of these release paths.

The sustained loss of all AC power leading to core melt may initiate release paths of various time scales. The probability of occurrance of these release paths will be further evaluated, and preventive measures such as more reliable decay heat removal systems, or mitigative measures such as a filtered vented containment system to prevent the resultant rupturing of the containment building will be required as necessary.

Delayed air pathway releases due to a slow overpressurization of containment may be generated as a result of loss of containment heat removal, hydrogen burning, or due to a buildup of gases from a molten fuel - containment basemat interaction. These slower release paths will be examined further, and mitigative or preventive design and procedural measures will be required as necessary.

The slow liquid pathway release results from fuel melting through the reactor vessel and through the containment basemat or the containment walls. This release path will be examined further, and mitigative or preventive design and procedural measures such as core retention devices will be required as necessary.

Recognizing the length of time that may be required to implement some or all of the severe accident mitigation features (probably one to two years), the staff has evaluated a number of interim operational actions that should be implemented at these high population density sites for this period of time. Additionally, the staff is undertaking a concerted effort to accelerate current outstanding generic and plant specific licensing actions at these plants.

The general objective then is to define design or procedural measures that significantly reduce the likelihood and/or mitigate the consequences of an accident more severe than the current design bases at the Indian Point and Zion nuclear plant sites. Measures are to be identified that significantly reduce the probability of an event or the source term magnitude of such an accident, or that result in significant additional time to respond to an accident at these sites.

Risk analysis may be helpful in establishing general concepts of appropriate action, but will not be used quantitatively to rule out positive plant improvements. The general approach will be to pursue actively those design features that contribute favorably toward the prevention as well as the mitigation of the consequences of a severe accident. Where reliance is placed on the response of the external population, the time required should be commensurate with the evacuation times estimated by the Federal Emergency Management Agency (FEMA), as available.

<u>Objective</u>

The objective of this project is to perform analyses and assessments in three specific areas directly related to the above program in order to aid in the licensing decision making which is to take place during the next 18 months.

B. <u>Work Required</u>

The contractor shall provide all necessary personnel, equipment, and facilities to provide technical assistance to the NRC in the area of licensing associated with the Zion and Indian Point activity. Specific tasks to be included in this program and associated effort are:

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TASK I CORE MELT ACCIDENT ANALYSIS

The contractor shall perform state-of-the-art analyses and assessments for Light Water Reactors (LWRs), particularly for the Indian Point Nuclear Units 2 and 3 and for the Zion Nuclear Units 1 and 2 in the core-melt related areas indicated in the following four subtasks:

Subtask: a. Molten Pool and Debris Bed Heat Transfer

This includes analyses related to the thermal-hydraulics of core debris penetration (molten or solid) into either containment materials (steel, concrete) or core retention system materials (e.g., magnesium oxide, graphite). Special emphasis in the area of molten pool heat transfer shall be devoted to phenomena associated with freezing/re-melting, penetration of a hot solid mass of core debris, concentration/density gradients, and gas generation effects from concrete interaction. The coolability of debris beds for the aforementioned water reactors will also be assessed, including a detailed study of the scenarios and conditions necessary to form a coolable debris bed.

Subtask: b. Thermophysical Properties

This includes preparation of a data file on all relevant thermophysical properties. The properties which will be compiled are those which are especially relevant to performing consequence evaluations of core meltdown events. This subtask shall complement but not be duplicative of the DOE Safety Analysis Coordinated Reactor Data (SACRD) program.

Subtask: c. Core Melt Computer Code Applications

This includes implementation and application of computer codes (e.g., GROWS, CORCON, WESCHL) being developed, particularly as part of RES programs, for analyzing core melt front penetration by either molten material or a hot solid mass of material. These codes shall be applied to Zion and Indian Point for performing core meltdown consequence evaluations. Modifications to existing computer codes shall also be included where this is deemed to be appropriate.

Subtask: d. Post-Accident Core Retention and Containment Systems

This includes a review and independent evaluation of heat transfer analytical methods used by reactor designers in developing proposed core retention/containment system approaches. Special emphasis shall be devoted to performing sensitivity analyses for determining the relative importance of various thermal/hydraulic parameters, and examining the effectiveness of various proposed core retention/containment system approaches in accommodating low probability accidents.

In the performance of subtasks a through d above, the contractor shall maintain close interaction with the BNL effort on post-accident containment analyses.

State-of-the-art technical information relevant to the above, in particular, data and analyses generated by contractors sponsored by NRC's Office of Nuclear Regulatory Research, shall be used by the contractor as appropriate.

TASK II. ADVANCED DECAY-HEAT REMOVAL SYSTEMS (ADHR)

This activity involves the evaluation of advanced decay-heat removal systems for PWRs that have the potential to improve the overall reliability of decay heat removal under operational and accident conditions. The evaluation should consider such items as potential improvement to existing systems (such as increasing the PORV relieving capacity, improvements to existing Auxiliary Feedwater Systems (AFW), and upgrading HPI and recirculation pumps), as well as the addition of a completely diverse system. This could be a dedicated high pressure RHR system or a comparable feedwater system, either of which could be "hardened" to protect against aircraft impacts, toxic fires and sabotage. A system such as the last one is now part of the German Standard PWR plant. The German System is located in a separate building partially below grade. This effort should be focused on the Indian Point Nuclear Plants Units 2 and 3 and the Zion Plants Units 1 and 2 and shall include the following subtasks:

Subtask: a. Survey and Evaluation

Evaluate existing designs (e.g., the German bunkered system) and/or other conceptual designs and options that can improve the reliability of the decay-heat removal in the Z/IP plants. Consider changes/add-ons, as well as totally diverse paths from the presently existing ones. Establish design criteria and requirements and evaluate the usefulness and/or the potential improvements to reliability of proposed ADHR systems.

Subtask: b. Application to Z/IP

Based on the results of subtask a, determine appropriate approaches for backfitting ADHR systems to Z/IP plants. Based on safety goals and design bases (to be determined by NRC) propose design criteria and requirements, and conceptual designs which are appropriate for the Z/IP plant/site.

Subtask: c. Impact of ADHR Systems on Overall Safety

Using deterministic as well as quantitative-probabilistic (at least in a relative sense) assessments determine contribution to safety of the potential candidate ADHR systems backfitted to the Z/IP plants, including an estimation of the uncertainties involved in such assessments.

TASK III. RISK REDUCTION CRITERIA FOR A FILTERED-VENTED CONTAINMENT

There have been several concepts suggested for use as engineered safety features aimed at mitigating the consequences/effects of degraded core/core melt accidents. Of primary interest is the use of a filtered-vented containment system (FVCS) in conjunction with methods to cope with hydrogen generation/accumulation for maintaining containment integrity following these potential accidents. The ultimate decision to require these features depends upon their potential for risk reduction.

In this task, potential acceptance criteria for employment of a FVCS in the Z/IP plants, including the effects of hydrogen generation/accumulation will be derived. These criteria will be formulated in terms of improved safety potential. From the criteria, the requirements for designing these engineered features will be obtained. The various proposed concepts and designs will be examined to determine whether or not such requirements can be met and/or what design changes could be made to meet them.

Included in this task is the consideration that will be given for both generic (such as steam explosions) and site specific (such as earthquakes) effects. In the latter context, various design bases such as the OBE and SSE used for other safety and non-safety grade features will be explored for filtered-vented concepts, as well as their interaction.