Task A-43

CONTAINMENT EMERGENCY SUMP PERFORMANCE

Lead NRC Organization:	Division of Reactor Safety Research (RES)
Implementation of Results:	Division of Licensing (NRR)
Lead Supervisor:	L.H. Sullivan, Acting Assistant Director for Water Reactor Safety Research
NRR Cognizant Supervisor:	Karl Kniel, Chief, Generic Issues Branch, DST
Task Manager:	A.W. Serkiz, Separate Effects Research Branch
NRR Lead Reviewer	F. Orr, RSB/DSI
Applicability:	PWRs and BWRs
Projected Completion Date:	September 1983

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1. DESCRIPTION OF PROBLEM

Following a loss-of-coolant accident (LOCA) in a PWR, water discharging from the break collects on the containment floor. During the initial portion of the LOCA, emergency core cooling systems (ECCS) and containment spray systems (CSS) draw coolant from a large tank. When a low level is reached in the tank, the ECCS and CSS pumps are realigned to draw coolant from the containment floor (containment emergency coolant sumps). This latter (or long-term) phase is called the recirculation mode. Thus, the containment and sumps become a key flow link for the safety systems in providing for long-term cooling to dissipate reactor decay heat and control of containment conditions.

The importance of the flow link formed by the containment and sump to the operation of the safety systems during recirculation has been recognized for some time. Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems" provides sump design guidelines and a section of Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors" describes a pre-operational test intended to demonstrate adequate NPSH for recirculation pumps and vortex control.

Plant reviews and containment sump tests have identified a number of possible flow conditions which could degrade safety system(s) operation during the recirculation mode. A deficiency in NPSH causes pump cavitation, which in turn, can produce flow instabilities, pump vibration and possible equipment failure. Excessive air entrainment due to inadequate sump hydraulic design(s) can produce similar effects. Experience with the application of these Regulatory Guides has revealed a number of deficiencies. It is now recognized, for example, that a test performed according to the provisions of Regulatory Guide 1.79 would not consider several parameters critical to sump performance. Safety is assured for new plants by requiring that successful tests are conducted prior to operation. This has resulted in the need for redesign and retesting by some utilities and rereview by the NRC staff.

The provisions of Regulatory Guide 1.82 are based, in part, on an assumption that debris will block not less than 50% of the sump screen area. This 50% blockage value is currently imposed as a preoperational test requirement. The appropriateness of this assumption requires quantitative assessment of the potential for blockage from debris (principally insulation) which might be generated as a result of a large pipe break.

Debris blown off during a LOCA and transported to the sump can alter the sump characteristics. Blockage of the approach paths and blockage of sump screens and trash racks can create conditions conducive to vortex formation. A vortex can simultaneously deliver entrained air and increase pressure losses to the pump inlet. Excessive blockage of trash racks or screens can result in pump cavitation. Flow from drains or pipe breaks near the sump can cause flow patterns conducive to vortex formation.

PWRs licensed prior to 1974 were not evaluated relative to the requirements set forth in Regulatory Guides 1.82 and 1.79; those plants were evaluated on a plant specific bases. Regulatory Guide 1.79 requires pre-startup tests to verify vortex control and acceptable pressure drops across screening and suction lines and valves. Regulatory Guide 1.82 provides criteria for the design of reactor building sumps.

Boiling water reactors also enter a recirculation mode following a LOCA and contain insulation which could present a potential blockage problem. Vortex formation is not considered a serious concern because of the large size of the suppression pool and the consequent low approach velocities. Accordingly, a smaller portion of this program will address only the blockage concern for boiling water reactors.

2. PLAN FOR PROBLEM RESOLUTION

This Task Action Plan is designed to provide guidelines and requirements applicable to the various licensing stages while proceeding with gathering of additional information required to achieve resolution of this unresolved safety issue. To accomplish these objectives, the work effort is sub-divided into the following subtasks:

(1) Summary of Recirculation Tests for PWRs (NRR)
(2) PWR Vortex Technology (NRR)
(3) Interim Plant Surveys (NRR)
(4) Experimental Studies of Sump Hydraulic Performance and Vortex Suppression Devices (DOE-Sandia/RES)
(5) Identification and Characterization of Insulation(s) Used in Representative Plants (RES)
(6) Estimation of Insulation Debris Resulting from Reactor Coolant Pipe Breaks (RES)
(7) Estimation of Debris Distribution within PWR Containments (RES)
(8) Assessment of Debris Motion During the Recirculation Mode (RES)
(9) Assessment of Tolerance of Safety Systems to Debris (RES)
(10) Development of Safety Evaluation Criteria, Implementation Documents, Design Guides, etc. required to resolve safety issue (NRR)

These subtasks are detailed below. The Offices with lead responsibility for carrying out and implementing these subtasks are indicated within the parentheses.

The responsibility for resolution of this safety issue rests with NRR. In order to best utilize NRC's capabilities and resources, these efforts will involve both NRR and RES staff, and subcontracted efforts. RES/RSR will serve as the Task Action Plan manager and will assume responsibility to develop work plans for subcontracting for and managing Subtasks 4, 5, 6, 7, 8 and 9. NRR will provide for overall technical cognizance, establishment of informational (or technical) requirements, and the review and assessment informational adequacy of Subtasks 4, 5, 6, 7, 8 and 9. NRR's principal effort will be initially directed at completing Subtasks 1, 2, and 3, and at a later date to concluding Task 10 (e.g., development of evaluation and design criteria, guides and resolution of the safety issue).

Subtask Descriptions

Subtask 1: Summary of Recirculation Tests for PWR's

The objective of this subtask is to document NRC experience in reviewing sump tests completed and to identify current interpretation of applicable Regulatory Guides. A NUREG report on this topic is scheduled to be issued in August 1981. This report will supplement current OL review practices.

Subtask 2: PWR Vortex Technology

Information and experience gained through plant sump tests has been summarized by the University of Iowa (Contract No. NRC-03-078-130) in a final draft report "PWR Vortex Technology". This report will be issued as a NUREG report and will supplement requirements set forth in Regulatory Guide 1.82 for CP and OL review activities. The Iowa Report and the report resulting from Subtask 1 will document the current experience and technology base.

Subtask 3: Interim Plant Surveys

The subtask will be initiated by NRR. A letter will be sent to several operating PWRs, generally licensed before 1975, requesting water sump and insulation information. The responses should provide plant specific data on sump location and design, and insulation utilized within containment. The sump and insulation information received will extend the survey undertaken under Subtask 5 and provide a data base for better assessing the significance of potential debris.

Subtask 4: Experimental Studies of Sump Hydraulic Design and Vortex Suppression Devices

This Subtask is directed at obtaining experimental data to determine: (a) the interrelationships and relative importance of sump flow and geometric design parameters on the hydraulic performance of containment recirculation sumps, and (b) examine the effective range of vortex suppression devices.

A DOE-sponsored program (on behalf of NRC) has been subcontracted to Alden Research Laboratory thru the Sandia Laboratories. This is an experimental program designed to pursue two principal areas. The first area is the effect of various sump design parameters on the inception of vortices and the experimental data obtained will provide a basis for evaluating containment sumps in older PWRs and formulation of recommended sump design criteria for new plants. The second principal area will be the evaluation of various vortex suppression techniques to identify their range of application. The Alden program and work scope is based on NRR's developed requirements and selection of contractor.

The Alden program was initiated in July 1979; facility shakedown testing was completed in August 1980; and the program is currently scheduled to complete testing in August 1982.

Subtask 5: <u>Identification and Characterization of Insulation Used in Representative Plants</u>

This subtask will survey and document the types, amounts, chemical and mechanical properties, mounting mechanisms and location of insulation currently utilized within reactor containments. Twelve reference plants will be selected for this study (nine PWRs and three BWRs) and the appropriate tabulations, material summations and descriptive plant drawings developed. The type of insulation will be identified by specification and manufacturer.

Contact will be made with the respective plant owners to obtain information related to insulation used and some site visits will be made to obtain confirmation of data received.

As indicated in Subtask 3, the operating plant information received will extend the data base and the combined information will provide a compilation of insulation utilized within containment to generate debris under large LOCA conditions. This then provides a decisional basis to determine the necessity of undertaking Subtasks 6, 7, 8 and 9 as described below.

Subtask 6: <u>Estimation of Quantity and Nature of Insulation Debris Resulting from Pipe Breaks</u>

Engineering analyses will be undertaken to estimate the amount and nature of insulation displaced from breaking primary system pipes by fracture, pipe whip and hydraulic forces. It is expected that initial analyses will provide bounding values as to amount and sizes of insulation debris generated at the postulated break locations. Considerations for the determinations of the displacement and break up of a given type of insulation material would include the nature of the initial pipe break, pipe whip, jet effects, and subsequent environmental effects. The displacement of insulation material from components adjacent to each postulated pipe break location will also be estimated. The currently postulated PWR pipe break locations will be utilized for the initial analyses. The results from the initial effort to estimate debris generation will be reviewed by NRR technical staff before proceeding into more extensive analytical programs or supplemental experimental programs. The estimated costs shown in Section 5 assume a follow up effort (or a two phase effort) being required to conclude estimates of the quantity and nature of LOCA generated insulation debris.

Subtask 7: <u>Estimation of Debris Distribution within the Containment</u>

Given the results of Subtasks 5 and 6, predictive methods will be developed to estimate the spatial distribution of LOCA-generated debris within the containment prior to recirculation. This effort will be undertaken with the first phase being development of analysis methods and sample calculations, these results then being reviewed for acceptability by NRR staff. The second phase will apply the analysis to break locations for a minimum number of representative plants.

Subtask 8: Assessment of Debris Motion and Redistribution During Recirculation

Based on the results of Subtasks 3, 5, 6 and 7, this effort will address the redistribution of debris during the recirculation mode. Engineering estimates will be made of the type, and amount of, debris which might block approach paths, sump screens, and penetrate to the safety systems. A limited experimental effort to evaluate debris flow characteristics is included in the costs contained in Section 5. If reevaluation of the 50 percent blockage assumption set forth currently in Regulatory Guide 1.82 is deemed necessary, Subtasks 6 thru 8 will provide the basis for this determination.

Subtask 9: Assessment of Tolerance of Safety Systems and Core to Debris

This subtask will utilize information developed under Subtasks 5, 6, 7, and 8 will address the susceptibility (or tolerance) of safety systems to entrained debris drawn from the sump. A description of anticipated debris, and attendant LOCA conditions will be compiled for representative plants and an assessment of component and system operability, and life will be carried out. The potential for core blockage will be assessed as part of this subtask.

It should be noted that Subtasks 6, 7 & 8 would be undertaken sequentially, with the proceeding tasks providing information for the specific assessments noted.

Subtask 10: Development of Safety Evaluation Criteria, Implementation Documents, Design and REG Guides Required to Resolve Safety Issue

Although all end products required to resolve this safety issue cannot be identified at this time, the following products are typically expected.

1) Recommendations for a new or revised Regulatory Guide addressing sump design.

2) Recommendations for a new or revised Regulatory Guide for preoperational sump testing.

3) Recommendations for a new Regulatory Guide addressing insulation usage inside containment.

4) Criteria for the reevaluation of containment sumps in operating reactors.

5) Revisions and/or additions to current Standard Review Plans.

6) A final NUREG report providing the staff's safety evaluation and conclusions regarding resolution of this currently unresolved safety issue.

3. BASIS FOR CONTINUED OPERATION AND LICENSING PENDING COMPLETION OF PROGRAM

The performance of containment emergency sumps for plants under review is determined from the evaluation of successful completion of preoperational tests performed in conformance of REG Guide 1.79, and the application of guidelines set forth in REG Guide 1.82. Over the past years (since 1974), the NRC staff has developed increasing confidence in the applicability of thorough preoperational recirculation tests. This experience is being documented (see Subtask (1)) and will be utilized in licensing evaluations and to provide a technical basis for this program. In addition, there are limits (such as the 50 percent blockage criteria) taken from existing REG Guides, which will be reevaluated as a result of this program. Therefore, preoperational testing will continue to be relied upon for demonstration of adequate hydraulic performance of emergency sumps and recirculation pumps prior to issuance of operating licenses for new plants.

However, the sump designs of older plants (pre-1974 OL issuance) were not tested in accordance with Regulatory Guide 1.79 or evaluated against the criteria specified in Regulatory Guide 1.82.

Therefore, the situation with respect to continued plant operation can be viewed as: (a) currently operating plants, (b) plants approaching or having recently received operating licenses and (c) plants approaching the CP stage. The related actions are as follows:

3.1 Currently Operating PWRs -

Operating PWRs, which have not been tested for adequate NPSH and vortex control, may be subject to cavitation or vortex formation. To obtain test data for these containment sump configurations, a series of full scale tests is being performed at the Alden Research Laboratory under Subtask 4. Preliminary results from this test program indicate that even though a severe vortex may be formed for some test configurations, the amount of air entrained in the recirculation pipe results in a void fraction of less than 5 percent. Typical air-water pump performance tests indicate that potential pump flow degradation under these conditions would be minimal. In view of the favorable results indicated by the test data obtained thus far, continued operation is justified pending completion of the preliminary test program.

A systematic review of the initial Alden test series (approximately 25 configurations) is planned for early 1981. These initial tests will include a preliminary assessment of typical vortex suppression devices. The data obtained in this test series will be examined specifically for any potentially significant inadequacies with respect to sump designs in operating plants.

3.2 PWRs Approaching OL, or Having Recently Received an OL

It is our judgement that plants in the OL review stage, or having recently received an OL, have demonstrated adequate sump performance through preoperational tests as described above, and that the pertinent requirements of REG Guide 1.82 have been met.

With respect to the amounts and types of insulation employed, the staff considers all materials which might be capable of being transported to the sump such that the potential for significant blockage of the containment sump screens is precluded. With regard to other potential sources of debris, periodic surveillance inspections are required to detect occurrences of degraded materials.

3.3 PWRs Approaching CP Stage

Licensing staff experience with recirculation tests has identified a number of potentially adverse conditions which could occur (e.g. vortex formation, need for vortex suppression devices, etc.) The reports resulting from Subtasks 1 and 2 will be provided to applicants in the CP stage and the specific plant designs reviewed accordingly.

In addition, applicants will be requested to review insulation utilized within containment, assess LOCA effects on generating debris, and to evaluate performance of safety systems during the recirculation mode.

3.4 BWR Containment Considerations

With regard to BWR containment and ECCS designs, the concern addressed by this task Action Plan is limited to the potential for degraded ECCS performance as a result insulation debris following a LOCA. Specifically, insulation has not been identified and considered quantitatively relative to debris resulting from postulated pipe breaks.

This concern is not adjudged to be significant since - even if some insulation did reach the suppression pool, the likelihood of any insulation being drawn into an ECCS pump suction line is very small. The reason is that suction piping to ECCS systems is typically located 4-6 feet above the pool bottom and calculated approach velocities are very low, thereby permitting debris to settle out or float on the pool surface. In addition, BWR designs employ strainers within pump suction piping, and NPSH calculations for RHR pumps are based on an assumed 50 percent blockage.

Accordingly, continued licensing and operation of BWRs is acceptable pending completion of this program.

4.0 NRC TECHNICAL ORGANIZATIONS INVOLVED

TAP A-43 has been assigned to the Separate Effects Research Branch of RES/RSR. The Task Manager will be responsible for the conduct of Subtasks 4, 5, 6, 7, 8 and 9 as described in Section 2. The assigned NRR Lead Reviewer and NRR Cognizant Supervisor are responsible for the conduct of Subtasks 1, 2, 3 and 10.

4. NRR Technical Organizations Involved

The following NRR branches will provide technical support and input: (Estimates in man-months)

	FY 81	FY 82	FY 83
Generic Issues Branch (GIB)	1.0	1.0	1.2
Reactor Systems Branch (RSB)	2.0	2.0	2.5
Containment Systems Branch (CSB)*	1.0	1.0	1.0
Mechanical Engineering Branch (MEB)*	1.0	1.0	1.0

The principal functions that assigned individuals will be required to provide are: provide problem definition and technical requirements, write interim and final guides and criteria, review technical findings and analyses obtained from subcontracted effort, determine technical acceptability of derived analysis techniques and limiting calculations, etc. Assigned staff will also be members of the TAP A-43 Technical Review Group. NRR staff will also be responsible for performing the effort described under Subtasks 1, 2, 3 and 10.

4.2 RES Technical Organizations Involved

Technical support will be provided by RSR and Sandia Laboratories: (Estimates in man-months)

	FY 81	FY 82	FY 83
Reactor Safety Research**	10	8	8
Sandia	24	30	30

The RES branches shown above will provide A-43 Technical Review Group members and assist the Task Manager in planning effort for, and evaluating results obtained, under Tasks 4, 5, 6, 7, 8 and 9.

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*This effort will be substantially reduced if Subtasks 6-9 are not necessary.

**RSR support will come principally from the Separate Effects Research Branch; they will call upon Metallurgy and Materials Branch and Mechanical Engineering Research Branch as required.

5.0 TECHNICAL ASSISTANCE

The estimated costs associated with Subtasks 4, 5, 6, 7, 8 and 9 are shown in Table 1 and include the contingencies noted. Should extensive analyses and/or experimental effort be required to substantiate findings regarding debris generation, and/or distribution of debris, a very significant cost increase will take place. Such costs are not included in the cost estimates shown in Table 1: The total estimated NRC subcontracted effort costs are as follows:

FY 1980	FY 1981	FY 1982	FY 1983
185,000	365,000	215,000	150,000

Assuming the absence of major analysis and experimental effort required for debris generation estimation, it appears that A-43 should be concluded in September 1983.

Subtask 4 is currently under contract through DOE-Sandia, and it is expected that Subtasks 5, 6, 7, 8 and 9 will be subcontracted through existing RES contracts or National Labs, if possible. Since the majority of subtasks are related to insulation employed, and the effects of LOCA generated debris on containment emergency sumps, use of industrial contractors knowledgeable in containment and reactor equipment design and installation (such as AEs) would be the most cost beneficial to the Government - as opposed to National Laboratories with limited experience in these areas.

Sandia, in addition to managing the Alden sump hydraulics research program, will provide the NRC with independent data analysis and evaluation capability, plus assistance in developing information for use in preparing design guides and evaluation criteria.

The Alden effort and Sandia technical assistance efforts are detailed below.

A. Contractor: Alden Research Laboratory
Funds Required: \$200K FY 1979; \$742K FY 1980; \$500K FY 1981; \$600K FY 1982

This contract is a DOE funded (on behalf of the NRC) experimental program which is under way at Alden Research Laboratory and addresses the issue of adequate sump or suppression pool functioning in the recirculation mode. The objective of that work is to provide the data needed to develop criteria for design, testing and evaluation of plant sumps. Parametric tests will be conducted at Alden to identify regimes where vortex formation or air entrainment present potential problems for sump pump performance.

The second portion of the Alden program is to develop vortex suppression techniques. This work will focus on those plant geometries found to be marginal when compared with the data base developed in the initial testing phase.

TABLE 1

ESTIMATED COSTS FOR TASKS 1, 2, 3, 4, 5, 6, 7, 8 and 9

(Costs in Thousands of Dollars)

	FY 1979	FY 1980	FY 1981	FY 1982	FY 1983
tasks 1 & 2 (Iowa Study)	18			-	-
btask 4 (Alden Contract)	200	742	500	600	-
btask 5 (Insulation Characterization)	-	65	85	-	-
ubtask 6 (Est. of Debris Generation)	-	-	} 85	-	-
ubtask 7 (Est. of Debris Distribution)	-	-		-	-
ubtask 8 (Assessment of Debris Motion)	-	-	45	-	-
S k 9 (Tolerance of Safety System)	-	-	-	65	-
Subtask Totals	218	807	715	665	0
Sandia Technical Assistance	0	120	150	150	150
TAP-43 Total	218	927	865	815	150
NRC Funding Requirements	18	185	365(Est)	215(Est)	150(est)
DOE Funding Requirements	200	742	500(Est)	400(Est)	0

Sump testing started in August 1980 with completion of experiments estimated to be August 1982; a final report is scheduled for mid-1983.

B. Contractor: Sandia Laboratories
Funds Required: $120K FY 1980: $150K FY 1981; $150K FY 1982, $150K FY 1983

This is a technical assistance contract, wherein Sandia staff will work with Alden staff to establish a satisfactory data base, evaluate results being obtained, and carryout independent data analyses. Also, Sandia will assist in preparation of technical data for use in criteria for avoidance of vortex formation, air entrainment and recirculation pump inlet head degradation.

In addition, use will be made of Sandia staff in subcontracting for, and evaluating results, obtained under Subtasks 5, 6, 7, 8 and 9.

6. Interactions with Outside Organizations

A. Utilities

Contacts will be made with utilities owning the "reference" plants selected for the generic insulation survey described in Subtask 5. The generic plant information will be acquired on a cooperative basis without resorting to formal request for information. The results of this generic survey (carried out with the assistance of Burns and Roe) will be used for estimating debris generation and potential effects on long term cooling.

B. Advisory Committee on Reactor Safeguards

These A-43 activities will be coordinated with the appropriate ACRS subcommittee. Significant information will be provided to the subcommittee as it becomes available and meetings will be scheduled at appropriate times as the task progresses.

7.0 ASSISTANCE REQUIRED FROM OTHER NRC OFFICES

A. The Office of Inspection and Enforcement

A liaison will be established with I&E and a request for I&Es involvement will be formulated. Their aid in arranging for site visits, development and preparation of recirculation test criteria, and development of evaluation criteria for operating plants is expected.

B. Office of Standards

The guidance and assistance of this office will be utilized in implementing the results of this program by preparing REG Guides.

8. POTENTIAL PROBLEMS

8.1 The principal potential problem is limited funding resources required to carry out subtasks 6, 7, 8 and 9 if these are determined to be required following concluding Subtasks 3 and 5. The costs shown in Table 1, represent current estimates assuming early achievement of adequate analysis models and that a limited number of plants will be adequate for arriving at meaningful conclusions. Should the initial results of subtask 6 and 7 show a strong plant specific dependence, then decisions will have to be made regarding continuing a generic evaluation, or pursuing a plant specific evaluation.

8.2 Performance of subtasks 1 thru 3 by NRR will require participation from members of DSI and DL over the next several months. Unconditional assignment of selected personnel will be required.

8.3 Subtasks 6 through 9 represent the development of new analyses deemed necessary to support, verify, or correct current practices and recommendations. Some of these subtasks (or elements thereof) may be difficult to model and could grow excessively unless carefully planned in advance and then constrained to the minimum necessary to resolve the safety issue. This will require work scope definition in advance, and NRR acceptance before the effort is undertaken. This poses a potential schedule impact due to the time required to obtain NRR concurrence on the work plans for Subtasks 6, 7, 8 and 9.

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APPENDIX A

TASK A-42

PIPE CRACKS IN BOILING WATER REACTORS

Lead NRR Organization: Division of Operating Reactors (DOR)

Lead Supervisor: L. C. Shao, Acting Assistant Director for Engineering Programs

Task Manager: C. Y. Cheng, DOR:EB

Applicability: General Electric Boiling Water Reactors

Projected Completion Date: December 1979

NOTE: Resolution of this USI is documented in NUREG-0313, Rev. 1 dated July 1980.

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1. DESCRIPTION OF PROBLEM

Leaks and cracks in the heat-affected zones (HAZs) of welds that join austenitic stainless steel piping and associated components in BWRs have been observed since mid-1960s. Prior to September 1974, all affected piping was Type 304 stainless steel with diameters of eight inches or less. All the cracks were attributed to intergranular stress corrosion cracking (IGSCC) due to the combination of high local stress, sensitization of material, and high oxygen content in the water.

During the last quarter of 1974, a number of incidents of IGSCC in weld HAZs of 4-inch diameter recirculation bypass lines and in 10-inch diameter core spray lines were again observed. Following these occurrences, the NRC formed a Pipe Cracking Study Group (PCSG) to (a) investigate the cause of cracks, (b) make an interim recommendation for operating plants, and (c) recommend corrective actions to be taken by future plants. The study Group published its report (NUREG-75/067) in October 1975 which contains several recommendations to reduce the incidence of IGSCC in sensitized stainless steel piping. Following staff review of the Study Group's recommendations, the staff issued an implementation document (NUREG-0313) which established staff positions consistent with the recommendations of the Study Group. The staff has been in the process of implementing these positions over the last couple of years for operating plants and for plants under review for an operating license.

Since 1975, IGSCC has continued to be found in recirculation bypass and core spray lines. Incidents of IGSCC have also been observed in some stainless steel recirculation riser piping up to twelve inches in diameter and in large diameter (>20 inches) recirculation piping in foreign countries. Cracks in these large recirculation lines had not been observed prior to 1975. These incidents, together with the reported questions concerning the reliability of ultrasonic inspections (UT), led to the activation of a new PCSG by NRC in September 1978.

The new Study Group was specifically chartered to reexamine the conclusions and recommendations of the 1975 PCSG report in view of cracks recently discovered in large diameter pipes. Particular attention was given to the significance of cracking found in large recirculation lines, to evaluate the capability of nondestructive examination (NDE) methods to detect IGSCC and, in addition, to assess the significance of the safe-end cracking at Duane Arnold relative to similar material and design aspects at other facilities.

The 1978 Study Group completed its evaluation and published the NUREG-0531 report in February 1979. The most important finding of this investigation was that the conclusions and recommendations reached in NUREG-75/067 by the previous PCSG and the implementation document, NUREG-0313, are still valid. The present Study Group not only reaffirmed the conclusions and recommendations reached by the previous group but also presented some new ideas to reduce the potential for IGSCC based on the operating experience since 1975 and the recent pipe cracking in large diameter pipes. In addition, the present Study Group has addressed IGSCC in safe-ends and has reached conclusions and recommendations concerning them which were not discussed by the previous Study Group. Because of these new ideas and issues addressed by the 1978 PCSG, the implementation document NUREG-0313 needs to be updated to incorporate the latest recommendations made by the present Study Group.

2. PLAN FOR PROBLEM RESOLUTION

A. Approach

The problem will be resolved by identifying the new conclusions and recommendations reached by the present PCSG by carefully studying and comparing the conclusions and recommendations made in NUREG-75/067, NUREG-0313, and NUREG-0531. The implementation document NUREG-0313 will then be revised to incorporate those new recommendations which can be implemented immediately. For those new recommendations which will require further study before it can be implemented, a plan for establishing the staff position on each recommendation will be proposed.

B. End Product

The end product of this activity will be a NUREG report documenting the updated staff position on material selection and processing guidelines for BWR piping based on recommendations made by the present PCSG. This report will be issued approximately in Mid-August 1979.

C. Tasks

C-1. Revision of NUREG-0313, "Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping"

Review and identification of those new conclusions and recommendations in NUREG-0531 which can be implemented immediately. The specific effort will include updating the implementation document NUREG-0313 to incorporate these new recommendations. This subtask will be accomplished in Mid-August 1979.

C-2. Staff Recommendation of Follow-on Efforts to Reduce the Potential for IGSCC in BWR Piping

Those conclusions and recommendations of NUREG-0531 which would require further study before the staff position can be established will be identified. In addition, a plan for establishing such a position will be recommended. This subtask will also be completed approximately in Mid-August 1979. However, the technical activities for these follow-on efforts will definitely not be completed within the time span specified for this activity.

3. BASIS FOR CONTINUED PLANT OPERATION AND LICENSING PENDING COMPLETION OF TASK

For new plants or plants under construction and operating plants, we have concluded that, pending completion of this task, continued plant operation and licensing do not constitute an undue risk to the health and safety of the public for the following reasons:

. Although the augmented inservice inspection programs required by NRC cannot detect all IGSCC, it has demonstrated to be effective in locating most instances of IGSCC prior to cracks propagating through the wall.

- The leak detection system employed as a monitoring system has been effective in alerting the plant operators of primary system leakage that could result from a through-wall crack.

- Sudden failure or significant loss-of-coolant is not expected from through-wall cracks prior to a period of leakage.

- Should a large through-wall crack develop, go undetected by NDE inspections, and by continuous leak detection devices, and subsequently should a rupture of the line occur causing a loss-of-coolant accident, the design of a nuclear power plant is such that protection is still provided for the public health and safety.

To summarize, the various NRC actions taken to date ensure that IGSCC does not pose an immediate safety problem to operating plants and thus constitute an acceptable basis for continued plant operation and licensing.

4. NRC TECHNICAL ORGANIZATION INVOLVED

A. Engineering Branch (EB), Division of Operating Reactors, has the overall lead responsibility to see this TAP to its completion. This includes review and evaluation of the subject NUREG reports to establish the implementation guidelines with particular emphasis on operating plants, and final issuance of a NUREG report. In addition, EB will have the lead responsibility of identifying long-term follow-on efforts and recommending plans for establishing the implementing guidelines for thise issues.

Manpower Estimates: 4 man-months FY 1979

B. Materials Engineering Branch (MTEB), Division of Systems Safety, has the lead responsibility of establishing the implementation guidelines for new plants and plants under construction. MTEB will have direct input to the revision of NUREG-0313. MTEB will also identify long-term follow-on efforts and recommend plans for establishing staff position on these issues.

Manpower Estimates: 3.5 Man-Months FY 1979

5. TECHNICAL ASSISTANCE

No technical assistance is needed for the present tasks. However, technical assistance may be required for the identified follow-on efforts.

6. ASSISTANCE REQUIREMENTS FROM OTHER NRC OFFICES

No assistance from other NRC offices is required for Subtasks C-1 and C-2. However, some assistance may be needed for the follow-on efforts identified under Subtask C-2. All research and developmental programs aiming to increase or maintain the integrity of BWRs piping will definitely assist us in establishing the implementation guidelines for the follow-on efforts. Specifically,

A. Office of Standards Development

Structures and Components Standards Branch/DES is currently funding EG&G to develop a Regulatory Guide on "UT of Austenitic Stainless Steel Piping."

This guide will provide a UT performance standard or procedure which will significantly increase the detection capability for IGSCC in austenitic stainless steel piping.

B. Office of Nuclear Regulatory Research

Metallurgy and Materials Branch/RES is currently funding the Pacific Northwest Laboratories to study the "Reliability of Non-destructive Examination" aimed to pinpoint the strengths and weaknesses of NDE and recommend the appropriate experimental programs to increase the reliability of flaw detection.

7. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

No major interactions with outside organizations are anticipated for the subtasks. However, an extensive interaction with outside organizations will be necessary for the follow-on efforts. This interaction involves information exchanges with licensee, GE, industry research institutes, and national labs that are active in research and development of methods to reduce the potential for IGSCC or to detect the occurrence of IGSCC. An information exchange with foreign regulatory and inspection organizations is also expected.

8. POTENTIAL PROBLEMS

No difficulties have been anticipated in achieving this task. However, some delay in achieving the follow-on efforts, if the task is expanded, might be expected because of the long-term nature of the problem and the necessary extensive interactions with other organizations.