NUREG-0471

GENERIC TASK PROBLEM DESCRIPTIONS

Category B, C, and D Tasks

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

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Program Support Branch
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
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FOREWORD

This document contains information relating to Category B, C, and D generic technical activities. The Category B, C and D generic technical activities described herein were identified and placed in their priority categories by the Office of Nuclear Reactor Regulation (NRR) within the context of its "Program for the Resolution of Generic Issues Related to Nuclear Power Plants." This program is described in NUREG-0410.

The specific information provided for each task includes the reactor type to which the generic issue applies, the NRC division with lead responsibility and a description of the problem to be addressed by the task. Also provided in this document is a listing of Category A generic technical activities and definitions of Priority Categories A, B, C and D. Detailed Task Action Plans for Category A generic technical activities are contained in NUREG-0371.

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PRIORITY CATEGORY DEFINITIONS

Category A:

Those generic technical activities judged by the staff to warrant priority attention in terms of manpower and/or funds to attain early resolution. These matters include those the resolution of which could (1) provide a significant increase in assurance of the health and safety of the public, or (2) have a significant impact upon the reactor licensing process.

Category B:

Those generic technical activities judged by the staff to be important in assuring the continued health and safety of the public but for which early resolution is not required or for which the staff perceives a lesser safety, safeguards or environmental significant than Category A matters.

Category C:

Those generic technical activities judged by the staff to have little direct or immediate safety, safeguards or environmental signficance, but which could lead to improved staff understanding of particular technical issues or refinements in the licensing process.

Category D:

Those proposed generic technicl activities judged by the staff not to warrant the expenditure of manpower or funds because little or no importance to the safety, environmental or safeguards aspects of nuclear reactors or to improving the licensing process can be attributed to the activity.

LIST OF TECHNICAL ACTIVITIES

Category A Tasks

Task No.	Title
A-1	Water Hammer
A-2	Asymmetric Blowdown Loads on PWR Primary Coolant Systems
A-3	Westinghouse Steam Generator Tube Integrity
A-4	Combustion Engineering Steam Generator Tube Integrity
A-5	Babcock & Wilcox Steam Generator Tube Integrity
A-6	Mark I Short Term Program
A-7	Mark I Long Term Program
A-8	Mark II Containment Pool Dynamic Loads
A-9	ATWS
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A-14	Flaw Detection
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning
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A-17	Systems Interactions in Nuclear Power Plants
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A-35	Adequacy of Offsite Power Systems
A-36	Control of Heavy Loads Near Spent Fuel

Task No.	Title
A-37	Turbine Missiles
A-38	Tornado Missiles
A-39	Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containments
A-40	Seismic Design Criteria - Short Term Program

CATEGORY B GENERIC TASK

PROBLEM DESCRIPTIONS

Task No	Title
B-1	Environmental Technical Specifications

Applicability	Lead Division
LWRs	DSE

Problem Description

Current NRC regulations and practice require that certain operating requirements, Technical Specifications, be made part of each operating license. These Technical Specifications comprise an Appendix A which deals with safety features and an Appendix B which deals with environmental concerns. The non-radiological portion of Appendix B traditionally derives from information in the FES and other relevant sources. They are proposed by the applicant and reviewed and approved by the staff on a case-by-case basis. Based on several years of staff experience with facility licensing and a better understanding of EPA and NRC responsibilities in the area of water quality regulation, we believe that the development of Standardized Environmental Technical Specifications (SETS) is appropriate. Standardized Technical Specifications will result in more efficient use of staff and applicant resources and more uniform requirements and performance standards for licensees. This task will include development of ETS for two scheduled plants, Three Mile Island 2 and McGuire 1 and 2 to be used as models for the development of SETS, and coordination with the Advisory Committee on Standardized Environmental Technical Specifications and with the Office of Standards Development in the development of Regulatory Guide 1.48. The final product will be SETS published either as a NUREG report or as part of Regulatory Guide 4.8.

Task No.	Title
B-2	Forecasting Electricity Demand

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

This task involves efforts directed at improving the NRC staff's capability to forecast electricity demand for the purpose of evaluating applicant's need for power forecasts in individual licensing cases. The task will include (1) developing the capability for forecasting electricity demand on a regional basis, (2) a reexamination of the NRC's current licensing approach to the need for power issue and (3) an analysis and comparison of the economic penalties of having a plant come on-line sooner than needed versus having the plant come on-line later than needed.

Task No.	Title	
B-3	Event Categorization	

Applicability	Lead Division
LWRs	DSS

Problem Description

There are several inconsistencies in event categorization between the GDC, SRP, Standard Format and applicant submissions. In addition, categorization by other groups such as ANSI and ANS is not always consistent with NRC positions. In several cases, applicants have proposed that certain events be categorized as accidents (which would permit limited fuel damage) whereas the staff categorizes them as anticipated transients. This task will categorize the postulated transients and accidents and define acceptance criteria for the various categories. The result will be an improved licensing process and possible relief from current restrictive requirements for some licensees.

Task No.	Title	
B-4	ECCS Reliability	

Applicability	Lead Division
LWRs	DSS

Problem Description

Numerical reliability goals and methods of analysis have not been established by the NRC. The current basis for plant licensing continues to be NRC regulations which require, among other things, that the consequences of a LOCA be suitably mitigated despite the postulated subsequent occurrence of a series of independent malfunctions, including the loss of offsite power and postulated single failures in mitigating safety features. As the state of the art on probabilistic evaluation techniques become more widely accepted, increasing concern has been expressed regarding the adequacy of the current licensing basis. This task will address these concerns by assessing the reliability of current U.S. and foreign emergency core cooling system designs.

Task No.	Title
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments

Applicability	Lead Division
All Reactor Types	DSS

Problem Description

Ductility of Two Way Slabs and Shells - There is a general lack of information related to the behavior of two-way reinforced concrete slabs loaded dynamically in biaxial membrane tension, flexure and shear. This task involves developing a more dependable and realistic procedure for evaluating the design adequacy of Category I reinforced concrete slabs subject to a postulated LOCA or high energy pipe break.

More specifically, this task will determine with sufficient accuracy the influence of biaxial membrane tension on the resistance function and the permissible ductility ratio of two-way slabs loaded in flexure and shear. Since the response of the slab to the postulated loading conditions will likely be in the nonlinear range because of the simultaneous application of the severe, time dependent pressure load and concentrated jet force, the analysis performed must encompass the nonlinear range. The task will provide the following specific information:

(1) A summary of the existing state-of-the-art on the subject resulting from a literature search.

(2) The relationship between ductility of one-way slabs and two-way slabs.

(3) The ductility of two-way slabs under shear and flexure separately and under combined loading conditions, including the biaxial membrane tensile force.

(4) Recommendations relative to avoidance of shear failure that could be utilized in practical design applications.

(5) A comparison of solutions obtained by analytical methods with applicable tests performed on two-way slabs.

Buckling Behavior of Steel Containments - The structural design of a steel containment vessel subjected to unsymmetrical dynamic loadings may be governed by the instability of the shell. For this type of loading, the current design verification methods, analytical techniques, and the acceptance criteria may not be as comprehensive as they should be. Section III of the ASME Code does not provide detailed guidance on the treatment of buckling of steel containment vessels for such loading conditions. Regulatory Guide 1.57 recommends a minimum factor of safety of two against buckling for the worst loading condition provided a detailed rigorous analysis, considering inelastic behavior, is performed. On the other hand, the 1977 Summer Addenda of the ASME Code permits three alternate methods, but requires a factor of safety between 2.0 and 3.0 against buckling depending upon the applicable service limits.

Due to the lack of a uniform well defined approach to the problem, this generic task has been established to assess the buckling criteria for containment shells. This task has the following specific objectives:

1. To review and assess the assumptions and methodology presently used in the buckling analysis of steel containment shells,
2. To establish general standard design and acceptance criteria for the dynamic/static stability of steel containment shells, particularly for steel containments subjected to unsymmetrical internal or external dynamic loads,
3. To evaluate the computer programs presently used in the buckling analysis and design of steel containment sheels by developing benchmark problems to verify these programs, and
4. To perform selective detailed reviews of typical containment designs to assess the effect that any new licensing requirements may have on different types of containments.

Task No.	Title
B-6	Loads, Load Combinations, Stress Limits

Applicability	Lead Division
LWRs	DSS

Problem Description

The designer of pressure vessels and piping system components must consider (1) the individual and combined loads that will act on each component due to normal operating conditions, system transients and postulated low probability events (accidents and natural phenomena) and (2) the stress limits to be used in evaluating structural integrity and component operability when subjected to these loads. This task will evaluate the need for and develop additions or modifications to current guidance and staff technical positions in Regulatory Guide 1.48 and the SRP that will (1) address areas not currently addressed (e.g., higher stress limits recently adopted by the ASME for use with Class 2 and 3 components), (2) identify and eliminate any overly conservative requirements and (3) provide more detailed guidance where needed.

Task No.	Title
B-7	Secondary Accident Consequence Modeling

Applicability	Lead Division
PWRs	DSE

Problem Description

This task will develop more reliable models and associated computer capability than currently available to the staff for assessing the radiological consequences of accidents that could result in the release of radioactivity through secondary systems.

Task No.	Title
B-8	Locking Out of ECCS Power Operated Valves

Applicability	Lead Division
LWRs	DOR

Problem Description

This is an ACRS generic concern. The physical locking out of electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS has been required by the staff, based on the assumption that a spurious electrical signal at an inopportune time could activate the valves to the adverse postion; e.g., closed rather than open, or opened rather than closed. While such an event has a finite probability, another probability exists that the valves might be adversely positioned due to operator error.

This task will involve a reevaluation of the staff requirement using a systems approach, and considering such items as (1) the evaluation of the probability of a spurious signal; (2) time required to reactivate the valve operator; (3) status of signal lights when the circuit breaker is open; (4) can the valve be locked out in an improper position due to a faulty indicator; (5) are there other designs improving reliability witout lock-out; (6) what are the advantages and disadvantages of corrective action by an alert operator in case of incorrect positioning vis-a-vis a system with power locked out.

Task No.	Title
B-9	Electrical Cable Penetrations of Containment

Applicability	Lead Division
LWRs	DSS

Problem Description

Some prototype electrical penetration failures have occurred to date. In addition, failures of low voltage penetration modules have occurred at a licensed facility. It was originally postulated that failures of the low voltage penetration modules was due to electrical short circuits caused by collection of moisture in fissures (cracks) in the epoxy insulator/sealant. However, results of the laboratory anlaysis indicated that failures were caused by heating of the conductors at the connection splices within the penetration module. The heating resulted from high contact resistance due to epoxy intrusion into an area of the connector splice which was not insulated during the manufacturing process. The accumulation of carbon deposits over a period of time, resulting from the heating process, created a conductive path (short circuit) between adjacent conductors in the pentration modules. This task will reevaluate current licensing criteria for the design and qualification testing of electrical penetrations in the reactor containment (provided in IEEE-317-1972 and Regulatory Guide 1.63) in light of concerns raised by these failures. A determination will be made as to whether additional or improved criteria are necessary and if so, improved criteria will be developed.

Task No.	Title
B-10	Behavior of BWR Mark III Containment

Applicability	Lead Division
BWRs Utilizing Mark III Containment Design	DSS

Problem Description

This is an ACRS generic concern. Evaluation and approval is required of various aspects of the Mark III containment design which differ from the previously reviewed Mark I and Mark II designs. This task involves the completion of the staff evaluation of the Mark III containment and documentation of the method used to validate the analytical models and assumptions needed to predict the containment pressures in the event of a LOCA.

Task No.	Title
B-11	Subcompartment Standard Problems

Applicability	Lead Division
LWRs	DSS

Problem Description

The calculations of differential pressures that occur in containment subcompartments from a loss-of-coolant event require a complex fluid dynamic analysis to assure that the subcompartment design pressures are not exceeded. To check the various industry computer codes used for the analyses, a standard problem has been issued to the reactor vendors and A/Es so that their models and calculational methods can be evaluated. This task involves the review and evaluation of the subcompartment standard problem analyses supplied by vendors and architect engineers to determine the validity of their models. Changes in the computer codes utilized by the NRC staff could also result from this task.

Task No.	Title
B-12	Containment Cooling Requirements (Non-LOCA)

Applicability	Lead Division
LWRs	DSS

Problem Description

The rationale for normal and post-accident containment cooling will be reviewed to determine the adequacy of the design requirements imposed on the containment ventilation systems. By reviewing typical designs the staff will develop a basic understanding of the consequences of a loss of normal containment cooling, including the impact, if any, on the operability of safety systems and control systems. Specifically, the task will establish whether or not (1) the normal ventilation system is essential to achieve a safe cold shutdown, (2) a failure in the system could cause an accident, and (3) the system is required to mitigate accidents.

Task No.	Title
B-13	Marviken Test Data Evaluation

Applicability	Lead Division
LWRs	DSS

Problem Description

Test data from the Marviken containment tests have been obtained for the purpose of validating containment pressure codes currently used for pre-forming independent calculations related to licensing reviews. The Marviken data are containment pressure responses from a full-scale blow-down using a pressure suppression type containment. This task will correlate the Marviken data and compare the results with existing computer programs.

Task No.	Title
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA

Applicability	Lead Division
LWRs	DSS

Problem Description

Consistent with the guidelines of Regulatory Guide 1.7, the prevention of local hydrogen concentrations in excess of 4% by volume after a LOCA requires mixing of the containment atmosphere. Preliminary calculations indicate that lower flammability limits will not be exceeded in the drywell of Mark III containments. However, additional evaluation and the development of a generalized calculational model applicable to all plants and conditions is needed.

Task No.	Title
B-15	CONTEMPT Computer Code Maintenance

Applicablity	Lead Division
LWRs	DSS

Problem Description

The CONTEMPT computer code is used by the NRC staff to perform independent containment analyses. This task involves the maintenance and revision of the CONTEMPT code to accommodate new containment designs or new problem areas as they are defined.

Task No.	Title
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

Applicability	Lead Division
LWRs	DSS

Problem Description

This task has been incorporated in Task A-18, "Pipe Rupture Design Criteria."

Task No.	Title
B-17	Criteria for Safety-Related Operator Actions

Applicability

LWRs

Lead Division

DSS

Problem Description

Current plant designs are such that reliance on the operator to take action in response to certain transients is necessary. In addition, some current PWR designs require manual operations to accomplish the switch over from the injection mode to the recirculation mode following a loss-of-coolant accident. The required time for the ECCS realignment operations is dependent on pipe break size and the operation must be accomplished before the inventory in the borated water storage tank is depleted.

This task will involve the development of a time criterion for safety-related operator actions and will include a determination of whether or not automatic ECCS realignment will be required.

Task No.	Title
B-18	Vortex Suppression Requirements for Containment Sumps

Applicability	Lead Division
PWRs	DSS

Problem Description

A number of applicants for operating licenses have been performing sump tests to demonstrate ECCS operability during the recirculation phase following a postulated lossof-coolant accident. These tests have shown that vortex formation is not well behaved, and as a result, the sump designs in some cases were found not to be adequate. This task will develop criteria for sump design with respect to vortex formation and define criteria for sump testing.

Task No.	Title
B-19	Thermal-Hydraulic Stability

Applicability	Lead Division
LWRs	DSS

Problem Description

Demonstrating the thermal-hydraulic stability of a reactor is an essential element in the thermal-hydraulic design. Instabilities can result in fuel failures from premature departure from nucleate boiling or excessive hydraulic loads. This task involves the development of the analytical methods necessary for the staff to perform independent calculations to check vendor analyses of thermal-hydraulic stability.

Task No.	Title
B-20	Standard Problem Analysis

Applicability	Lead Division
LWRs	DSS

Problem Description

Most vendors, in the conduct of internal audits of emergency core cooling performance computer codes, have discovered errors in coding and/or logic which had significant effects on the prediction results of approved models. This task involves the use of standard problems to evaluate the predictive accuracy of these complex computer codes and to detect errors to the extent that the errors affect the results of code predictions.

Task No.	Title
B-21	Core Physics

Applicability	Lead Division
LWRs	DSS

Problem Description

The NRC staff has a variety of technical activities ongoing or planned related to core physics. For the most part these activities are directed at improving the NRC staff's analytical and computer capabilities for performing independent analyses related to such areas as reactor kinetics predicting static core physics charactersitics, core parameters for transient analyses and MCPR and DNBR. This task is to coordinate all staff reactor physics efforts into a single program with clearly defined objectives. Additional generic tasks may be identified as a result of this effort.

Task No.	Title
B-22	LWR Fuel

Applicability	Lead Division
LWRs	DSS

Problem Description

Individual reactor fuel rods sometimes fail* during normal operation, and many rods are expected to fail during severe accidents releasing activity to the surroundings and providing a source for releases from the plant. Failures during some accidents could be severe enough to fragment the cladding and disperse fuel pellets into the coolant, but regulations require that the coolable rod-like geometry must be maintained. Behavioral characteristics, such as rod bowing and densification, also have a strong effect on plant-limiting conditions. Thus, fuel behavior during normal operation and postulated accidents must be predictable in order to set operating limits, to limit activity releases and to insure no more than acceptable degradation of the fuel system. The objective of the reactor fuels task is to assure that such predictions are reliable.

The fuel system includes the fuel assembly structure and all the components of individual fuel and poison rods (burnable poison rods, PWR control rods, and BWR cruciform control blades). Failures in recent years

*A fuel failure is defined as the loss of the first fission product barrier; i.e., the loss of hermeticity of the cladding.

have been by cladding hydriding, fuel densification, and pellet/cladding interaction (PCI). Channel box wear has been a problem in BWRs, for example, and in 1976 nearly half the burnable poison rods in a new PWR failed as the result of a design inadequacy. These failures during normal operation are highly undesirable and they clearly represent modes of failure that must be taken into account in accident analyses.

To improve the predictability of fuel performance, a number of problems must be solved and certain technical studies must be continued. Individual activities planned are listed in Table 1 and fall in 3 general classifications.

1. Evaluation of Design Criteria

   The ground rules for fuel design, such as cladding failure limits and rod pressure criteria, must be changed from time to time as earlier criteria are shown to be inadequate (either non-conservative or excessively conservative). In addition, manufacturers find that earlier "criteria of convenience" can no longer be met with new fuel designs, and they formally request revisions -- 3 such cases are currently under review. There has been little change in fuel design criteria in the last 5 years and recent operating experience and research results indicate the need for improvement.

2. Evaluation of Performance Predictions

With the fuel densification crisis in 1972 came the realization that simplistic predictions of fuel performance (temperatures, stored energy, gas pressures, etc.) were inadequate. All vendors now have generic fuel performance codes that are used in licensing. Since these codes are complicated and have strong effects on plant-limiting conditions (e.g., peak clad temperature in LOCA), they must be reviewed, audited, and periodically checked against new results of confirmatory research. Such research does not always show that the models are conservative (e.g., fission gas release at high bunrups); therefore, problems arise that invalidate previously approved codes. These activities are directed at solving identified problems.

3. Evaluation of Operating Experience

Fuel performance predictions must be compared with actual experience of fuels in commercial reactors as a final test of the safety analysis procedures. This not only includes special surveillance for new fuel designs but also includes a general sampling of routine operations. Advances, such as failure detection methods, which could enhance reactor safety, are also investigated in this category.

Table 1

FUELS PROGRAM ACTIVITIES

Evaluation of Design Criteria

Pellet/Cladding Interactions (PCI)

Fuel Behavior During Design-Basis Accidents

Fuel Cladding Design Limits

Evaluation of Performance Predictions

Fuel Rod Bowing

Fuel Rod Performance Codes

ECCS Materials Behavior

Radioactive Fission Gas Release

Behavior of Waterlogged Fuel

Fission Gas Release

Evaluation of Operating Experience Fuel Operational

Experience

Surveillance of New Fuel Assembly Designs

Fuel Rod Failure Detection

Task No.	Title
B-23	LMFBR Fuel

Applicability	Lead Division
LMFBRs	DSS

Problem Description

This task covers a spectrum of technical efforts related to the staff review of LMFBR fuel designs. The efforts include (1) evaluating licensing requirements for the behavior of stainless steel cladding and hexcans in an LMFBR and establishing fuel damage criteria, (2) developing and maintaining a flexible and well verified thermal performance code (GAPCON-FAST) for audit calculations, testing of separate effects models, and parametric studies, and (3) evaluating the effect of cladding attack by fission products on the mechanical properties and performance of stainless steel cladding.

Task No	Title
B-24	Seismic Qualification of Electircal and Mechanical Equipment

Applicability	Lead Division
All Reactor Types	DSS

Problem Description

The seismic qualification of complex electrical and mechanical components has undergone rapid changes in the past several years. Revisions in industry standards (IEEE-Standard 344) have resulted in a need to audit the qualification programs of NSSS vendors and architect engineers to ascertain if components previously qualified using older requirements meet the new requirements. This task will include (1) revising Standard Review Plan Sections 3.9.2 and 3.10 to combine pertinent material, (2) continuing the generic audit of NSSS vendors and architect engineers and (3) compiling a listing of all equipment considered to be adequately qualified and identifying the level for which the qualification is acceptable.

Task No.	Title
B-25	Piping Benchmark Problems

Applicability	Lead Division
All Reactor Types	DSS

Problem Description

Applicants are required to provide confirmation of the adequacy of computer programs used in the structural analysis and design of piping systems and components. This presently consists of applicants providing and the staff reviewing, brief descriptions of the computer programs used and solutions to simple textbook problems. In order to better provide assurance of the reliability of these programs, this task will consist of the staff developing benchmark problems and solutions to these problems for use in the review of applications for construction permits. The case-by-case review will then consist of requesting that the applicant submit solutions to the problems and comparing the applicant supplied solutions to the staff solutions.

Task No.	Title
B-26	Structural Integrity of Containment Penetrations

Applicablity	Lead Division
All Reactor Types	DSS

Problem Description

Containment penetration assemblies provide a means to maintain the integrity of the containment pressure boundary and prevent overstressing of the penetration nozzle due to thermal stresses. A typical penetration assembly may consist of a flued head, a guard pipe, an expansion bellows and an impingement ring. The flued head may be fabricated from a forging which may be welded into the process line or may be welded to the outer surface of the process piping. This task involves an evaluation to assess the adequacy of specific containment penetration designs from the point of view of structural integrity and inservice inspection requirements.

Specifically, the task involves two areas. The first is an independently performed stress analysis of the various penetrations produced as an integral fitting and welded into the process line, or penetrations which are welded to the outside circumference of the process line. The model will consider the applicable requirements of Section III, Subsections NC and NE of the ASME Code, NRC stress criteria, any existing fabrication residual stresses, and the mechanical loadings resulting from normal plant operation, from postulated pipe breaks, and from seismic events. The second area involves a determination that the configuration and accessibility of the welds in the proposed design and the procedures proposed for performing volumetric examination will permit the inservice examination requirements of Section XI of the ASME Code to be met.

A computer code will be developed that will enable the staff to perform reviews of specific containment penetration designs proposed by vendors for the purpose of developing staff conclusions for application in case-by-case reviews.

Task No.	Title
B-27	Implementation and Use of Subsection NF

Applicability	Lead Division
All Reactor Types	DSS

Problem Description

Since the adoption by the ASME Code, Section III, of Subsection NF on component supports, technical review has been limited to conformance of the information provided in the application and commitment by the applicants to component support design in accordance with the provisions in NF.

Certain deficiencies in the use of NF, however, have been identified primarily by NRC Code Committee members on the Working Group on Component Supports and its Task Forces. Examples of these deficiencies are:

(1) The absence of definitive criteria to be used in defining the jurisdictional boundary between a load carrying building structure designed by AISC rules which do not contain inservice inspection requirements and an attached NF component support having NF inservice inspection requirements.

(2) As the design limits for Class 1 liner type component supports presently appear in the Code, the allowable stresses exceed those

permitted for other Code designed components. If these limits are approached repeatedly in the component support, the support could fail by fatigue.

It is anticipated that some of the identified deficiencies will soon be addressed and corrected by revisions to the Code. This task will develop a Branch Technical Position that will assess the remaining deficiencies for use by the staff in case reviews of component supports.

Task No.	Title
B-28	Radionuclide/Sediment Transport Program

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

As a result of Appendix I and the Liquid Pathway Generic Study, the staff has had to take a more realistic look at the effects of sediment (surface waters) and aquifer materials (groundwater) on radionuclide transport through the hydrosphere. To accomplish this objective, it is necessary that the staff have available for its use radionuclide/sediment transport models that have been field verified. This task will accomplish this objective through model development and verification in the five following areas:

1. Radionuclide/Sediment Transport Model (Non-Tidal Rivers)
2. Radionuclide/Sediment Transport Model (Tidal Rivers, Estuaries, Oceans, and Large Lakes)
3. Unified Transport Model Utilization
4. Development of Design Curves for Parameters Needed in Models of Radionuclide Transport in Groundwater Systems
5. Three-Dimensional Multiaquifer Radionuclide Transport Model

Task No.	Title
B-29	Effectiveness of Ultimate Heat Sinks

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

This task involves the following:

(1) Confirmation of currently used mathematical models for prediction of Ultimate Heat Sink performance by comparing model performance with field data.

(2) Better guidance regarding the criteria for weather record selection to define Ultimate Heat Sink design basis meteorology. At the present time, the guidance for selection of the weather records for highest temperature and highest evaporation of the heat dissipator is only vaguely defined. Undoubtably, these records will be defined differently for cooling ponds, spray ponds, and cooling towers. The analysis carried out as part of this task will be based on the present generation of NRC developed UHS models, combined with long weather records from National Weather Service files and it will be supplemented by additional field performance data as it is gathered.

Task No.	Title
B-30	Design Basis Floods and Probability

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

This task involves the preparation of a paper detailing the bases for design basis flood events utilized by the staff in case reviews, including Probable Maximum Floods, Hurricanes, Tsunamis, Seiches, Seismically Included Dam Failures, and Combinations of Lesser Events. Additionally, descriptions of probability estimates, including potential errors, will be prepared for the principal flood producing events. This material is being prepared to respond to a request of the ACRS to provide them with a better understanding of the staff's approach to design basis floods.

Task No.	Title
B-31	Dam Failure Model

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

During licensing reviews, the need has arisen on several occasions to have a staff model to predict the failure discharge hydrograph due to erosional failures of earthen dams. No known model presently exists for such evaluations and, accordingly, the staff and the applicants have been forced to conservatively postulate complete and instantaneous failure of the dam.

This task will develop an analytical model, or nomograph, to predict erosion rates and patterns of failure for an earthen embankment for a given initiating mode (e.g., overtopping, cracking). The model will be validated using existing failure data. Since it is recognized that there is a lack of high quality data, the model may require features to coservatively increase or decrease the initiating conditions to indicate sensitivity of the structure to varying initiators. Using the rate, pattern, and initial pool conditions, failure discharge hydrographs can then be estimated for use in case reviews.

Task No.	Title
B-32	Ice Effects on Safety-Related Water Supplies

Applicability	Lead Division
Reactors in Northern U.S.	DSE

Problem Description

Additional information is needed by the staff concerning the potential effect of extreme cold weather and ice buildup on the reliability of various plant water supplies. Of particular concern are phenomenon that could impact the proper operation of safety-related systems (e.g., the Ultimate Heat Sink) and thereby impair the operator's ability to shutdown the plant and provide adequate core cooling.

Experience gained during the severe weather conditions that existed during the past winter indicated that a more thorough understanding of the potential effects of severe ice conditions is necessary to ensure that design and operation of safety-related water supplies will ensure adequate operation of the systems.

Task No.	Title
B-33	Dose Assessment Methodology

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

This task involves the maintenance and improvement of calculational capabilities for assessing doses to individuals from radiation and radioactive effluents from normal operation and from radioactive releases from postulated accidents.

Task No.	Title
B-34	Occupational Radiation Exposure Reduction

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

Compilation of occupational radiation exposure reports from operating reactors has shown that exposures to station and contractor personnel has generally been increasing over the past seven years for both PWR's and BWR's. The overriding problem at LWR's at present with respect to occupational exposures is system and equipment maintenance. The exposures connected with this function are higher than expected due to unexpetedly high activation and fission product depositions, lower than expected equipment reliability, unanticipated failures in steam generators (PWRs) and piping (BWRs) and the continuing necessity to perform contact maintenance in most cases. The violation of 10 CFR Part 20 limits has in general been avoided by licensees during high exposure tasks by obtaining the temporary services of transient workers, both skilled and unskilled. In some cases this has been done instead of spending sufficient effort on reducing exposures. If this practice continues, shortages in skilled tradesmen are expected as the nuclear industry grows. However the most unfortunate consequence of this approach, i.e., using large numbers of people rather than better designs (as the case of earlier facilities) and operating procedures, is the unnecessary exposure and risk that could have been minimized.

This task will provide an improved basis for the staff to review reactor plant designs and projected operations to assure that occupational radiation exposure is maintained as low as is reasonably achievable. The following areas will be studied under this task:

(1) Activated corrosion product reduction at LWRs.

(2) Reducing exposure from maintenance and inspection of reactor plants.

(3) Shielding source terms.

(4) Compilation of exposure data.

(5) Development of inplant radiation dose rates.

(6) Evaluation of neutron streaming at PWRs.

Task No	Title
B-35	Confirmation of Appendix I Models for "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors"

Applicability	Lead Division
LWRs	DSE

Problem Description

This task involves evaluating information from semiannual operating reports, inplant measurements programs and topical reports and to revise models for calculating releases of radioactive materials in effluents from PWRs and BWRs. This task is expected to improve the accuracy and realism of current staff models by using the best available data to develop model revisions.

Task No.	Title
B-36	Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

This task involves developing revisions to current guidance and staff technical positions regarding ESF and normal ventilation system air filtration and adsorption units. This involves developing revisions to Regulatory Guide 1.52 and Branch Technical Position ETSB 11.2 to address technical advances that have shown that some current positions are unjustifiably conservative, some are unnecessary and in some cases additional positions are necessary.

Task No.	Title
B-37	Chemical Discharges to Receiving Waters

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

In accordance with our licensing responsibilities under the National Environmental Policy Act (NEPA), we assess the impact of discharges of chemicals to surface waters. The objective of this assessment is to afford a weighing of impacts of the proposed action and a comparison of alternative actions rather than to provide absolute protection to surface waters. In accordance with Section 511 of the FWPCA, effluent limitations are set by EPA and not by NRC under NEPA. However, in accordance with the Second Memorandum of Understanding between EPA and NRC, NRC has the lead in the conduct of the environmental impact assessment for nuclear stations and must assess the limits placed on discharges under the FWPCA. NRC must continue to determine and consider the impact of discharges in its licensing decisions.

The NRC assessment typically considers ambient water quality and addresses the needs of specifically identified users of the impacted waterbody. The assessments usually have not resulted in a quantitative determination of impact but have rather involved a subjective determination of acceptability based on a comparison of projected water quality to published criteria for protection of water users. Such criteria are not always formulated to assure the absence of impact and there are substances discharged for which criteria do not exist. Furthermore, the absolute determination of acceptability does not afford a quantitative comparison of alternative actions. This task will provide additional insight into impact of chemical discharges and provide procedures for quantifying the magnitude of any such impacts. This improvement in NRC procedures for impact assessment will provide a clearer division between NRC responsibilities under NEPA and EPA responsibilities under the FWPCA.

There are three specific water quality effects which have been questioned more frequently recently and which will be studied initially. These are:

1. Environmental significance of condenser tube copper in cooling water discharges.
2. Impact of increased total dissolved solids in receiving waters.
3. Significance of chlorinated organic compounds produced during condenser chlorination.

Task No.	Title
B-38	Reconnaissance Level Investigations

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

NRC environmental information needs for licensing fall into the categories of: (a) detailed site-specific investigations at a preferred site, and (b) reconnaissance level information to support alternative site assessment and selection, including early site review and to identify monitoring program needs. This proposed technical activity would generate a technical report which would form the basis of a staff position paper providing guidance to applicants concerning the need, applicability, proper utilization, scope, and content of an adequate reconnaissance level investigation. This guidance is needed because of the requirement for an applicant to demonstrate how environmental considerations were factored into the alternative site selection process, the emerging importance of early site reviews and the efficiencies inherent in standardizing procedures used during the site selection process.

Task No.	Title
B-39	Transmission Lines

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

This task involves two activities:

(1) The United States Department of Interior (DOI) is preparing a handbook which will set forth practices for siting and managing transmission line corridors for the betterment of wildlife. This project will be national in scope and will involve coordination with both Federal and private agencies and organizations. NRC has been requested to participate in an interagency committee to oversee the preparation of this manual. This manual will probably form the basis for many comments on future NRC draft environmental statements which discuss impacts of electrical transmission lines as part of our NEPA responsibilities. It will contain sufficient technical detail to allow formation and implementation of specific management plans for every geographic area in the United States. It will most certainly affect the route selection and management practices for transmission system rights-of-way throughout the nation. This manual will not be repetitious of any existing Federal guidelines for routing or managing rights-of-way. It will, therefore, provide a

variety of specific routing and managing actions not presently available to NRC environmental reviewers.

(2) In addition, NRC in an effort to work together with other Federal agencies to streamline NEPA reviews has been working with the Rural Electirfication Administration (REA) to develop a single environmental review process involving all transmission systems of joint concern. This process will result in reduction of redundant environmental impact reviews.

Task No.	Title
B-40	Effects of Power Plant Entrainment on Plankton

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

The effects of entrainment on phyto- and zooplankton populations are often minimal and occasionally beneficial. Numerous studies of the effects of entrainment on plankton organisms, phytoplankton and zooplankton, have shown impacts to be minimal and/or not significant. Studies have also shown that even when entrainment mortality is high, the overall impacts may be minimal due to the fast reproductive and recovery time for many species - a few hours for some phytoplankters to several days for zooplankton.

It has not been uncommon for utilities to have undertaken exhaustive and sometimes unnecessary preoperational and operational environmental monitoring programs. In view of the above points, it may be possible to reduce or eliminate studies of certain planktonic elements, perhaps on a site or regional basis. A study of these matters will form the basis for a staff position on monitoring requirements of plankton and entrainment programs. If the state-of-the-art as defined in the study is adequate, perhaps intensive studies can be reduced, saving time and expense for both utilities and the NRC staff.

Task No.	Title
B-41	Impacts on Fisheries

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

This task involves studies related to the impacts of power plant operation on fishery resources. Possible studies to be undertaken are described below: (1) In recent years an attempt has been made at a number of facilities to evaluate the effects of plant operation on fishery resources using advanced modelling techniques. This approach has resulted in the expenditure of a considerable amount of money by utilities to collect quantitative data on populations of individual species. We have developed methodologies that provide for fairly good estimates on impingement and entrainment losses; however, the modelling efforts have not been successful in assessing the significance of plant operation on the fish populations in surrounding waterbodies. As an alternative to advanced modelling, this study would evaluate the kinds of data that could be collected in routine far field monitoring programs on aspects of fish populations other than numerical estimates of population size and attempt to relate these characteristics to the losses sustained by impingement and entrainment. Such characteristics as size and age distribution, growth rates, condition factor, incidence of abnormalities and fish parasites, gut analysis, as well as other standard techniques used in applied fishery management would be examined. This could provide an effective alternative to the anaytical modelling techniques now employed with doubtful reliability.

(2) Observations of large impingement losses of threadfin shad (Dorosoma petenense) at nuclear plants are associated with periods of low water temperature which puts the threadfin shad in a stressed or perhaps dying condition. It has been espoused that the power plants are harvesting threadfin shad which would have otherwise succumbed to natural mortality and thus is not adversely impact the population. However, quantitative evidence in support of this supposition is lacking and the potential impacts to the reservoir fish community due to removal of viable but debilitated forage have not been determined to a degree of confidence such that the loss can be declared insignificant. The information available is inconsistent and does not reconcile conflicting data and/or expert opinions concerning the significance of threadfin shad losses. To supplement available information, a combination of laboratory and field investigations is required.

Extension impingement and surveillance monitoring is being required at many of these plants due chiefly to the high levels of threadfin shad impingement. If it can be confirmed that the impacts are not significant to the shad population, then the ETS requirements for long-term studies of secondary effects on the sport and commercial species due to impingement could be lessened or completely dropped. Concomitant with this action would be a reduction in cost of the licensee's monitoring program.

An ultimate goal is to use the study results as a basis for a staff position paper which can be referenced in future case work. The format of the final report should be compatible with this potential use of the study results. Although we cannot estimate the number of new cases wherein threadfin shad impingement may be of concern, the projection of power need for the TVA system alone suggests numerous possiblities. Thus, a reduction of staff effort in the environmental impact assessment for new cases could be achieved via a generic staff position on the matter.

(3) Sources of entrainment mortality during passage through condensor

A) Field Studies. During assessment of significance of environmental impact of condensor cooling systems on entrained organisms, it is customary to assume 100% mortality because of our lack of knowledge of actual effects. This conservative approach may overestimate the predicted impact of facility operation. Design modifications, such as addition of cooling towers, may be required to mitigate predicted impacts. In some cases, the effects of entrainment may be predicted to be so significant as to compromise the suitability of the site. Certain facilities, such as Brunswick and Indian Point, are contesting the need for cooling system modifications on the basis that the impacts of entrainment will not be as great as predicted in the EIS. This study would provide needed information concerning the mortality of certain selected organisms during passage through a simulated condensor system under controlled conditions and thus provide improved validity to EIS predictions. Further, it will identify and quantify the sources of mortality as to biocide, changes in hydrostatic pressure during passage, temperature shocks, and/or mechanical damage.

B) Laboratory Studies. This study would differ from the field study in that it would be performed at a facility still under construction by the Oak Ridge National Laboratory. The laboratory facility would be larger than the New York one and would not be portable. It would have the additional capability of identifying and quantifying the effects of pump passage in addition to the parameters identified earlier.

(4) The potential for entrainment and impingement is a continuing concern with conventional shoreline and approach canal cooling system intakes. Submerged pipe intakes, particularly of the perforated pipe design, suggest an attractive alternative for minimizing these potential ecological impacts; however, insufficient knowledge is presently available on whether there is generic applicability of such a system. This activity would review the literature and operating experience at facilities equipped with submerged pipe intake systems. Some degree of conceptual evaluation may also be required due to a paucity of data for large capacity systems of this type. Desired results would include determination of generic applicability to varying habitats and of trade-offs in potential impacts from one to another trophic level, species, or lifestage.

(5) The process of assessing and predicting potential impacts on aquatic systems from the construction and operation of nuclear generating stations places considerable emphasis on fish populations of the primary water source used for plant cooling. When an analysis reveals a potentially significant adverse impact on fish populations the NRC staff must address measures that

can mitigate this adverse impact. Two measures which have potential for mitigating adverse impacts on fish populations are replacement using a fish hatchery and habitat restoration to increase natural fish reproduction. Mitigation of an adverse impact, however, is not at this time considered a feasible alternative by the EPA. An ecological evaluation of these two mitigation measures will provide a basis for a staff position and a useful reference document for future NRC impact statements.

Task No.	Title
B-42	Socioeconomic Environmental Impacts

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

As part of the cost-benefit analysis of nuclear power plant licensing applications, the NRC is required to assess likely socioeconomic impacts of power plant construction and operation on local communities and the surrounding region. This task encompasses several studies to improve the staff's ability to forecast socioeconomic impacts for preparation of Environmental Statements and hearing testimony. Areas to be studied include:

(1) Nuclear Power Station Construction Labor Force Mobility and Residential Choices.

(2) Visual Change within a Region Due to Alternative Closed Cycle Cooling Systems and Associated Socioeconomic Impacts.

(3) Impacts of Coastal and Offshore Nuclear Generating Stations on Recreational and Tourist Behavior at Adjacent Coastal Sites.

Task No.	Title
B-43	Value of Aerial Photographs for Site Evaluation

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

The technique of aerial photography has a long established and proven utility for earth resource inventory and evaluation. Applicants for nuclear construction permits are becoming aware of this and are making increasing use of aerial photographs in their environmental reports. The uncertainties with the methodology at present relate to: (1) photo interpretation techniques and the extent to which existing regulatory guidance can be met using this method; (2) fine tuning of the interplay between aerial photography and ground truthing needed to meet licensing requirements; (3) quantification of presumed cost advantages of this method, and (4) relative information return from different films, photographic scales and seasons of coverage. This task will examine existing regulatory guidance and produce a list of items which might be fulfilled in whole or in part from aerial photographic information. Field tests on actual sites will be carried out to determine the information return from photographs in relation to regulatory requirements and in relation to conventional ground based data collection efforts. The results will give the staff a documentary basis for accepting aerial photographic inventories and resource evaluation in environmental reports and for revising existing guidance for making environmental surveys.

Task No.	Title
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

In the performance of NEPA obligations to evaluate alternatives to the proposed action, the staff must reach a conclusion as to the comparative costs of generating power among the feasible alternatives. While alternatives other than coal are treated in the staff's analysis, coal represents by far the most feasible alternative and requires detailed cost comparisons equivalent to those performed for nuclear. For the several years, the staff has used a computer code known as CONCEPT to obtain forecasts of plant capital costs. This code was developed by ORNL based upon design specifications and cost estimates done by United Engineers and Constructors. This task involves maintaining and developing improvements to the CONCEPT code so that it remains up-to-date for use in projections of power plant capital cost, front-end cost and generating cost forecasts.

Task No.	Title
B-45	Need for Power - Energy Conservation

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

This task is included in Task No. B-2.

Task No.	Title
B-46	Costs of Alternatives in Environmental Design

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

Frequently regulatory changes are made in the applicant's proposal for design and/or operation of systems or subsystems based on perceived needs to mitigate impacts on the environment. Also, differences in design and/or operation are an integral part of the treatment of alternatives in the EIS. The cost of such changes or alternatives, if calculated, are determined on an ad hoc basis. However, this cost is not always calculated, and many times they are not calculated on a consistent basis. A more consistent and comprehensive analysis of the cost of various design and operating modes appears to be warranted so that there is a reasonable and documented rationale for determining such costs. Such costs would also have to include costs of redesign. Once experience is gained in this area, consideration would be given to expanding the study to the cost of making changes because of changing safety criteria, both from a redesign standpoint as well as from a "backfit" point of view.

Task No.	Title
B-47	Inservice Inspection of Supports Class 1, 2, 3 and MC Components

Applicability	Lead Division
LWRs	DOR

Problem Description

Recent results from inspections of various structural components in the torus support systems of operating BWRs have indicated several inconsistencies between the design drawings and the "as built" hardware, including missing support struts, out of tolerance weld dimensions, unwelded regions and unsupported columns. In addition, a limited number of separate inspections have been performed on PWR steam generator supports. The results of these inspections revealed several cracked support bolts.

In view of the above, additional investigation of BWR and PWR component support systems should be undertaken to determine if similar deficiencies and "off design" conditions exist generally in operating plants. This investigation should determine the extent of support system deficiencies, and whether the deficiencies are service induced or are the result of faulty construction. Determination of the extent and nature of the deficiencies is necessary to define the possible safety significance and to provide guidance for further appropriate staff action regarding inservice inspection of supports.

Task No.	Title
B-48	BWR Control Rod Drive Mechanical Failures

Applicability	Lead Division
BWRs	DOR

Problem Description

Surface cracks have been discovered in control rod drive internal parts at some operating BWR plants. This cracking, although only observed to be localized in nature, if propogated could potentially affect the capability of the control rod drive to perform its design function. Cracking has been identified in the following control rod drive internal parts: the collet retaining tube (CRT), the poison tube, and the index tube.

The cracking in the collet retaining tubes was located near a transition in tube thickness approximately six inches down the tube. The cracks appear to have initiated on the outer tube surface and to have propogated circumferentially in the areas between the flow holes of the tube. The cracking in the index and piston tubes have occurred in the creviced, sensitized and non-nitrided threaded areas of these tubes.

The extent of the cracking observed at operating BWR plants to date has not been severe enough to impair the ability of the control rod drives to operate in their required capacity. As an interim precautionary measure, additional technical specification requirements were established for operating reactors which require shutdown within 48 hours of detecting a rod unless investigation demonstrates that the cause of the failure was not due to a separated collet tube. Another technical specification that has long been in effect for operating reactors requires exercising fully or partially withdrawn rods once every week as a part of the corrosion prevention program. This requirement also provides demonstration of CRT integrity.

Further investigation of the cracking phenomenon and evaluation of the proposed CRD modifications to preclude such cracking in the future are necessary.

Task No.	Title
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments

Applicability	Lead Division
LWRs	DOR

Problem Description

General Design Criterion 53, "Provisions for Containment Testing and Inspection," requires in part that the reactor containment be designed to permit: (1) periodic inspection of all important areas, and (2) an appropriate surveillance program. 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," requires a general inspection of the surfaces of the containment prior to any Type A test to uncover any evidence of structural deterioration.

Containment designs typically utilize any one of the following structural materials: steel, steel lined reinforced concrete, steel lined prestressed concrete. To date the only detailed criteria that have been developed for inservice inspection of containments relate to tendon surveillance for pre-stressed concrete containments. These criteria are contained in Regulatory Guides 1.35 and 1.90 which address ungrouted and grouted tendons respectively. These Regulatory Guides deal primarily with the pre-stressing hardware; no detailed inservice inspection criteria exist for the steel liner or other portions of the containment. Similarly, there are no criteria for inservice inspection of steel containments or steel lined reinforced concrete containments. In view of this, detailed and comprehensive criteria need to be developed for performing inservice inspections of all types of containments.

In addition, the long term corrosion problems of reinforcements and of the steel liner in contact with concrete in concrete containments, or the corrosion of the steel surface in contact with the water in BWR suppression chambers, have yet to be adequately analyzed. Long term studies of these corrosion phenomena need to be undertaken to develop criteria and requirements to prevent corrosion in all types of containments.

Task No.	Title
B-50	Post-Operating Basis Earthquake Inspection

Applicability	Lead Division
All Reactor Types	DOR

Problem Description

Section V(a)(2) of Appendix A to 10 CFR Part 100 states that licensees will be required to shut down their plants in the event of an earthquake if vibratory ground motion exceeds that of the OBE. Prior to restart the licensee must demonstrate to the NRC that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public. In order to determine the capability of a plant to resume operation following an OBE an adequate inspection of the plant and site area must be performed. The requirements for this post-OBE inspection are stated in Section 3.7.4.II.4 of the Standard Review Plan. Since neither the Regulations nor the Standard Review Plan provide details on the extent of such inspections, this task will develop an acceptable inspection procedure.

Task No.	Title
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components

Applicability	Lead Division
All Reactor Types	DOR

Problem Description

In the design of nuclear power plants, inelastic response of equipment and components due to severe transients from low probability events is permitted in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NA, Appendix F. Local inelastic response is also permitted for structures under severe impact loads due to low probability events.

Assessment of inelastic analysis techniques applicable to equipment and components is the basic objective of this task. Inelastic analysis techniques for structures are under study as a part of NRR Category A Task No. A-40, "Seismic Design Criteria - Short Term Program."

Since inelastic analysis procedures acceptable under Appendix F of the ASME Boiler and Pressure Vessel Code permit Level D Service Limits and since Level D Service Limits allow large inelastic strains, it is particularly important that properly qualified analysis techniques are used, and that their limitations are properly understood.

Task No.	Title
B-52	Fuel Assembly Seismic and LOCA Responses

Applicability	Lead Division
LWRs	DSS

Problem Description

The fuel assembly is a highly nonlinear structure which can be subjected to substantial loadings during seismic excitations and LOCA transients. The integrity of this assembly is critical for plant safety. Extensive work has been completed by NSSS vendors for the determination of fuel element response during the sustained vibrations of normal operation. However, there is a need to study the behavior of these assemblies during accident conditions. Future work will include a determination as to whether or not existing geometry can adequately satisfy the functional design criteria. Vibration interaction between components of the assembly and reactor pressure vessel will be investigated.

Subsequent to its approval as a Category B task, this task has been included within the scope of Task A-2, Asymmetric Blowdown Loads on PWR Primary Coolant Systems.

Task No.	Title
B-53	Load Break Switch

Applicability	Lead Division
All Reactor Types	DSS

Problem Description

Plant designs which utilize generator load circuit breakers to satisfy the requirement for an immediate access circuit stated in GDC-17 must be prototype tested to demonstrate functional capability.

This task involves the preparation of a staff position to clarify and document the prototype testing requirements for generator load circuit breakers and associated circuitry used to provide an immediate access circuit in accordance with GDC-17. This technical position, when completed, will be incorporated in the Standard Review Plan.

Task No.	Title
B-54	Ice Cond nser Containments

Applicability	Lead Division
Westinghouse PWRs That Use the Ice Condenser Containment Design	DSS

Problem Description

This task involves two staff efforts associated with the ice condenser containment concept.

(1) Verification of the established design margin for ice condenser containments to the satisfaction of the ACRS, and

(2) Reviewing the surveillance programs for ice inventory and functional performance testing at operating facilities to determine whether the surveillance frequencies should be increased or other action should be taken. Operating experience from the D.C. Cook plant has indicated that sublimation and ice melting are causing a loss of ice inventory and related functional performance problems.

Task No.	Title
B-55	Improved Reliability of Target-Rock Safety-Relief Valves

Applicability	Lead Division
BWRs	DOR

Problem Description

Target rock power operated relief valves are used to limit the pressure rise in boiling water reactor (BWR) transients and to automatically depressurize the primary system in the event of certain loss-of-coolant accidents. In the past, a significant number of Target Rock power operated relief valves installed in BWRs inadvertently opened and subsequently failed to reseat until the primary system was essentially depressurized. The primary cause of valve maloperation was due to erosion of the setpoint pilot valve seat, resulting in the leakage of steam into the second and third stages of the valve actuator. There is a strong correlation between pilot leakage and the amount of "simmer margin" (SRV setpoint pressure minus system operating pressure) for the valve. Virtually all of the SRV blowdowns occurred on valves with less than 100 psi "simmer margin." For the short term, General Electric (GE) proposed that licensees (1) grind out the throat diameters of Target Rock SRVs which will allow the valve setpoint to be raised 35 psi and (2) montior SRV tailpipe temperatures (from existing thermocouples) to detect valve leakage prior to an inadvertent actuation. For the long-term, GE (in conjunction with Target Rock) has proposed a redesigned valve actuator for Target Rock SRVs. The most significant change would be the elimination of the second stage so that any leakage past the pilot would exhaust rather than cause a pressure buildup to the actuating pressure in the second stage. GE and Target Rock have manufactured four prototype valves with this modification and have begun a testing program. This task involves monitoring the current programs and developing generic positions for use in the review of individual plants.

Task No.	Title
B-56	Diesel Reliability

Applicability	Lead Division
All Reactor Types	DOR

Problem Description

An examination of Licensee Event Reports on the experience with diesel generators (1969-1975) indicates that the emergency onsite diesel generators at operating plants have an average reliability of about 0.94 compared with the NRC's reliability goal of 0.99. The reliability of the diesel generator is strongly dependent on the interaction of the following factors: design, testing and operational requirements, operational history, inspections, maintenance, and the personnel qualifications of operators.

The lack of detail regarding the failures reported in the Licensee Event Reports make it difficult for the NRC to establish the causes of the reported failures. A comprehensive review to determine the underlying and recurring causes of the reported failures is necessary in order to enable the NRC to establish improved guidance and requirements to increase the reliability of the emergency onsite diesel generators.

Task No.	Title
B-57	Station Blackout

Applicability	Lead Division
All Reactor Types	DSS

Problem Description

This task will make a determination as to whether plants should be designed to accommodate a total loss of all ac power for a limited period of time.

Task No.	Title
B-58	Passive Mechanical Failures

Applicability	Lead Division
All Reactor Types	DSS

Problem Description

This task involves a review of valve failure data in a more systematic manner to confirm the staff's present judgment regarding the likelihood of passive mechanical valve failures, categorize these and other valve failures as to expected frequency, specify acceptance criteria and determine if and how the results of this effort should be applied in licensing reviews.

Task No.	Title
B-59	N-1 Loop Operation in BWRs and PWRs

Applicability	Lead Division
LWRs	DOR

Problem Description

The majority of the presently operating BWRs and PWRs are designed to operate with less than full reactor coolant flow. If a PWR reactor coolant pump or a BWR recirculation pump becomes inoperative, the flow provided by the remaining (N-1) loops is sufficient for steady state operation at a power level less than full power. Although the FSARs for the licensed BWRs and PWRs present (N-1) loop calculations showing allowable power and protective system trip setpoints, the staff has disallowed this mode of operation for most plants primarily due to insufficient ECCS analyses. Some Babcock & Wilcox (B&W) PWRs are authorized for long term operation with one reactor coolant pump out of service since they have submitted and the staff has approved the necessary ECCS, steady state, and transient calculations. The remaining BWR and PWR licensees have Technical Specifications which require shutdown within a fairly short time if one of the reactor coolant loops becomes inoperable.

A number of BWR and PWR licensees have recently requested authorization to operate with one of the coolant loops out of service. To grant authorization for this mode of operation, the staff should ensure the licensee has adequately evaluated the effects of partial flow on the plant steady state, transient and accident response.

The purpose of this task action plan is to develop a set of acceptance criteria and review guidelines for the (N-1) loop authorization requests. A report will be prepared summarizing the staff's criteria for each NSSS design.

Task No.	Title
B-60	Loose Parts Monitoring Systems

Applicability	Lead Division
LWRs	DSS

Problem Description

The presence of a loose (i.e., disengaged and/or drifting) object in the primary coolant system can be indicative of degraded reactor safety resulting from failure or weakening of a safety-related component. A loose part, whether it be from a failed or weakened component or from an item inadvertently left in the primary system during construction, refueling, or maintenance procedures, can contribute to component damage and material wear by frequent impacting with other parts in the system. A loose part can pose a serious threat of partial flow blockage with attendant departure from nucleate boiling (DNB) which in turn could result in failure of fuel cladding. In addition, a loose part increases the potential for control-rod jamming and for accumulation of increased levels of radioactive crud in the primary system.

The primary purpose of the loose - part detection program is the early detection of loose metallic parts in the primary system. Early detection can provide the time required to avoid or mitigate safety-related damage to or malfunction of primary system components.

Applicants for construction permits and operating licenses are required to commit to a loose-part detection program. The NRC has developed hardware criteria and programmatic (operational) criteria for loose-parts detection programs. These criteria are contained in a proposed NRC Regulatory Guide which has been distributed for industry comments.

Loose parts detection programs are also in effect at most operating facilities. While these programs do not fully comply with the criteria contained in the proposed Regulatory Guide, operating experience with such programs has been generally very good.

The purpose of this task is to resolve any outstanding issues related to the proposed Regulatory Guide, including the development of staff positions and guidance with respect to upgrading loose parts detection systems at operating facilities.

Task No.	Title
B-61	Analytically Derived Allowable ECCS Equipment Outage Periods

Applicability	Lead Division
LWRs	

Problem Description

Surveillance test intervals and allowable equipment outage periods in the Technical Specifications for safety-related systems are largely based on engineering judgement. Analytically based criteria are needed for the staff's use in confirming or modifying these surveillance intervals and allowable equipment outage periods.

Task No.	Title
B-62	Re-Examination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions

Applicability	Lead Division
LWRs	DOR

Problem Description

The methods used to establish safe operating limits for reactor cores were developed about ten years ago. At present, safety margins are reviewed utilizing previous staff judgements based on individual plant reviews. A uniform staff position needs to be developed for application to core performance reviews of new plants and to reloads and core modifications of operating plants.

Task No.	Title
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary

Applicability	Lead Division
LWRs	DOR

Problem Description

There are several systems connected to the reactor coolant pressure boundary that have design pressures that are considerably below the reactor coolant system operating pressure. The NRC staff has required that valves forming the interface between these high and low pressure systems have sufficient redundancy to assure that the low pressure systems are not subjected to pressures which exceed their design limits.

Recently, there has been discussion relative to the adequacy of the isolation of low pressure systems that are connected to the reactor coolant pressure boundary. Past reviews have concentrated on insuring isolation of the residual heat removal system, which is a low pressure system on almost all PWRs and BWRs. Current reviews of license applications for new plants, that is, our CP/OL evaluations, are based on guidelines set forth in the Standard Review Plan (SRP). However, these guidelines were not available during the reviews of the plants which are currently operating.

To assess the isolation capabilities of low pressure systems, this task involves the review of a representative operating plant for each NSSS vendor. Each low pressure system connected to the reactor coolant pressure boundary and penetrating the containment will be examined.

Task No.	Title
B-64	Decommissioning of Reactors

Applicability	Lead Division
All Reactor Types	DOR

Problem Description

The Code of Federal Regulations 10 CFR 50.82 provides for licensees to terminate their licenses. The Commission may require information from the licensee to demonstrate that the methods and procedures to be used for decontamination and for disposal of radioactive materials provide reasonable assurance that the dismantling and disposal will not be inimical to the common defense and security or to the health and safety of the public. 10 CFR 50.33(f) includes the requirement that operating license applicants show that they possess or have reasonable assurance of obtaining funds necessary to cover the "estimated costs of permanently shutting the facility down and maintaining it in a safe condition."

Since 1960 about 50 research-type reactor facilities and 15 small power and test reactors have been decommissioned in accordance with the above regulations. In addition, the NRC reviews the general plans for decommissioning and financial arrangements as a part of their review of operating license applications. Based on acceptable findings, including this area, the NRC has issued operating licenses to the utilities. As a result of the need for increased guidance to the industry in this area, the staff published in June, 1974 a Regulatory Guide (1.86) on the "Termination of Operating Licenses for Nuclear Reactors." This guide includes methods and procedures considered acceptable by the staff for the termination of operating reactors.

Because of the increasing number of reactors being licensed to operate and the increased interest in decommissioning, additional guidance is needed on this topic. In addition, several utilities are planning the replacement of major pieces of equipment. These efforts can be considered as "mini-decommissionings." Therefore, the safety and environmental acceptance criteria need to be defined as does the expected costs and funding alternatives for equipment and total facility decommissioning.

Task No.	Title
B-65	Iodine Spiking

Aplicability	Lead Division
LWRs	DSE

Problem Description

The calculated radiological consequences for some postulated design basis accidents are highly dependent on the magnitude of the iodine spike postulated to occur following the transient. These calculations in turn determine the coolant activity limits allowed in the technical specifications. This task will develop and confirm a model for the iodine spiking phenomena. Procurement of data from operating plants and the development of a fuel release model for predicting the magnitude of the spikes will provide an understanding of this phenomenon which is not presently available. Improved knowledge of this topic will allow setting of the coolant activity limits at realistic levels. In addition, this could provide the basis for more realistic accident calculations.

Task No.	Title
B-66	Control Room Infiltration Measurements

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

A key parameter affecting control room habitability under the conditions described in General Design Criteria 19 and Standard Review Plan 6.4 is the magnitude of control room air infiltration rates. Current estimates of these rates are based on data relating to buildings that are substantially different than typical nuclear power plant control room buildings. Additional experimentally-measured air exchange rates of operating reactor control rooms are needed to develop an improved data base. This task will support the following requirements:

" Criterion 19 of Appendix A to 10 CFR Part 50 (General Design Criteria for Nuclear Power Plants) requires that control rooms be accessable and habitable under both normal and accident conditions.

" Safety Standard Review Plan 6.4 Habitability Systems and SRP 9.4.1 Control Room Area Ventilation Systems specify a review of the control room habitability systems with respect to the functional performance required to maintain a habitable control room area in the event of postulated accidents.

Task No.	Title
B-67	Effluent and Process Monitoring Instrumentation

Applicability	Lead Division
LWRs	DSE

Problem Description

Monitoring of radioactivity in gaseous and liquid effluent streams from nuclear power plants is required for several purposes: (a) assessment of the adequacy of process and waste treatment systems, (b) the control of releases of radioactivity to the environment so that they do not exceed the limits of 10 CFR 20 and 10 CFR 50, Appendix I, and (c) the evaluation of environmental impact. This task involves improving current guidance to applicants and reviewers in the areas of radiation monitoring for process and effluent systems and reviewing the effluent monitoring systems for selected operating BWRs and PWRs to determine their effectiveness in meeting the effluent release limits of 10 CFR Parts 20 and 50.

Task No.	Title
B-68	Pump Overspeed During a LOCA

Applicability	Lead Division
LWRs	DSS

Problem Description

There is a potential for BWR recirculation pumps or PWR main coolant pumps to overspeed during a LOCA, resulting in the potential for missile generation. This task involves the conduct of analytical and experimental work to determine whether or not destructive overspeeds could be attained and to determine if corrective actions are necessary.

Task No.	Title
B-69	ECCS Leakage Ex-Containment

Applicability	Lead Division
LWRs	DSE

Problem Description

Following a plant transient, or accident, provisions are required for long-term decay heat removal. Redundancy of components is required as necessary to assure that a failure in the residual heat removal (RHR) system will not impair the ability to maintain the plant in a safe shutdown condition. Should such a failure involve leakage of primary coolant (as might result, for example, in the event of a pump seal failure), the potential exists for a release of radioactivity to the environment. Any resultant airborne releases are typically controlled either by provisions for filtration of airborne activity or location of RHR equipment within leaktight cells. Contaminated liquid releases (primary coolant not volatilized when leaked) is typically collected in a sump for subsequent transfer and cleanup through the radwaste system. Some guidance is provided in various standard review plans for identifying and controlling leakage from RHR equipment (cf. SRP 6.3.3, 9.3.3, 11.2, 15.6.5, and App B). However, there are no specific provisions to assure that the interface requirements for equipment and procedures are adequately developed.

For example, a seal failure might result in leakage of tens of gallons per minute of primary coolant. Any leaked fluid cannot be pumped to radwaste storage tanks unless storage capability is available. Post-event recovery operations require kowledge of the levels of airborne radioactivity and the degree of contamination of primary coolant. Accordingly, radiation monitoring equipment must be suitably located to provide operators with information needed to determine appropriate recovery operations.

In the event of a severe accident, such as a loss-of-coolant accident, or any other event which could lead to significant cladding failures, the levels of radioactivity in the coolant could be high. Such a situation would require effective control of any resultant leakage. Our present review procedure does not require dose calculations for passive failures, such as pump seal failures, if charcoal adsorbers are provided for the secondary containment atmosphere exhaust. Furthermore, the procedure does not require a systematic review by appropriate secondary review branches to assure that the highly-radioactive post-LOCA water leakage will be handled properly during all phases of the post-accident recovery operation. A systematic review should include review of emergency operating procedures and technical specifications that are necessary to assure the capability to control leakage in a manner that will meet both the first objective (assuring performance of the cooling functions) and the second objective (preventing excessive offsite doses). Because of the inaccessibility of the equipment under post-LOCA conditions and the manual operations involved in aligning equipment for loop functions and isolating excessively leaking components, advanced planning of the steps involved in controlling the probable leakages for the required long-term loop configurations should be set out in emergency operating procedures. Technical specifications governing loop boundary integrity, leak detection equipment, isolation equipment and leakage control equipment should be established, including limiting conditions for operation and surveillance requirements.

While existing equipment and procedures may permit a successful post-accident recovery operation, the current standard review plan does not provide an explicit basis for confirming that these objectives will be met.

Task No.	Title
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps

Applicability	Lead Division
PWRs	DSS

Problem Description

Offsite power system frequency decay, depending on the rate of decay, could provide an electrical brake on the reactor coolant pump motors that could slow the pumps faster than the assumed flywheel coastdown flow rates normally used in analyzing loss-of-flow accidents. Task A-35 will determine a maximum credible frequency decay rate to be used in this task. This task will determine if any additional measures are necessary to protect against a frequency decay event.

Task No.	Title
B-71	Incident Response

Applicability

Lead Division

DOR

Problem Description

Present NRC actions taken in response to a serious incident are directed from an Incident Response Center (IRC). To implement an adequate response, it is necessary that the IRC be equipped with appropriate communications services, information handling and evalution aids, pre-approved action guidelines, and technical and management personnel resources. The present IRC, which is manned during the course of an incident by a team of NRC management and technical staff, has all of these to some degree. A joint NRR/IE paper dated July 23, 1976, addresses whether practical and useful short-term and long-term improvements can be made. The paper (1) discusses the practicality and need for various IRC resources as a function of the goals of the NRC response capability and the time sequences of a spectrum of incident scenarios judged typical of those that are at least theoretically possible, and (2) makes recommendations with regard to NRR and IE actions to improve the IRC. The paper was reviewed and discussed by the Directors of NRR and IE, and the implementation recommendations agreed to in principle and a commitment of manpower was made by each of the Directors. IE has had and now has work proceeding in the areas for which they are responsible. NMSS also has responsibility for contributing to the incident management effort.

The subject addressed by this task is the identification of those actions that will implement the work for which NRR is responsible.

Task No.	Title
B-72	Health Effects and Life-Shortening from Uranium and Coal Fuel Cycles

Applicability	Lead Division
LWRs	DSE

Problem Description

Current practice in health impact assessments is to convert radiation exposure estimates into estimates of health effects, such as cancer deaths, illness, and life-shortening. However, the models presently being used, such as those in WASH-1400, GESMO, current NRC case related testimony, and EPA assessments, all suffer from similar weaknesses. A major common weakness, which appears amenable to solution, is related to the correct treatment of competing risks among populations with life expectancies, age, and sex distributions that vary with time. Since the staff is currently attempting to assess health effects in the future (e.g., Year 2000 and beyond), it is reasonable to expect significant changes in current population statistics. To make such an assessment, a demographic model is required which extrapolates the current population into the future, correctly allowing for competing risks of mortality from various causes (e.g., accidents, heart disease, and cancer). Failure to do so results, for example, in hypothetical cancer deaths for people who would statistically die from other causes. In the absence of better predictive models, it is not possible to even evaluate the uncertainty associated with the use of the current simplified methods for estimating health effects and consequent life-shortening. Uncertainties in the use of current models are greatly magnified when attempting to make comparisons of health effects for the coal and nuclear fuel cycles.

Current health effects models generally are used for estimating long-term impacts. Chronic exposure may be the primary determinant of the number of deaths for a given period for a given pollutant. However, in the case of non-radiological pollutants from the coal fuel cycle, short-term fluctuations leading to acute exposures may determine the time of death and consequent life-shortening. Current evaluations of the coal fuel cycle generally fail to account for short-term mortality, disease and illness. In addition, short-term effects from chemical pollutants are generally dependent on the prior history of chronic (long-term) exposure.

Current models generally assume linear dose-response relationships even when evidence exists for real or practical thresholds, or where experimental data support a non-linear dose response relationship.

This task involves the development of models to address these problems so that health effects (morbidity and mortality) can be assessed for both the coal and uranium fuel cycles as completely as current data permit and on a comparable basis.

Task No.	Title
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel

Applicability	Lead Division
LWRs	DSS

Problem Description

This is an ACRS generic concern. This task involves assessing the need for and, if necessary, developing criteria for acceptable vibration monitoring systems to provide early warning of excessive vibration inside the reactor vessel.

CATEGORY C GENERIC TASK

PROBLEM DESCRIPTIONS

Task No.	Title
C-1	Assurance of Continuous Long-Term Integrity of Seals on Instrumentation and Electrical Equipment

Applicability	Lead Division
All Reactor Types	DPM

Problem Description

Certain classes of instrumentation incorporate seals. When safety-related components within containment must function during post-LOCA accident conditions, their operability is sensitive to the ingress of steam or water. If the seals should become defective as a result of personnel errors in the maintenance of such equipment, such errors could lead to the loss of effective seals and the resultant loss of equipment operability. The establishment of a basis for confidence that sensitive equipment has a seal during the lifetime of the plant is needed. If current equipment and practices are found to be inadequate, criteria involving a testable design and/or quality assurance procedures adequate to assure continued seal integrity may be required.

Task No.	Title
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure

Applicability	Lead Division
LWRs	DSS

Problem Description

Inadvertent operation of containment sprays can result in a rapid depressurization of the containment building. Where containment external design pressure may be exceeded many plants have been provided with vacuum breakers or control system interlocks to prevent the containment external design pressure from being exceeded. The depressurization of the containment is a transient behavior and can take place in a short time period. This task involves the development of a code to be used for the analysis of containment pressure response (both with and without the effects of vacuum breakers or control systems) for the inadvertent spray accident.

Task No.	Title
C-3	Insulation Usage Within Containment

Applicability	Lead Division
LWRs	DSS

Problem Description

Various kinds of insulation are being used on piping and components inside the containment of a nuclear power plant. Of concern is its behavior under pipe break accident conditions with regard to the potential for blocking vent paths in subcompartments and impairing the effectiveness of the containment emergency sumps. The purpose of this task will be to gain a better understanding of how insulation might behave under pipe break accident conditions. This infomation will then be used to assess the assumptions made regarding insulation behavior for subcompartment analysis and the need for revising containment emergency sump design requirements.

Task No.	Title
C-4	Statistical Methods for ECCS Analysis

Applicability	Lead Division
LWRs	DSS

Problem Description

Appendix K of 10 CFR 50 specifies the requirements for LWR ECCS analysis. These requirements presently call for specified conservatisms to be applied to certain mo-els and correlations used in the analysis to account for data uncertainties at the time Appendix K was written. The resulting conservatism in the calculated peak clad temperature however, has never been compared against the uncertainty in peak clad temperature obtained from a realistically calculated (best estimate) LOCA.

In order to assess the safety margin in the Appendix K requirements, the peak clad temperature requirement (2200°F) will be equated to an uncert y level of a realistic claculation. This will be accomplished by analytical analysis utilizing best estimate LOCA analysis codes in which certain input parameters are simultaneously varied about their uncertainty distribution functions such that a resulting uncertainty distribution function in peak clad temperature is obtained. It is then possible to express the conservatism of the 2200°F cladding temperature limit in terms of probability and/or standard deviations from the most probable peak clad temperature.

The statistical methods for ECCS analysis will provide a probabilistic quantification of the safety margin imposed by 10 CFR 50 Appendix K ECCS safety evaluation requirements. The results of this program will be used to aid the staff in the review of changes to vendor ECCS models and in performing staff audit calculations of ECCS performance.

Task No.	Title	
C-5	Decay Heat Update	

Applicability	Lead Division
LWRs	DSS

Problem Description

This task involves following the work of research groups in determining best estimate decay heat data and associated uncertainties for use in LOCA calculations. The results of this task could be incorporated in future revisions of the current regulations regarding ECCS performance.

Task No.	Title	
C-6	LOCA Heat Sources	

Applicability	Lead Division
LWRs	DSS

Problem Description

The contributers to LOCA heat sources, along with their associated uncertainties, and the manner in which they are combined have an impact on LOCA calculations. An evalution of the combined effect of power density, decay heat, stored energy, fission power decay and their associated uncertainties with regard to calculations of LOCA heat sources is needed. This task will involve the review of vendor's data and approaches for determining LOCA heat sources and developing staff positions as needed.

Task No.	Title
C-7	PWR System Piping

Applicability	Lead Division
PWRs	DOR

Problem Description

Combinations of fabrication, stress and environmental conditions have resulted in isolated instances of stress corrosion cracking of low pressure schedule 10 type 304 stainless steel piping systems. Although these systems are not part of the reactor coolant pressure boundary, they are safety-related; e.g., the containment spray system. The incidence of cracking has been restricted to thin wall, low pressure, low flow systems. These cracks have occurred adjacent to the weld zones of the thin-walled piping after approximately three to five years of service and were identified by volumetric examination, by leak detection systems, or by visual inspection. In each of the cracking events that have occurred to date, the affected piping was determined to have been sensitized and, therefore, particularly vulnerable to corrosive attack.

Current licensing criteria preclude the use of sensitized piping in safety-related piping systems and place increased emphasis on the use of corrosion-resistant material in such systems. The purpose of this task is to continue to evaluate operating experience to determine if augmented inservice inspection requirements should be established to further enhance the reliability of such piping systems.

Task No.	Title
C-8	Main Steam Line Leakage Control Systems

Aplicability	Lead Division
BWRs	DOR

Problem Description

Operational experience has indicated that there is a relatively high fialure rate and variety of failure modes for components of the main steam isolation valve leakage control systems (MSIV-LCS) in certain operating BWRs. Experience from surveillance testing and reported in recent LERs is being compiled by DOR to serve as a basis for identifying design improvements and preparing recommendations to DSS and DSE for consideration in terms of future revisions to Regulatory Guide 1.96, changes to the Standard Review Plan, and implementation on pending licensing cases.

Task No.	Title
C-9	RHR Heat Exchanger Tube Failures

Applicability	Lead Division
LWRs	DOR

Problem Description

RHR heat exchangers are designed to bring the reactor to a safe cold shutdown condition and to maintain the core in a coolable geometry following a postulated loss-of-coolant accident. In the recent past, there have been several RHR heat exchanger tube failures at BWRs. Since the pressure control system on the river water (shellside) piping system maintains the pressure of the river water in the shellside of the RHR heat exchanger greater than the primary coolant pressure in the tubeside of the RHR heat exchanger during plant cooldown operations, a leak in the tubes would result in back leakage of river water into the primary loop. The objective of maitnaining the pressure in the shell side greater than that in the tubeside is that in the event of a tube failure there would be no leakage of redioactive fluids into the environment. This task will investigate the cause of these tube failures and the design of the pressure control system to assure that adequate long term core cooling capability is available.

Task No.	Title
C-10	Effective Operation of Containment Sprays in a LOCA

Applicability	Lead Division
LWRs	DSE

Problem Description

This task will respond to a concern of the ACRS about the effectiveness of various containment sprays to remove airborne radioactive materials which could be present within the containment following a LOCA. This concern has been expanded to include the possible damage to equipment located inside containment due to an inadvertent actuation of the sprays.

This task involves assisting the industry in writing an ANSI Standard on the design of containment spray systems, developing a topical report on the technological bases for spray washout models, and in managing contrac to evaluate the ability of different spray solutions to remove the radioiodines and radioactive particulates released to containment during a postulated LOCA.

Draft 7 of ANSI N581, "BWR and PWR Containment Spray System Design", has been reviewed by the staff. Following resolution of the NRC comments and issuance of this standard by ANSI, a regulatory guide endorsing this standard will be developed.

Task No.	Title
C-11	Assessment of Failure and Reliability of Pumps and Valves

Applicability	Lead Division
LWRs	DOR

Problem Description

The operating experience of nuclear power plants indicates that a number of valves, valve operators and pumps fail to operate as specified in the technical specifications either under testing conditions or when they are called upon to perform. The operating experience is documented by the Office of Management Information and Program Control (MIPC) publications in a monthly report of LERs sorted by components which include pumps, valves, and valve operators. Most of these occurrences relate to valve leakage, valve actuation, and safety/relief valve operation outside their operational bounds. The main steam isolation, safety and solenoid valves caused the most frequent abnormal occurrences in safety-related systems. Valve malfunctions can cause forced outage of operating plants. It is noted that about 10% of all outage time can be attributed to the malfunction of the critical pumps and valves within the plant. Of primary interest are outages caused by the main steam isolation and safety/relief valves.

Several adverse operating experiences concerning main steam isolation valves (MSIVs) were reported to the Office of Inspection and Enforcement following operational tests and spurious closures at various PWR plants. This led to a staff investigation of MSIVs at all operating and near operating facilities. This investigation has recently been completed.

The principal activity under this task will be the evaluation of active pumps and valves with respect to their operability and reliability under accident loading, i.e., loss of coolant accident and safe shutdown earthquake.

Task No.	Title
C-12	Primary System Vibration Assessment

Applicability	Lead Division
PWRs	DOR

Problem Description

Structural damage to the primary system, including the reactor pressure vessel and internals, associated piping and steam generator tubing in PWR's, can be caused by vibrations of sufficient magnitude. These vibrations can be either flow-induced or the result of operation of the pumps to which primary system piping is attached. There have been a number of instances where components internal to the reactor coolant pressure boundary have come loose as the result of flow-induced vibration and been carried through the primary system by the coolant flow.

Excessive core barrel movement, caused by flow-induced vibration, may lead to many detrimental effects including damage to reactor internals and interference with control rod movement. Problems resulting from excessive core barrel movement have been encountered at Palisades and possibly other operating plants.

Structural damage due to flow-iduced vibration of steam generator tubing has also been encountered. Anti-vibration bars are currently utilized to minimize tube vibration. However, fretting has occurred due to deficient design and material selection for the anti-vibration bars.

Piping systems are also susceptible to forced vibration as a result of pump vibration during operation. If a natural frequency of the connected piping is very nearly the same as the driving frequency of the pump there is then the possibility, depending on the amplitude of vibration, for fatigue failures in the system, particularly at the nozzle where the stresses will be highest.

Preoperational testing of reactor internals, piping systems and mechanical equipment is conducted during startup functional testing to assure structural and functional integrity per Section 3.9.2 of the Standard Review Plant and Regulatory Guide 1.20. However, vibration frquency shifts are possible during operation as a result of component and/or component support wear or degradation. Also, vibration effects for the long term may not have been properly assessed during startup testing.

Inservice inspection during the life of the plant and possible visual and audible detection of vibration during plant operation may be necessary in order to arrest structural damage already incurred or, if the vibration were to continue, might occur at some future time. This vibration assessment could lead to modifications in the design of system components or component support arrangements of system operation sequences.

Task No.	Title
C-13	Non-Random Failures

Applicability	Lead Division
LWRs	DSS

Problem Description

This is an ACRS generic concern. It formerly was referred to as common mode failure of identical components exposed to identical or nearly identical conditions or environments. The concern now has been expanded to include other types of multiple failures for which the consequences and probabilities cannot be predicted by application of the single failure criterion, such as use of the same sensors or components for both control and protection systems, sequential multiple failures due to a domino effect, or simultaneous multiple failures due to a single fault. This task has been included in Tasks A-9, A-30, A-35, B-56 and B-57.

Task No.	Title
C-14	Storm Surge Model for Coastal Sites

Applicability	Lead Division
Reactors Located on Coastal Sites	DSE

Problem Description

The staff is required to estimate the design basis water levels for each site. For coastal and estuarine sites, the design basis water level is often caused by a storm surge, which results from the wind and pressure fields of an intense storm acting on the water.

The primary tool used by the staff for estimating storm surge has been the "bathystrophic" model as developed by the U.S. Army Corps of Engineers, Coastal Engineering Research Center (CERC). This model is based on the bathystrophic approximation, relating sea surface slope to wind stress, bottom stress, and pressure gradient, with a correction for corriolis force due to along-shore currents. The model has served its intended purposes well, but is now obsolete. Bigger and faster computers are now capable of solving multidimensional dynamic equations which account for many effects not included in the bathystrophic model. The multidimensional dynamic mathematical models can account for irregular shorelines, while the shape of the shoreline is not considered at all by the bathystrophic model.

True long wave dynamics are simulated by multidimensional dynamic mathematical models, but are completely neglected by the bathystrophic models. These two effects are especially important when estimating storm surges in semienclosed areas such as Long Island Sound (i.e., Jamesport, Shoreham, Millstone, etc.).

This task will develop a replacement for the bathystrophic model so that our evaluation of storm surge reflects state-of-the-art techniques.

Task No.	Title
C-15	NUREG Report for Liquid Tank Failure Analysis

Applicability	Lead Division
LWRs	DSE

Problem Description

Standard Review Plan 15.7.3 requires an analysis of the consequences of failure of tanks containing radioactive liquids outside containment. This task involves the development of a NUREG report that will describe a consistent and acceptable method for analyzing the effects of a failure of a radioactive liquid waste tank.

Task No.	Title
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection

Applicability

	Lead Division
Power plants with large cooling lakes located in regions of prime agricultural land	DSE

Problem Description

Interpretations of NEPA require that environmental impact assessment include land use impacts and alternatives in nuclear power plant licensing cases. The staff has performed both economic and non-economic land resource assessments in compliance with these NEPA requirements. Recent licensing cases have questioned the adequacy of the staff's resource evaluative methods with respect to large land areas required for sites and cooling lakes. The primary issue concerning the staff's assessment is that neither economic analyses nor resource assessment as currently performed provides a convincing rationale for preemption of high quality land in view of continued population pressures, predicted impending lags in world-wide agricultural food production and probable increasing international demands on the United States for exports of agricultural products.

Food and fiber production and distribution rank with energy production and utilization as vital world problems now and for the forseeable future. These problems are inextricably linked since energy production facilities can be consumers of large land areas while energy is a prime requirement for even modest levels of agricultural production. Thus, land use is and probably will remain a key siting issue in nuclear plant licensing.

This task will involve the conduct of a confirmatory exploration of net energy techniques to determine their suitability for application to environmental licensing assessment under NEPA. A problem of immediate licensing concern is the conflict in land use which occurs when power plants with large cooling lakes are sited in regions of prime agricultural land.

Task No.	Title
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

Applicability	Lead Division
All Reactor Types	DSE

Problem Description

There are no current criteria for acceptability of solidification agents. This task involves the development of criteria for acceptability of radwaste solidification agents to properly implement a process control program for the packaging of diverse plant wastes for shallow land burial.

CATEGORY D PROPOSED GENERIC TASKS

PROBLEM DESCRIPTIONS

Task No.	Title
D-1	Advisability of a Seismic Scram

Applicability	Lead Division
All Reactor Types	DPM

Problem Description

The ACRS has recommended that studies be made of techniques for seismic scram and of the potential safety advantages and potential disadvantages of prompt reactor scram in the event of strong seismic motion, say more than one-half the safe shutdown earthquake. Various suitable techniques have been identified and exist, but thus far only limited studies have been reported on the pros and cons of seismic scram.

Task No.	Title
D-2	Emergency Core Cooling System Capability for Future Plants

Applicability: LWRs

Lead Division: DSS

Problem Description

This is an ACRS generic concern. It involves exploration of diverse means of obtaining ECCS capability.

Task No.	Title
D-3	Control Rod Drop Accident

Applicability	Lead Division
BWRs	DSS

Problem Description

This is an ACRS generic concern. It involves assessing the uncertainties in calculations of the control drop accident including the choice of negative reactivity insertion rate due to a scram and the potential differences between a two-dimensional and a three-dimensional calculation.

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