UNRESOLVED SAFETY ISSUES SUMMARY

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UNRESOLVED SAFETY ISSUES SUMMARY

AQUA BOOK

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OFFICE OF NUCLEAR REACTOR REGULATION
U. S. NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

FOREWORD

THE UNRESOLVED SAFETY ISSUES SUMMARY IS DESIGNED TO PROVIDE THE MANAGEMENT OF THE NUCLEAR REGULATORY COMMISSION WITH A QUARTERLY OVERVIEW OF THE PROGRESS AND PLANS FOR COMPLETION OF GENERIC TASKS ADDRESSING UNRESOLVED SAFETY ISSUES REPORTED TO CONGRESS PURSUANT TO SECTION 210 OF THE ENERGY REORGANIZATION ACT OF 1974 AS AMENDED. THIS SUMMARY UTILIZES DATA COLLECTED FROM THE OFFICE OF NUCLEAR REACTOR REGULATION, OFFICE OF NUCLEAR REGULATORY RESEARCH, AND THE NATIONAL LABORATORIES AND IS PREPARED BY THE OFFICE OF NUCLEAR REACTOR REGULATION.

THE DEFINITION OF WHAT CONSTITUTES COMPLETION OF AN UNRESOLVED SAFETY ISSUE (USI) INCLUDES THE IMPLEMENTATION OF THE TECHNICAL RESOLUTION. THIS IS IN ACKNOWLEDGEMENT OF THE FACT THAT REAL SAFETY BENEFITS OCCUR ONLY AFTER THE IMPLEMENTATION HAS TAKEN PLACE. IMPORTANT ELEMENTS OF THIS IMPLEMENTATION PHASE ARE:

(1) THE PROVISION OF A PUBLIC COMMENT PERIOD FOLLOWING THE ISSUANCE OF A DRAFT NUREG REPORT INCORPORATING THE STAFF'S TECHNICAL RESOLUTION FOLLOWED BY A DISCUSSION AND DISPOSITION OF THE COMMENTS RECEIVED IN A FINAL NUREG REPORT.

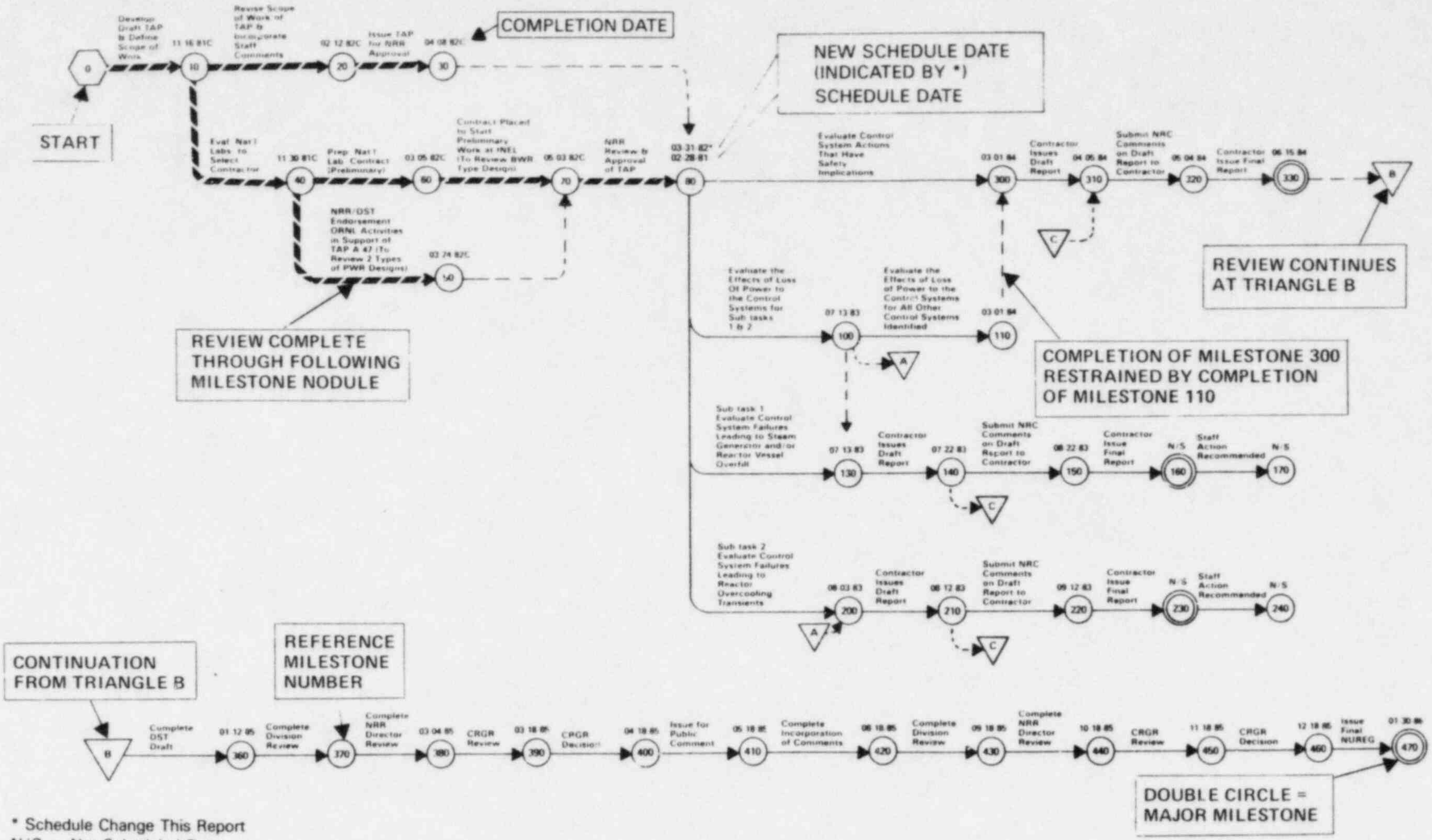
(2) THE PROVISION FOR INCORPORATION OF THE TECHNICAL RESOLUTION INTO THE NRC'S REGULATIONS, STANDARD REVIEW PLAN, REGULATORY GUIDES, OR OTHER NRC OFFICIAL GUIDANCE OR REQUIREMENTS, AS APPROPRIATE.

(3) THE PROVISION FOR APPLICATION OF THE TECHNICAL RESOLUTION TO INDIVIDUAL OPERATING PLANTS IN THE FORM OF HARDWARE OR DESIGN CHANGES, TECHNICAL SPECIFICATION CHANGE, AND/OR CHANGE TO OPERATING PROCEDURES AND TRAINING, AS APPROPRIATE.

THE MILESTONE CHARTS FOR EACH USI SHOW THE CURRENT SCHEDULE AS OF THE DATE OF PUBLICATION. IF A MILESTONE DATE HAS CHANGED SINCE THE LAST REPORT, THE OLD DATE WILL BE SHOWN WITH THE NEW DATE IMMEDIATELY ABOVE IT. THE NEW DATE WILL BE MARKED WITH AN ASTERISK WITH A FOOTNOTE INDICATING THAT A SCHEDULE CHANGE HAS BEEN MADE. THE PROGRAM STATUS TABLE WHICH BEGINS ON PAGE 3 OF THIS NUREG SHOWS THE COMPLETION DATE STATED IN THE LATEST APPROVED TASK ACTION PLAN AND THE CURRENT SCHEDULED COMPLETION DATE. THE MILESTONE AT THE END OF EACH ACTION PLAN WHICH REPRESENTS THE INITIATION OF THE IMPLEMENTATION PROCESS BOTH WITH RESPECT TO INCORPORATION OF THE TECHNICAL RESOLUTION IN THE NRC OFFICIAL GUIDANCE OR REQUIREMENTS AND ALSO THE APPLICATION OF CHANGES TO INDIVIDUAL OPERATING PLANTS. THE SCHEDULE FOR IMPLEMENTATION WILL NOT NORMALLY BE INCLUDED IN THE TASK ACTION PLAN(S) FOR THE RESOLUTION OF A USI SINCE THE NATURE AND EXTENT OF THE ACTIVITIES NECESSARY TO ACCOMPLISH THE IMPLEMENTATION CANNOT NORMALLY BE REASONABLY DETERMINED PRIOR TO THE DETERMINATION OF A TECHNICAL RESOLUTION. THE PROGRESS AND STATUS FOR IMPLEMENTATION OF UNRESOLVED SAFETY ISSUES FOR WHICH A TECHNICAL RESOLUTION HAS BEEN COMPLETED ARE REPORTED SPECIFICALLY IN A SEPARATE TABLE PROVIDED IN THIS SUMMARY. MORE DETAIL ON THE STATUS OF IMPLEMENTATION IN PROGRESS ON A SPECIFIC UNRESOLVED SAFETY ISSUE WHERE THE TECHNICAL RESOLUTION REQUIRES CHANGES TO INDIVIDUAL OPERATING PLANTS IS PROVIDED IN NUREG-0748, "OPERATING REACTORS LICENSING ACTIONS SUMMARY" WHICH IS PUBLISHED MONTHLY.

KARL KNIEL, CHIEF OF THE GENERIC ISSUES BRANCH, DIVISION OF SAFETY TECHNOLOGY, OFFICE OF NUCLEAR REACTOR REGULATION IS RESPONSIBLE FOR MANAGING THE GENERIC TASKS INCLUDED IN THIS SUMMARY.

EXAMPLE PAGE



* Schedule Change This Report
N/S = Not Scheduled Date

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ABBREVIATIONS

AAB	ACCIDENT ANALYSIS BRANCH (FORMER NRR BRANCH)
AB	ADMINISTRATION BRANCH, TRAINING AND ADMINISTRATION STAFF (IE)
AC	ALTERNATING CURRENT
ACRS	ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
AD	ASSISTANT DIRECTOR
ADB	ANALYSIS AND DEVELOPMENT BRANCH, DIVISION OF REACTOR SAFETY RESEARCH (RES)
AEB	ACCIDENT EVALUATION BRANCH, DIVISION OF SYSTEMS INTEGRATION (NRR)
AEOD	OFFICE OF THE ANALYSIS AND EVALUATION OF OPERATIONAL DATA
AIF	ATOMIC INDUSTRIAL FORUM
APTS	ACTION PLAN TRACKING SYSTEM
ARL	ALDEN RESEARCH LABORATORY
ASB	AUXILIARY SYSTEMS BRANCH, DIVISION OF SYSTEMS INTEGRATION (NRR)
ASME	AMERICAN SOCIETY OF MECHANICAL ENGINEERS
ASTM	AMERICAN SOCIETY OF TESTING MATERIALS
ATWS	ANTICIPATED TRANSIENTS WITHOUT SCRAM
BG&E	BALTIMORE GAS AND ELECTRIC COMPANY
B&W	BABCOCK AND WILCOX COMPANY
BNC	BROOKHAVEN NATIONAL CONSERVATORY
BNL	BROOKHAVEN NATIONAL LABORATORY
BOP	BALANCE OF PLANT
BWR	BOILING WATER REACTOR
CE	COMBUSTION ENGINEERING, INCORPORATED
CMEB	CHEMICAL ENGINEERING BRANCH, DIVISION OF ENGINEERING (NRR)
CFR	CODE OF FEDERAL REGULATIONS
CP	CONSTRUCTION PERMIT
CPB	CORE PERFORMANCE BRANCH, DIVISION OF SYSTEMS INTEGRATION (NRR)
CR	CONTRACTOR REPORT
CRGR	COMMITTEE TO REVIEW GENERIC REQUIREMENTS
CSB	CONTAINMENT SYSTEMS BRANCH, DIVISION OF SYSTEMS INTEGRATION (NRR)
DC	DIRECT CURRENT
DE	DIVISION OF ENGINEERING (NRR)
DEDROGR	DEPUTY EXECUTIVE DIRECTOR FOR REGIONAL OPERATIONS AND GENERIC REQUIREMENTS
DFO	DIVISION OF FACILITY OPERATIONS (RES)
DHFS	DIVISION OF HUMAN FACTORS SAFETY (NRR)
DHRS	DECAY HEAT REMOVAL SYSTEMS
DL	DIVISION OF LICENSING (NRR)
DOE	U. S. DEPARTMENT OF ENERGY
DOR	DIVISION OF OPERATING REACTORS (FORMER NRR DIVISION)
DRA	DIVISION OF RISK ANALYSIS (RES)
DSI	DIVISION OF SYSTEMS INTEGRATION (NRR)
DSS	DIVISION OF SYSTEMS SAFETY (FORMER NRR DIVISION)
DST	DIVISION OF SAFETY TECHNOLOGY (NRR)
E	ENGINEERING
EB	ENFORCEMENT BRANCH, ENFORCEMENT AND INVESTIGATIONS STAFF (IE)
ECC	EMERGENCY CORE COOLING
EEB	ENVIRONMENTAL ENGINEERING BRANCH, DIVISION OF ENGINEERING (NRR)
EFPY	EFFECTIVE FULL-POWER YEARS
EG&G	EDGERTON, GERMESHAUSEN & GRIER
EP	EMERGENCY PREPAREDNESS
EPRI	ELECTRIC POWER RESEARCH INSTITUTE
EQB	EQUIPMENT QUALIFICATION BRANCH, DIVISION OF ENGINEERING (NRR)
FIN	FINANCIAL
FSTF	FULL-SCALE TEST FACILITY
FW	FEEDWATER
FY	FISCAL YEAR
GSB	GEOSCIENCES BRANCH, DIVISION OF ENGINEERING (NRR)
GE	GENERAL ELECTRIC
GIB	GENERIC ISSUES BRANCH, DIVISION OF SAFETY TECHNOLOGY (NRR)
HFEB	HUMAN FACTORS ENGINEERING BRANCH, DIVISION OF HEALTH, SITING AND WASTE MANAGEMENT (RES)
HSST	HEAVY SECTION STEEL TECHNOLOGY
ICBR	INSTRUMENTATION AND CONTROL BRANCH, DIVISION OF FACILITY OPERATIONS (RES)
ICSB	INSTRUMENTATION AND CONTROL SYSTEMS BRANCH, DIVISION OF SYSTEMS INTEGRATION (NRR)
IE	OFFICE OF INSPECTION AND ENFORCEMENT
IEEE	INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS
INEL	IDAHO NUCLEAR ENGINEERING LABORATORY
IP	INDIAN POINT
IREP	INTEGRATED RELIABILITY EVALUATION PROGRAM
ISI	IN-SERVICE INSPECTION
LANL	LOS ALAMOS NATIONAL LABORATORY
LER	LICENSEE EVENT REPORT
LLNL	LAWRENCE LIVERMORE NATIONAL LABORATORY
LOCA	LOSS-OF-COOLANT ACCIDENT
LPP	LEAD PLANT PROGRAM
LTP	LONG TERM PROGRAM
LWR	LIGHT-WATER REACTOR
MARK I-III	CONTAINMENT TYPES FOR BOILING WATER REACTORS
MEB	MECHANICAL ENGINEERING BRANCH, DIVISION OF ENGINEERING (NRR)
MIT	MASSACHUSETTS INSTITUTE OF TECHNOLOGY
MSLB	MAIN STEAM LINE BREAK
MTEB	MATERIALS ENGINEERING BRANCH, DIVISION OF ENGINEERING (NRR)
NDE	NON-DESTRUCTIVE EXAMINATION
NRC	NUCLEAR REGULATORY COMMISSION
NREP	NEUTRON RESONANCE ESCAPE PROBABILITY
NRR	OFFICE OF NUCLEAR REACTOR REGULATION
NSS	NUCLEAR STEAM SYSTEM
NUREG	NUCLEAR REGULATORY REPORT (PREPARED IN-HOUSE)
NUREG/CR	NUCLEAR REGULATORY REPORT (PREPARED BY CONTRACTOR)
OL	OPERATING LICENSE
ORAB	OPERATING REACTORS ASSESSMENT BRANCH, DIVISION OF LICENSING (NRR)

ABBREVIATIONS

ORB	OPERATING REACTORS BRANCH, DIVISION OF LICENSING (NRR)
ORNL	OAK RIDGE NATIONAL LABORATORY
OSD	OFFICE OF STANDARDS DEVELOPMENT (FORMER NRC OFFICE)
OTSG	ONCE-THROUGH STEAM GENERATOR
PASNY	POWER AUTHORITY OF THE STATE OF NEW YORK
PDA	PRELIMINARY DESIGN APPROVAL
PNL	PACIFIC NORTHWEST LABORATORY (BATTELLE)
PRA	PROBABILISTIC RISK ASSESSMENT
PSB	POWER SYSTEMS BRANCH, DIVISION OF SYSTEMS INTEGRATION (NRR)
PSU	PLANT SYSTEMS UNIT (AEOD)
PTRB	PROCEDURES AND TEST REVIEW BRANCH, DIVISION OF HUMAN FACTORS SAFETY (NRR)
PWR	PRESSURIZED WATER REACTOR
RAB	RADIOLOGICAL ASSESSMENT BRANCH, DIVISION OF SYSTEMS INTEGRATION (NRR)
RCIC	REACTOR CORE ISOLATION COOLING
REF	REFERENCE
RES	OFFICE OF NUCLEAR REGULATORY RESEARCH
RFP	REQUEST FOR PROPOSAL
RHR	RESIDUAL HEAT REMOVAL
RPV	REACTOR PRESSURE VESSEL
RM	OFFICE OF RESOURCE MANAGEMENT
RRAB	RELIABILITY AND RISK ASSESSMENT BRANCH DIVISION OF SAFETY TECHNOLOGY (NRR)
RRRC	REGULATORY REQUIREMENTS REVIEW COMMITTEE
RS	REACTOR SAFETY (FORMER NRR BRANCH)
RSB	REACTOR SYSTEMS BRANCH, DIVISION OF SYSTEMS INTEGRATION (NRR)
RSSMAP	REACTOR SAFETY STUDY METHODOLOGY APPLICATION PROGRAM
RV	REACTOR VESSEL
SAI	SCIENCE APPLICATIONS, INC.
SCC	STRESS-CORROSION CRACKING
SGEB	STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH, DIVISION OF ENGINEERING (NRR)
SEP	SYSTEMATIC EVALUATION PROGRAM
SEPB	SYSTEMATIC EVALUATION PROGRAM BRANCH, DIVISION OF LICENSING (NRR)
SER	SAFETY EVALUATION REPORT
SG	STEAM GENERATOR
SGWH	STEAM GENERATOR WATER HAMMER
SNL	SANDIA NATIONAL LABORATORY
SQUG	SEISMIC QUALIFICATION UTILITIES GROUP
SRP	STANDARD REVIEW PLAN
SRV	SAFETY RELIEF VALVE
SSE	SAFE SHUTDOWN EARTHQUAKE
SSPB	STANDARDIZATION AND SPECIAL PROJECTS BRANCH DIVISION OF LICENSING (NRR)
STP	SHORT-TERM PROGRAM
TAP	TASK ACTION PLAN
TER	TECHNICAL EVALUATION REPORT
TH	THERMAL HYDRAULICS
TM	TASK MANAGER
TMI	THREE MILE ISLAND
UCLA	UNIVERSITY OF CALIFORNIA, LOS ANGELES
USI	UNRESOLVED SAFETY ISSUE
W	WESTINGHOUSE ELECTRIC CORPORATION
WH	WATER HAMMER

PROGRAM STATUS

USI NO.	TITLE	SCHEDULED COMPLETION DATE FROM LATEST APPROVED TASK ACTION PLAN	CURRENT SCHEDULED COMPLETION DATE	REMARKS
A-3, A-4, A-5	STEAM GENERATOR TUBE INTEGRITY	MAY 1984	NOT SCHEDULED	A COMMISSION BRIEFING WAS HELD ON SEPTEMBER 10, 1984. THE COMMISSION APPROVED SECY-84-139 WITH THE EXCEPTION THAT A REVISED GENERIC LETTER TO PWR LICENSEES BE PREPARED BY FEBRUARY 12, 1985 FOR COMMISSION APPROVAL. THIS GENERIC LETTER WAS ISSUED TO PWR LICENSEES ON APRIL 17, 1985. COMMENTS ON THE LICENSEES' STEAM GENERATOR PROGRAMS AND THE STAFF'S REPORT, NUREG-0844, ARE DUE WITHIN 60 AND 90 DAYS, RESPECTIVELY.
A-17	SYSTEMS INTER-ACTIONS IN NUCLEAR POWER PLANTS	MARCH 1986	MAY 30, 1986	ALL TECHNICAL WORK ON USI A-17 TASKS IS ESSENTIALLY COMPLETE AND THE RESOLUTION PACKAGE IS IN DRAFT FORM. IT IS EXPECTED THAT THE DRAFT PACKAGE WILL BE CIRCULATED FOR INTERNAL REVIEW DURING THE MONTH OF JUNE.
A-40	SEISMIC DESIGN CRITERIA	JANUARY 1985	NOT SCHEDULED*	THE NRC STAFF INTERNAL REVIEW HAS BEEN COMPLETED. A VALUE/IMPACT ANALYSIS HAS BEEN PREPARED AND A CRGR SUBMITTAL PACKAGE IS UNDER REVIEW BY THE ACTING DIRECTOR OF THE DIVISION OF ENGINEERING.
A-43	CONTAINMENT EMERGENCY SUMP PERFORMANCE	SEPTEMBER 30, 1984	DECEMBER 31, 1985*	ALL TECHNICAL SUPPORT (NUREG/CR) REPORTS HAVE BEEN ISSUED. NUREG-0897 AND NUREG-0869 ALONG WITH SRP SECTION 6.2.2 WERE ISSUED FOR PUBLIC COMMENT IN MAY 1983. THE PUBLIC COMMENT PERIOD ENDED IN JULY 1983 AND THE COMMENTS RECEIVED WERE UTILIZED IN THE PREPARATION OF THE REVISED CRGR SUBMITTAL OF JUNE 14, 1984. THE REGULATORY ANALYSIS HAS BEEN REVISED TO REFLECT COMMENTS RECEIVED FROM THE JULY 11, 1984 CRGR MEETING AND FOLLOWUP STAFF EVALUATIONS.
A-44	STATION BLACKOUT	MAY 1985	DECEMBER 30, 1986*	THE PROPOSED RULEMAKING PACKAGE WAS SENT TO THE COMMISSION ON MAY 6, 1985. NUREG-1032, "EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS, TECHNICAL FINDINGS RELATED TO USI A-44," IS BEING PREPARED TO BE ISSUED FOR PUBLIC COMMENT. A NUCLEAR UTILITY GROUP ON STATION BLACKOUT SUBMITTED IN MAY 1985, A PROPOSAL TO NRC FOR THE RESOLUTION OF USI A-44. THE PROPOSAL IS BEING REVIEWED BY THE STAFF.
A-45	SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS	FEBRUARY 1986	FEBRUARY 28, 1986	PLANT VISITS FOR THE PURPOSE OF OBTAINING MISSING INFORMATION RELATIVE TO DHR SYSTEMS ANALYSES HAVE TAKEN PLACE AT POINT BEACH, TURKEY POINT, QUAD CITIES, ARKANSAS NUCLEAR NO. 1, TROJAN, COOPER AND ST. LUCIE.

*SCHEDULE CHANGE THIS REPORT

PROGRAM STATUS

USI NO.	TITLE	SCHEDULED COMPLETION DATE FROM LATEST APPROVED TASK ACTION PLAN	CURRENT SCHEDULED COMPLETION DATE	REMARKS
A-46	SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS	DECEMBER 1984	DECEMBER 30, 1985*	WORK ON ALL TASKS IS ESSENTIALLY COMPLETE WITH THE EXCEPTION OF TASK 4. AN INTERIM REPORT WHICH SUMMARIZES THE STATUS OF WORK ACCOMPLISHED ON A-46 WAS ISSUED AS NUREG-1018 IN OCTOBER 1983. THE A-46 CRGR PACKAGE (INCLUDING DRAFT NUREG-1030) WAS APPROVED BY THE DIRECTOR OF NRR ON OCTOBER 31, 1984 AND SENT TO CRGR FOR REVIEW AND APPROVAL IN MAY 1985. DELAY IN COMPLETING THE CRGR REVIEW WILL RESULT IN A SCHEDULE SLIP OF APPROXIMATELY 2 MONTHS.
A-47	SAFETY IMPLICATIONS OF CONTROL SYSTEMS	APRIL 1986	APRIL 1, 1986	DRAFT REPORT ON THE SAFETY IMPLICATIONS OF CONTROL SYSTEMS OF A B&W PWR DESIGN WAS SUBMITTED BY ORNL IN OCTOBER 1984. PNL RISK ASSESSMENT DRAFT REPORT (REV. 2) ON CONTROL SYSTEMS FAILURES FOR GE, WESTINGHOUSE, AND B&W DESIGNS HAVE BEEN SUBMITTED BY PNL FOR STAFF REVIEW. A PRELIMINARY DRAFT REPORT ON SAFETY IMPLICATIONS OF CONTROL SYSTEMS OF A CE PWR DESIGN WAS SUBMITTED BY ORNL ON APRIL 30, 1985. A FINAL DRAFT IS SCHEDULED TO BE SUBMITTED BY MAY 31, 1985.
A-48	HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT	JUNE 1986	DECEMBER 30, 1986*	WORK ON THIS USI IS LIMITED TO THE GENERIC RESOLUTION OF HYDROGEN CONTROL AND EQUIPMENT QUALIFICATION FOR ICE CONDENSER AND BWR MARK III CONTAINMENTS. A COMMISSION PAPER REGARDING HYDROGEN CONTROL FOR MARK III AND ICE CONDENSER CONTAINMENT WAS REVIEWED AND ENDORSED BY THE CRGR ON JUNE 1, 1983. THE COMMISSION PAPER WAS FORWARDED ON TO THE COMMISSION ON AUGUST 26, 1983, AND ADDITIONAL INFORMATION PROVIDED ON DECEMBER 28, 1983. ON JANUARY 25, 1985, THE COMMISSION ISSUED THE HYDROGEN FINAL RULE (50 FR 3498). THE RESULTS OF THE LARGE SCALE HYDROGEN BURN TESTS CONDUCTED AT THE NEVADA TEST SITE SHOW POTENTIAL CHALLENGE TO EQUIPMENT SURVIVABILITY. THE STAFF'S PRELIMINARY EVALUATION OF THE DATA INDICATED THAT THE POSTULATED HYDROGEN BURN. FURTHER EVALUATION OF THE DATA IS PLANNED.
A-49	PRESSURIZED THERMAL SHOCK	DECEMBER 31, 1985	MARCH 31, 1986	NRC STAFF PROPOSED PTS RULE WAS APPROVED BY THE COMMISSION IN JANUARY 1984. THIS NEW PTS RULE WAS PUBLISHED FOR PUBLIC COMMENT ON FEBRUARY 7, 1984. THE PROPOSED FINAL RULE, TAKING THE PUBLIC COMMENTS INTO ACCOUNT, WILL BE SUBMITTED TO THE COMMISSION FOR APPROVAL IN MARCH 1985.

*SCHEDULE CHANGE THIS REPORT

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS
A-1	WATER HAMMER	MARCH 15, 1984	NUREG-0927, REV. 1 NUREG-0933, REV. 1 SRP SECTIONS 3.9.3, REV. 1 3.9.4, REV. 2 5.4.6, REV. 3 5.4.7, REV. 3 6.3, REV. 2 9.2.1, REV. 3 9.2.2, REV. 2 10.3, REV. 3 10.4.7, REV. 3	THE REVISED SRP SECTIONS WILL BE USED ONLY FOR REVIEW OF "CUSTOM PLANT" CONSTRUCTION PERMIT APPLICATIONS, AND FOR STANDARD PLANT APPLICATIONS DOCKETED AFTER THE ISSUANCE OF THESE SRP SECTION REVISIONS, WHICH ARE INTENDED FOR REFERENCING IN CONSTRUCTION PERMIT APPLICATIONS. (FORWARD FIT IMPLEMENTATION ONLY.)
A-2	ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEMS	DECEMBER 1980	NUREG-0609	THE WESTINGHOUSE OWNERS GROUP "LEAK BEFORE BREAK" CONCEPT WAS APPROVED. THE SER WAS THE SUBJECT OF GL-84-04 DATED FEBRUARY 1, 1984. TACS FOR THE 16 AFFECTED PLANTS WERE CLOSED. SERs FOR THE 6 CE PLANTS (4 CE OWNERS GROUP PLANTS I.E., CALVERT CLIFFS 1 & 2, PALISADES, AND MILLSTONE 2 AND 2 PLANT-SPECIFIC SUBMITTALS, I.E., MAINE YANKEE AND ST. LUCIE) WERE COMPLETED IN SEPTEMBER 1984. THE SERs ARE ON "HOLD" WHILE MANAGEMENT CHECKS THE LEGAL ASPECTS OF THE ASSUMPTION THAT LOCA AND SSE LOADS NEED NOT BE COMBINED. THE REMAINING 6 PWRs HAVE INDICATED INTENTION TO USE "LEAK BEFORE BREAK" CONCEPT, BUT HAVE NOT SUBMITTED SCHEDULES. THESE PLANTS ARE BEAVER VALLEY, PRAIRIE ISLAND 1 & 2, KEWAUNEE, SALEM 1 AND TROJAN. MPA ITEM D-10 WILL BE CLOSED WHEN THESE REMAINING SERs ARE ISSUED. THERE IS NO FIRM SCHEDULE AT PRESENT.

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS
A-6	MARK I SHORT TERM PROGRAM	DECEMBER 1977	NUREG-0408	COMPLETE - ALL PLANT-UNIQUE ANALYSES AND EQUIPMENT MODIFICATIONS AS REQUIRED WERE REVIEWED AND ACCEPTED AND APPROPRIATE TECHNICAL SPECIFICATION CHANGES WERE MADE.
A-7	MARK I LONG TERM PROGRAM	JULY 1980 AUGUST 1982	NUREG-0661 NUREG-0661, SUPPL. NO. 1 SRP SECTION 6.2.1.1C	LICENSEES ARE IN THE PROCESS OF OR HAVE INSTALLED MODIFICATIONS TO MEET THE COMMISSION'S ORDER DATE FOR EACH OPERATING PLANT. MORE THAN THREE QUARTERS OF THE PLANTS AFFECTED HAVE COMPLETED THESE MODIFICATIONS. THE LICENSEES HAVE SUBMITTED PLANT-UNIQUE ANALYSES TO THE STAFF FOR POST-IMPLEMENTATION AUDIT REVIEW FOR COMPLIANCE WITH THE ACCEPTANCE CRITERIA CONTAINED IN APPENDIX A TO NUREG-0661. OUR CONTRACTORS, BNL AND THE FRANKLIN RESEARCH CENTER (FRC), ARE REVIEWING THESE SUBMITTALS. BNL HAS COMPLETED ITS REVIEW FOR ALL THE PLANTS AND FRC HAS COMPLETED ITS REVIEW FOR 16 PLANTS. SEE MULTIPLANT ACTION ITEM D-01 IN NUREG-0748.
A-8	MARK II CONTAINMENT POOL DYNAMIC LOADS	AUGUST 1981	NUREG-0808 SRP SECTION 6.2.1.1C	THE REQUIREMENTS RECOMMENDED IN NUREG-0808 ARE BEING IMPLEMENTED DURING THE OPERATING LICENSE REVIEW FOR EACH PLANT WITH A MARK II CONTAINMENT. THESE REQUIREMENTS HAVE ALSO BEEN INCLUDED AS AN ADDITION TO THE APPROPRIATE SECTION OF THE STANDARD REVIEW PLAN.

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS
A-9	ATWS	SEPTEMBER 1980	NUREG-0460, VOL. 4 PROPOSED RULE 46FR57521 FINAL RULE 49FR57521	THE TECHNICAL FINDINGS FOR THIS ISSUE HAVE BEEN PUBLISHED IN NUREG-0460, "ANTICIPATED TRANSIENTS WITHOUT SCRAM FOR LIGHT WATER REACTORS," VOL. 4. A PROPOSED RULE BASED ON THIS WORK PLUS ADDITIONAL ANALYSIS WAS PUBLISHED FOR COMMENT. THE COMMENTS RECEIVED WERE ADDRESSED AND A FINAL RULE WAS AFFIRMED BY THE COMMISSION IN NOVEMBER 1983. THE FINAL RULE WAS PUBLISHED ON JUNE 26, 1984. GUIDANCE FOR IMPLEMENTATION IS INCLUDED IN THE FINAL RULE.
A-10	BWR FEEDWATER NOZZLE CRACKING	NOVEMBER 1980	NUREG-0619	RESPONSES FROM LICENSEES TO AN IMPLEMENTATION LETTER HAVE BEEN RECEIVED AND RECOMMENDED TREATMENT OF THESE RESPONSES HAVE BEEN SUBMITTED TO NRC MANAGEMENT. ADDITIONAL INFORMATION HAS BEEN REQUESTED OF LICENSEES. ALL PLANTS HAVE RECEIVED LETTERS ACCEPTING THEIR PROPOSED MODIFICATION PLANS. VERMONT YANKEE'S OPERATION IS SUCH THAT NO FEEDWATER NOZZLES NEED BE INSTALLED. LASCROSSE, BIG ROCK POINT, AND DRESDEN 1 DO NOT HAVE SUSCEPTIBLE PLANT SYSTEM CONFIGURATIONS AND ARE CONSIDERED COMPLETE WITH REGARD TO THIS ACTION. HUMBOLDT BAY, BY VIRTUE OF ITS STATUS (SHUTDOWN, NO FORSEEABLE RESTART) IS ALSO CONSIDERED COMPLETE. COMPLETE - SEE MULTIPLANT ACTION ITEM B-25 IN NUREG-0748.
A-11	REACTOR VESSEL MATERIALS TOUGHNESS	OCT. 15, 1982	NUREG-0744 VOLS. I AND II	GENERIC LETTER 82-26 TRANSMITTED THIS NUREG REPORT. NO FURTHER ACTION IS CONTEMPLATED.

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS
A-12	STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS	OCT. 11, 1983	NUREG-0577, REVISION 1 SRP SECTION 5.3.4	THE PROPOSED A-12 RESOLUTION IMPLEMENTATION WILL APPLY TO NEW CONSTRUCTION ONLY, THROUGH A NEW SRP SECTION 5.3.4, WITH NO BACKFITTING. SRP SECTION 5.3.4 HAS BEEN REVISED BASED ON PUBLIC COMMENTS RECEIVED AND FURTHER REVIEW BY THE STAFF AND CRGR. THE REVISED SRP SECTION 5.3.4 IS PART OF AN ISSUANCE PACKAGE WHICH IS UNDERGOING FINAL REVIEW PRIOR TO SUBMITTAL TO THE EXECUTIVE DIRECTOR FOR OPERATIONS.
A-24	QUALIFICATION OF CLASS 1E SAFETY RELATED EQUIPMENT	AUGUST 1981	NUREG-0588 NEW RULE 48FR2729	EQB HAS THE LEAD IN IMPLEMENTING THE POSITIONS IDENTIFIED IN THE REPORT. SEE MULTIPLANT ACTION ITEM B-60 IN NUREG-0748.
A-26	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION	SEPTEMBER 1978	NUREG-0224 SRP SECTION 5.2	ALL PLANTS WERE REQUESTED TO PROVIDE AN OVER-PRESSURE PREVENTION SYSTEM THAT WOULD BE USED WHENEVER THE PLANT WAS IN A COLD SHUTDOWN CONDITION. ALL PWRS IMPLEMENTED THEIR SYSTEMS WITH PRELIMINARY APPROVAL FROM THE NRC, AND A COMPLETE REVIEW TOOK PLACE ON A POST-IMPLEMENTATION BASIS. ONE LICENSING ACTION REMAINS TO BE COMPLETED. SEE MULTIPLANT ACTION ITEM B-04 IN NUREG-0748.
A-31	RESIDUAL HEAT REMOVAL REQUIREMENTS	MAY 1979	REGULATORY GUIDE 1.139 SRP SECTION 5.4.7	RRRC APPROVED IMPLEMENTATION PLAN OF JANUARY 31, 1978 IS BEING IMPLEMENTED ON NTOLS DURING THE REVIEW PROCESS. NO BACKFIT TO OPERATING REACTORS IS PLANNED.

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS
A-36	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	JULY 1980	NUREG-0612 SRP SECTION 9.1.5	IMPLEMENTATION REQUIREMENTS WERE ISSUED TO ALL LICENSEES BY LETTER DATED DECEMBER 22, 1980. THE LETTER REQUESTED; INTERIM ACTIONS TO BE COMPLETED IN 90 DAYS, A PHASE I ACTION (REPORT, CONFIRMATION AND JUSTIFICATION) IN SIX MONTHS AND PHASE II (SPECIFIC REQUIREMENTS) IN NINE MONTHS. ALL LICENSEES HAVE RESPONDED TO THE DECEMBER 22, 1980 GENERIC LETTER AND THEIR RESPONSES ARE BEING EVALUATED. AS OF FEBRUARY 22, 1985, ALL BUT 1 PHASE I REVIEWS HAVE BEEN COMPLETED AND IT IS ANTICIPATED TO COMPLETE THE LAST ONE IN FY-85. MPA C-15 HAS BEEN ESTABLISHED FOR PHASE II. IT IS EXPECTED TO CLOSE OUT PHASE II IN MAY 1985.
A-39	DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT	MARK I -02-29-80 MARK II -09-30-82 MARK III-10-14-82	NUREG-0763 NUREG-0783 NUREG-0802 SRP SECTION 6.2.1.1C	GENERIC LETTERS TRANSMITTING THESE NUREGS TO BWR APPLICANTS AND LICENSEES HAVE BEEN ISSUED. IMPLEMENTATION ON MARK I PLANTS IS PART OF USI A-7. IMPLEMENTATION ON MARK II AND MARK III PLANTS IS BEING PERFORMED DURING THE OPERATING LICENSE REVIEW FOR EACH PLANT.
A-42	PIPE CRACKS IN BOILING WATER REACTORS	JULY 1980	NUREG-0313, REV. 1	IN FEBRUARY 1981, NUREG-0313, REV. 1 WAS ISSUED TO ALL HOLDERS OF BWR OPERATING LICENSES OR CONSTRUCTION PERMITS AND TO ALL APPLICANTS FOR OPERATING LICENSES. BY JULY 1, 1981, THE APPLICANTS/LICENSEES WERE TO PROVIDE THEIR PROGRAM FOR REPLACEMENT OF SERVICE SENSITIVE LINES AND WELDS, THEIR PROGRAM FOR AUGMENTED INSERVICE INSPECTION, THEIR PROGRAM FOR IMPROVING THE WATER CHEMISTRY ENVIRONMENT AND INCORPORATION OF ADEQUATE LEAK DETECTION CAPABILITY. BASED ON OUR REVIEW OF THE INSPECTIONS AND OTHER ACTIONS TAKEN BY BWR LICENSEES TO DETECT AND MINIMIZE INTER-GRANULAR STRESS CORROSION CRACKING, WE HAVE SENT LETTERS TO ALL OPERATING BWR LICENSEES ADVISING THEM THAT A-42 HAS BEEN SATISFACTORILY RESOLVED FOR THEIR FACILITIES. THIS ISSUE IS NOW CONSIDERED COMPLETE.

RESEARCH SUMMARY

	UNRESOLVED SAFETY ISSUE	RES PERSONNEL DIRECTLY SUPPORTING USI	RESEARCH PROGRAMS PERFORMED PRIMARILY TO SUPPORT USI	RESEARCH PROGRAMS RELEVANT TO USI RESOLUTION
1.	USI A-3,4,5 Steam Generator Tube Integrity	None	None	The PNL Steam Generator Integrity Program is deeloping models to predict failure under burst and collapse pressures of defective steam generator tubing. The BNL Stress Corrosion Cracking Program is developing data and models to predict stress corrosion cracking of Inconel 600 steam generator tubing.
2.	USI A-17 Systems Interaction	D. Rasmuson	None	PRA research on "dependent failures" (e.g., ARMIEP).
3.	USI A-43 Containment Sump Performance	L. Porse	DOE funded design and construction of full scale sump test facility at Alden Research Laboratory to investigate a wide range of sump design parameters (complete). RES funded SNL data evaluation of ARL tests, and concluding test at ARL (complete).	None
4.	USI A-44 Station Blackout	P. Baranowsky	RES managed DST funded program at ORNL to review onsite and offsite AC power reliability, and at SNL for accident sequence analysis.	The SASA program analyzed specific accident sequences for station blackout events.

RESEARCH SUMMARY

UNRESOLVED SAFETY ISSUE	RES PERSONNEL DIRECTLY SUPPORTING USI	RESEARCH PROGRAMS PERFORMED PRIMARILY TO SUPPORT USI	RESEARCH PROGRAMS RELEVANT TO USI RESOLUTION
5. USI A-45 Shutdown Decay Heat Removal Requirements	None	SNL performed a value/impact study of alternate decay heat removal concepts.	SNL is performing the Severe Accident Risk Rebaselining and Risk Reduction Program (SARRP) to incorporate insights from ongoing programs in a rebaselining of reactor risks.
6. USI A-46 Seismic Qualification of Equipment in Operating Plants	None	Southwest Research Institute is evaluating methods to qualify both mechanical and electrical equipment to determine if older methods provide adequate assurance for seismic events.	The Seismic Equipment Qualification Program at INEL deals with the same subject matter as A-46 but on a longer time frame.
7. USI A-47 Safety Implications of Control Systems	D. Basdekas A. Dipalo	ORNL is conducting a review of 2 of the 4 LWR designs under review by this program to assess safety implications of control systems. This includes a review of power supplies to assess potential multiple control system failures. PNL is performing PRA and cost benefit studies on potentially significant control system failures.	None

RESEARCH SUMMARY

UNRESOLVED SAFETY ISSUE	RES PERSONNEL DIRECTLY SUPPORTING USI	RESEARCH PROGRAMS PERFORMED PRIMARILY TO SUPPORT USI	RESEARCH PROGRAMS RELEVANT TO USI RESOLUTION
8. USI A-48 Hydrogen Control	P. Worthington W. Farmer	The SNL Hydrogen Behavior Program provides: (a) computer code development to analyze hydrogen events (HECTR), (b) analysis of data developed at the Nevada large scale test, (c) experimental data on burn dynamics from the FLAME test facility, and (d) investigation of potential hydrogen detonation in containment. The SNL Hydrogen Mitigation and Mixing Studies investigates means such as water spray on aerosols to mitigate the effects of hydrogen combustion.	
9. USI A-49 Pressurized Thermal Shock	C. Johnson M. Vagins J. Reyes L. Shotkin N. Randall	The ORNL Integrated PTS Program coordinates efforts to confirm the bases for the PTS rule-specified screening criterion and supporting Reg Guide. RES is supporting several efforts to investigate thermal mixing of cooling water with warmer water in certain critical areas of the pressure vessel. RES also supports several efforts performing detailed thermal-hydraulics calculations to support the integrated ORNL program.	The Heavy Section Steel Technology (HSST) program investigates fracture-mechanics and materials behavior of pressure vessel steels. There are numerous other efforts that relate to neutron spectra, dosimetry and non-destructive examination techniques.

WEST., CE & B&W STEAM GENERATOR TUBE INTEGRITY (A-3, A-4, A-5)

AS OF WEEK ENDING MAY 17, 1985

KEY PERSONNEL

TASK MANAGER*

P. NORIAN X27487

Paul Norian

NRR ANALYST

JUDY BUTTS X24822

TASK REVIEWERS

NAME	BRANCH
E. MURPHY	ORAB/DL
C. PARSZEWSKI	CEB/DE
J. RAJAN	MEB/DE
B. TUROVLIN	CEB/DE
F. ODAR	ADB/RSR
F. AKSTULEWICZ	ORB5/DL

SCHEDULED COMPLETION

1978 ANNUAL REPORT: Early 1980

CURRENT: Not Scheduled

• PROBLEM DESCRIPTION

Pressurized water reactor steam generator tube integrity can be degraded by corrosion induced wastage, cracking, reduction in tube diameter (denting) and vibration induced fatigue cracks. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Westinghouse and Combustion Engineering steam generator tubes have suffered degradation due to wastage and stress corrosion cracking. Both types of degradation have been decreased by conversion from phosphate to an all-volatile secondary water treatment. Degradation due to denting which leads to primary side stress corrosion cracking continues to be a problem.

B&W's once-through steam generators (OTSG's) were relatively free of trouble prior to the first tube leak incident at Oconee Unit 3 in July, 1976. Since then, all three Oconee units have experienced tube leak incidents. The leaks at the Oconee units are the result of cracks of unknown origins propagated in the circumferential direction by flow induced vibration and have been limited to tubes located adjacent to the open tube inspection lane.

A second form of degradation characterized as an erosion-cavitation phenomena has been observed at Oconee and other B&W units.

*The staff contact for the Division of Licensing's integrated steam generator program is Emmett Murphy, X27457.

• RES INTERFACE INFORMATION

A. RES has funded, at the request of NRR, a major confirmatory program at PNL. The activity of this program consists of tests to verify the burst and cyclic strengths of degraded steam generator tubes and the leak rate data.

B. RES is funding a program addressing the factors which determine Inconel 600 susceptability to stress corrosion cracking in primary water. Metal condition, chemistry, temperature, stress and environment will be considered.

• ACRS INTERFACE INFORMATION

The current status of this program was discussed with the ACRS Metal Components Subcommittee on January 28, 1983 and September 12, 1983. A meeting with the full ACRS was held on October 13, 1983.

• TECHNICAL ASSISTANCE CONTRACTS

The following technical assistance contracts are generic in nature and will be applicable to the three Category "A" Technical Activities (A-3, A-4, and A-5) related to PWR steam generators.

A SANDIA – Provide statistical analysis of steam generator tube failures in operating reactors in order to establish the bases for the sampling plan for inservice inspection. Completed.

B BNL – Provide necessary computer code and perform parametric evaluation of effects of tube failures concurrent with MSLB. Completed.

C BNL – Provide technical consultation and assistance to review information in areas of water chemistory and corrosion analysis, stress and/or burst strength calculations. Completed.

D PNL – Provide cost/benefit evaluation of ISI plans. Completed.

E PNL – Evaluate environmental consequences of multiple tube failures concurrent with MSLB. Completed.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
2314	PNL	$75,000	$75,000
B2316	PNL	$95,000	$95,000

• POTENTIAL PROBLEMS

The ACRS letter dated October 18, 1983 stated that the proposals should be recommended industry actions and not new requirements.

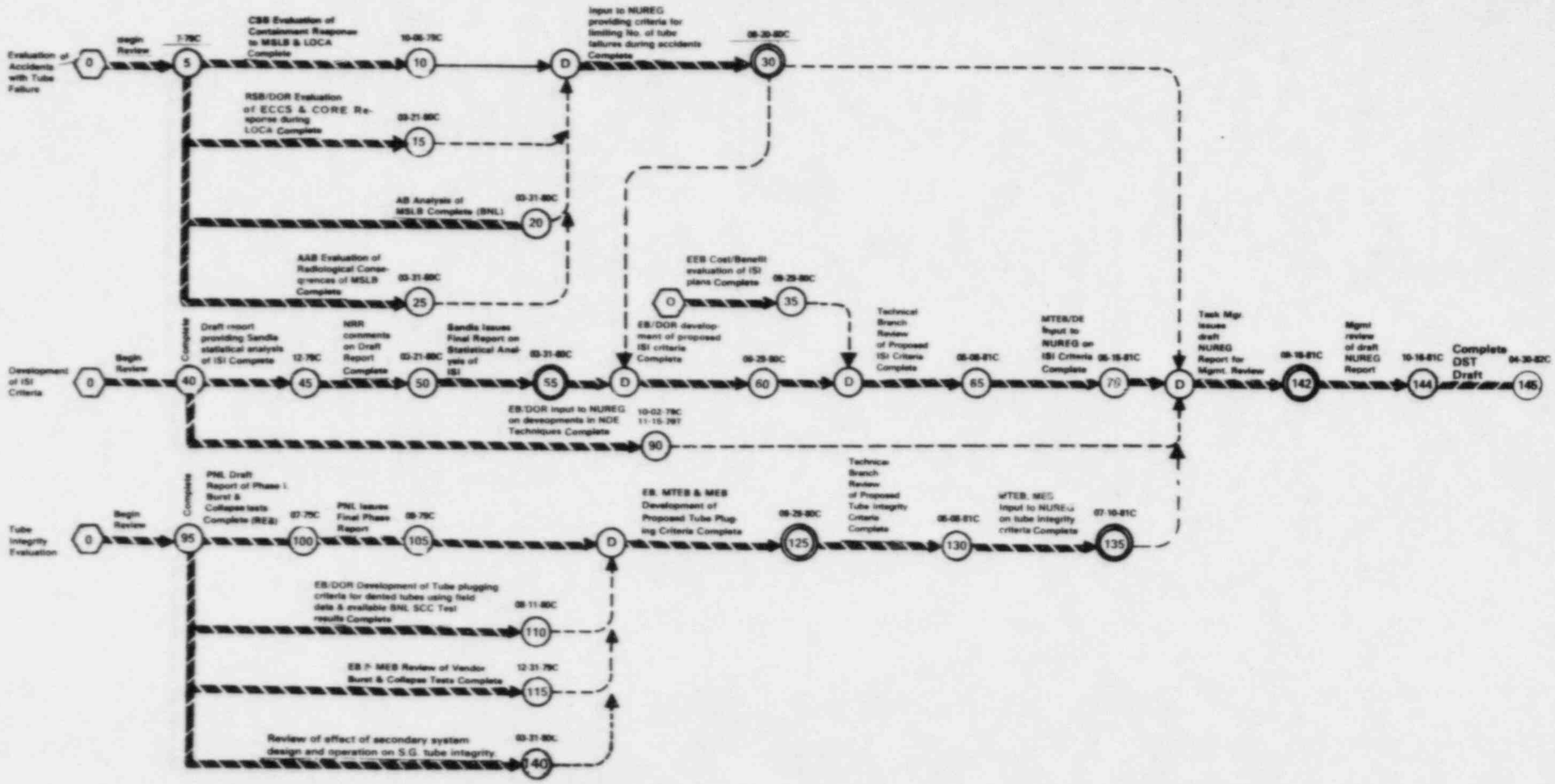
• STATUS SUMMARY

The NRC has formed a Task Force under the Division of Licensing to prepare its proposed requirements regarding steam generator tube integrity. These requirements will include new concerns resulting from the Ginna tube failure (such as loose parts in the secondary system and plant response to SG tube failures) and also corrosion related failure mechanisms. The recommendations prepared by the staff under USI A-3, 4, 5 were primarily concerned with corrosion mechanisms such as wastage and denting. Consequently, as discussed with the Commission on June 30, 1982, the requirements from the USI program will be incorporated in the overall set of requirements being developed to address tube failures.

CRGR meetings were held on September 14, 1983 and October 24, 1983. An ACRS meeting was held on October 13, 1983. A Commission briefing was held on September 10, 1984. Additional information was sent to the Commission in SECY-84-13B, dated November 5, 1984.

A generic letter to PWR licensees was issued on April 17, 1985. Comments on the licensees' steam generator programs and the staff's report, NUREG-0844, are due within 60 and 90 days, respectively.

WEST., CE & B&W STEAM GENERATOR TUBE INTEGRITY (A-3, A-4, & A-5)



FINAL REPORT AND SCHEDULE SUPERCEDED BY TASK REPORT — SEE STATUS SUMMARY

SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS (A-17)

AS OF WEEK ENDING MAY 17, 1985

KEY PERSONNEL

TASK MANAGER
DALE THATCHER X29640

Dale Thatcher

NRR ANALYST
JUDY BUTTS X24822

TASK REVIEWERS

NAME	BRANCH
E. CHELLIAH	RRAB/DST
C. MORRIS	RRAB/DST
F. COFFMAN	RRAB/DST

SCHEDULED COMPLETION

1978 ANNUAL REPORT: Phase I - 09-79

CURRENT: 05-30-86

• PROBLEM DESCRIPTION

The design of a nuclear power plant is accomplished by groups of engineers and scientists organized into engineering disciplines and into scientific disciplines. The reviews performed by the designers include interdisciplinary reviews to assure the functional compatibility of the plant structures, systems, and components. Safety reviews and accident analyses provide further assurance that system functional requirements will be met. These reviews include failure mode analyses.

The NRC review and evaluation of safety systems is accomplished in accordance with the Standard Review Plan (SRP) which assigns primary and secondary review responsibilities to organizational units arranged by plant systems or by disciplines. Each element of the SRP is assigned to an organizational unit for primary responsibility and, where appropriate, to other units for secondary responsibilities.

Thus, the design and analyses by the plant designers, and the subsequent review and evaluation by the NRC staff take into consideration the interdisciplinary areas of concern and account for systems interaction to a large extent. Furthermore, many of our regulatory criteria are aimed at controlling the risks from systems interactions. Examples include the single failure criterion and separation criteria.

Nevertheless, there is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems.

The problem to be resolved by this task is to identify where the present design, analysis, and review procedures may not acceptably account for potentially adverse systems interaction and to recommend the regulatory action that should be taken.

• RES INTERFACE INFORMATION

The Division of Risk Analysis has been consulted during the development and execution of this plan.

• ACRS INTERFACE INFORMATION

A meeting with the combined ACRS Subcommittees on Reliability and Risk Assessment and Extreme External Phenomena was held on 03/13/83 to describe the status of the program.

A meeting with the ACRS Subcommittee on Probabilistic Assessment was held on July 6, 1983. Subsequently, the ACRS wrote a letter critical of the staff program.

A meeting was held on November 19, 1983 with the full committee for the purpose of discussing the revised staff program.

\# An ACRS Subcommittee meeting was held on November 14, 1984 for the purpose of discussing the status of the A-17 program. It is anticipated that an additional ACRS meeting will be requested by the staff to discuss the draft technical resolution.

• TECHNICAL ASSISTANCE CONTRACTS

LLNL - LLNL performed a systems interaction review of a portion of the Indian Point-3 plant using the Digraph Matrix method.

BNL - BNL performed a systems interaction review of a portion of the Indian Point-3 plant using Fault Tree combined with a Failure Mode and Effect Analysis.

ORNL - ORNL reviewed a number of information sources (including LERs) to gather information on experienced and hypothesized system interaction events. From this information, an evaluation was made to establish trends and patterns among the events.

ORNL - ORNL is also investigating search methods which could be used to uncover system interaction events. A draft letter report was submitted January 17, 1986.

ORNL - ORNL is estimating potential risk significance and cost for resolution of adverse systems interactions.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
A-0446	LLNL	$1,000K	$1,000K
A-3725	BNL	$1,000K	$1,000K
# B-0789	ORNL	$ 870K	$ 800K

• POTENTIAL PROBLEMS

\# It is anticipated that the schedule slip approximately 3 month for all remaining milestones.

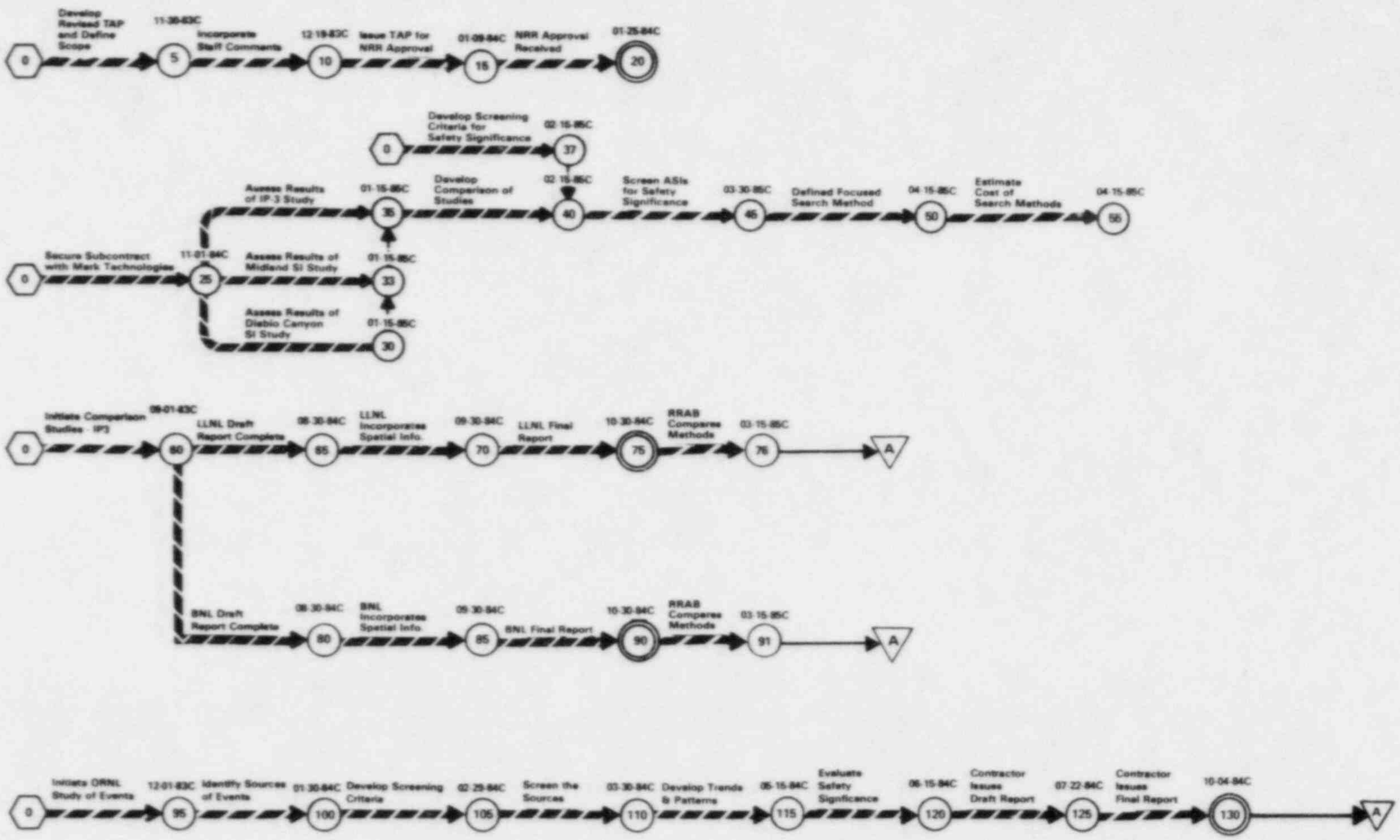
• STATUS SUMMARY

Responsibility for resolution of USI A-17 was transferred to the Generic Issues Branch of the Division of Safety Technology in September 1983 and a full-time Task Manager was assigned. The Task Action Plan has subsequently been revised and has been approved by the Director, NRR.

\# All technical work on A-17 tasks is essentially complete and the resolution package is in draft form. It is expected that the draft package will be circulated for internal review during the month of June.

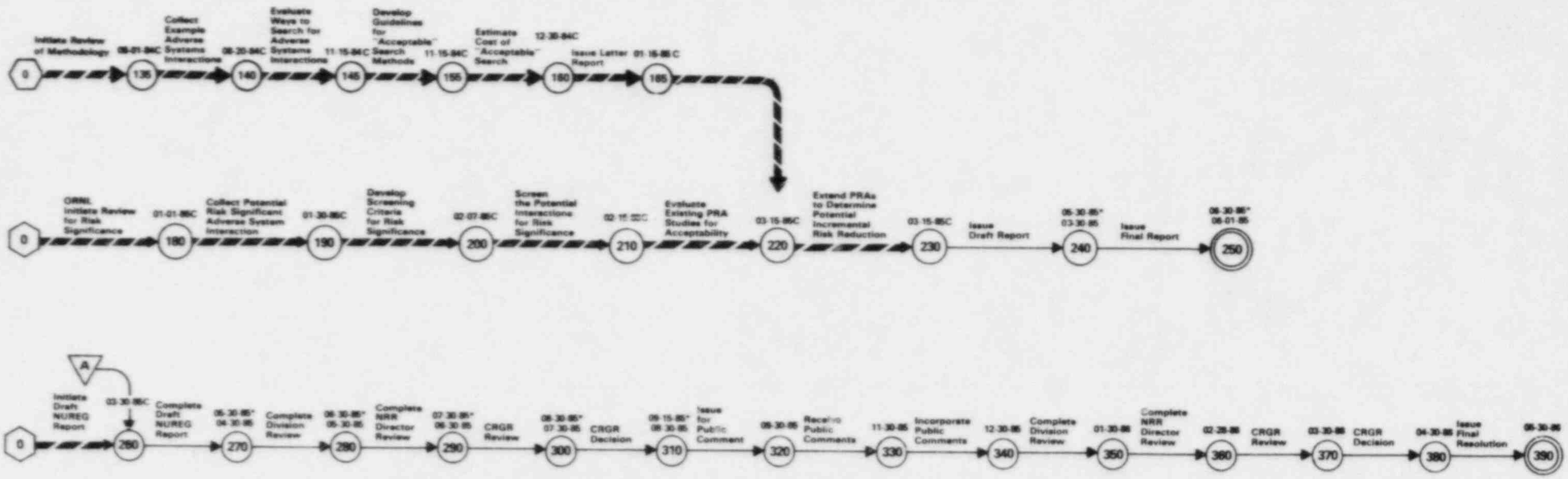
 # INDICATES A CHANGE IN INFORMATION

SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS (A-17) Continued



* Schedule Change This Report.

SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS (A-17) Continued



N/S = Not Scheduled
* Schedule Change This Report

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

AS OF WEEK ENDING MAY 17, 1985

KEY PERSONNEL

TASK MANAGER
SYED SHAUKAT X24216

Syed K. Shaukat

NRR ANALYST
JUDY BUTTS X24822

TASK REVIEWERS

NAME	BRANCH
N. CHOKSHI	SGEB/DE
L. REITER	GSB/DE
P. SOBEL	GSB/DE
G. BAGCHI	EQB/DE
T. CHENG	SEPB/DL

SCHEDULED COMPLETION

1978 ANNUAL REPORT: PHASE I – 1979; PHASE II – 1981

CURRENT: #NOT SCHEDULED

• PROBLEM DESCRIPTION

The seismic design process required by current NRC criteria includes the following sequence of events:

(a) Define the magnitude or intensity of the earthquake which will produce the maximum vibratory ground motion at the site (the safe shutdown earthquake or SSE).
(b) Determine the free-field ground motion at the site that would result if the SSE occurred.
(c) Determine the motion of site structures by modifying the free-field motion to account for the interaction of the site structures with the underlying foundation soil.
(d) Determine the motion of the plant equipment supported by the site structures.
(e) Compare the seismic loads, in appropriate combination with other loads, on structures, systems, and components important to safety, with the allowable loads.

While this seismic design sequence includes many conservative factors, certain aspects of the sequence may not be conservative for all plant sites. At present, it is believed that the overall sequence is adequately conservative. The objective of this program is to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified. In this manner, this program will provide additional assurance that the health and safety of the public is protected, and if possible, reduce costly design conservatisms by improving (1) current seismic design requirements, (2) NRR's capability to quantitatively assess the overall adequacy of seismic design for nuclear plants in general.

• RES INTERFACE INFORMATION

None.

• ACRS INTERFACE INFORMATION

None.

• TECHNICAL ASSISTANCE CONTRACTS

Lawrence Livermore National Laboratory (LLNL), under contract to RES, reviewed all reports by 04-30-79. LLNL report on recommendations for changes to the seismic design criteria was completed on 12-28-79. (NUREG/CR-1161).

LLNL has performed the value/impact analysis on proposed requirements developed from the A-40 technical findings. LLNL report was completed and issued in August 1984 as NUREG/CR-3480.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
A-0441	LLNL	$135 K	$135 K

• POTENTIAL PROBLEMS

The remaining milestones will be scheduled upon approval of the CRGR package by the Acting Director/DE.

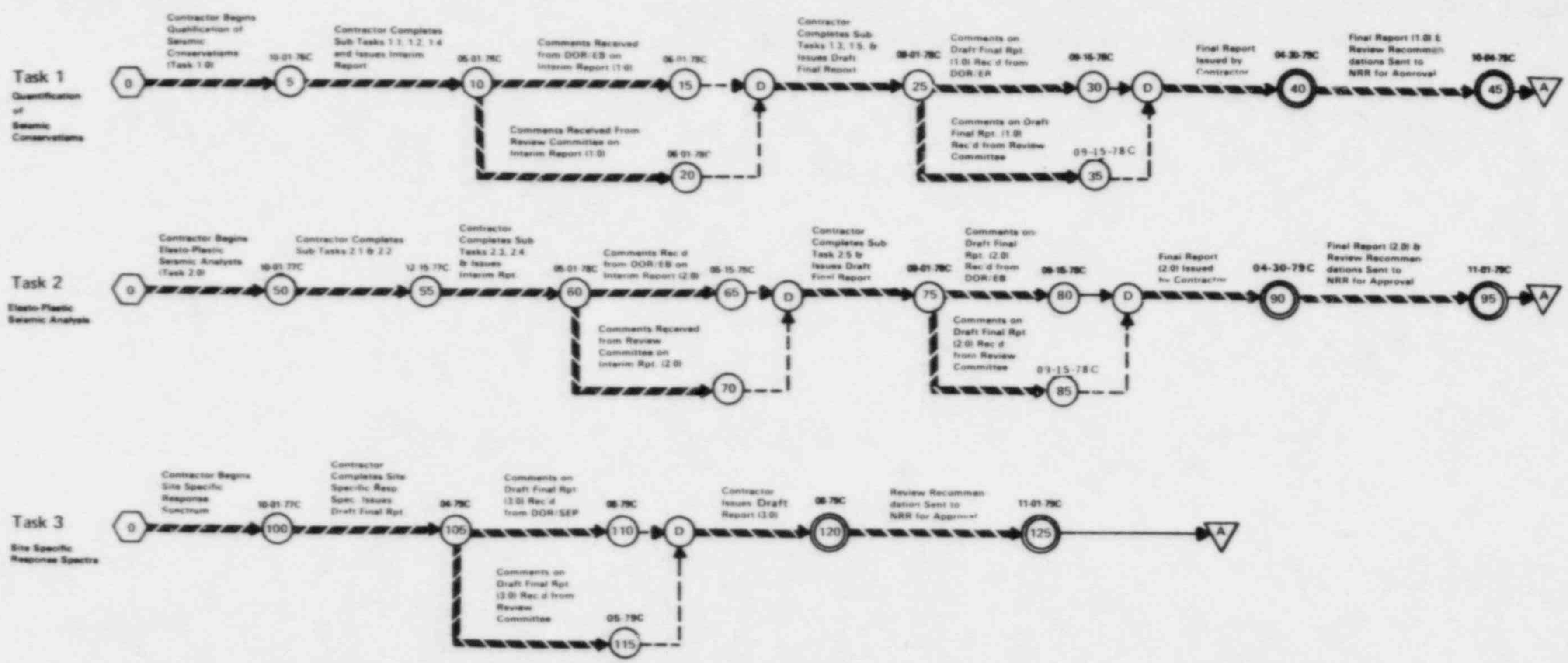
• STATUS SUMMARY

The CRGR package is under review by the Acting Director of Engineering.

#INDICATES A CHANGE IN INFORMATION

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

Phase 1

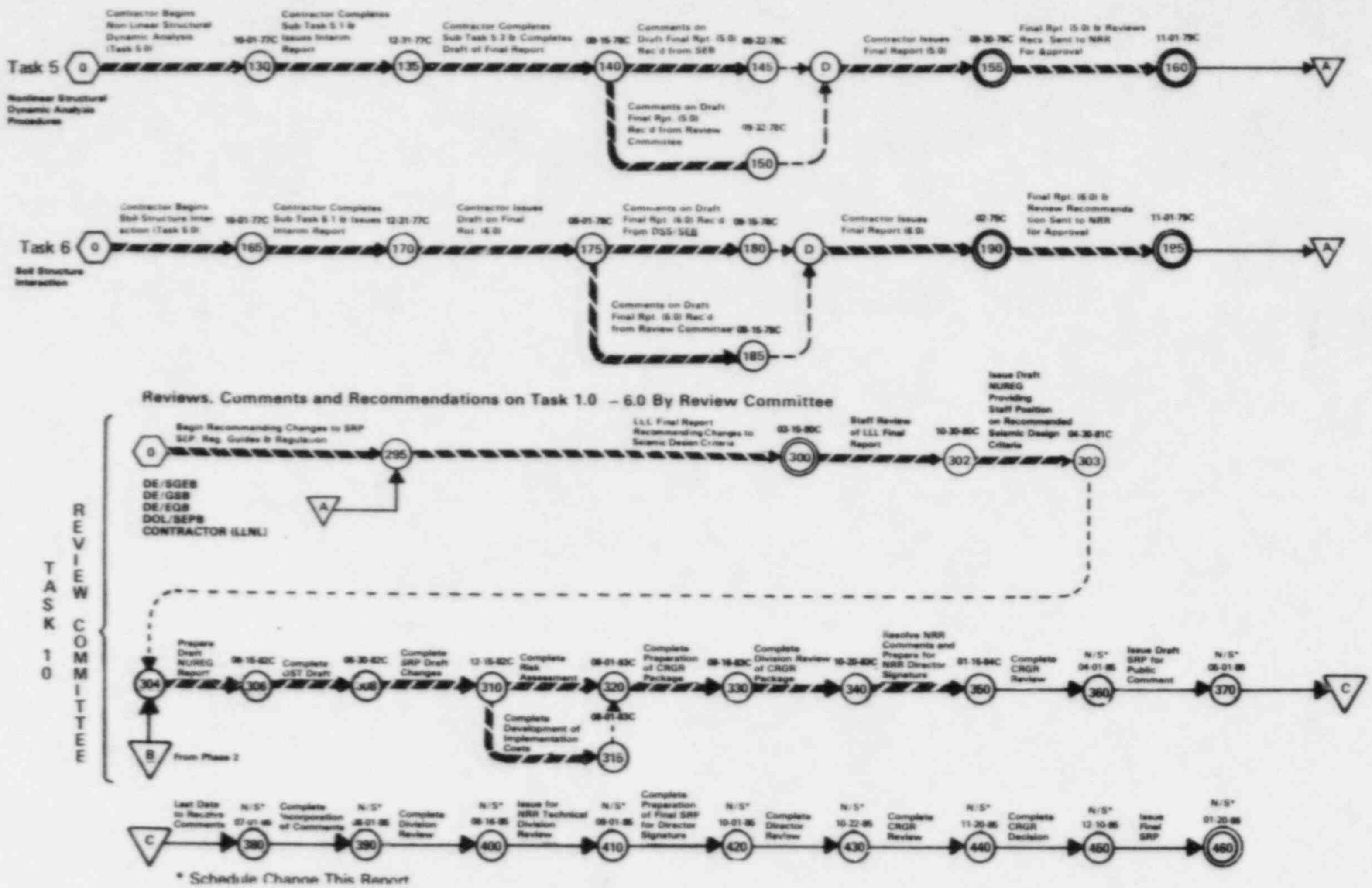


Task 4 Deleted

Seismic After Shocks

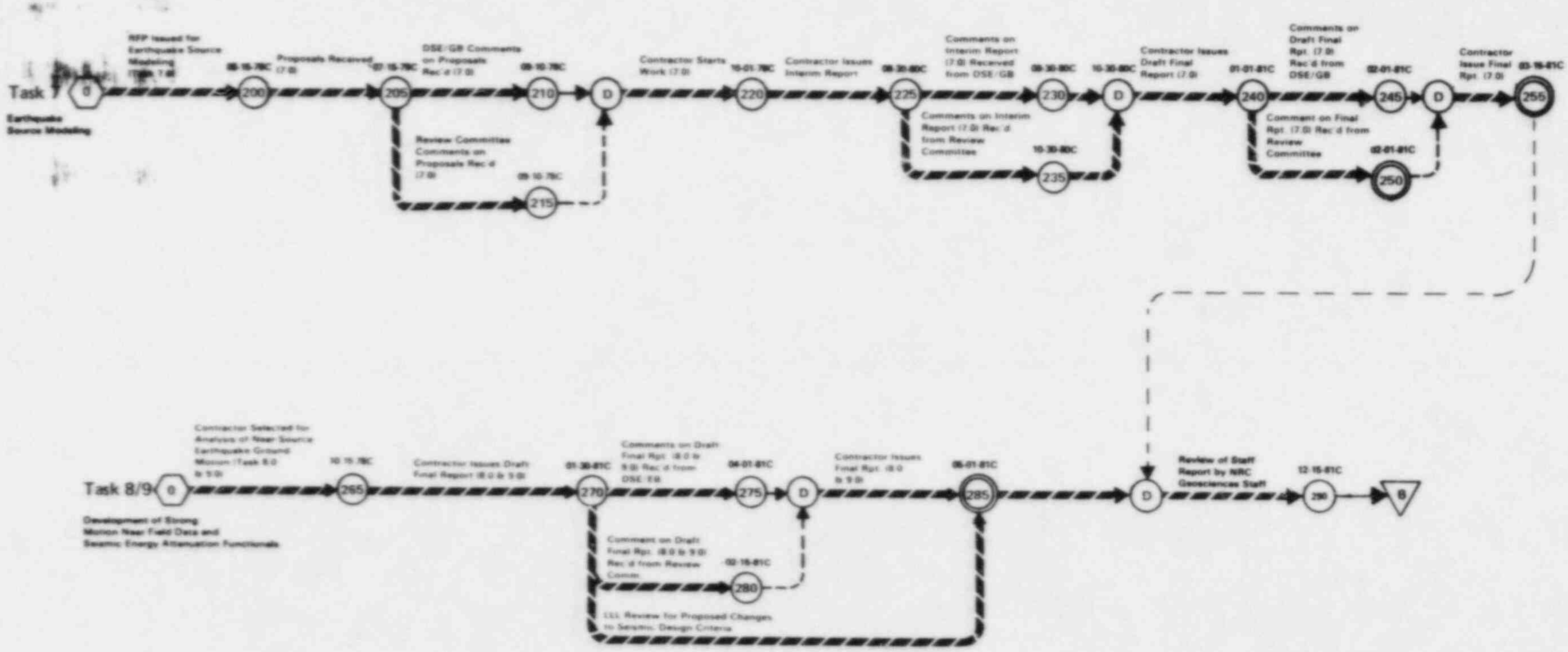
SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)
CONTINUED

(Phase 1 Cont.)



Phase 2

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)
CONTINUED



CONTAINMENT EMERGENCY SUMP PERFORMANCE (A-43)

AS OF WEEK ENDING MAY 17, 1985

KEY PERSONNEL

TASK MANAGER

ALECK W. SERKIZ X24217

AWSerkiz

NRR ANALYST

JUDY BUTTS X24822

TASK REVIEWERS

NAME	BRANCH
S. DIAB	RSB/DSI
J. KUDRICK	CSB/DSI

SCHEDULED COMPLETION

ORIGINAL April 1982

CURRENT 12-31-85

• PROBLEM DESCRIPTION

Following a Loss of Coolant Accident (LOCA) in a PWR, water flowing from the break in the primary system would collect on the floor of containment. During the injection mode, water for core cooling and containment spray is drawn from a large supply tank. When the water reached a low level in the tank, pumps are realigned to draw from the containment. This is called the recirculation mode wherein water is drawn from the containment floor or sump and pumped to the primary system or containment spray headers. This program addresses the safety issue of adequate sump or suppression pool function in the recirculation mode. It is the objective of this program to develop improved criteria for design, testing, and evaluation which will provide better assurance that emergency sumps will function to satisfy system requirements.

The principal concerns are somewhat interrelated but are best discussed separately. One deals with the various kinds of insulation used on piping and components inside of containment. The concern being that break-initiated debris from the insulation could cause blockage of the sump or otherwise adversely affect the operation of the pumps, spray nozzles, and valves of the safety systems.

The second deals with the hydraulic performance of the sump as related to the hydraulic performance to safety systems supplied therefrom. Preoperational tests have been performed on a number of plants to demonstrate operability in the recirculation mode. Adverse flow conditions have been encountered requiring design and procedural modifications to eliminate them. These conditions, air entrainment, cavitation, and vortex formation, are aggravated by blockage. If not avoided or suppressed, they could result in pump failure during the long term cooling phase following a LOCA.

The concerns relative to debris, blockage, and hydraulic performance also apply to boiling water reactors during recirculation from the suppression pools, and will also be addressed.

• RES INTERFACE INFORMATION

None; USI A-43 being managed by the Generic Issues Branch (GIB).

• ACRS INTERFACE INFORMATION

The ACRS has been periodically briefed on the resolution status of A-43 and the comments received, etc. Committee members raised questions regarding BWR recirculation pump bearings and seals ingesting particulates and the effect thereof. In addition, the Committee cautioned against hasty or generalized application of the leak-before-break concept to other issues without a very thorough analysis and review. ACRS views have been incorporated into revised documents.

• TECHNICAL ASSISTANCE CONTRACTS

FIN No. A1237, "Containment Emergency Sump Performance", and FIN No. A1296, "Technical Assistance for Resolution of USI A-43", are being funded by RES and NRR respectively. This work is managed by the GIB Task Manager and these combined efforts are expected to be concluded in FY 84.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

\# FIN Nos. A-1237 and A-1296 have been closed.

• POTENTIAL PROBLEMS

Further reviews by CRGR could lead to additional assessments, rework and schedule slippages.

• STATUS SUMMARY

The regulatory analysis has been revised to reflect comments received from the 07/11/84 CRGR meeting and followup staff evaluations.

\# INDICATES A CHANGE IN INFORMATION

CONTAINMENT EMERGENCY SUMP PERFORMANCE (A-43)

USI A-43 RELATED REPORTS:

1. Weigand, G. G., et. al. "A Parametric Study of Containment Emergency Sump Performance," SAND82-0624/NUREG/CR-2758, Sandia National Laboratories, Albuquerque, NM, July 1982.

2. Padmanabhan, M. and Hecker, G. E., "Assessment of Scale Effects on Vortexing, Swirl, and Inlet Losses in Large Scale Pump Models," NUREG/CR-2760, Alden Research Laboratory, Worcester Polytechnic Institute, Holden, MA, June 1982.

3. Padmanabhan, M., "Hydraulic Performance of Pump Suction Inlet for Emergency Core Cooling Systems in Boiling Water Reactors," NUREG/CR-2772, Alden Research Laboratory, Worcester Polytechnic Institute, Holden, MA, June 1982.

4. "Results of Vertical Outlet Sump Tests," Joint ARL/Sandia Report, ARL47-82/SAND82-1286/NUREG/CR-2759, September 1982.

5. Padmanabhan, M., "Results of Vortex Suppressor Tests, Single Outlet Sump Test and Miscellaneous Sensitivity Tests," NUREG/CR-2761, Alden Research Laboratory, Worcester Polytechnic Institute, Holden, MA, September 1982.

6. Reyer, R., et. al. "Survey of Insulation Used in Nuclear Power Plants and the Potential for Debris Generation," NUREG/CR-2403, Burns and Roe, Inc., Oradell, NJ, October 1981.

7. Kolbe, R., and Gahan, E., "Survey of Insulation Used in Nuclear Power Plants and the Potential for Debris Generation," NUREG/CR-2403, Supplement 1, Burns and Roe, Inc., Oradell, NJ, May 1982.

8. Wysocki, J. J. et. al. "Methodology for Evaluation of Insulation Debris," NUREG/CR-2791, Burns and Roe, Inc., Oradell, NJ, September 1982.

9. Kamath, P., Tantilo, T., and Swift, W., "An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions," NUREG/CR-2792, Creare, Inc., Hanover, NH, September 1982.

10. Brocard, D. N., "Buoyancy, Transport and Head Loss of Fibrous Reactor Insul...on," NUREG/CR-2982, SAND82-7205, November 1982.

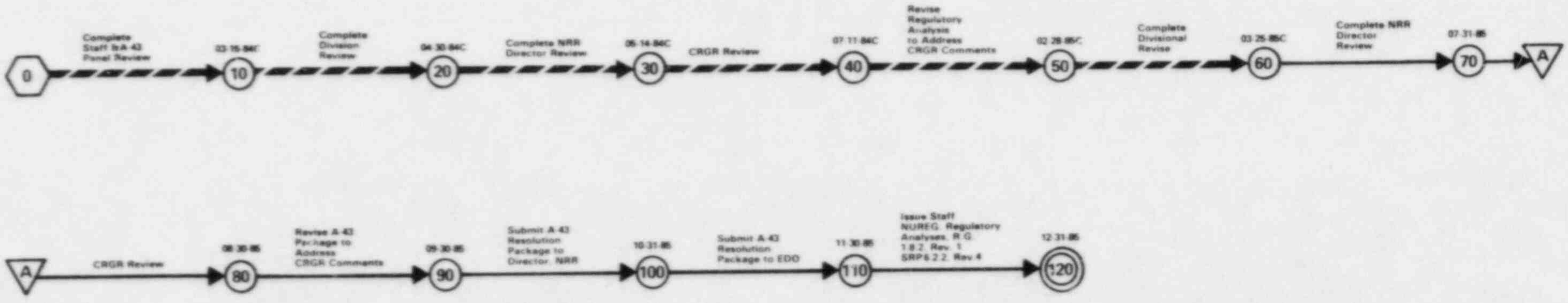
11. Durgin, W. W., Noreika, J., "The Susceptability of Fibrous Insulation Pillows to Debris Formation Under Exposure to Jet Flows," NUREG/CR-3170, SAND83-7008, March 1983.

12. Wysocki, J.J., "Probabilistic Assessment of Recirculation Sump Blockage Due to Loss-of-Coolant Accidents," NUREG/CR-3394, SAND83-7116, Volumes 1 and 2, July 1983.

13. Brocard, D. N., "Transport and Screen Blockage Characteristics of Reflective Metallic Insulation Materials," NUREG/CR-3616, SAND83-7471, ARL-124-83/M398F January 1984.

14. NUREG-0897, For Comment, "Containment Emergency Sump Performance, Technical Findings Related to Unresolved Safety Issue A-43," April 1983.

15. NUREG-0869, For Comment, "USI A-43 Resolution Positions," April 1983.



N/S = Not Scheduled

STATION BLACKOUT (A-44)

AS OF WEEK ENDING MAY 17, 1985

KEY PERSONNEL

TASK MANAGER
ALAN RUBIN X28303

Alan M. Rubin

NRR ANALYST
JUDY BUTTS X24822

TASK REVIEWERS

NAME	BRANCH
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L. ENGLE	ORB/DL
O. CHOPRA	PSB/DSI
D. LANGFORD	RSB/DSI
A. BUSLIK	RRAB/DST

SCHEDULED COMPLETION

ORIGINAL JUNE 1982

CURRENT # 12-30-86

• PROBLEM DESCRIPTION

Electric power for safety systems at nuclear power plants is supplied by two redundant and independent divisions. Each of these electrical divisions includes an offsite alternating current (A.C.) source, an onsite A.C. source (usually diesel-generators), and a direct current (D.C.) source. Appendix A to 10 CFR 50 defines a total loss of offsite power as an anticipated occurrence, and as such, it is required that an independent emergency onsite power supply be provided at nuclear power plants.

The unlikely, but possible loss of A.C. power (that is, the loss of A.C. power from the offsite source and from the onsite source) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems which do not require A.C. power supplies, and on the ability to restore A.C. power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event and that the consequences of this event may be unacceptable, for example, severe core damage may result.

• RES INTERFACE INFORMATION

RES is providing technical assistance for the resolution of A-41.

• ACRS INTERFACE INFORMATION

Station Blackout is related to a number of ACRS concerns regarding the reliability of power
systems. A number of presentations have been made to the ACRS as work on USI A-44 has progressed.

The most recent presentation on the current status, updated technical findings, and proposed resolution of USI A-44 was made to the ACRS Subcommittee on Electrical Systems on February 26, 1985. The Subcommittee presented its report to the full ACRS at a meeting on March 8, 1985. In a letter dated March 12, 1985, from the ACRS to the EDO, the ACRS reaffirmed its support of the staff's proposed resolution of USI A-44.

• TECHNICAL ASSISTANCE CONTRACTS

ORNL FIN B0744 $749K –
Evaluate expected frequency and duration of offsite (preferred) power losses at nuclear power plants.

Estimate the reliability and evaluate the dominant factors affecting the reliability of emergency A.C. power supplies.

Perform statistical correlation and trend analysis of diesel generator data.

NUREG/CR-2989, "Reliability of Emergency AC Power Systems at Nuclear Power Plants," was published in July 1983.

NUREG/CR-3992, "Collection and Evaluation of Complete and Partial Losses of Offsite Power at Nuclear Power Plants," was published in February 1985.

SNL FIN A1302 $300K –
Evaluate the risks posed by station blackout accidents and assess the effectiveness of safety improvements in reducing those risks.

Evaluate risk reduction and costs of various fixes and to provide input for value/impact analysis.

NUREG/CR-3226, "Station Blackout Accident Analyses (Part of NRC Task Action Plan A-44)," was published in May 1983.

	FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED*
#	B0744	ORNL	$749K	$746K
	A1302	SNL	$300K	$294K

*As of March 31, 1985

• POTENTIAL PROBLEMS

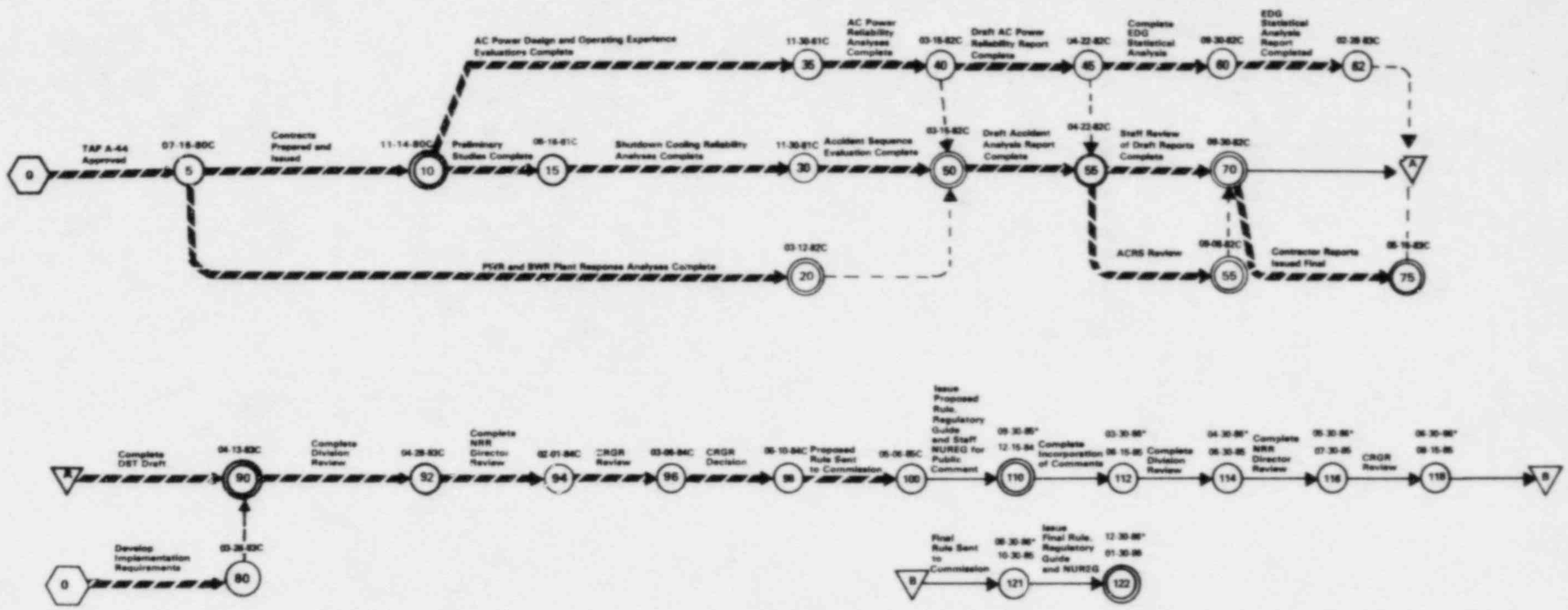
• STATUS SUMMARY

The staff's proposed recommendations to resolve A-44 based on the technical findings, were reviewed by NRR and RES divisions. This review resulted in the recommendation to proceed with proposed rulemaking, in conjunction with a new Regulatory Guide, to resolve A-44. The proposed technical resolution has been reviewed by the Director, NRR, and forwarded to CRGR. Meetings were held with CRGR in March and April 1984 to review the proposed resolution. CRGR recommended that the proposed rule, the proposed Regulatory Guide and the draft staff NUREG-1032 be issued for public comment after making modifications to reflect CRGR comments.

The technical basis and recommendations were revised based on updated data on loss of offsite power experienced at nuclear power plants.
These revisions are included in the proposed rulemaking package which was sent to the Commission on May 6, 1985. NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to USI A-44," has been published for public comment. A Nuclear Utility Group on Station Blackout submitted in May 1985 a proposal to NRC for the resolution of USI A-44. This proposal is being reviewed by the staff.

# INDICATES A CHANGE IN INFORMATION

STATION BLACKOUT (A-44)



NOTE: Milestone 75 – Accident Analysas Report and EDG Reliability Report were published in 1983. The Draft Loss of Offsite Power Report was published in July 1984, and the final report was published in February 1985.

* Schedule Change This Report.

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45)

AS OF WEEK ENDING MAY 17, 1985

KEY PERSONNEL

TASK MANAGER

ANDREW MARCHESE X24712

A.R. Marchese

NRR ANALYST

JUDY BUTTS X24822

TASK REVIEWERS

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# C. LIANG	RSB/DSI
F. ROSA	ICSB/DSI
M. SRINIVASAN	PSB/DSI
E. McPEEK	SSPB/DL
D. DiIANNI	ORB-4/DL
M. CUNNINGHAM	DRA/RES
R. FRAHM	RRAB/DST
P. HEARN	ASB/DSI

SCHEDULED COMPLETION

ORIGINAL 10-30-85

CURRENT 02-28-86

NOTE: A schedule change is under development.

• PROBLEM DESCRIPTION

Task A-45 was approved as a USI by the NRC in December 1980.

Although many improvements to the steam generator auxiliary feedwater system were required of the reactor manufacturers by the NRC following the TMI-2 accident, the staff feels that providing an alternative means of decay heat removal could substantially increase the plants' capability to deal with a broader spectrum of transients and accidents and potentially could, therefore, significantly reduce the overall risk to the public. Consequently, Task A-45 will investigate alternative means of decay heat removal in PWR plants, including but not limited to using existing equipment where possible. This Unresolved Safety Issue will also investigate the need and possible design requirements for improving reliability of decay heat removal systems in boiling water reactors (BWRs).

The overall purpose of Task A-45 is to evaluate the adequacy of current licensing design requirements, in order to ensure that nuclear power plants do not pose an unacceptable risk due to failure to remove shut-down decay heat. The objective will be to develop a comprehensive and consistent set of shutdown cooling requirements for existing and future LWRs, including the study of alternative means of shutdown decay heat removal and of diverse "dedicated" systems for this purpose.

The main objectives of the program are as follows:

- Determine the safety adequacy of decay heat removal systems in existing power plants for achieving both hot shutdown and cold shutdown conditions.
- Evaluate the feasibility of alternative measures for improving decay heat removal systems, including diverse alternatives dedicated to the decay heat removal function.
- Assess the value and impact of the most promising alternative measures.
- Develop a plan for implementing any new licensing requirements for decay heat removal systems.

The interrelation and relative timing of each of the program sub-tasks are shown on the schedule network.

• RES INTERFACE INFORMATION

Close coordination and cooperation will be required on Task A-45 between NRR and RES. RES assistance will be required from the Divisions of Risk Analysis and Accident Evaluation. The Division of Risk Analysis will provide technical input from their Sandia Laboratory Program on Alternate Decay Heat Removal Concepts, technical evaluations relative to reliability and risk assessment for shutdown decay heat removal systems, and input from Task A-44, "Station Blackout," relative to shutdown cooling systems. The Division of Accident Evaluation will provide technical input relative to the transient response of existing and improved shutdown decay heat removal systems to transient events and small LOCAs. This will also include performing (in-house, contractors) detailed thermal-hydraulics analyses where required to support existing and improved decay heat removal systems behavior under transient and accident conditions.

• ACRS INTERFACE INFORMATION

The Task Manager participated in briefing the Full Committee on November 2, 1984 on the results of an NRC team visit through five European countries to discuss their approach to decay heat removal systems and plant protection against sabotage.

- Task Manager briefed the full committee on August 9, 1984 on the overall status of USI A-45.
- ACRS Subcommittee on Decay Heat Removal Systems (DHRS) met on January 22, 1985, and Task Manager provided the subcommittee with an update of the USI A-45 program.
- Further meetings with the full committee and subcommittee on DHRS will be held as the work on USI A-45 progresses and certain pre-determined milestones are completed.

• TECHNICAL ASSISTANCE CONTRACTS

Implemented a technical assistance contract on May 10, 1982 with Sandia (FIN A1308) to provide overall project management, technical direction and integration for the entire Task A-45 program, including selection and management of subcontractors.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
# A1308	Sandia	$4,658*	$4,504

* Includes the following funding which has been committed to support subcontracting:

#	UCLA	$ 291K
	ORNL	$431K
	B&R	$ 92K
#	LANL	$107K
	ASAI	$ 143K
#	SAI	$ 288K
	AE Support	$384K
	DHR Tech.	
#	Support	$ 92K
	SEA	$ 30K
#	SAI/JRB	$ 210K
	SMA	$ 227K

• POTENTIAL PROBLEMS

\# The following are the major schedular problems that
\# have recently confronted the USI A-45 program:

- Plant visits (9 sites) have taken longer to arrange than estimated.
- Individual plant fault tree and event tree analyses are requiring more detail (support systems, containment systems, recovery actions) and taking longer to complete (based on updated plant information) than originally estimated.
- Information required to perform an integrated special emergency evaluation (fire, flood, seismic and sabotage) has increased significantly and has been difficult to obtain.
- Time for Sandia to place supporting subcontracts (based on competitive bids) has taken longer than anticipated.
- Staff review of interim milestone reports has required more time than originally estimated.
- Additional Commission, ACRS, and staff comments and concerns have added to the scope of the A-45 program.

• STATUS SUMMARY

\# Plant visits for the purpose of obtaining missing information relative to DHR system analyses have taken place at Point Beach, Turkey Point, Quad Cities, Arkansas Nuclear No. 1, Trojan, Cooper and St. Lucie. Arranging for the site visits took longer than originally estimated.

Responses have been prepared to: (1) Chairman Palladino's memorandum of August 21, 1984 requesting that we address ACRS comments contained in a memorandum to W.J. Dircks, dated August 14, 1984; and (2) Commissioner Roberts' questions on the A-45 program contained in his memorandum to W.J. Dircks, dated September 7, 1984.

During this reporting period, a foreign trip report was issued on January 15, 1985 in connection with visits to certain European countries to discuss decay heat removal systems and related sabotage issues.

Because of the above schedular problems, the near-term A-45 plan of approach is as follows:

- Hold existing schedule
- Develop A-45 resolution package based on two plant analyses
- Complete balance of plant analyses on adjusted schedule (8 month extension)
- Modify resolution package as necessary based on results of additional plant analyses.

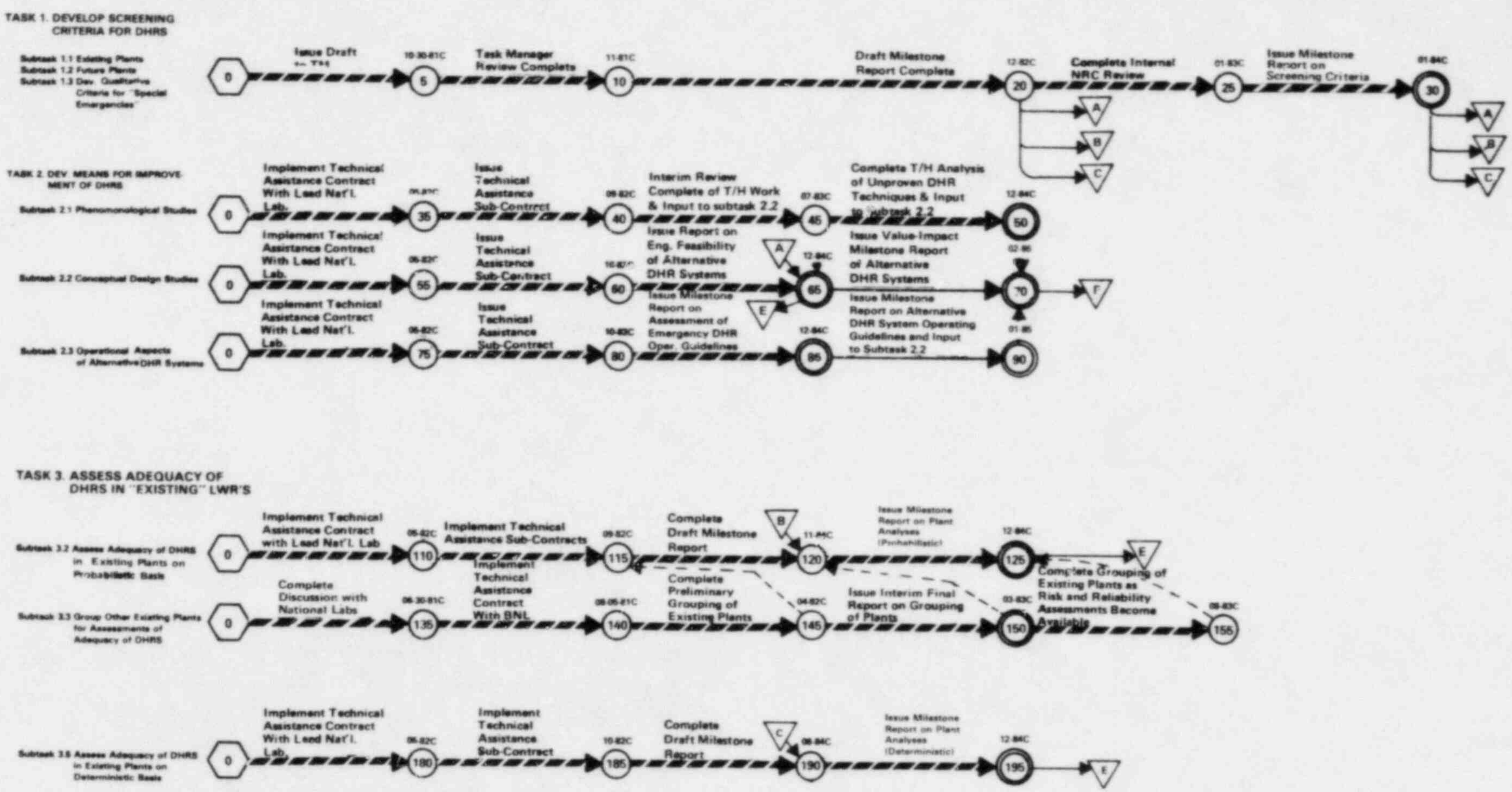
\# The following two USI A-45 program Internal Reports have been placed in the Public Document Room (PDR):

- "Qualitative Screening Questions for Examining Light Water Reactor Decay Heat Removal Capabilities," Sandia National Laboratories, January 1984 (Revised June 1984)
- "Preliminary Assessment of Decay Heat Removal (DHR) Capability of Operating and Soon-To-Be-Operating Light Water Reactors," Oak Ridge National Laboratory September 1984

\# Complete plant analyses draft reports for the Point Beach and Quad Cities facilities were issued in March 1985, and an internal staff review is being performed.

# INDICATES A CHANGE IN INFORMATION

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45)



SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45)
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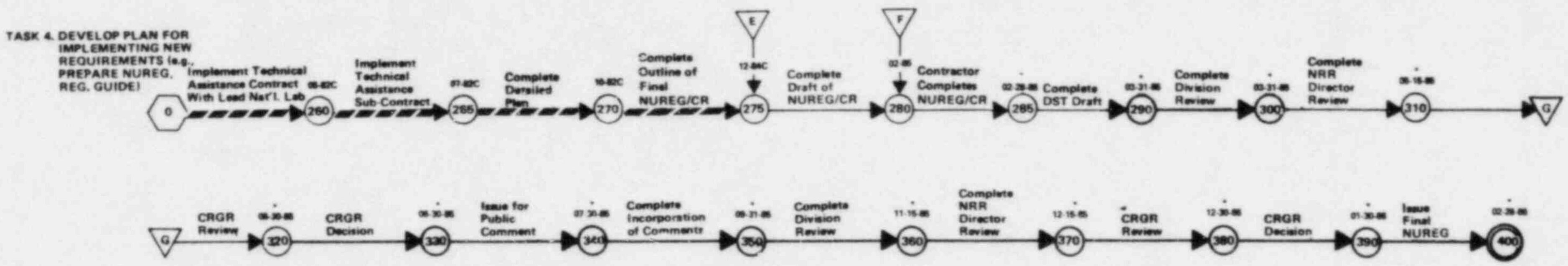


* A request for schedule change is under development.

SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)

AS OF WEEK ENDING MAY 17, 1985

KEY PERSONNEL

TASK MANAGER
T.Y. CHANG X27486

NRR ANALYST
JUDY BUTTS X24822

TASK REVIEWERS

NAME	BRANCH
ARNOLD LEE	FOB/DE
PEI-YING CHEN	SEPB/DL
JOHN KNOX	PSB/DSI
FRANK SKOPEC	RAB/DSI
KULIN DESAI	RSB/DSI
HAROLD POLK	SGEB/DE
GUSTAAF GIESE-KOCH	GSB/DE
GERALD WEIDENHAMER	MSEB/RES

SCHEDULED COMPLETION

ORIGINAL 12-15-83

CURRENT 12-30-85

• PROBLEM DESCRIPTION

Task A-46 was approved as a USI by the NRC December, 1980.

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this Unresolved Safety Issue is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shutdown the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions. Also, explicit guidelines will be established for use in requalifying equipment whose seismic qualification was found to be inadequate.

A breakdown of the tasks is as follows:

- Task 1 Identification of Seismic-Sensitive Systems and Equipment
- Task 2 Assessment of Adequacy of Existing Seismic Qualification
- Task 3 Development and Assessment of In-Situ Testing Methods to Assist in Qualification of Equipment
- Task 4 Seismic Qualification of Equipment Using Seismic Experience Data
- Task 5 Development of Methods to Generate Generic Floor Response Spectra
- Task 6 Document Results of USI A-46 and Prepare Final Report

• RES INTERFACE INFORMATION

Part of a RES contract with Southwest Research Institute (SWRI) is concerned with developing methodology to correlate various seismic qualification tests and is designated Task 2 for A-46. This work is essentially complete. SWRI issued related reports in June and November 1983.

• ACRS INTERFACE INFORMATION

The status of A-46 was presented to the ACRS Subcommittee in March 1983. In March and July 1983, respectively, the Seismic Qualification Utility Group (SQUG) also made two seperate presentations to the ACRS Subcommittee and Full Committee on their pilot program to establish the feasibility of using seismic experience data in conventional power plants to demonstrate the adequacy of similar equipment installed in operating nuclear power plants. The ACRS, in their comments, indicated that the SQUG apporoach was in line with the ACRS recommendations made in January 1983 and should be encouraged. However, ACRS believes that more work is required to establish the operability of equipment during and after an earthquake, and more data will be required to support conclusions drawn concerning the seismic resistance of the equipment investigated. The status of A-46 was presented to the ACRS Subcommittee again in March and April 1984, and to the ACRS Full Committee in May 1984.

• TECHNICAL ASSISTANCE CONTRACTS

Tasks 1 and 5 were performed by Brookhaven National Laboratory and are essentially complete. NUREG/CR-3357 on Task 1 was issued in June 1983. NUREG/CR-3266 on Task 5 was issued in September 1983. A draft guideline on Task 1 was issued in September 1983.

Task 3 was performed by Idaho National Engineering Laboratory and is now complete. NUREG/CR-3875 on Task 3 was issued in June 1984.

Task 4 has been studied independently by Lawrence Livermore National Laboratory (LLNL) and by the Seismic Qualification Utility Group (SQUG). Results of the LLNL study were published in NUREG/CR-3017 dated August 1983. In addition, Sandia National Laboratories (SNL) is providing assitance in (1) editing and publishing SSRAP report, and (2) resolving public comments.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
A0423	LLNL	$75K	$75K
A6474	INEL	$286K	$283K (est)
A3387	BNL	$324K	$320K
# A1318	SNL	$ 50K	$ 25K

• POTENTIAL PROBLEMS

Delay in completing the CRGR review will result in a schedule slip of approximately 2 months.

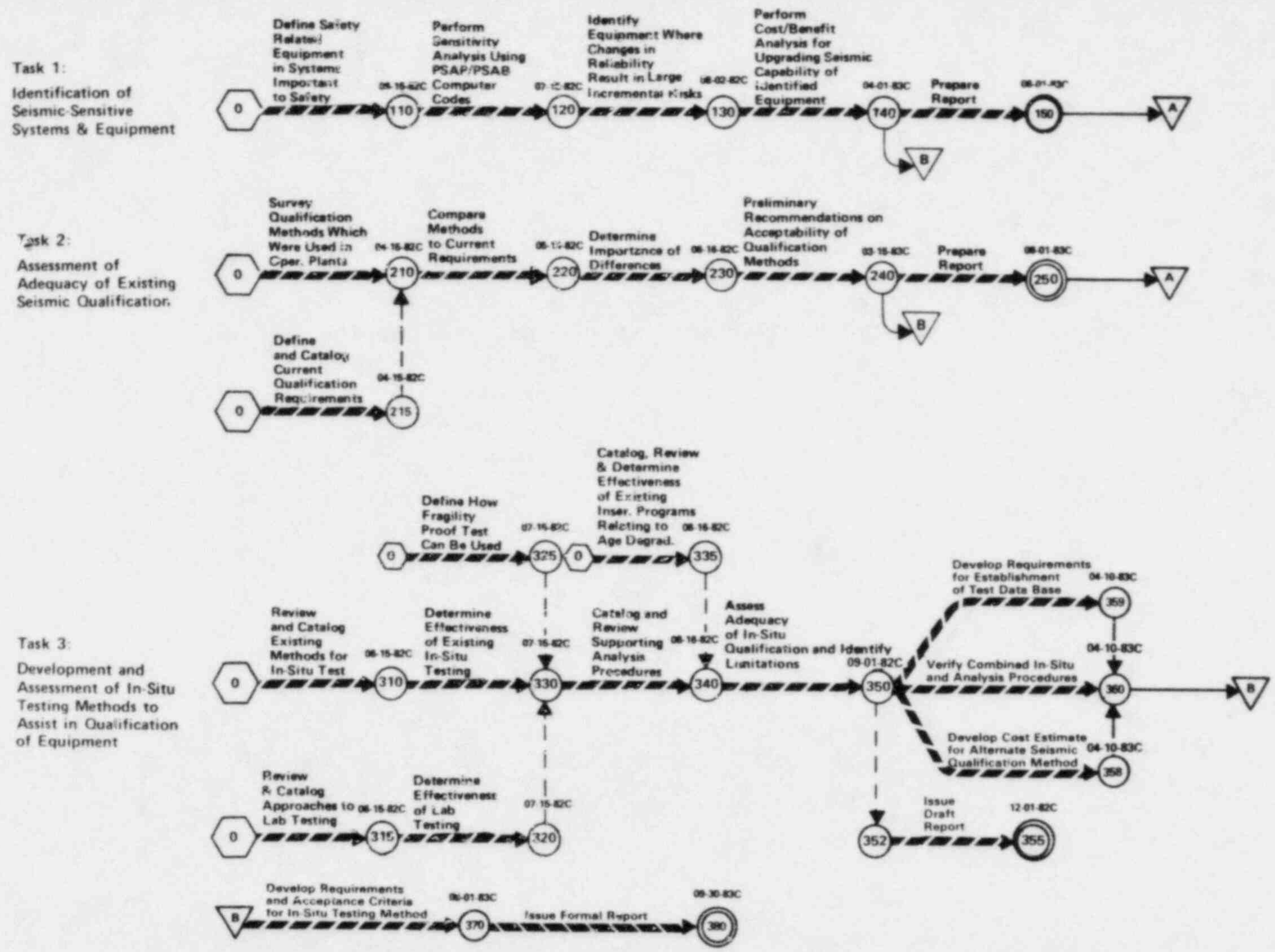
• STATUS SUMMARY

Work on all tasks is essentially completed by the contractors with the exception of Task 4. The SQUG formed an independent Senior Seismic Review and Advisory Panel (SSRAP) in June 1983 to make recommendations for use of seismic experience data. The NRC staff has been working very closely with the SQUG and the SSRAP and will continue to do so. This activity is an important element in the resolution of A-46. The A-46 schedule was approved by NRC management on July 18, 1983. SSRAP issued its report in February 1984 and then updated it in August 1984.

The USI A-46 CRGR package was approved by the Director of NRR on October 31, 1984 and sent to CRGR for review on November 1, 1984. A meeting was held with CRGR On December 3, 1984. The CRGR decided they did not have enough information to make a decision. The CRGR package is being revised and additional information incorporated prior to
re-submitting for CRGR review in May 1985.

# INDICATES A CHANGE IN INFORMATION

SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)



SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)

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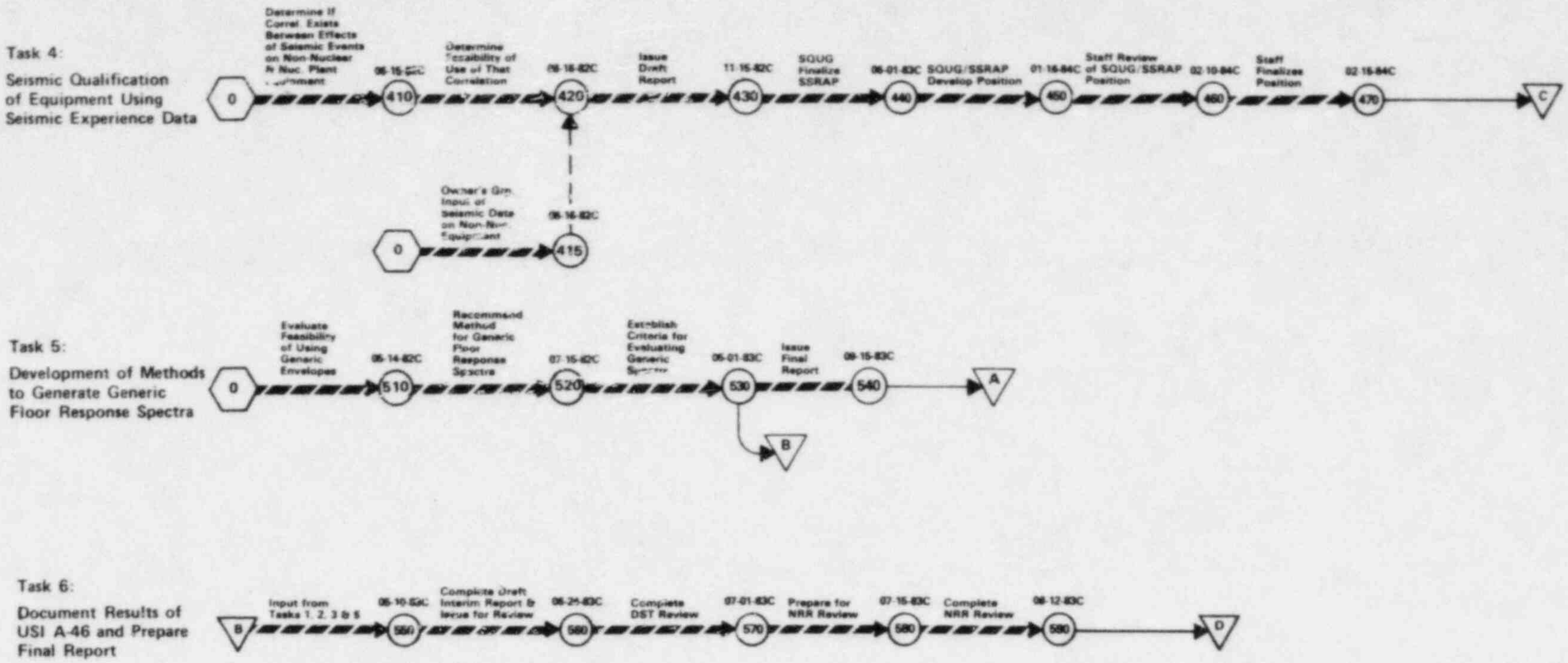
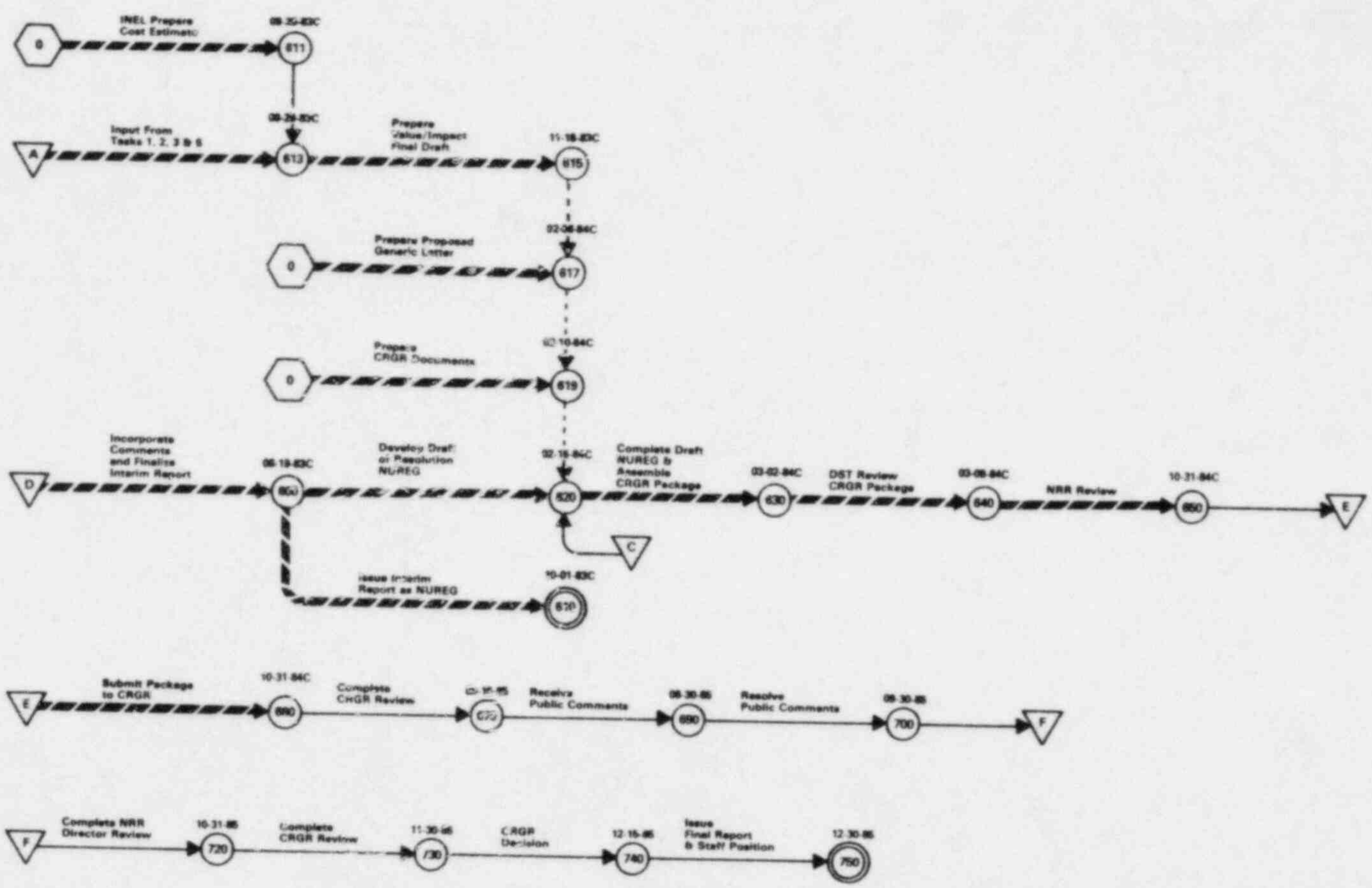


SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)
CONTINUED

Task 6
(Continued)



SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)

AS OF WEEK ENDING MAY 17, 1985

KEY PERSONNEL

TASK MANAGER
ANDREW SZUKIEWICZ X24713

Szukiewicz

NRR ANALYST
JUDY BUTTS X24822

TASK REVIEWERS

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# J. MAUCK	ICSB/DSI
E. CHELLIAH	RRAB/DST
# E. LANTZ	RSB/DSI
M. CHIRAMAL	PSU/AEOD
J. T. BEARD	ORAB/DL
W. KENNEDY	PTRB/DHFS

SCHEDULED COMPLETION

ORIGINAL 01-30-86

CURRENT 04-01-86

• PROBLEM DESCRIPTION

Task A-47 was approved as a USI by the NRC in December 1980.

Instrumentation and control systems utilized by nuclear plants are composed of safety grade protection systems and non-safety grade control systems. Safety grade systems are used to (1) trip the reactor when specified parameters exceed allowable limits; and (2) protect the core from overheating by initiating ECCS systems. Non-safety grade control systems are used to maintain the plant within prescribed parameters during shutdown, startup and normal load varying power operation. Non-safety grade systems are not relied on to perform any safety functions during or following postulated accidents, but are used to control plant processes. Although non-safety grade control system failures are not likely to result in accidents or transients that could lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed. Concerns have been identified in which a failure or malfunction of the non-safety grade control system can (1) potentially cause steam generator or reactor vessel overfill; or (2) can lead to a transient that could cause severe vessel overcooling. In addition, there is the potential for control system failures to result in plant conditions which may result in unacceptable risk.

The purpose of USI A-47 is to perform a systematic evaluation of non-safety grade control systems that are typically used during normal plant operations and to identify control systems whose failure could (1) cause transients or accidents identified in the FSAR analysis to be potentially more severe than previously analyzed; (2) adversely affect any assumed or anticipated operator action during the course of an event; (3) cause technical specification safety limits to be exceeded; or (4) cause transients or accidents to occur at a frequency in excess of those established for abnormal operational transients and design basis accidents. Specific subtasks of this issue are to study steam generator/reactor vessel overfill and overcool transients to determine the need for preventive and/or mitigating design measures. The objective of this USI is to evaluate the need for requiring control systems changes in operating reactors and to verify the adequacy of licensing requirements and to propose, if needed, additional criteria and guidelines.

• RES INTERFACE INFORMATION

Close coordination is required on USI A-47 between NRR and RES. RES assistance is required from the Division of Engineering Technology and the Division of Risk Analysis and Operations.

The Division of Engineering Technology through contract assistance with ORNL will provide final evaluation reports of their studies of non-safety grade control system failures for B&W and CE reference plant designs.

The Division of Risk Analysis through contract assistance with PNL will provide the risk analyses and value impact analyses associated with significant control system failure scenarios identified during the review of each of the four reference plant designs.

• ACRS INTERFACE INFORMATION

The ACRS Subcommittee on Electrical Power Systems (Dr. Kerr) met on the issue on 1/23/81 to initiate an approximate six month study at the request of Chairman Ahearne.

Status of the activities identified in TAP A-47 was discussed with the ACRS Subcommittee on
December 21, 1982, November 16, 1983, June 1, 1984 and November 14, 1984.

• TECHNICAL ASSISTANCE CONTRACTS

A Technical Assistance (T.A.) contract with ORNL (FIN B-0467) was established through NRR/RES. RES will conduct a review of two different types of PWR designs (one B&W plant, Oconee, and one CE plant, Calvert Cliffs), and perform the activities
identified in Tasks 1 through 7 of Task Action Plan A-47. A separate T.A. contract to perform the review on one BWR type design, Browns Ferry, and one PWR type design, (Westinghouse) H.B. Robinson, was established with INEL (FIN A-6477). The technical assistance contracts will perform the following tasks:

1. **Evaluate Control System Failures Leading to Steam Generator and/or Reactor Overfill Transients**
2. **Evaluate Control System Failures Leading to Reactor Overcooling Transients**
3. # **Evaluate (All Others) Control System Failures That Have Safety Implications**
4. **Evaluate the Effects of Loss of Power to the Control Systems**

A technical assistance contract with INEL and CREARE was established (FIN No. A-6477) to estimate the water hammer potential in steam lines from steam generator overfill events and to develop probability estimates of steam line damage due to steam generator overfill events (Task 8 of TAP A-47). A technical assistance contract with PNL was established through NRR/RES (FIN No. B-2386) to perform a risk analysis and a value impact analysis on the control system failure scenarios that have been identified by INEL and ORNL as safety significant (Task 9 of TAP A-47).

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
# A-6477	INEL	$1410K	$1365K
B-0467 and # B-0816	ORNL	$3650K	$2950K

• POTENTIAL PROBLEMS

Due to the unavailability of the BG&E simulator for the Calvert Cliffs 1 evaluation, a computer model had to be developed for the Calvert Cliffs design. Schedule slips have incurred in (1) issuing the final draft report on the CE reference plant design and (2) on completing the risk analysis and the value impact analysis for the CE design. It is anticipated that the schedule will slip about 6 months for all remaining milestones.

• STATUS SUMMARY

Draft Report on the Safety Implications of Control Systems of a B&W PWR design was submitted by ORNL in October 1984.

PNL Risk Assessment Draft Report (Rev. 2) on Control System Failures for the General Electric Design was submitted for staff review on February 25, 1985.

PNL Risk Assessment Draft Report (Rev. 2) on Control System Failures for the Westinghouse Design was submitted for draft review on March 22, 1985.

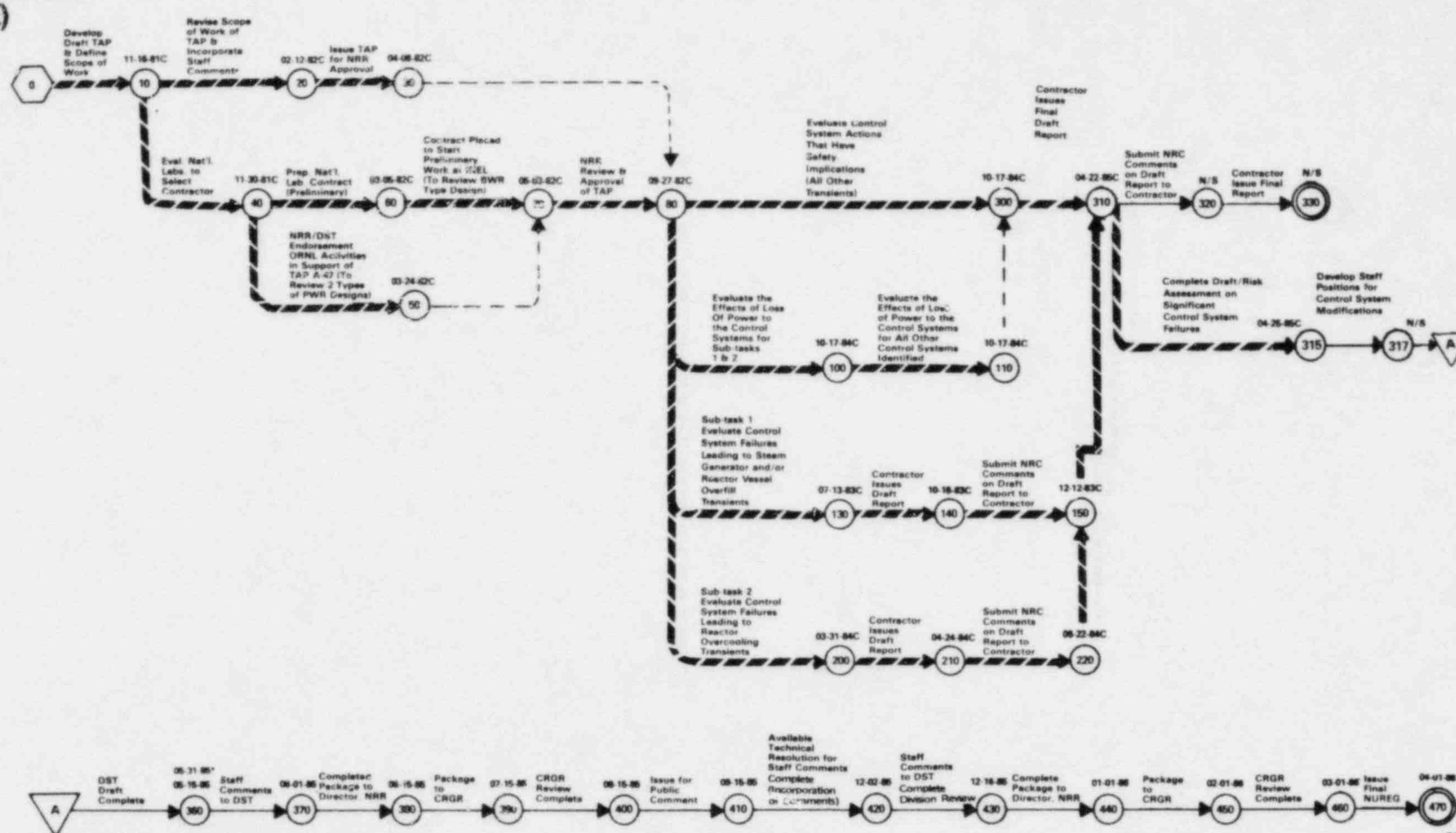
PNL Risk Assessment Draft Report (Rev. 2) on Control System Failures for the Babcock & Wilcox Design was submitted for staff review on April 22, 1985.

A preliminary draft report on the Safety Implications of Control Systems of a CE PWR Design was submitted by ORNL on April 30, 1985. A final draft is scheduled to be submitted on May 31, 1985.

# INDICATES A CHANGE IN INFORMATION

B&W-PWR PLANT REVIEW (OCONEE)

SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)



* Schedule Change This Report.

CE-PWR
PLANT REVIEW
(CALVERT CLIFFS)

SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)

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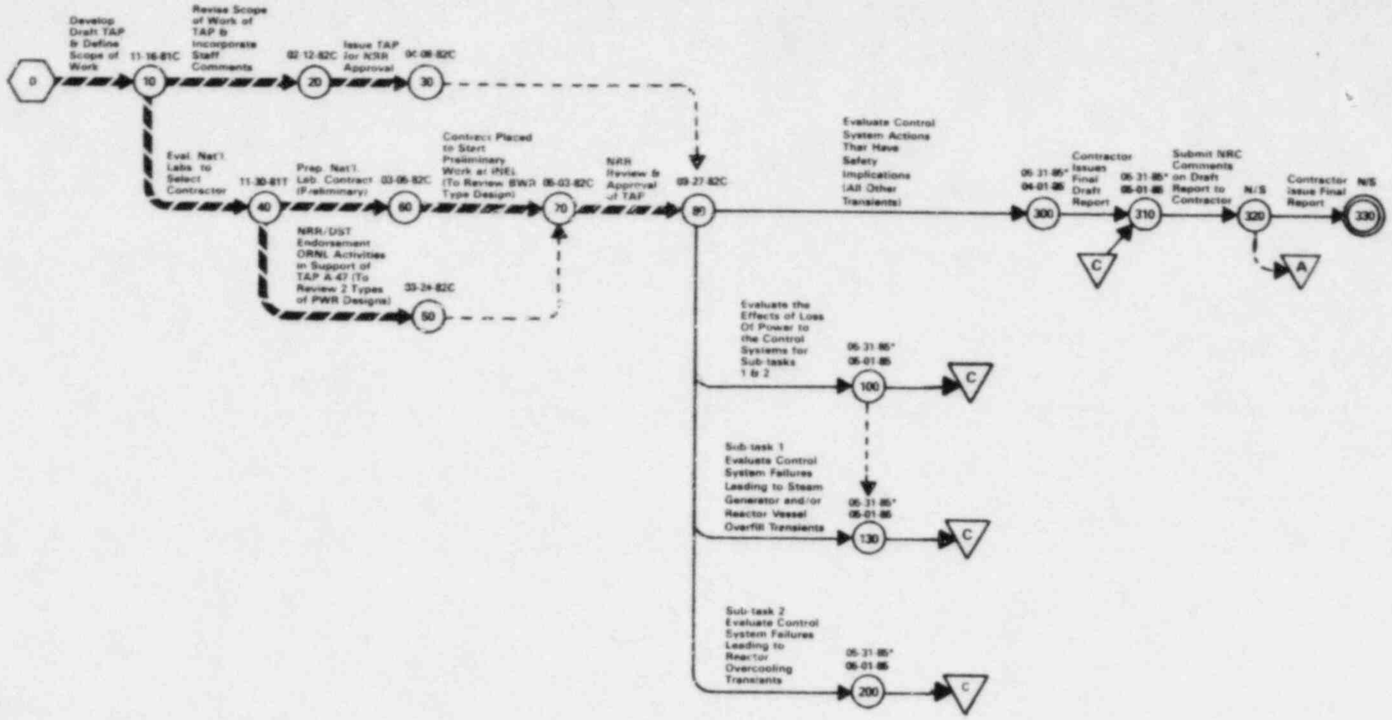


NOTE: This schedule is contingent on the availablity of the BG&E simulator for ORNL use, per agreement between RES and BG&E. Simulator studies are to begin in March 1985.

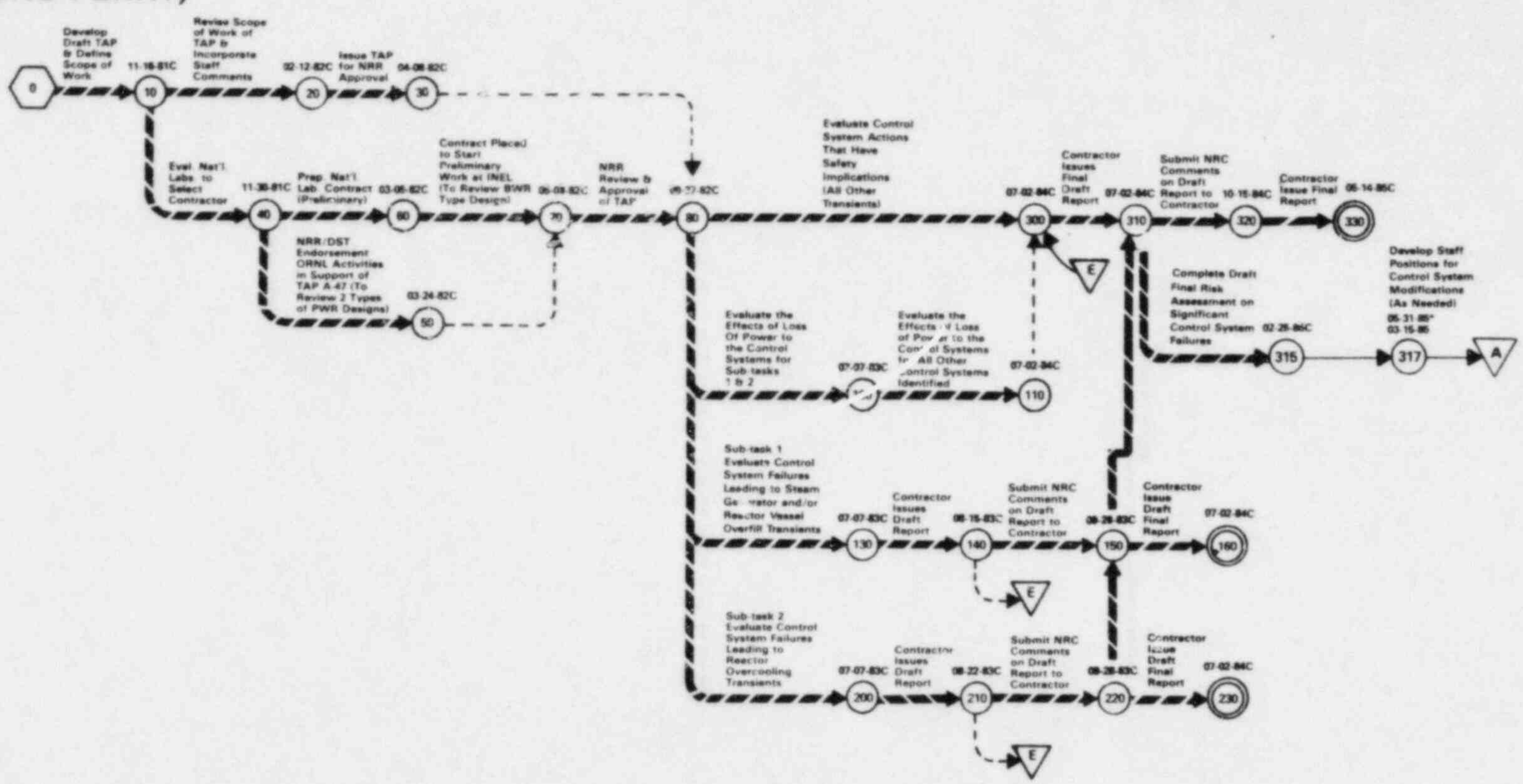
* Schedule Change This Report.

N/S = Not Scheduled.

GE-BWR PLANT REVIEW (BROWNS FERRY)

SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)

CONTINUED



* Schedule Change This Report.

N/S = Not Scheduled.

WESTINGHOUSE-PWR PLANT REVIEW

SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)

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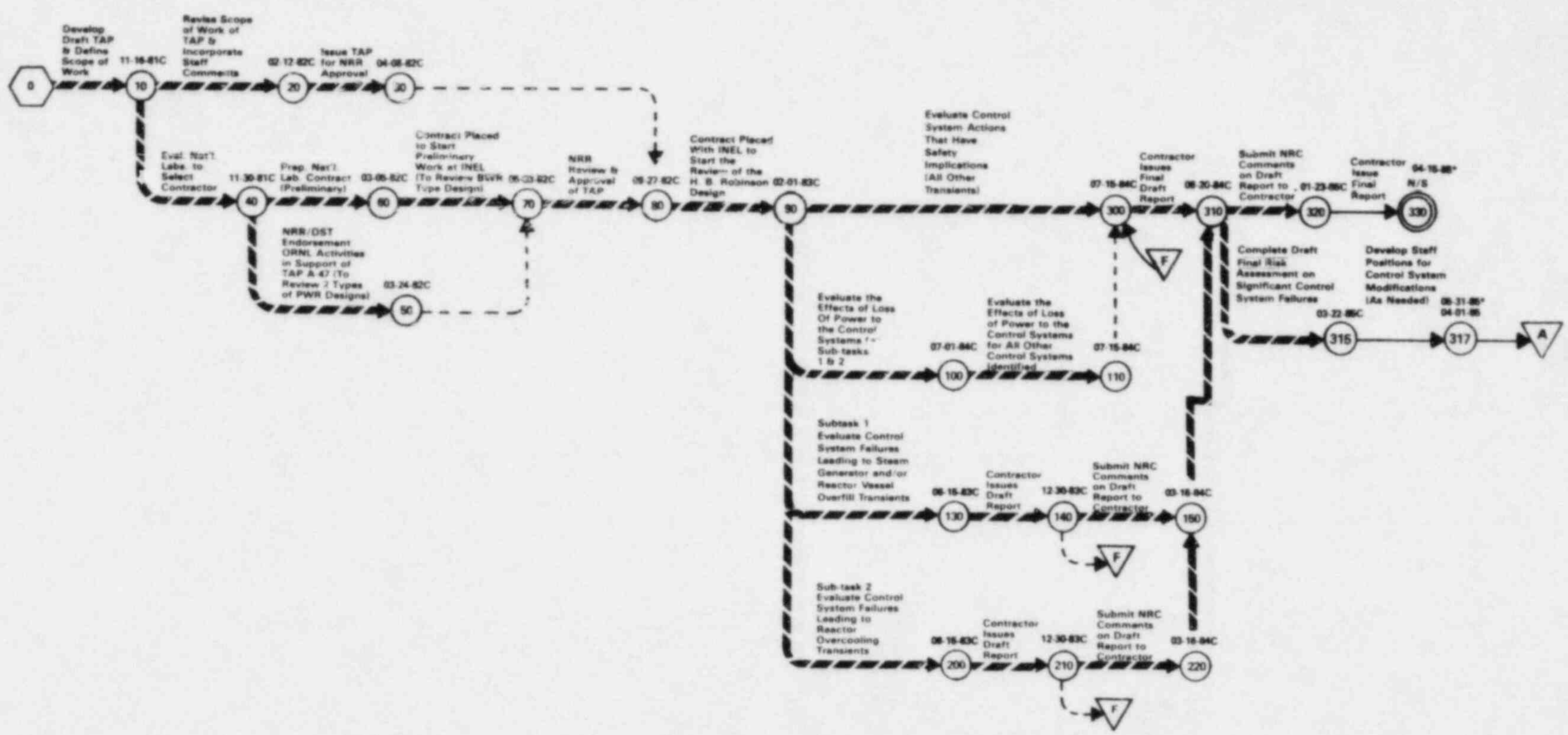


* Schedule Change This Report.

N/S = Not Scheduled.

HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT (A-48)

AS OF WEEK ENDING MAY 17, 1985

KEY PERSONNEL

TASK MANAGER

TSUNG MING SU X27477

NRR ANALYST

JUDY BUTTS X24822

TASK REVIEWERS

NAME	BRANCH
# TOM GREENE	PSRB/DHFS/NRR
CHARLES TINKLER	CSB/DSI/NRR
# JACK ROSENTHAL	RSB/DSI/NRR
HUKAM GARG	EQB/DE/NRR
KRYSZTOF PARCZEWSKI	CEB/DE/NRR
RICHARD CLEVELAND	RSCB/DST/NRR
# CARL STAHLE	DL/NRR
PAT WORTHINGTON	SAB/DAT/RES
MORTON FLEISMAN	RAB/DRA/RES
HAROLD POLK	SGEB/DE/NRR

SCHEDULED COMPLETION

ORIGINAL 06-30-85

CURRENT # 12-30-86

• PROBLEM DESCRIPTION

Task A-48 was approved as a USI by the NRC in December 1980.

Postulated reactor accidents which result in a degraded or melted core can result in generation and release to the containment of large quantities of hydrogen. The hydrogen is formed from the reaction of the zirconium fuel cladding with steam at high temperatures and/or by radiolysis of water. Experience gained from the TMI-2 accident indicates that we may want to require more specific design provisions for handling larger hydrogen releases than currently required by the regulations particularly for smaller, low pressure containment designs.

The scope of this USI is limited to the generic resolution of hydrogen control and equipment qualification for ice condenser and BWR containments, and is based on the licensing case review for these containments.

• RES INTERFACE INFORMATION

There are extensive research programs related to the hydrogen issue sponsored by RES. The results of those research programs will be incorporated into licensing decisions as appropriate.

• ACRS INTERFACE INFORMATION

TO BE DEVELOPED

• TECHNICAL ASSISTANCE CONTRACTS

To be developed when the generic summary report is considered.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

• POTENTIAL PROBLEMS

The state-of-the-art has substantial uncertainties. Therefore, there is a potential for new findings which may impact the current schedule.

The 1/4 scale tests sponsored by the Mark III Owners Group has slipped for several months. Since there are a number of open issues on the test program, the schedule for the tests cannot be established until the open issues are resolved.

• STATUS SUMMARY

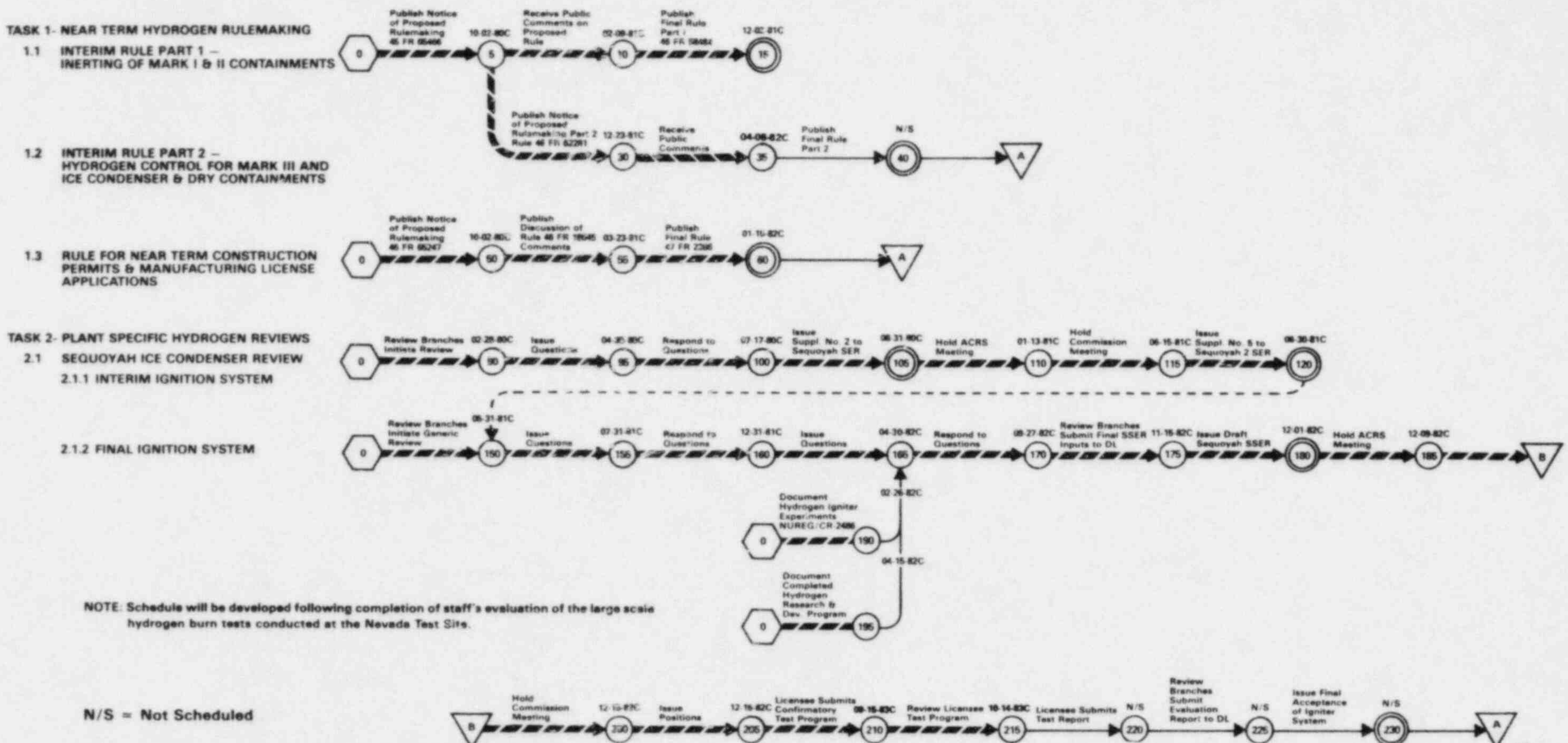
The Task Action Plan (TAP) was approved on 12/03/82, and a detailed schedule has been developed as shown on the following pages.

A Commission Paper regarding hydrogen control for Mark III and ice condenser containments was reviewed and endorsed by CRGR on June 1, 1983. The Commission Paper was forwarded to the Commissioners on August 26, 1983. Additional information was provided on December 28, 1983 to justify the staff position on the
Commission Paper. On January 25, 1985, the Commission issued the hydrogen final rule (50 FR 3498).

The results of the large scale hydrogen burn tests conducted at Nevada Test Site show potential challenge to equipment survivability. The staff's preliminary evaluation of the data indicated that the equipment required for safe shut down will survive following a postulated hydrogen burn. Further evaluation of the data is planned.

# INDICATES A CHANGE IN INFORMATION

HYDROGEN CONTROL MEASURES & EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT (A-48)

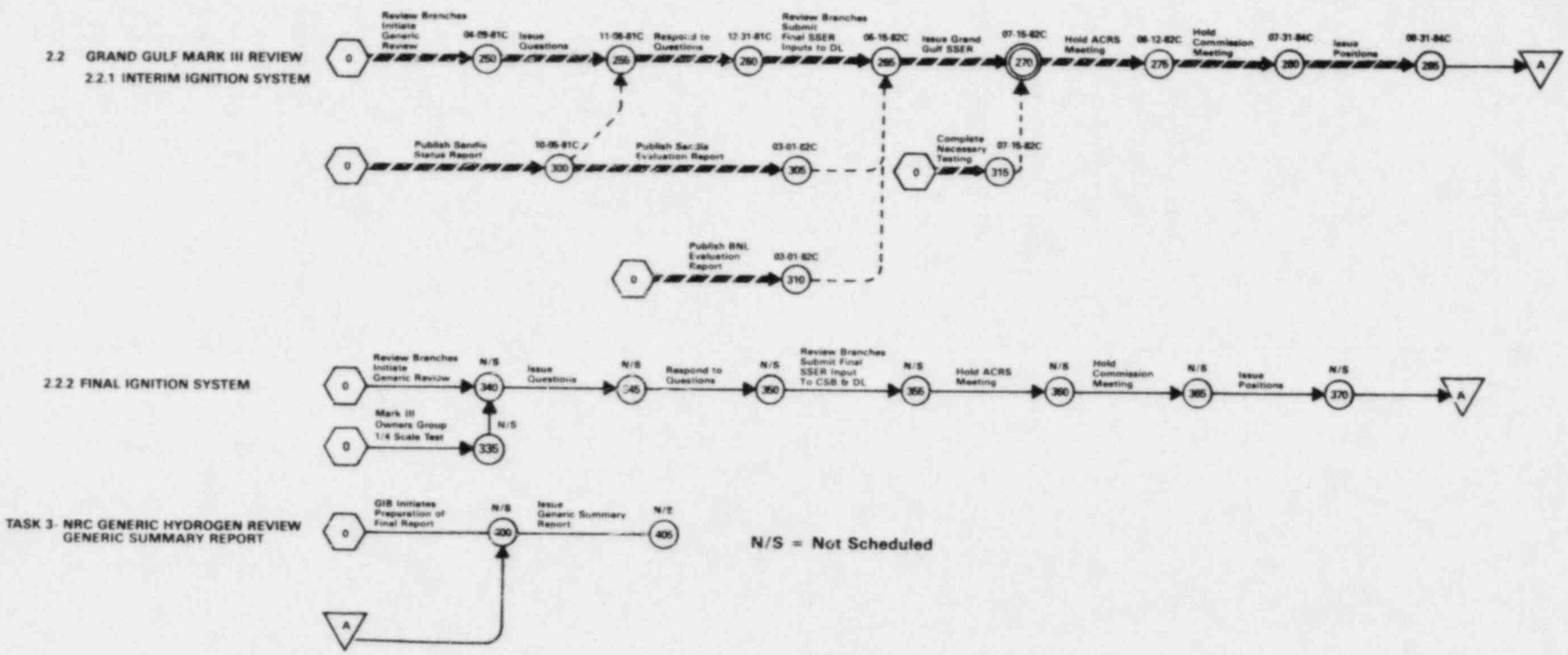


NOTE: Schedule will be developed following completion of staff's evaluation of the large scale hydrogen burn tests conducted at the Nevada Test Site.

N/S = Not Scheduled

HYDROGEN CONTROL MEASURES & EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT (A-48)

CONTINUED



PRESSURIZED THERMAL SHOCK (A-49)

AS OF WEEK ENDING MAY 17, 1985

KEY PERSONNEL

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JUDY BUTTS X24822

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SCHEDULED COMPLETION

ORIGINAL Not Determined

CURRENT 03-31-86

• PROBLEM DESCRIPTION

This task was designated a USI by the NRC in December 1981.

Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The fracture toughness sensitivity to radiation induced change is increased by presence of certain materials such as copper. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessels inner surface, that crack could grow to a size that might threaten vessel integrity.

Severe pressurized overcooling events are improbable since they require multiple failures and improper operator performance. However, certain precursor events have happened that could have potentially threatened vessel integrity if additional failures had occurred and/or if the vessel had been more highly irradiated. Therefore, the possibility of vessel failure due to a severe pressurized overcooling event cannot be ruled out.

• RES INTERFACE INFORMATION

A major portion of the work is being performed under a contract with Oak Ridge National Laboratory through the Division of Risk Analysis, RES (FIN # B0468).

Other major contributors are:

Primary System Integrity Research Program through the Division of Engineering Technology, RES and

Code Applications Program through the Division of Accident Evaluation, RES

• ACRS INTERFACE INFORMATION

Meetings have been held and will be scheduled as necessary with the Subcommittee on Metallic Components and with the full ACRS. The latest Subcommittee meeting was held on May 17, 1984.

The latest ACRS Committee meetings on this subject were held on October 12, 1984 and February 7, 1985.

• TECHNICAL ASSISTANCE CONTRACTS

Contract (B-2510) issued to PNL. PNL will perform sensitivity studies using the VISA code, and investigate vessel failure modes due to PTS.

The following RES contracts are providing technical assistance to the PTS program. These are in addition to the technical assistance contracts which were initiated to specifically address the PTS issue and listed in the table below.

FIN NO.	LAB	DESCRIPTION
G-1047	Purdue	Mixing Calculations
A-4070	Creare	Mixing Experiments
A-3266	BNL	T-H Calculation Comparisons
A-7306	LASL	SOLA Mixing Calculations
A-7315	LASL	TRAC T-H Calculations
A-6047	INEL	RELAP T-H Calculations
B-0468	ORNL	Integrated PTS Study
B-0119	ORNL	HSST Experiments
B-8900	ENSA	Struct. Integrity
B-6290	NSRDC	Spectrum Shape
B-7026	USNA	Rapid J-R Curve
B-8942	Gundremming	
B-2853	PNL	Visa Development
B-5988	HEDL	Dosimetry
B-0415	ORNL	P. V. Simulation
B-6224	NBS	Dosimetry
B-2289	PNL	NDE
B-2467	PNL	NDE
B-2088	PNL	Accoustic

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
#B-2510	PNL	$1031K	$857K
A-7272	LANL	$580K-FY83	$580K
A-3701	BNL	$200K	$200K

• POTENTIAL PROBLEMS

• STATUS SUMMARY

The Task Action Plan for A-49 was approved and issued on March 26, 1982.

NRC Staff PTS recommendations (SECY-82-465) were approved by the Commission in December 1982. A new rule for PTS resolution was published for public comment on February 7, 1984. The proposed final rule, taking the public comments into account, will be submitted to the Commission for approval during March 1985.

The TAP has been extensively revised (Revision 1) to reflect division of the program into two parts, a short term program, "Development of a PTS Rule" and a long term program that will provide "Guidance and Acceptance Criteria for the Analyses to be Required by the PTS Rule." The revised TAP describes issuance of the new rule, confirmatory studies now underway to support the new rule, and the plant-specific analyses and other requirements that will be included in the new rule.

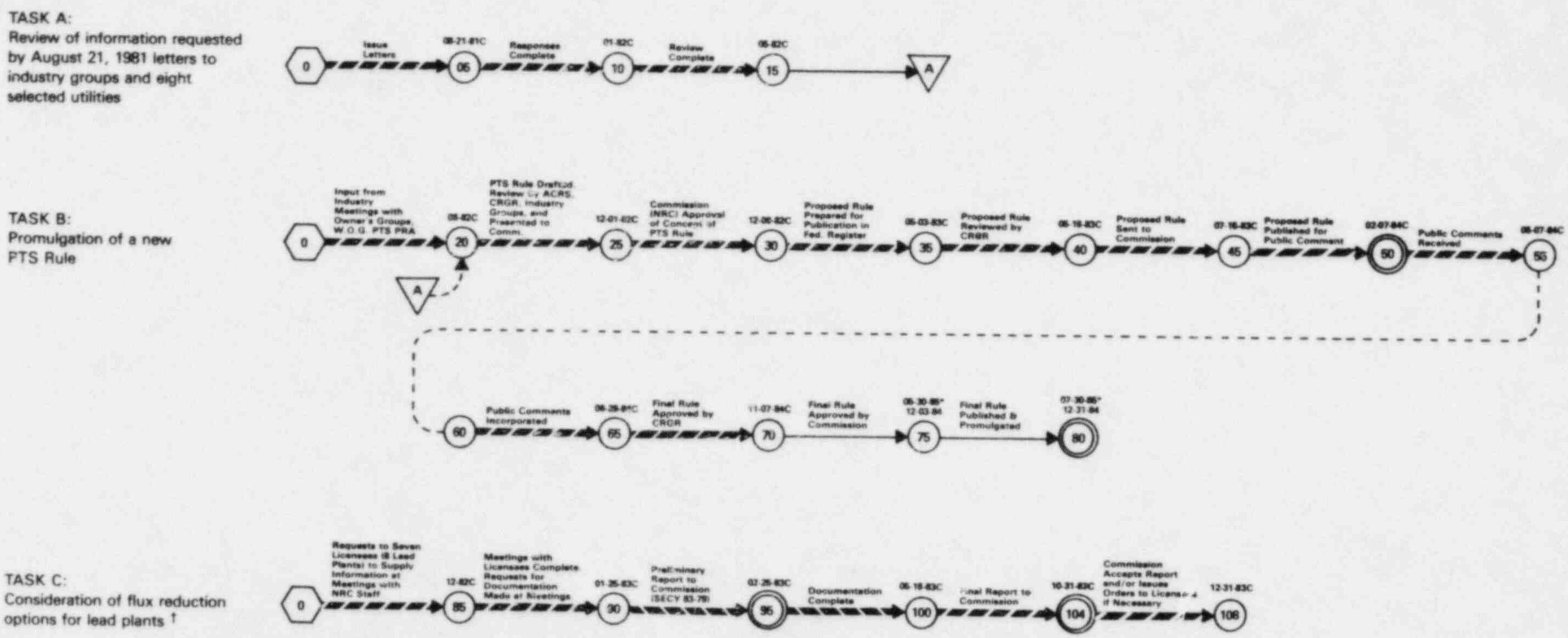
A status report regarding flux reduction efforts was sent to the Commission on February 25, 1983 (SECY-83-79). The final report was submitted on October 28, 1983 (SECY-83-443).

Revision 1 of the TAP as described above was approved on November 22, 1983 by the Director of NRR. Revision 2, containing minor scheduled changes, was submitted to the Director of NRR for approval in March 1984.

# INDICATES A CHANGE IN INFORMATION

PRESSURIZED THERMAL SHOCK (A-49)

SHORT TERM PROGRAM



* Schedule Change This Report

† PTS Rule also requires consideration of flux reduction option for all PWRs. This Task (C) is such consideration in the immediate future to prevent preclusion of this option for the oldest (lead) plants.

Task 1:

Development of a Revised Regulatory Guide 1.99

Draft revision of the trend curves in Reg. Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials": This task is no longer considered to be necessary for completion of A-49. Its scheduled completion is a longer term item than A-49, and adequate guidance regarding this subject is contained within Task (B). A detailed schedule for this task is therefore not presented.

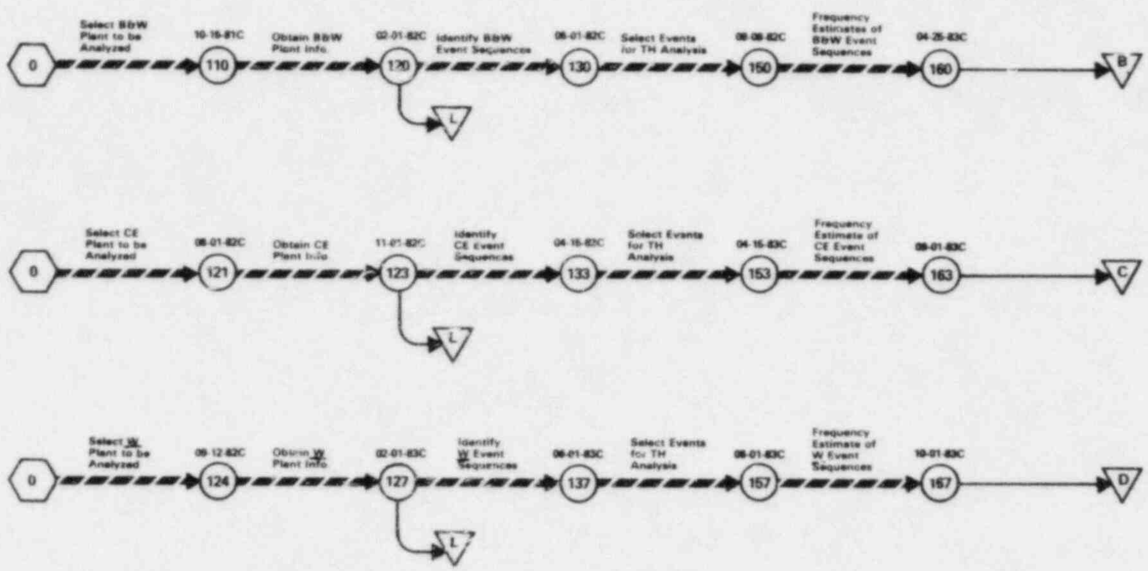
Task 2:

Ongoing Program to Improve Procedures and Operator Training

This program is ongoing separate from the A-49 PTS effort and is much broader than PTS, considering PTS as one of the many types of incidents for which procedures and training should be improved, on a combined/integrated basis. Generic Letter 82-33 contains a description of the overall program and schedule. The PTS effort cannot and should not be separated from the overall effort, and so a detailed PTS schedule is not presented here. The ongoing program will be completed and applied to each plant, however, on a schedule compatible with completion of the final PTS resolution for each plant (i.e., before acceptance of plant specific analyses required by the PTS rule, Task (B) above.)

Task 3:

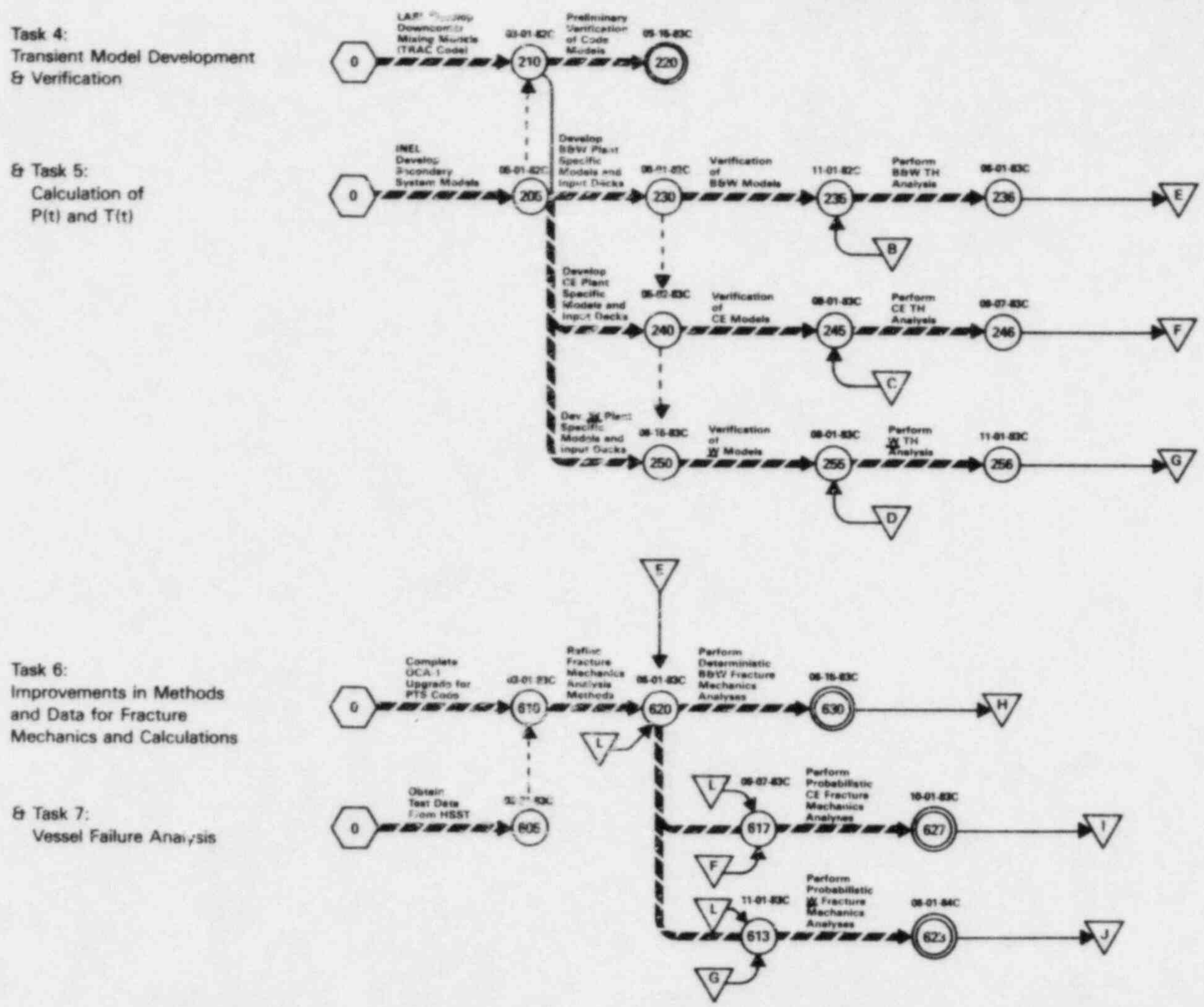
Determination of Event Sequences to be Considered



PRESSURIZED THERMAL SHOCK (A-49)

LONG TERM PROGRAM

(CONTINUED)



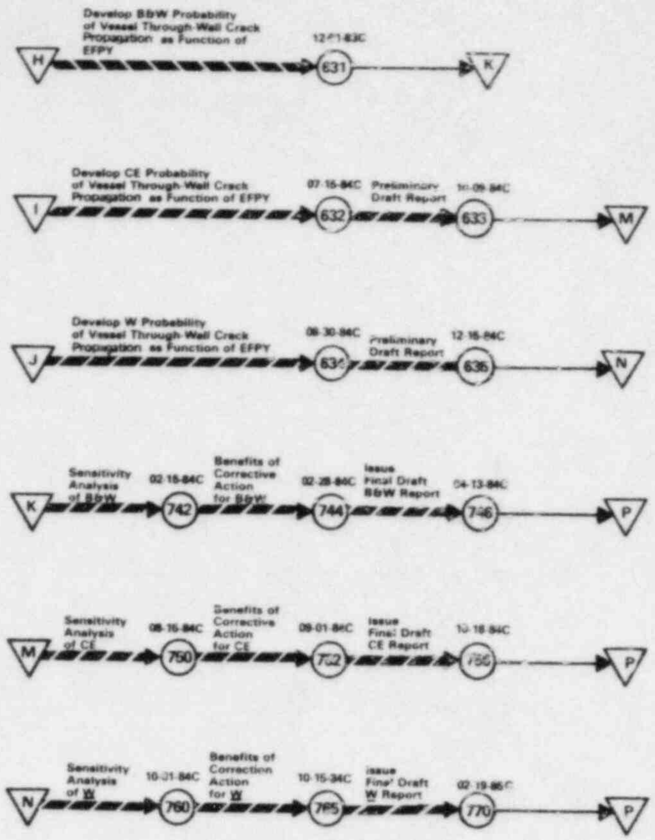
PRESSURIZED THERMAL SHOCK (A-49)
LONG TERM PROGRAM
(CONTINUED)

Task 8:
Integration of Results

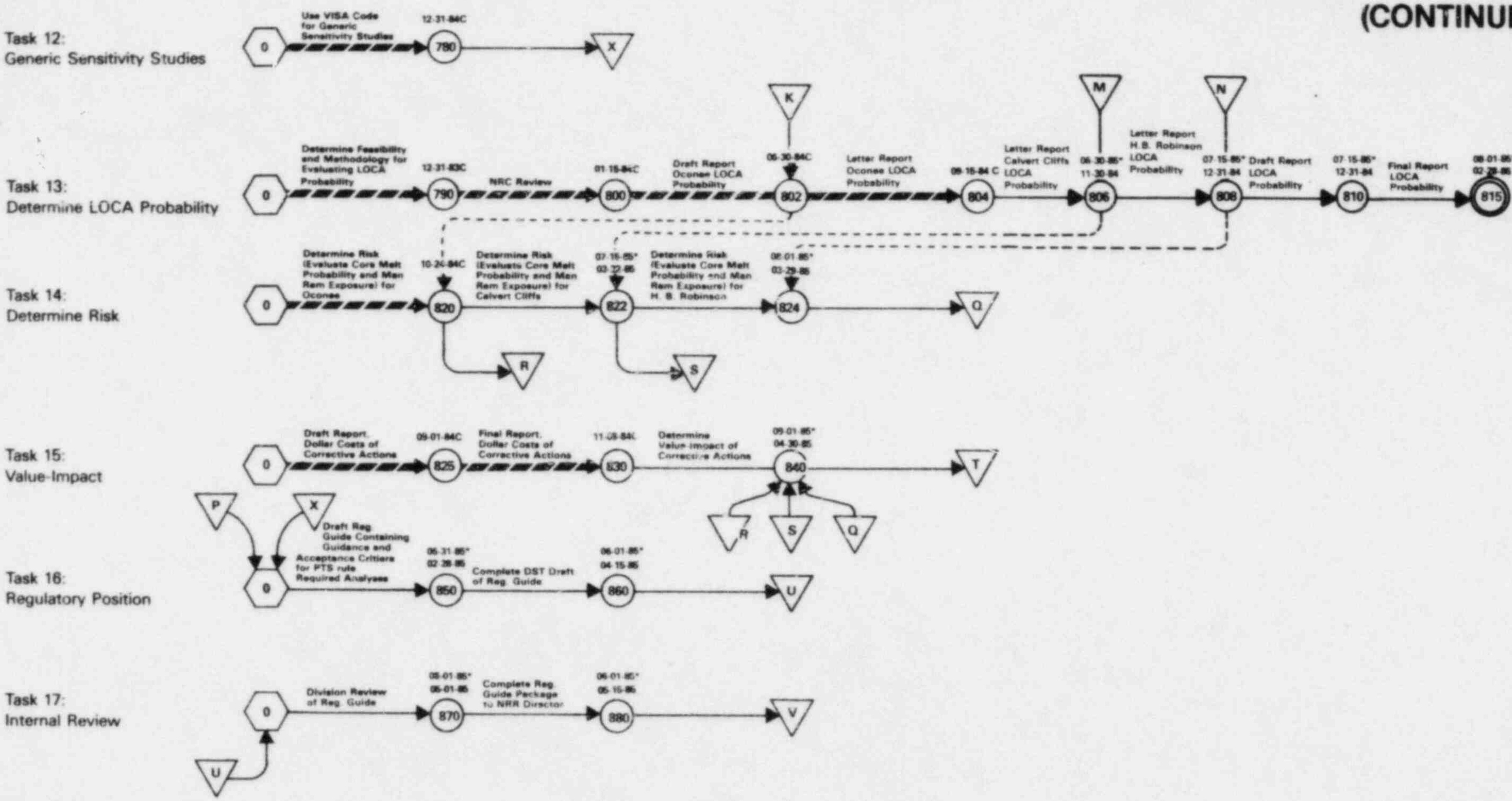
Task 9:
Plant-Specific Sensitivity Studies, Benefits of Corrective Actions, and Draft Final Report for B&W Plant

Task 10:
Plant-Specific Sensitivity Studies, Benefits of Corrective Actions, and Draft Final Report for CE Plant

Task 11:
Plant-Specific Sensitivity Studies, Benefits of Corrective Actions, and Draft Final Report for W Plant



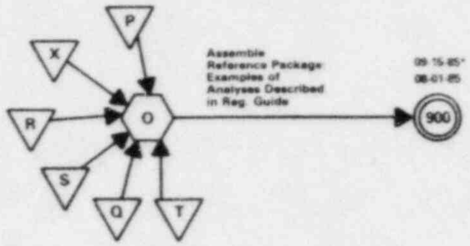
PRESSURIZED THERMAL SHOCK (A-49)
LONG TERM PROGRAM
(CONTINUED)



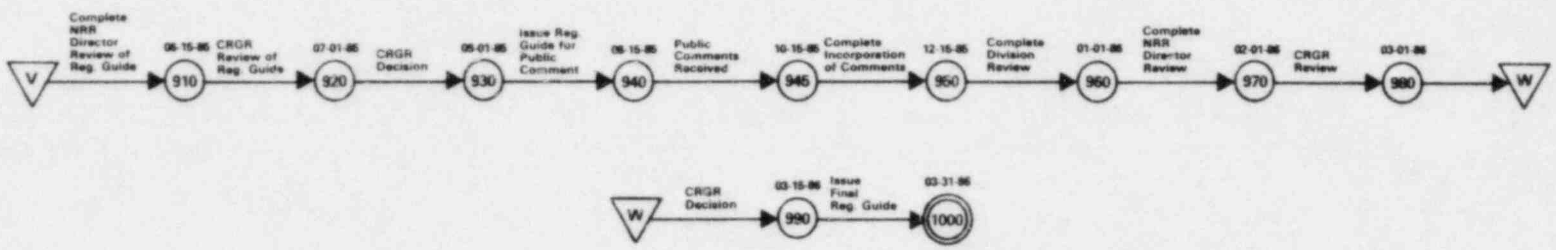
* Schedule Change This Report.

PRESSURIZED THERMAL SHOCK (A-49)
LONG TERM PROGRAM
(CONTINUED)

Task 18:
Assemble
References



Task 19:
Issue Final
Technical Resolution



* Schedule Change This Report.

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BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

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12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

Provides an overview of the status of the progress and plans for resolution of the generic tasks addressing "Unresolved Safety Issues" as reported to Congress.

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