UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

December 12, 1986

MEMORANDUM FOR: Robert M. Bernero, Director
Division of BWR Licensing

FROM: John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing

SUBJECT: NOVEMBER 20-21, 1986, TRIP TO GENERAL ELECTRIC

On November 20 and 21, 1986, I. Villalva, resident inspector for La Crosse, Region III; J. Donohew, Project Manager for Oyster Creek; and myself visited the General Electric (GE) facilities at San Jose, California. The purpose was to discuss (1) the general capabilities GE has at San Jose which support licensees and NRC in nuclear power plant technical areas (November 20th) and (2) the Nuclear Measurement Analysis and Control (NUMAC) equipment GE is marketing to licensees (November 21st). Attachment 1 is a list of the individuals attending the meetings. Attachment 2 contains the handouts from the meeting on November 20th. Attachment 3 has the handouts on the NUMAC equipment from the meeting on November 21st.

The following is a summary of the significant items discussed during the two-day trip.

1.0 Meeting on November 20th

This meeting was held at GE's main facilities in San Jose at 175 Curtner Avenue. The agenda for the meeting is in Attachment 2. The individuals attending the meeting are listed in Attachment 1.

An introduction to its Nuclear Energy Business Operations (NEBO) and to its Family of Coordinated Utility Services (FOCUS) was made by GE. Handouts are in Attachment 2. These were discussed briefly. Several of GE's Nuclear Systems and Services Operations were also discussed. These are underlined in the associated handout enclosed in Attachment 2.

GE reactor performance improvement programs to improve plant operating flexibility, maneuverability, power output and capacity factor were discussed. (GE maintains files and records on performance with their performance indicator base being well over 100 items and probably close to 200.) These programs include upgrading the unit's licensed generating power level, expanding the core operating region, improving hardware reliability and reducing lost capacity due to out-of-service equipment. Two handouts are enclosed in Attachment 2.

In the discussion on hydrogen water chemistry, GE explained its system to monitor crack growth in the reactor coolant system. This is the crack verification system (CVS). It monitors growth in an existing crack. This is being used by BWR licensees in plant tests to determine the effectiveness of hydrogen water chemistry. An extensive discussion took place on GESSAR, GE's approved Topical for reloads.

Atm 15-1
XRD-8-2
GE

There was a tour of the pipe test lab, the Atlas SRV/Fuel Design Testing facility and the maintenance training center. The pipe test lab can test 1500 welds at a time, and 4" and 12" diameter pipes. The Atlas facility tests the thermal hydraulics of new fuel designs and safety relief valves (SRV). The Maintenance Training Center has a mockup of a complete operating floor, including the core, spent fuel pool, and fuel handling equipment, and the lower part of a BWR reactor vessel. These facilities have been used by licensees to prepare for work in radiation areas in their plants.

2.0 Meeting on November 21, 1986

This meeting was held at a GE facility downtown in San Jose on its NUMAC equipment. Handouts on the equipment are in Attachment 3. The individuals attending the meeting are listed in Attachment 1.

The NRC stated that it wanted to be more familiar with the GE NUMAC product line because at least two licensees (La Crosse and Big Rock Point) are interested in this equipment. GE stated that its NUMAC equipment was the following: microprocessor based with one for equipment operation and one for the user interface, as the visual display; modular design with redundant power supplies built-in; fully automatic with self testing on-line; digital data processing after an analog-to-digital data transformation; user friendly including the use of the microprocessor in the user interface; and common spares for Class 1E and non-Class 1E uses. GE stated the equipment is of high quality and high reliability and the benefits to the licensee are reduced surveillance time, reduced spare inventory and improved performance. NRC and GE discussed the training offered a licensee in the use of the NUMAC equipment. This training by GE stressed the training of the licensee's engineers to understand how the equipment functioned and displayed data.

There was a tour of the testing facility, where different NUMAC equipment was operated with dummy signals. The equipment demonstrated to NRC was the rod worth minimizer, log count rate meter, process radiation monitor and hydrogen water chemistry control monitor.

3.0 Conclusions

The GE facility has capabilities that licensees and NRC could use in the future, These include the pipe testing facility, Atlas SRV/Fuel Design Testing facility, Maintenance Training Center and Crack Verification System, GE is working to assist licensees to (1) have better nuclear measurement capabilities with NUMAC and (2) improve power generation of existing nuclear power plants through upgrading the licensed power level, expanding the core operating region, improving hardware reliability and reducing generation time lost to out-of-service equipment.

The NUMAC equipment demonstrated to NRC did appear to exhibit improvements over existing equipment in nuclear plants. The professed higher reliability, self testing on-line, modular design with built-in redundant power supplies, common spares for Class 1E and non-Class 1E uses, and being user friendly are qualities that appear to be focused to meet the needs of the nuclear industry.

Original signed by

John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing

Enclosures:
1. List of Attendees
2. GE Handouts from Meeting on 11/20/86
3. GE Handouts from Meeting on 11/21/86

cc w/enclosures:
R. Houston
G. Lainas
B. D. Liaw
W. Hodges
E. Adensam
W. Butler
D. Muller

DISTRIBUTION: w/ enclosures
Central File
NRC PDR
BWD#1 Reading
JZwolinski
JDonohew
IVillalva, R-III

w/list of enclosures 2&3 only
OGC-BETH (Info only)
EJordan
BGrimes
ACRS (10)
CJamerson

DBL:BWD#1	DBL:BWD#1	DBL:BWD#1
CJamerson	JDonohew:ac	JZwolinski
12/11/86	12/11/86	12/12/86

ENCLOSURE 1

MEETINGS BETWEEN NRC AND GENERAL ELECTRIC (GE)

NOVEMBER 20-21, 1986

November 20, 1986	Company
J. Zwolinski	NRC/NRR/DBL
I. Villalva	NRC/Region III
J. Donohew	NRC/NRR/DBL
R. Hill	GE
B. Smith	GE
G. Nortin	GE
J. Embley	GE
L. Youngborg	GE
J. Klapproth	GE
J. Post	GE
T. Lee	GE
C. Van Damm	GE
J. Charnley	GE

November 21, 1986	Company
J. Zwolinski	NRC/NRR/DBL
I. Villalva	NRC/Region III
J. Donohew	NRC/NRR/DBL
C. Van Damm	GE
D. Reigel	GE
H. Tellsley	Dairyland Power Cooperative
R. Rowe	GE
F. Chao	GE

ENCLOSURE 2

MEETING ON NOVEMBER 20, 1986, WITH GE

Enclosures

Agenda

Nuclear Energy Business Operation Organization chart

Introduction to General Electric (GE) NEBO Services

FOCUS, Family of Coordinated Utility Services

General Electric Nuclear Systems and Services Operations

Reactor Performance Improvements

Plant Operation Performance Improvements by GE BWRs

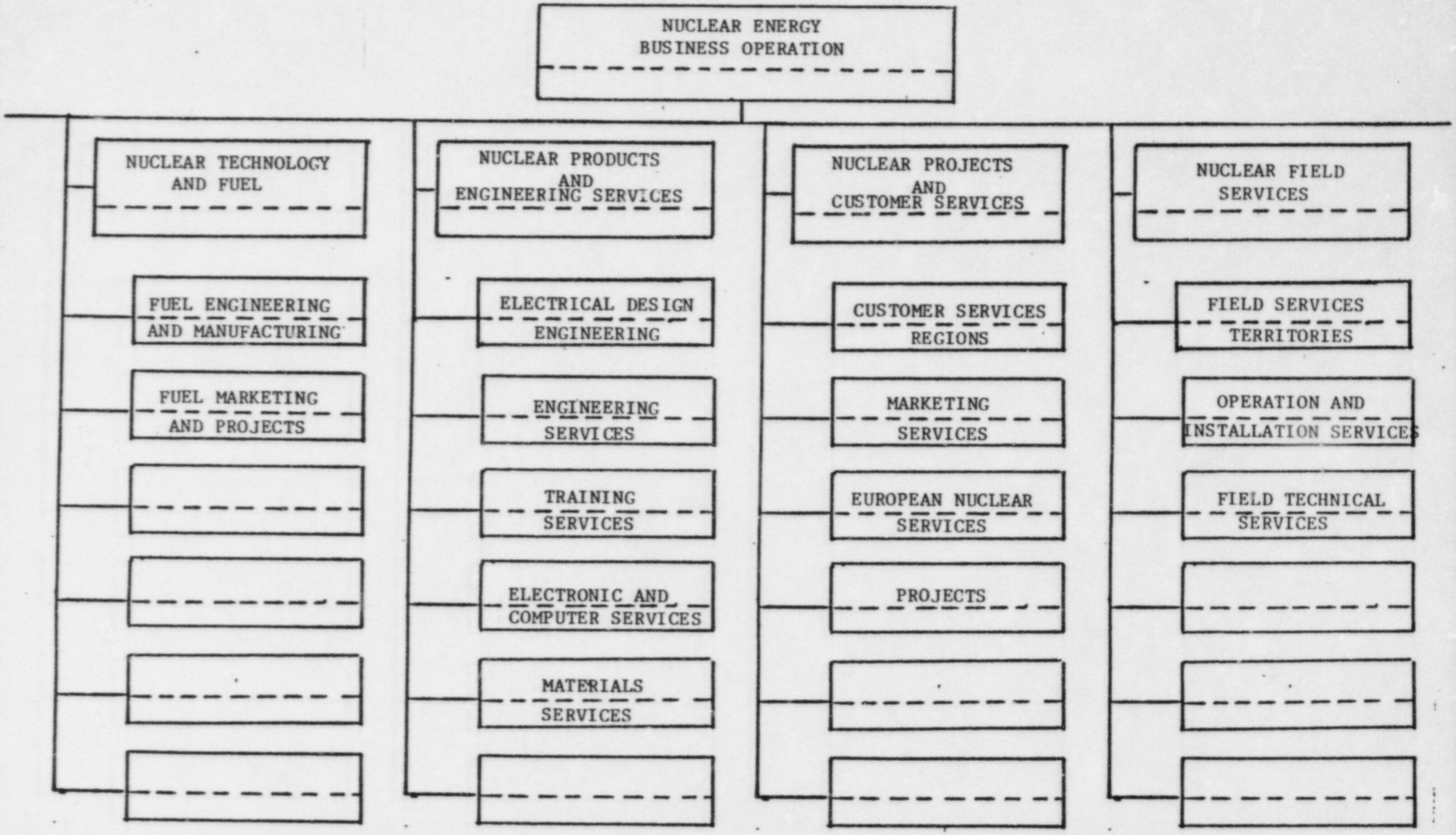
General Electric BWR/6 Thermal Power Upgrade Capability Assessment

Power Uprate

AGENDA

NRC/GE MEETING
November 20, 1986

8:30	Introduction	
		Agenda
		GE Organization
8:45	Services Overview	
		Services Procedure
		Executive Summary
9:15	Focus (Family of Coordinated Utility Services)	
9:45	Selected Specific Services	
		FSAR/TS Updating
		BWR Power Uprate
		Water Quality/Hydrogen Water Chemistry
		Instrumentation Improvement Services
		Core Performance Improvements
12:00	Lunch	
		Containment Vent Purge and Repressurization
		Emergency Procedure Guidelines
1:00	Bernero Meeting Summary	
1:15	GESTAR Review	
2:00	Open Discussion	
3:00	Test Facilities Tour (San Jose)	
		Pipe Test Lab
		Atlas
		Training Center


NUCLEAR ENERGY BUSINESS OPERATION
NUCLEAR TECHNOLOGY AND FUEL
FUEL ENGINEERING AND MANUFACTURING
FUEL MARKETING AND PROJECTS
NUCLEAR PRODUCTS AND ENGINEERING SERVICES
ELECTRICAL DESIGN ENGINEERING
ENGINEERING SERVICES
TRAINING SERVICES
ELECTRONIC AND COMPUTER SERVICES
MATERIALS SERVICES
NUCLEAR PROJECTS AND CUSTOMER SERVICES
CUSTOMER SERVICES REGIONS
MARKETING SERVICES
EUROPEAN NUCLEAR SERVICES
PROJECTS
NUCLEAR FIELD SERVICES
FIELD SERVICES TERRITORIES
OPERATION AND INSTALLATION SERVICES
FIELD TECHNICAL SERVICES

INTRODUCTION
TO
GENERAL ELECTRIC
NEBO
SERVICES

GUS NORTIN
MARKETING SERVICES
NOVEMBER 1986

OBJECTIVES

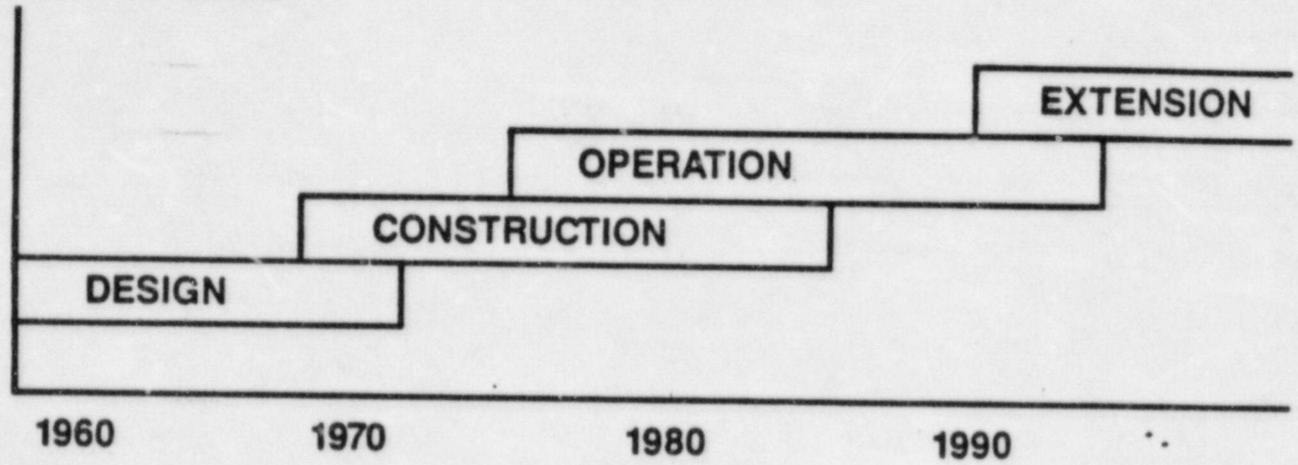
- ENHANCE NEBO PRACTICES OF IMPROVED CUSTOMER LISTENING/FEEDBACK TO GUIDE GE DECISIONS

- DEVELOP PACKAGED SOLUTIONS TO INDIVIDUAL CUSTOMER CONCERNS/PROBLEMS/NEEDS

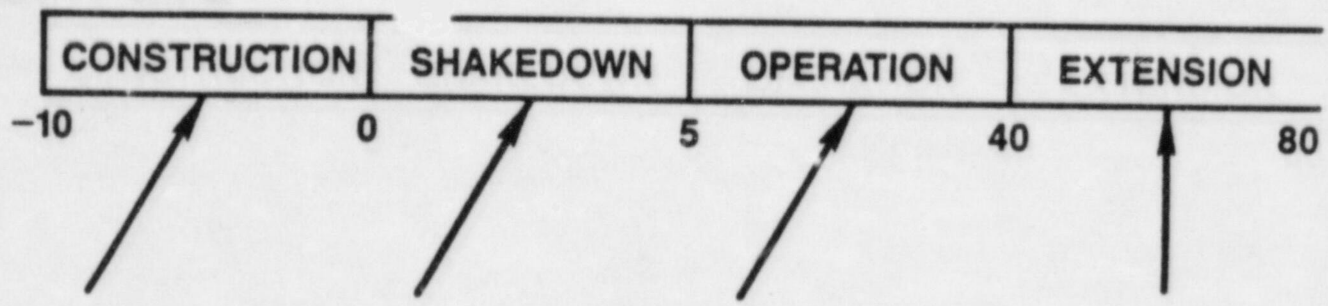
PROVIDE SERVICES THAT CUSTOMER WANTS/NEEDS

INDUSTRY AND PLANT LIFE

- INDUSTRY LIFE



- PLANT LIFE



- ASSEMBLE
- CHECK OUT
- DOCUMENT
- INITIAL SPARES

- OPERATIONS
- OUTAGES
- DOCUMENT CONTROL
- PLAN MODS

- PERFORMANCE
- REGULATORY BACKFITS
- TECHNOLOGICAL UPGRADE
- MAINTAIN/REPLACE
- MONITOR/INSPECT

- PROTECT KEY COMPONENTS
- REPLACE FOR PERFORMANCE & LONGEVITY
- ESTABLISH RELIABILITY OF SAFETY SYSTEMS

SERVICES
FROM
GENERAL ELECTRIC

GENERAL ELECTRIC WILL PROVIDE EQUIPMENT, SERVICES, AND CONSULTING TO HELP KEEP GE BWR's OPERATING SAFELY AND ECONOMICALLY.

GENERAL ELECTRIC WILL SUPPORT OPERATING BWR's BY PROVIDING SERVICES AIMED AT SERVING MAINTENANCE NEEDS, RESOLVING REGULATORY CONCERNS, AND ULTIMATELY EXTENDING OPERATING PLANT LIFE.

GE SERVICES FORM THE FRAMEWORK TO HELP ENSURE THAT BWRs MAINTAIN GOOD PERFORMANCE AND OPERATE IN SUCH A WAY THAT LONG TERM AGING EFFECTS ARE MINIMIZED.

OUR SCOPE

- MECHANICAL & ELECTRICAL PRODUCTS
- ENGINEERING SERVICES
- ELECTRONIC & COMPUTER SERVICES
- REUTER-STOKES (NUCLEAR INSTRUMENTATION)
- MATERIALS SERVICES
- QUALITY ASSURANCE SERVICES
- SAFETY AND LICENSING SERVICES
- FIELD SERVICES
- TRAINING SERVICES
- OPERATOR TRAINING SERVICES

GE SERVICES ARE UNIQUE, ARE TAILORED TO CUSTOMER NEEDS.

OUR SCOPE

	IDENTIFIED NO. OF SERVICES
• MECHANICAL & ELECTRICAL PRODUCTS	50
• ENGINEERING SERVICES	
- ELECTRICAL DESIGN	28
- ENGINEERING ANALYSES, DESIGN, AND CONSULTING	34
• ELECTRONIC & COMPUTER SERVICES	38
• REUTER-STOKES	15
• MATERIALS SERVICES	8
• QA SERVICES	14
• SAFETY AND LICENSING	24
• FIELD SERVICES	40
• TRAINING SERVICES (ENGINEERING, MAINTENANCE, AND I&C)	116
• OPERATOR TRAINING SERVICES	75
TOTAL:	442

MECHANICAL & ELECTRICAL PRODUCTS:

- RETROFITS FOR BETTER OPERATION
- REGULATORY BACKFITS
- TECHNOLOGICAL UPGRADES
- SOLVE PROBLEMS
- IMPROVE OUTAGES

SOME EXAMPLES ARE:

REFUELING PLATFORM UPGRADES,
IMPROVED JET PUMP BEAMS,
LASERTRAC,
FEEDWATER NOZZLE SURVEILLANCE INSTRUMENTATION SYSTEM, AND
REMOTE-OPERATED CRD HANDLING EQUIPMENT.

ENGINEERING SERVICES:

- DESIGN/DRAFTING SERVICES
- UPDATES
- DEDICATED SITE ENGINEERS
- SYSTEM IMPROVEMENTS
- OPERATIONAL ANALYSES
- REALISTIC/IMPROVED COMPUTER MODELS
- PLANT LIFE EXTENSION PROGRAMS
- SCRAM REDUCTION PROGRAMS
- IGSCC PROGRAM
- VARIOUS CONSULTING

SOME EXAMPLES ARE:

IMPROVED SETPOINT METHODOLOGY,
FOCUS,
SAFER/GESTR, AND
DECONTAMINATION CONSULTING.

ELECTRONIC AND COMPUTER SERVICES:

- OMNIBUS (FAMILY OF COMPUTER-BASED PRODUCTS)
  - GEDAC
  - GETARS
  - ERIS
  - GEPAC PLUS
  - SPDS
  - MONICORE & MONICORE PLUS
  - 3-D MONICORE
- DIGITAL CONTROLS
- IMPROVED EQUIPMENT
- INSTRUMENTATION FOR MEETING WATER LEVEL REQUIREMENTS
- IMPROVED NUCLEAR INSTRUMENTATION DESIGNS

IMPROVED AVAILABILITY, RELIABILITY, AND PERFORMANCE, REDUCED SURVEILLANCE TIME, AND MINIMIZED HALF-SCRAMS AND SPARE PARTS INVENTORY.

REUTER-STOKES

- IMPROVED NUCLEAR INSTRUMENTATION
  - GAMMA TIP
  - TIP SYSTEM, BWR/6 COMPONENTS
  - AUTO TIP
- IMPROVED LPRM
- WIDE RANGE NEUTRON MONITOR SYSTEM
  - MEETS REG. GUIDE 1.97 REQUIREMENTS

LPRM PROGRAM PACKAGES INCLUDE BOTTOM ENTRY LPRM, DISPOSAL SYSTEMS, CABLE REPLACEMENT, AND MANAGEMENT SERVICES.

MATERIALS SERVICES:

- INSTRUMENTATION REPAIR AND RETURN
- RENEWAL SPARES PROGRAM
- ENVIRONMENTALLY QUALIFIED HARDWARE - BASED ON UTILITY NEEDS
  - PROGRAM MANAGEMENT, LICENSING RESPONSES AND ASSISTANCE
  - ANALYSES
  - TESTING
  - SPECIFICATIONS

GE PROVIDES BOTH QUALIFIED NEW PRODUCTS AND SERVICES TO QUALIFY EQUIPMENT, INCLUDING PREDICTING OPERATING LIFE.

QUALITY ASSURANCE SERVICES:

- SOURCE INSPECTION - SUPPLIER SURVEILLANCE
- QUALITY AUDITS
- QA SEMINARS
- QA AND QC TRAINING
- UPDATING O&M MANUALS
- CONFIGURATION MANAGEMENT
- INSPECTION SYSTEMS
- DOCUMENT SYSTEMS

DRAWS ON GE PARTICIPATION IN INSTALLATION AND STARTUP OF EQUIPMENT IN OVER 50 BWR PLANTS.

LICENSING/REGULATORY SERVICES:

- PRA
- TECH SPEC UPDATES
- REG GUIDE ASSESSMENTS
- FSAR UPDATES
- LIVING SCHEDULES
- RADIOLOGICAL EVALUATIONS
- REG COMPLIANCE ASSESSMENTS
- CONSULTING SERVICES
- EMERGENCY OPERATING PROCEDURES AND TRAINING

GE HAS EXTENSIVE EXPERIENCE IN INTERACTING WITH REGULATORY AGENCIES.

FIELD SERVICES:

- VISUAL INSPECTIONS
- UT SYSTEMS
- OUTAGE PLANNING AND SCHEDULING SUPPORT
- REFUELING FLOOR COORDINATOR AND TECHNICAL DIRECTOR
- PROJECT MANAGEMENT
- PLANT/SYSTEMS OPERATIONS SUPPORT ENGINEERS
- OPERATIONS ENGINEER PROGRAM
- PREPARATION OF PROCEDURES
- SPECIALTY TOOLING/EQUIPMENT
- ALARA CONSULTATION

FOR MORE THAN THREE DECADES, GE HAS BEEN A SUPPLIER OF ENGINEERING AND TECHNICAL SERVICES FOR BWR'S. ALSO, GE HAS COMMITTED SIGNIFICANT RESOURCES TO THE DEVELOPMENT OF SPECIALIZED BWR FIELD SUPPORT SKILLS.

TRAINING SERVICES:

- MAINTENANCE TRAINING
  - CRD
  - MSIV
  - REFUELING FLOOR
  - RELIEF VALVE
  - DIVER
  - RECIRC PUMP SEALS

- ENGINEERING
  - SNE
  - STA
  - BWR CHEMISTRY
  - COMPUTER
  - PRA
  - HP
  - ALARA
  - DEGRADED CORE

- INSTRUMENTATION & CONTROL
  - NUCLEAR INST.
  - PROCESS
  - MINI-COURSES
  - REACTOR CONTROLS

GE INSTRUCTORS HAVE FULL ACCESS TO THE ENGINEERING RESOURCES OF GENERAL ELECTRIC'S NEBO STAFF.

OPERATOR TRAINING SERVICES:

- SIMULATOR QUALIFICATION
- TRAINING SYSTEM DEVELOPMENT
- ACADEMIC AND INTRODUCTORY LEVEL COURSES
- RO/SRO BWR SYSTEMS TRAINING
- NRC EXAM PREPARATION
- INSTRUCTOR TRAINING
- SRO SUPERVISORY SKILL PROGRAMS
- BWR OPERATING FUNDAMENTALS
- RO/SRO HOT LICENSE TRAINING
- POST-LICENSE RO/SRO TRAINING
- STA OPERATIONS TRAINING

SINCE 1968, GE HAS PROVIDED OPERATOR TRAINING TO OVER 2200 UTILITY OPERATORS ON BWR PRODUCT LINES BWR/1 THRU BWR/6.

PACKAGED SERVICES:

EXAMPLES:

- SPECIAL OUTAGE MANAGEMENT SERVICES
  - PIPE REPLACEMENT
  - DRYWELL SURVEYS
  - ULTRASONIC TESTING
  - VESSEL UT
  - SPECIALTY TOOLING
  - IHSI

PACKAGED SERVICES:

EXAMPLES:

- PLANT UPGRADE
  - OMNIBUS
  - NUMAC
  - DIGITAL CONTROLS
  - WRNM/IMPROVED LPRM
  - TIP UPGRADES

PACKAGED SERVICES:

EXAMPLES:

- FUELS MANAGEMENT SERVICES
  - CONTROL CELL CORE
  - BARRIER FUEL
  - FEEDWATER TEMPERATURE REDUCTION
  - IMPROVED CONTROL BLADES

Number	Recommendation	Comment	Program Manager	Experience
152A	NA-300 LPRM	• General Electric has developed a new, longer life NA-300 LPRM. Its advantages are: a. Diffused sensor coating which provides up to 20% longer life. b. New seal/purge which eliminates sensor/cable failures. c. Incorporates a quick-disconnect connector. d. Uses low carbon and low cobalt materials to reduce Intergranular Stress Corrosion Cracking and Rem exposure concerns. e. Directly replaces the NA-200 LPRM without electronic changes.	W.W. Phelan Reuter Stokes (216) 581-9400	
152B	NA-350 LPRM	• Bottom entry LPRM are standard on BWR/6 plants. General Electric has developed the bottom entry NA-350 LPRM for backfit into the earlier BWR plants. • The NA-350 LPRM includes the same longer life features incorporated into the NA-300 LPRM (see Recommendation Number 152A). • Individual Sensors can be replaced in the NA-350 LPRM. this reduces costs and Rem exposure. • A Bottom Entry Disposal System (BEDS) is required (see Recommendation Number 152C).	W.W. Phelan Reuter Stokes (216) 581-9400	

Number	Recommendation	Comment	Program Manager	Experience
152C	BOTTOM ENTRY DISPOSAL SYSTEM	• A Bottom Entry Disposal System (BEDS) is required to dispose of spent NA-350 LPRM sensors. • BEDS quickly and efficiently disposes of the LPRM lead cable and sensors.	W.W. Phelan Reuter Stokes (216) 581-9400	
152D	LPRM CABLE REPLACEMENT	• General Electric is developing a two step LPRM cable replacement program. The first step is to replace the present RG59 (soft) lead cable with mineral insulated (hard) cable. The second step is to route the mineral insulated cable above the Control Rod Drive flanges. This would eliminate disconnecting, rolling up, and reconnecting all of the LPRM lead cables each refueling outage. These steps would greatly reduce the LPRM cable failures. It would also greatly reduce the undervessel cable disconnection/reconnection effort each refueling outage.	W.W. Phelan Reuter Stokes (216) 581-9400	

Number	Recommendation	Comment	Program Manager	Experience
157	AUTO-TIP SYSTEM	• General Electric has designed an Automated Traversing Incore Probe (A-TIP) System. The system is a modification to the existing TIP System. The existing drive control units are replaced by A-TIP Control Units (ATCU). • A-TIP has three modes of operation: automatic, semi-automatic, and manual. • In automatic, all TIP are operated simultaneous by the master ATCU control room microprocessor. • Flux data is stored in individual ATCU memory and later forwarded to the host computer. • Redundant functions and diagnostic self-checking capability eliminate problems inherent in the old system designs.	W.W. Phelan Reuter Stokes (216) 581-9400	

FOCUS

FAMILY OF COORDINATED UTILITY SERVICES

PRESENTED TO:

REPRESENTATIVES OF

US NUCLEAR REGULATORY COMMISSION STAFF

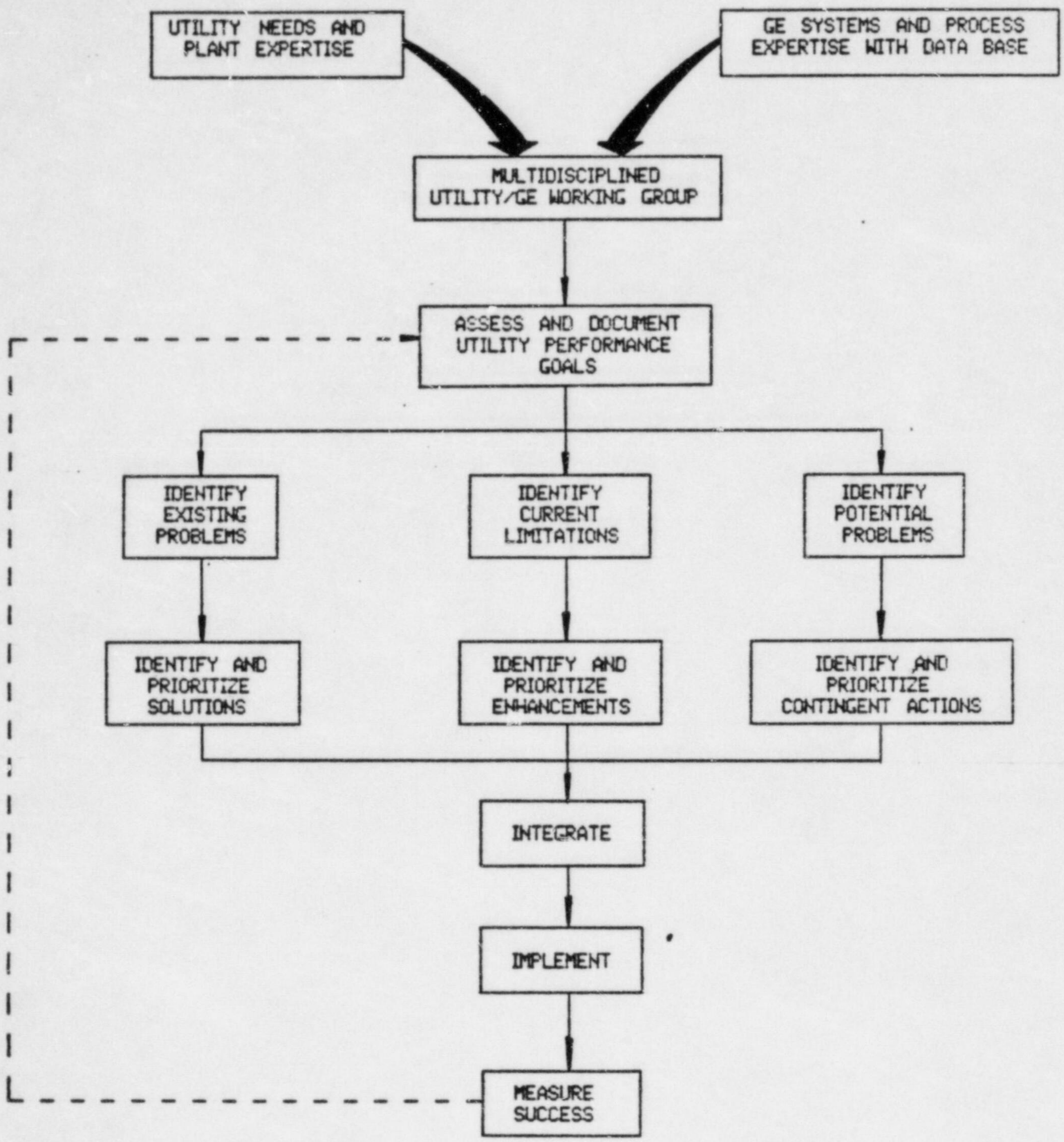
NOVEMBER 20, 1986

B. W. SMITH

GENERAL ELECTRIC COMPANY

NUCLEAR ENERGY OPERATIONS

FOCUS PROCESS



FOCUS

ELEMENTS

PERFORMANCE IMPROVEMENT	• IMPROVE ON PAST PERFORMANCE (CAPACITY FACTOR, ALARA, O&M COST, ETC.)
PLANT AGING	• FORESTALL FUTURE PROBLEMS
LIFE EXTENSION	• PREPARE FOR RELICENSING/RECERTIFICATION
PRODUCTIVITY IMPROVEMENT	• ADDRESS ORGANIZATIONAL/ADMINISTRATIVE CONCERNS
SAFETY ASSESSMENT	• EVALUATE EFFECTS OF MODIFICATIONS
REGULATORY COMPLIANCE	• IMPROVE REGULATORY PERFORMANCE
INTEGRATED LONG RANGE PLAN	• PRIORITIZE AND PLAN FOR ACTIONS COMMON TO ALL FOCUS ELEMENTS

FOCUS

RESULTS

- EFFECTIVELY COMBINES THE UNIQUE TALENTS OF THE UTILITY AND GE
- PROVIDES A DISCIPLINED, SYSTEMATIC APPROACH TO UTILITY GOAL ACHIEVEMENT
- IDENTIFIES AND PRIORITIZES ACTIONS NEEDED TO MEET GOALS
- FACILITATES UTILITY BUDGET PLANNING
- PROVIDES JUSTIFIABLE, DOCUMENTED BASIS FOR UTILITY SPENDING POLICY AND IMPLEMENTATION SCHEDULES

DEFINING, TRACKING,

AND ACHIEVING

PLANT PERFORMANCE GOALS

* APPLYING THE FOCUS PROCESS *

NOTE:

THE FOLLOWING PAGES WERE PRESENTED AT THE ASME-ANS NUCLEAR POWER CONFERENCE

IN JULY 1986

INTRODUCTION

- NUMEROUS DEMANDS ON UTILITY RESOURCES
  - REGULATORY ISSUES
  - ECONOMIC GENERATION IMPROVEMENTS
- EMPHASIS WILL VARY
  - WITHIN INDUSTRY
  - WITH TIME
- CORPORATE GOALS MAY INDICATE APPROPRIATE BALANCE
- ACHIEVING ALL GOALS REQUIRES A STRUCTURED PROCESS

A STRUCTURED PROCESS FOLLOWS

PLANT IMPROVEMENT PROCESS

- DEFINE "OPTIMUM" PERFORMANCE
- ASSESS CURRENT PERFORMANCE
- ESTABLISH IMPROVEMENT GOALS
- SET PRIORITIZATION METHOD
- PRIORITIZE IMPROVEMENTS
- RECONCILE BUDGET, GOALS AND SCHEDULE
- IMPLEMENT IMPROVEMENTS
- MEASURE SUCCESS

DEFINE "OPTIMUM" PERFORMANCE

- ACKNOWLEDGE PLANT AND UTILITY DIFFERENCES
  - OPERATING VS. CONSTRUCTION PLANT MIX
  - PLANT AGE
  - STATE REGULATORY BODIES
- ESTABLISH PLANT UNIQUE DEFINITION
  - DEVELOP INDEPENDENT PARAMETER LISTING
    - INDUSTRYWIDE MEASURABLE PARAMETERS
    - OTHER IMPORTANT PARAMETERS
  - ASSIGN PARAMETER WEIGHTING FACTORS

PERFORMANCE CRITERIA

INDUSTRYWIDE MEASURABLE PARAMETERS

NRC VIOLATIONS
NRC FINES
NRC OPEN ITEMS
PERSONNEL EXPOSURE
LIQUID RADIOACTIVE DISCHARGES
GASEOUS RADIOACTIVE DISCHARGES
RADWASTE
EMERGENCY PREPAREDNESS

INTERIM CYCLE CAPACITY FACTOR
SCHEDULED OUTAGE DAYS INCURRED
PLANNED VS. SCHEDULED OUTAGE DAYS
HEAT RATE

OTHER PARAMETERS

SITE SAFETY
OFF-SITE SAFETY
PUBLIC IMAGE
REGULATORY IMAGE
HUMAN FACTORS

ASSESS CURRENT PERFORMANCE

- GATHER NEEDED PARAMETRIC DATA
  - SPECIFIC PLANT
  - INDUSTRY
- EVALUATE PERFORMANCE
  - EACH PARAMETER AND PLANT
  - TWO OR THREE TIME PERIODS
- ESTABLISH INDUSTRY RANK
  - BY PARAMETER
  - OVERALL
- PERFORM MORE EXTENSIVE REVIEW IN KEY AREAS
  - CONSIDER USE OF MAJOR SUPPLIER EQUIPMENT DATA BASES
  - ASSESS ROOT CAUSES OF PROBLEMS

PARAMETER RANKINGS FOR ONE PLANT

(FROM POPULATION OF 22 DOMESTIC OPERATING BWR'S)

PARAMETER	RANK
REGULATORY CONFORMANCE	2
NRC VIOLATIONS	6
NRC FINES	11
NRC OPEN ITEMS	8
EXPOSURE	1
LIQUID DISCHARGES	4
GAS DISCHARGES	7
RADWASTE	3
EMERGENCY PREPAREDNESS	9
PLANT PERFORMANCE	18
CAPACITY FACTOR	19
SCHEDULED OUTAGE TIME	8
ACTUAL/PLANNED OUTAGE	1
HEAT RATE	21
OVERALL	14

CAPACITY FACTOR LOSSES

INDEX	CATEGORY	77	78	79	80	81	YEAR 82	83	84	85	AVE	AVE BWR
	REACTOR PRESSURE VESSEL											
1	FEEDWATER NOZZLES	0.	0.	0.	10.93	0.	0.	0.	0.	0.	1.22	0.62
2	CONTROL ROD DRIVE NOZZLE	2.39	0.67	0.	0.	0.	0.	0.	0.	0.	0.34	0.14
3	RECIRC NOZZLES	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.29
4	REACTOR VENT LINES	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.01
5	INSPECTION	0.96	0.	0.	0.	0.	0.	0.	0.	0.14	0.12	0.01
6	OTHER	0.06	0.	0.	2.87	0.	0.	0.	0.	0.	0.33	0.51
7	SYSTEM TOTAL	3.40	0.67	0.	13.80	0.	0.	0.	0.	0.14	2.00	1.56
	MAIN STEAM											
8	MAIN STM ISO VALVES	0.	0.	0.	0.	0.	0.	0.	0.87	0.	0.10	0.54
9	TARGET ROCK S/R VALVES	1.07	0.69	0.	0.41	0.	2.71	0.58	0.	0.	0.61	0.37
10	SAFETY & RELIEF VALVES	0.	0.	0.	0.12	0.34	0.	0.	0.	0.	0.05	0.12
11	INSTRUMENTATION	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.04
12	PIPING REPAIR	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.03
13	OTHER VALVES	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.02
14	OTHER	0.	0.	0.	0.	0.	0.	0.	0.11	0.	0.01	0.25
15	SYSTEM TOTAL	1.07	0.69	0.	0.52	0.34	2.71	0.58	0.98	0.	0.77	1.39
	REACTOR RECIRCULATION											
16	RECIRCULATION PUMPS	0.	0.	0.	1.82	2.25	2.19	0.08	0.	0.	0.70	0.71
17	RECIRCULATION PUMP MOTOR	0.	0.	0.	0.	9.45	0.20	0.28	0.	0.	1.10	0.14
18	MOTOR GENERATOR SETS	0.	0.16	0.	0.43	0.86	0.	0.	0.	0.	0.16	0.15
19	JET PUMPS	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.15
20	VALVES	0.83	0.86	0.	0.73	0.	0.	0.	0.	0.	0.27	0.32
21	FLOW CONTROL	0.	0.	0.01	0.	0.	0.	0.	0.	0.	0.00	0.04
22	PIPING	0.	0.	0.	0.	0.	0.	42.48	59.65	24.06	14.03	3.84
23	INSTRUMENTS & CONTROLS	0.	0.01	0.01	0.	0.	0.	0.	0.	0.	0.00	0.06
24	OTHER	0.	0.	0.	0.	0.01	0.	0.	0.	0.	0.00	0.18
25	SYSTEM TOTAL	0.83	1.03	0.02	2.98	12.58	2.38	42.84	59.65	24.06	16.27	5.59
	ROD CONTROL											
26	CRD CHANGEOUT	0.43	0.48	0.01	0.	0.	4.11	0.06	0.	4.12	1.02	0.39
27	SCRAM DISCHARGE VOLUME	0.	0.	0.	0.30	0.04	0.	0.	0.	0.	0.04	0.53
28	CONTROL ROD DRIVE TEST	0.27	0.	0.	0.	0.	0.	0.	0.	0.	0.03	0.12
29	RETURN LINES PIPING	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.04
30	PIPING SUPPORTS	0.	0.	0.	0.55	0.	0.	0.	0.	0.	0.06	0.09
31	CRD PUMP & MOTOR	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.02
32	NOTCH EXERCISING	0.	0.02	0.00	0.	0.	0.	0.	0.	0.	0.00	0.03
33	VALVES	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.04
34	OTHER	0.	0.02	0.	0.00	0.09	0.	0.	0.	0.27	0.04	0.09
35	SYSTEM TOTAL	0.71	0.52	0.01	0.85	0.13	4.11	0.06	0.	4.39	1.20	1.34

CAPACITY FACTOR LOSS COMMENTS

INDEX	DATE	% LOSS	(1)	COMMENTS
1	4/80	1.3661		FW NOZZLE CLAD REMOVAL.
1	5/80	8.4699		NO COMMENT
1	6/80	1.0929		NO COMMENT
2	5/77	0.2740		INSPECT CRD RETURN LINE NOZZLE. MANY INDICATIONS FOUND
2	5/77	2.1119		GRIND OUT CRACK INDICATIONS IN CRD RETURN LINE NOZZLE
2	10/78	0.1598		CRD NOZZLE INSPECTION
2	10/78	0.5137		CRD NOZZLE REPAIR
5	7/77	0.9589		VESSEL IN-SERVICE INSPECTION.
5	3/85	0.1370		RPV FLANGE ISI
6	5/77	0.0571		INSPECT RX VESSEL FLANGE
6	4/80	0.1366		INSTALL SERVICE PLATFORM FOR FW NOZZLE WORK.
6	4/80	2.7322		DRAIN/DECON RX CAVITY IN PREPARATION FORFW NOZZLE CLAD REMOVAL.
8	12/75	0.0913		RX SCRAM DURING SURVEILLANCE TESTING. MSIV CLOSUREON APRM HIGH FLUX SIGNAL. FLUX SPIKE RESULTED FROM PRESSURE TRANSIENT DURINGMSIV FAST CLOSURE TEST. ONLY ONE MSIV WAS BEING TESTED WHEN REACTOR SCRAMMED.
8	6/76	1.4458		NO COMMENT
8	2/84	0.0228	F	DECREASING POWER FOR MAINTENANCE OUTAGE
8	2/84	0.8083		MSIV LEAK RATE TESTS PER TECH SPEC REQUIREMENTS
8	2/84	0.0378	F	TESTING FOLLOWING MAINTENANCE OUTAGE
9	10/74	1.7130		S/R VALVE 71E OPEN SPONTANEOUSLY & COULD NOT BE CLOSED. WORN SECOND STAGE & P ILOT SEATS ALLOWED LEAKAGE. LAPPED SEAT.
9	11/74	2.5694		S/R VALVE 71L OPENED SPONTANEOUSLY & COULD NOT BE CLOSE. RX WAS MANUALLY SCRAMM ED. VALVE 71J HAD LEAK ALARM. ALL VALVES INSPECTED. FIVE VALVES REQUIRED R EPAIRS.VARING FROM LAPPING OF PILOT VAL-VE TO REPLACEMENT OF SECOND STAGE DISC.
9	6/75	0.1598		RV-2-71K K RELIEF VLV HAS EXCESS PILOT SEAT LEAKAGE.VLV OPENED DEPRESS RX.PILOT VLV LAPPED.
9	6/76	0.3529		RX SCRAM WHEN SAFETY RELIEF VALVE 71A LIFTED SPON-TANEOUSLY AND STUCK OPEN. BLOW DOWN TO 150 PSI OCCURRED. REMAINED SHUTDOWN TO INVESTIGATE AND REPAIR VAL VE. VALVE SHOWED STEAM CUTTING OF PILOT SEAT.
9	7/76	0.5806		NO COMMENT
9	11/76	0.5464		RX MANUALLY SCRAMMED WHEN A SAFETY-RELIEF 71F VALVE LIFTED. THE DEFECTIVED V ALVE WAS REPLACED ALONG WITH VALVES 71B, 71J, AND 71L, WHICH EXHIBITED HIGH TEMPERATURES VALVE 71F STUCK OPEN. REACTOR BLEW DOWN TO 600PSI
9	1/77	0.2511		INSPECT TARGET ROCK RELIEF VALVE DIAPHRAGMS.
9	1/77	0.8219		INSPECT AND TEST SAFETY-RELIEF VALVE WHICH LIFTED SPONTANEOUSLY DURING RESTART. REACTOR BLEW DOWN. RESULTANT FLUCTUATING WATER LEVEL SCRAMMED REACTORPRIOR TO M ANUAL SCRAM.
9	2/78	0.6849		REMOVE 5 SUSPECTED LEAKING PILOT VALVES ON S/R VALVES C, D, E, F & L). REPLAC E WITH REWORKED ASSEMBLIES. ONE COMPLETE ASSEMBLY (L), 4 BASE ASSEMBLIES .ONLY (C, D, E & L).
9	1/80	0.4098		REPLACEMENT OF 71G ADS SAFETY/RELIEF VALVE DIA- PHRAGM.

----

(1) F INDICATES A FORCED POWER REDUCTION COMMENT

ESTABLISH IMPROVEMENT GOALS

- REVIEW RANKING RESULTS
- NOTE POOR SCORES IN HIGHLY WEIGHTED AREAS
- SET IMPROVEMENT GOALS
  - NEXT MEASUREMENT PERIOD
  - LONGER RANGE

SET IMPROVEMENT PRIORITIZATION METHOD

- REVIEW DEFINITION OF OPTIMUM PERFORMANCE AND PERFORMANCE RANKINGS
- CONSIDER ADJUSTMENTS TO PARAMETER WEIGHTS
- ASSESS NET BENEFIT OF EACH IMPROVEMENT
  - PREDICT IMPACT ON EACH PARAMETER
  - NET IMPACT IS NET BENEFIT
- CALCULATE BENEFIT/COST RATIO
- UTILIZE BENEFIT/COST RATIO AS PRIORITY SCORE
- OBTAIN EXECUTIVE LEVEL ENDORSEMENT
- CONSIDER INSTALLATION ON SPREADSHEET-TYPE SOFTWARE

DEFINE, INTEGRATE AND PRIORITIZE IMPROVEMENTS

- DEVELOP SOLUTIONS WITH COGNIZANT PLANT PERSONNEL

- SEEK COMBINATION OPPORTUNITIES TO ADDRESS MULTIPLE ISSUES

- PRIORITIZE COMBINED IMPROVEMENTS

RECONCILE BUDGET, GOALS AND SCHEDULE

- MATCH NEEDED VS. AVAILABLE RESOURCES
- ESTABLISH TENTATIVE SCHEDULE
- PROJECT IMPROVEMENT WITH AVAILABLE RESOURCES
- ITERATE ON BUDGET, GOALS AND SCHEDULE FOR COMPATIBILITY

SUMMARY

- UTILIZED PLANT-UNIQUE DEFINITION OF PERFORMANCE
- BALANCED REGULATORY, ECONOMIC AND OTHER GOALS
- MEASURED KEY AREAS OF PLANT PERFORMANCE
- IDENTIFIED IMPROVEMENT NEEDS
- PRIORITIZED IMPROVEMENTS TO BEST MEET GOALS
- PROVIDED CHECKS AND BALANCES ON BUDGET, GOALS AND SCHEDULE
- OPTIMIZED ACHIEVEMENT OF GOALS

Number	Recommendation	Comment	Program Manager	Experience
102	**ION CHROMATOGRAPHY SERVICES**	• General Electric has developed a range of services and products for on-line water purity monitoring based on ion-chromatography (IC). • Services are designed to supplement a plant's protective water chemistry program by providing specific and continuous information on contaminate intrusion. • Includes both diagnostic and consultation services utilizing portable IC equipment and/or specification of analysis systems, training of operators, reduction of data, procedures expansion of existing capabilities, and sample conditioning. • Applicable to BWR, PWR, and fossil-fired steam generation systems. • Also, see Recommendation No. 108.	D.N. Rodgers X52977	More than 30,000 hours of related experience logged at PWRs More than 15,000 hours of related experience logged at BWRs. Sold Units to Hatch, Savannah River, Peach Bottom, Genkai (Japan).
103	**IMPROVED FLUID COUPLING CONTROLLER**	• A highly-reliable, direct replacement for the existing fluid coupling controller (scoop tube positioner controller) is available. • A Technical discussion between General Electric and BWR plant personnel is recommended. The discussion would cover the plant experience with their present scoop tube controls. Based on this discussion, a specific recommendation on control replacement could be made.	F.F. Witt X52881	Installed on six reactors in Japan, with up to four years operating experience.

Number	Recommendation	Comment	Program Manager	Experience
104	SIMULATED THERMAL POWER MONITOR	• New neutron flux scram circuitry is available to replace the present flow-referenced neutron flux scram circuits. • Will increase the margin to scram, improve plant availability. • Has been licensed at ten BWR plants and is standard equipment on all BWR/5 and 6 plants.	F.E. Holland X54340	Hatch, Browns Ferry, FitzPatrick, Chinshan
105	SERVICES FOR MEETING RPV WATER LEVEL INSTRUMENTATION REQUIREMENTS	• General Electric is available to assist utilities in whatever course of action is taken to meet the NRC requirements of Regulatory Guide 1.97 and Generic Letter No. 84-23. This includes rerouting of the reference legs, design of systems to prevent reference leg overheating, and determination of whether the utility's present system or an alternate fix is adequate.	F.E. Holland X54340	
105A	COMPENSATED WATER LEVEL INSTRUMENTATION SYSTEM (CWLIS)	• The Compensated Water Level Instrumentation System was designed to meet the Category I requirements of Regulatory Guide 1.97 and addresses the NRC's Inadequate Core Cooling Concern of high drywell temperature. • Use of the CWLIS with some operator action may be acceptable to the NRC requirements specified in Generic Letter No. 84-23.	F.E. Holland X54340	Nine Mine Pt. 1 , Oyster Creek, Chinshan

Number	Recommendation	Comment	Program Manager	Experience
106	ANALOG TRIP SYSTEM	• The Analog Trip System is a direct replacement for the various pressure, level, flow, and temperature switches in the plant. • Units require less maintenance. • Will reduce scrams since units have superior drift characteristics and less sensitivity to mechanical shock. • Are seismically qualified to IEEE 344-1975 and environmentally qualified to IEEE 323-1974.	G.V. Dain X51205	All BWR/6, FitzPatrick, Nine Mile Pt. 1, Chinshan 1 & 2, Tokai 2, Hatch 1 & 2

Number	Recommendation	Comment	Program Manager	Experience
110	GAMMA TIP SYSTEM	• Gamma sensitive probes are being used at seven operating BWR plants and have been purchased by six other plants. • Data from these plants confirm that the Gamma TIP System monitors core power more accurately than the Thermal TIP System. More accurate core monitoring results in additional thermal margin. This can be used to simplify reactor operation and to attain more efficient, lower-cost core loadings. • Reduced maintenance time at one plant has resulted in an annual reduction in the radiation exposure to the maintenance personnel.	W.W. Phelan Reuter Stokes (216) 581-9400	
111	TIP SYSTEM UPDATE	• BWR/6 components can be retrofitted into existing BWR/2-5 operating plant TIP Systems to improve operation and reduce both maintenance and personnel exposure. • The major equipment items are: a. Purge air control cabinet. b. Indexing mechanism. c. TIP control unit. d. Drive mechanism.	W.W. Phelan Rueter Stokes (216) 581-9400	

Number	Recommendation	Comment	Program Manager	Experience
112	SAFETY/RELIEF VALVE OPEN/CLOSE MONITOR	• The General Electric open/close monitor provides a positive indication (not susceptible to ambiguous readings from leaking safety/relief valves). • Switches are qualified for 40 years in-containment. • Switches are not susceptible to cross-talk from other valves and S/RV.	F.E. Holland X54340	
113	HYBRID CONTROL ROD	• A new control rod design (a hybrid design) with improved lifetime is available. • Hybrid control rods, on the average, have 40 percent improvement in lifetime over the B_4C control rod. • Is compatible with existing core designs, is a direct replacement (same nuclear worth). • Contract can be negotiated with the utility to provide cost incentives to the utility in return for agreeing to a long-term supply contract with NEBO. Individual terms and conditions can vary, price is indexed to labor and material indexes to provide escalation protection.	K.W. Brayman X56587	Most BWRs

Number	Recommendation	Comment	Program Manager	Experience
132	LOW-LOW SET/LOWERED MSIV WATER LEVEL TRIP	• Plant-specific analyses have demonstrated the capability of Low-Low Set (LLS) relief logic and Lowered MSIV Water Level Trip to mitigate postulated thrust and torus loads associated with subsequent actuations of a safety/relief valve in a BWR plant with a Mark I containment. • Advantages of installing LLS/Lowered MSIV Water Level Trip include a reduction of safety/relief valve challenges and cycles, decreased fatigue duty on the containment, reduced safety/relief valve maintenance, reduced probability of "sticking open" safety/relief valves, and improved plant availability by reducing spurious MSIV trips and scrams.	C.A. Walker X54277	Cooper, Duane Arnold, Hatch 1 & 2
133	FEEDWATER NOZZLE SURVEILLANCE INSTRUMENTATION SYSTEM (FNSIS)	• General Electric has developed a thermal detection system which compares outside nozzle wall temperature with supply pipe outside wall temperature. • The system accurately monitors and records leakage rates past the inboard seals of the double-piston-ring feedwater sparger. • Installation of the system should relieve in-vessel test requirements imposed by NUREG-0619.	K.W. Hess X51343	

GENERAL ELECTRIC

NUCLEAR SYSTEMS AND SERVICES OPERATIONS

Number	Recommendation	Comment	Program Manager	Experience
145	HPCI TURBINE OVER-SPEED TRIP	• An improved mechanical overspeed trip assembly is available. This assembly includes a new improved tappet assembly, a new piston, and modification to (or replacement of) the value body. • For additional information see SIL Numbers 353 and 392.	P.F. Kachel X51307	
146	RCIC STARTUP TRANSIENT IMPROVEMENT	• An overspeed trip often occurs when the RCIC turbine is started. • General Electric has developed a steam bypass modification which greatly aids RCIC turbine startup.	M.S. Laurent X51325	Hatch, Fermi, Grand Gulf, Kuosheng 1 & 2
147	GEMAC 5000 REPLACEMENTS (TOSMAC-500)	• General Electric has available a supply of plug-compatible GEMAC 5000 replacement units. These instruments would be dimensionally and functionally replaceable with the existing plant instruments. • BWR plant reliability would be improved by use of the replacement product line.	S.N. Patel X56688	Hatch 1 & 2, Browns Ferry, Millstone
148	LPRM MANAGEMENT SERVICES	• General Electric has developed an LPRM Management Service that provides BWR plant operators with the necessary level of data plus special analytical techniques to develop a replacement schedule that maximizes the life of LPRM.	W.W. Phelan Reuter Stokes (216) 581-9400	

REACTOR PERFORMANCE IMPROVEMENTS

(T. LEE)

REACTOR PERFORMANCE IMPROVEMENTS

The BWR Reactor Performance Improvement Package is a family of programs that provides improved plant operating flexibility, maneuverability and capacity factor to the BWR owners. It includes features to uprate the unit's generating power output, expand the operating region, improve system hardware reliability and provide insurance against lost capacity due to out of service equipment. These are achieved via design and licensing evaluations, technical specification modifications and minor setpoint and hardware changes.

These increased plant operating states have been demonstrated by analysis, testing and operation in various plants. Most of these improvements have been licensed in both the U.S. and overseas.

Significant insight was gained as a consequence of performing this analysis and testing for the Leibstadt plant. Leibstadt is the lead BWR 6 Mark III plant in Switzerland, which has implemented many of these improvement programs. The ability of Leibstadt to operate in these extended operating states was established by performing extensive analytical studies and was confirmed via special tests performed during the startup test program and by monitoring plant performance during normal cycle operation. Plant operating experience during Cycle 1 and particularly in Cycle 2 has demonstrated that these programs have resulted in significant improvements in plant operating flexibility and capacity factor.

Based on the performance to date, the reactor performance improvement programs are achieving the expected gains in operating maneuverability and simplicity and capacity factor at the plants which have implemented them.

AGENDA

PROGRAM ELEMENTS

LICENSING STATUS

LEIBSTADT EXPERIENCE

SUMMARY

NOMENCLATURE/ACRONYM

POWER UPRATE (PU)

MAXIMUM EXTENDED OPERATING DOMAIN (MEOD)

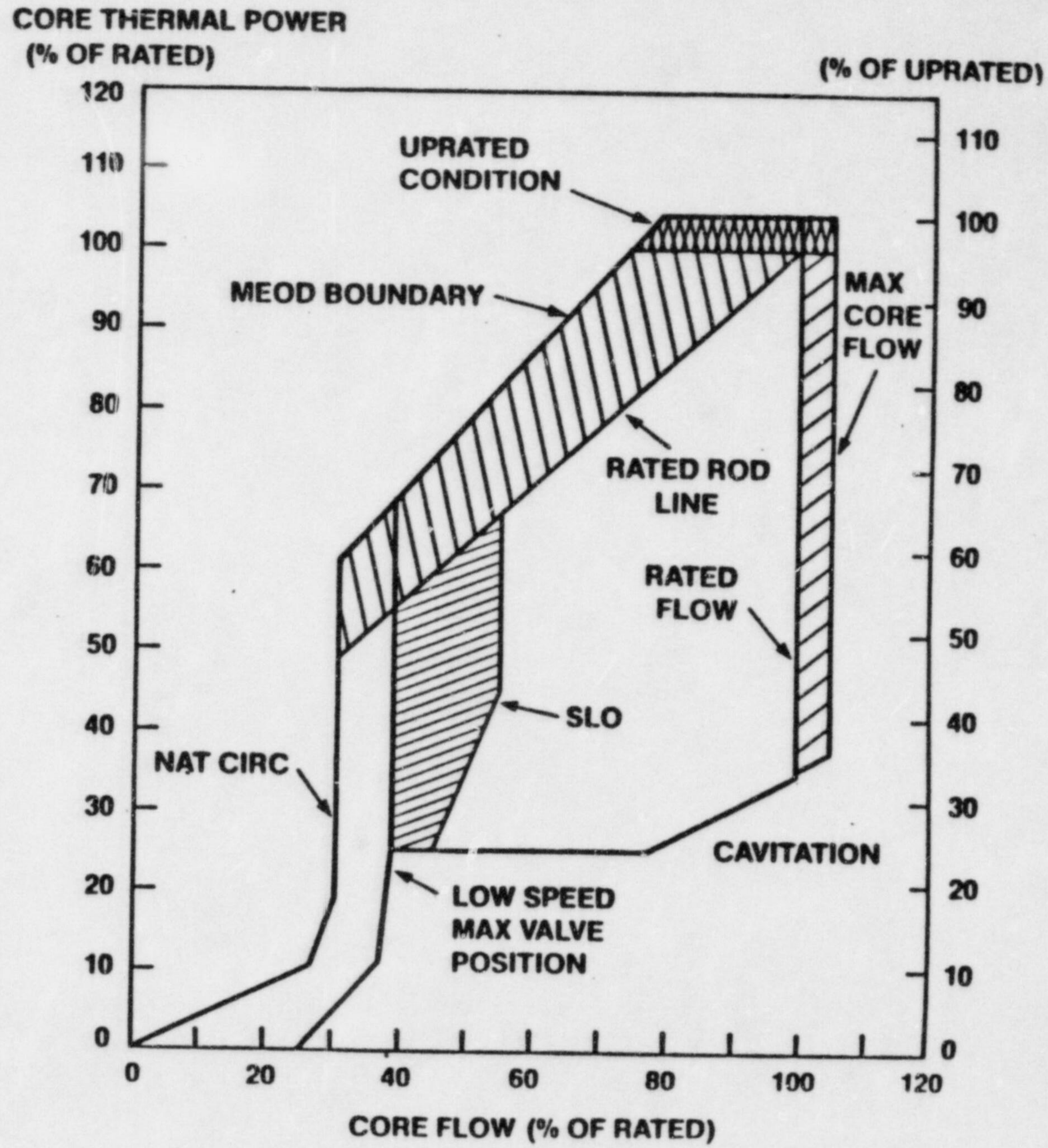
- EXTENDED LOAD LINE LIMIT ANALYSIS (ELLLA)
- INCREASED CORE FLOW ANALYSIS (ICFA)
- APRM/RBM TECH SPEC MODIFICATION (ARTS)

FINAL FEEDWATER TEMPERATURE REDUCTION (FFWTR)

OUT OF SERVICE (OS) EQUIPMENT

- FEEDWATER HEATER(S) (FWH)
- MAIN STEAM ISOLATION VALVE(S) (MSIV)
- SAFETY RELIEF VALVE(S) (SRV)
- RECIRCULATION LOOP (SLO)

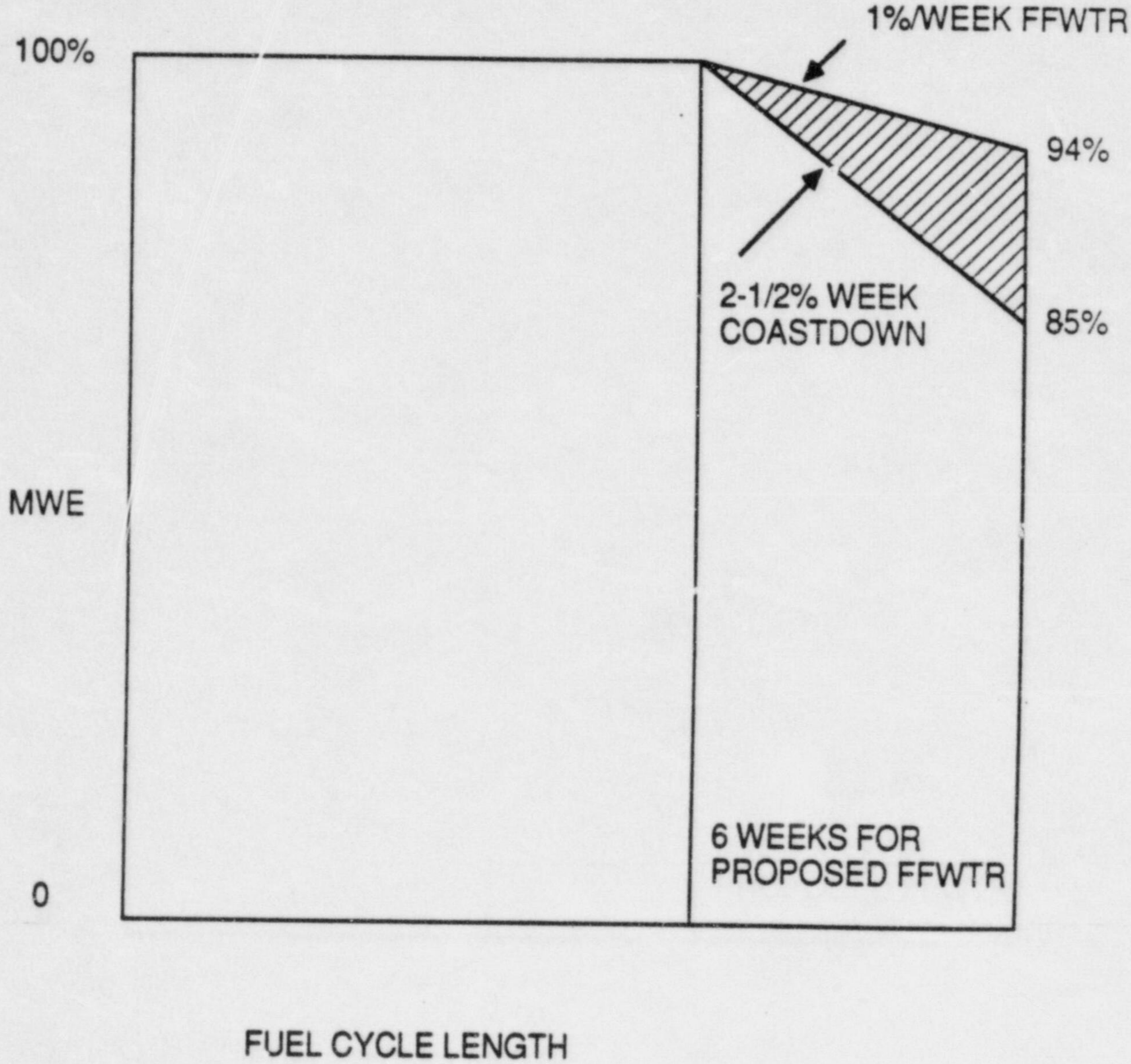
KKL 104.2% POWER UPRATE OPERATING MAP



Significant Increase in Allowed Operating States Achieved via

- **Design & Licensing Evaluations**
- **Technical Specification Changes**
- **Minor Setpoint and Hardware Adjustments**

CAPACITY FACTOR IMPROVEMENT DUE TO PROPOSED 250°F FFWTR=0.7%



TYPICAL REACTOR PERFORMANCE IMPROVEMENT PROGRAM PACKAGE ELEMENTS

POWER UPRATE

- OPERATION AT 104.2% THERMAL POWER
- 5% INCREASE IN STEAM FLOW PRODUCTION
- MAJOR UPRATE BEYOND 5% IN THE FUTURE

MAXIMUM EXTENDED OPERATING DOMAIN

- RATED POWER ALLOWED FROM 75% TO 105% CORE FLOW
- APRM AND RBM HARDWARE AND TECH SPEC MODIFICATIONS

FINAL FEEDWATER TEMPERATURE REDUCTION

- RATED THERMAL POWER ALLOWED FROM 420°F TO 250°F FEEDWATER TEMPERATURE

OUT OF SERVICE EQUIPMENT

- RATED POWER ALLOWED WITH FWHOS
- INCREASE ALLOWABLE OUT OF SERVICE TIMES WITH SRVOS
- APPROXIMATELY 95% POWER ALLOWED WITH MSIVOS
- APPROXIMATELY 70% POWER ALLOWED WITH SLO

> OPERATING STATES EXTENDED TO ACHIEVE MAXIMUM OPERATING FLEXIBILITY WITHOUT MAJOR HARDWARE MODIFICATION.

POWER UPRATE

OBJECTIVE

- UPRATE THE UNIT POWER TO THE DESIGN RATING OF 105% STEAM FLOW (104.2% THERMAL POWER)

- ALLOW OPERATION OF OTHER IMPROVEMENT PROGRAMS AT THE UPRATED POWER

8

BENEFITS

- PROVIDE ADDITIONAL GENERATING CAPACITY

SIGNIFICANT UPRATE BEYOND 5% IS FEASIBLE IN THE FUTURE WITH FUEL AND PLANT MODIFICATIONS AND TESTING CONFIRMATION

MAXIMUM EXTENDED OPERATING DOMAIN

DEFINITION

- MEOD FEATURES = ELLLA + ICFA + ARTS

OBJECTIVE

- RATED POWER OPERATION OVER TARGET FLOW RANGE OF 75% - 105% FLOW (OR BEYOND)
- ENHANCE THE ROD BLOCK MONITOR (RBM) SYSTEM (FOR PRE-BWR 6 PLANTS)
- INCREASE MARGINS TO THERMAL LIMITS
- SIMPLIFY OPERATIONS

FEATURES

- EXTENDED LOAD LINE REGION (ELLR)
- INCREASED CORE FLOW REGION (ICFR)
- ELIMINATION OF ARPM TRIP SETDOWN
- NEW RBM SYSTEM AND CORRESPONDING TECH SPEC MODIFICATIONS

MAXIMUM EXTENDED OPERATION DOMAIN

BENEFITS

INCREASED MANEUVERABILITY, FLEXIBILITY AND CAPACITY FACTORS

- REACTIVITY COMPENSATION
- FASTER STARTUP
- ELIMINATE APRM SETDOWN
- LOAD FOLLOWING
- ELIMINATE UNNECESSARY RBM ROD BLOCKS

FLOW CONTROL SPECTRAL SHIFT

PUMPING POWER BENEFITS

OTHER FEATURES

OPTION - AUTOMATIC FLOW DEMAND LIMITER

COMBINED WITH FFWTR TO MAXIMIZE FUEL CYCLE ECONOMICS

FINAL FEEDWATER TEMPERATURE REDUCTION

OBJECTIVE

- ALLOW RATED THERMAL POWER OPERATION WITH REDUCED FEEDWATER TEMPERATURE
- TO EXTEND AN OPERATING CYCLE
- TARGET RATED FEEDWATER TEMPERATURE OF 250°F

BENEFIT

- IMPROVED CAPACITY FACTOR RELATIVE TO EOC COASTDOWN
- IMPROVED FUEL CYCLE ECONOMICS
- COMBINED WITH MEOD TO MAXIMIZE BENEFIT

OUT OF SERVICE EQUIPMENT

OBJECTIVE

ALLOW POWER OPERATION WITH VARIOUS EQUIPMENT OUT OF SERVOCE

OUT OF SERVICE EQUIPMENT INCLUDES:

- FEEDWATER HEATER (S)
- MAIN STEAM ISOLATION VALVE (S)
- SAFETY RELIEF VALVE (S)
- RECIRCULATION LOOP

BENEFIT

IMPROVED OPERATING AND MAINTENANCE FLEXIBILITY

INCREASED AVAILABILITY

PROTECTION AGAINST COSTLY PLANT SHUTDOWN - INSURANCE POLICY

REACTOR PERFORMANCE IMPROVEMENTS

LICENSING SERVICE

LICENSING ANALYSIS REPORT

SUGGESTED TECH SPEC MODIFICATIONS

LICENSING SUPPORT
- PRESENTATION/DISCUSSION
- QUESTIONS/ANSWERS
- 10CFR 50-59 EVALUATIONS
- SIGNIFICANT HAZARD CONSIDERATION

FSAR UPDATE

CURRENT LICENSING STATUS

104.2% POWER UPRATE

- DUANE ARNOLD LICENSED AND OPERATING
- LEIBSTADT PROVISIONALLY LICENSED AND OPERATING
- SEVERAL BWR PLANTS COMPLETED FEASIBILITY STUDY

MAXIMUM EXTENDED OPERATION DOMAIN

- LEIBSTADT, GRAND GULF, AND KUO SHENG LICENSED AND OPERATING
- PERRY FSAR APPROVED
- MANY BWR PLANTS LICENSED AND OPERATING WITH ELLLA AND ICFA
- HATCH, DUANE ARNOLD, MONTICELLO AND KKM LICENSED AND OPERATING WITH ARTS

FINAL FEEDWATER TEMPERATURE REDUCTION

- SEVERAL BWR PLANTS LICENSED AND OPERATING

OUT OF SERVICE EQUIPMENT - FWH, MSIV, SRV, SLO

- PERRY AND GRAND GULF FOR FWHOS
- SEVERAL BWR PLANTS LICENSED AND OPERATED WITH SLO
- WORK IN PROGRESS FOR SEVERAL BWR PLANTS

PLANT LICENSING STATUS

PLANT	PRODUCT LINE	PU	MEOD	ELLLA	ICFA	ARTS	FFWTR W/ICFA	FWHOS	SLO*
DOMESTIC PLANTS									
DRESDEN 2&3	3								X
MILLSTONE	3			X					X
MONTICELLO	3			X		X			X
PILGRIM	3			X	X		X		X
QUAD CITIES 1&2	3								X
BROWN'S FERRY 1-3	4				X				X
BRUNSWICK 1&2	4								X
COOPER	4								X
DUANE ARNOLD	4	X		X		X			X
FERMI 2	4								X
FITZPATRICK	4								X
HATCH 1&2	4			X	X	X	X		X
HOPE CREEK 1	4								X
PEACHBOTTOM 2&3	4				X		X		X
SUSQUEHANNA 1&2	4			X					X
VERMONT YANKEE	4								X
LIMERICK 1&2	4								X
HANFORD 2	5								X
LA SALLE 1&2	5								X
NMP2	5								X
GRAND GULF 1	6		X					X	X
PERRY 1	6		X				X	X	X
RIVERBEND 1	6								X
CLINTON 1	6								X
INT'R'N'TL PLANTS									
NUCLENOR	3								X
CAORSO	4								X
CHINSHAN 1&2	4			X					X
FUKUSHIMA 2	4								
MUHLEBERG (KKM)	4			X	X	X	X		
COFRENTES	6						X		
KUO SHENG 1&2	6		X						
LEIBSTADT (KKL)	6	X	X				X		X

* ANALYSIS COMPLETED--LICENSING EITHER COMPLETED OR UNDER IMPLEMENTATION

BWR/3 MAJOR (10-12%) POWER UPRATE LICENSING STATUS

FEASIBILITY STUDY COMPLETED FOR TWO BWR/3 PLANTS

GE-NRC MEETING FEBRUARY, 1986

- PROVIDED OVERVIEW TO STAFF
- STAFF GENERALLY RECEPTIVE
- OBTAINED NRC OPINIONS ON LICENSING ISSUES

MEETING WITH MR. BERNERO MARCH, 1986

- IDENTIFIED NECESSARY LICENSING STEPS AND ANALYSES
- DEVELOPED ACTION TO ESTABLISH LICENSING PATH AND PROCESS

LICENSING ACTION PLAN DEVELOPED FOR IMPLEMENTATION

LEIBSTADT OPERATING EXPERIENCE WITH REACTOR PERFORMANCE IMPROVEMENT PROGRAMS

(

LEIBSTADT PLANT DEFINITION

- UTILITY: KERNKRAFTWERK LEIBSTADT
- LOCATION: SWITZERLAND
- BWR PRODUCT LINE: BWR6 MARK III
- POWER RATING: 3012 MWTH, 950 MWE
- RATED CORE FLOW: 88.5 MLBM/HR
- VESSEL SIZE: 238 IN. DIAMETER
- NUMBER OF BUNDLES: 648
- GE-6 FUEL DESIGN
- CONVENTIONAL FUEL LOADING (NON-CCC)
- POWER DENSITY: 54.1 KW/l
- STATUS: COMPLETED CYCLE 1 & CYCLE 2

LEIBSTADT REACTOR PERFORMANCE

IMPROVEMENT PROGRAMS

MAXIMUM EXTENDED OPERATING DOMAIN (MEOD)

- OPERATION WITH EXPANDED POWER/FLOW REGION
- RATED POWER ALLOWED FROM 75% TO 105% CORE FLOW

SINGLE LOOP OPERATION (SLO)

- EXTENDED OPERATION WITH ONE RECIRC LOOP UP TO 70% POWER

FINAL FEEDWATER TEMPERATURE REDUCTION

- OPERATION WITH REDUCED FEEDWATER HEATING TO EXTEND THE FUEL CYCLE
- RATED THERMAL POWER ALLOWED FROM 420°F TO 250°F FEEDWATER

POWER UPRATE

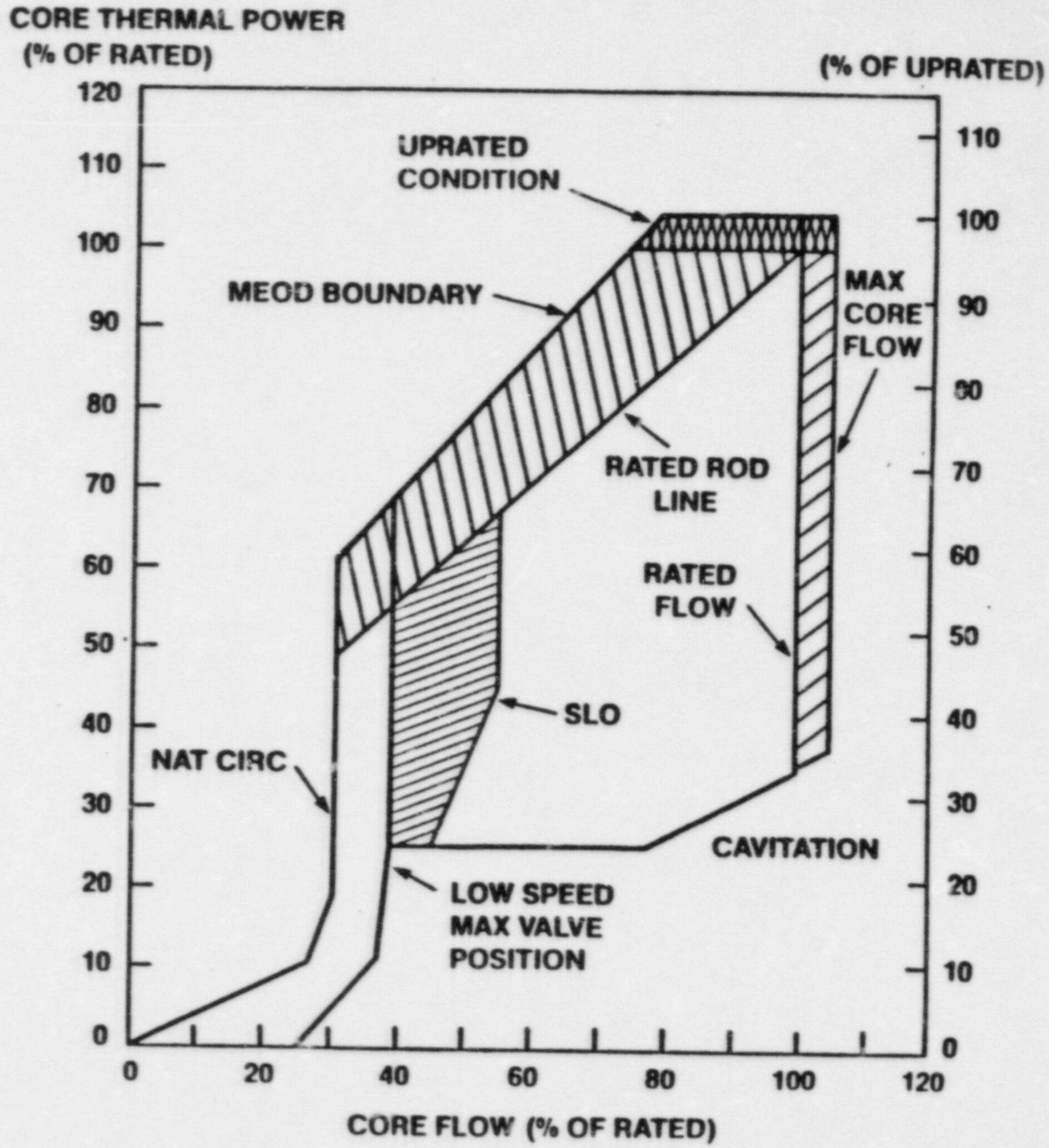
- OPERATION AT 104.2% THERMAL POWER

OPERATING STATES EXTENDED TO ACHIEVE MAXIMUM OPERATING FLEXIBILITY WITHOUT MAJOR HARDWARE MODIFICATION

KKL 104.2% POWER UPRATE OPERATING MAP

Significant Increase in Allowed Operating States Achieved via

- Design & Licensing Evaluations
- Technical Specification Changes
- Minor Setpoint and Hardware Adjustments



KKL LICENSING STATUS

MEOD--LICENSED

SLO--LICENSED

FFWTR--LICENSED

POWER UPRATE

- 102% POWER LICENSED
- 104.2% POWER TESTING
- INITIAL DEMONSTRATION AT BEGINNING OF CYCLE-2 APPROVED
- EXTENDED DEMONSTRATION TEST FOR REMAINDER OF CYCLE-2 APPROVED
- LICENSING REVIEW AND DISCUSSION IN PROGRESS FOR CYCLE-3 UPRATE OPERATION
- PROVISIONAL LICENSE FOR CYCLE-3 104.2% POWER OPERATION

KKL OPERATING EXPERIENCE

STARTUP TEST DEMONSTRATION

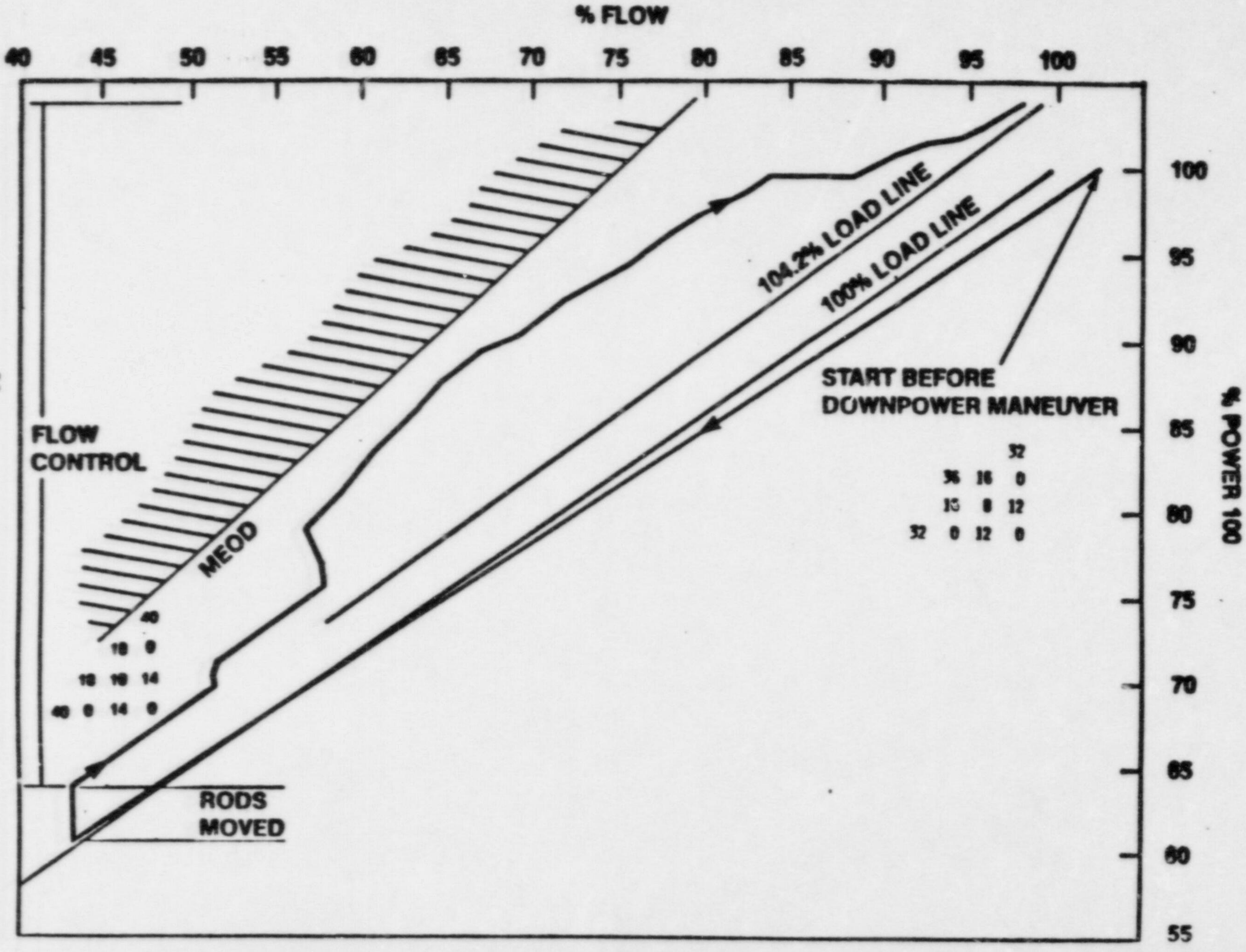
CYCLE 1 OPERATING PARAMETER

- 98% AVAILABILITY; 96% CAPACITY FACTOR
- 12% MARGIN ON MCPR
- 2% MARGIN ON KW/FT
- 3% MARGIN ON MAPRAT
- 90% TO 102% CORE FLOW RANGE AT RATED POWER

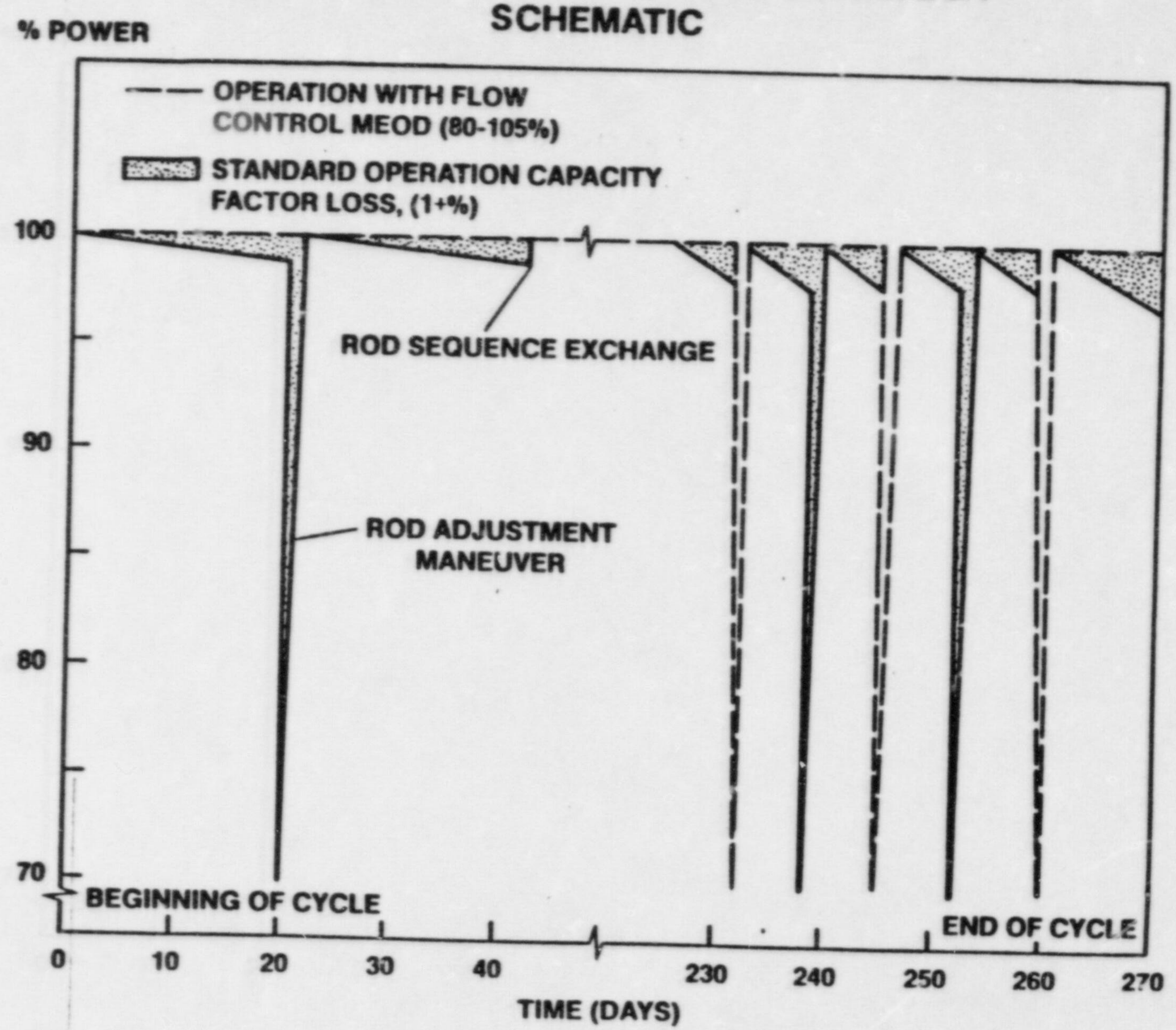
CYCLE 2 OPERATING PARAMETER

- 99% AVAILABILITY; 98% CAPACITY FACTOR
- 6% MARGIN ON MCPR
- 2% MARGIN ON MLHGR
- 4% MARGIN ON MAPRAT
- 75% TO 104% CORE FLOW RANGE AT RATED POWER
- 104.2% OPERATING POWER
- FULLY UTILIZED EXPANDED OPERATING MAP FOR STARTUP AND RESTART

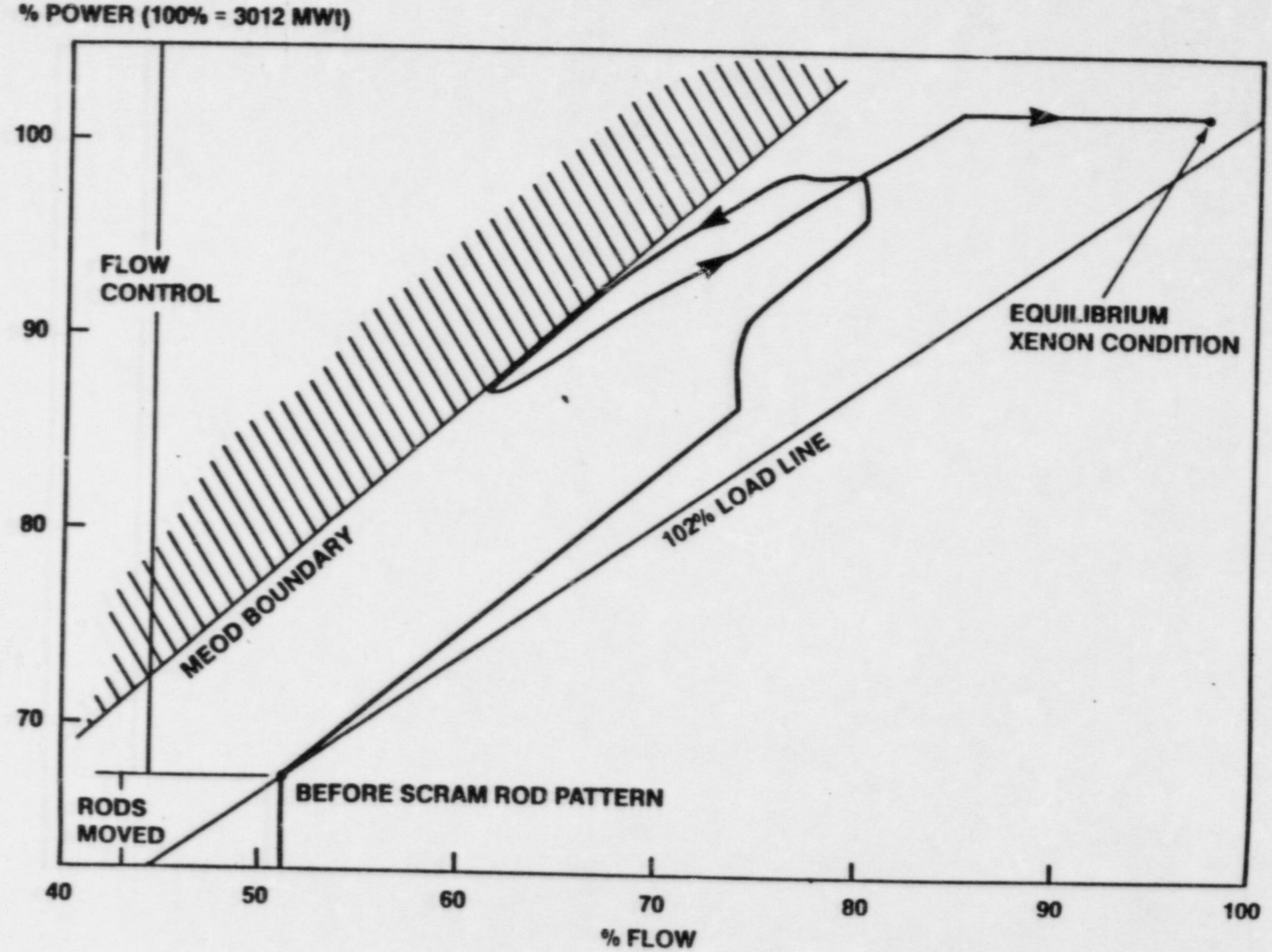
KKL ROD ADJUSTMENT TO REACH 104.2% POWER FOR TESTING



POWER OUTPUT FOR STANDARD vs MEOD FLOW CONTROL OPERATING STRATEGY
SCHEMATIC



KKL HIGH XENON RESTART LOAD LINE



LEIBSTADT EXPERIENCE SUMMARY

ANALYSIS, TESTING, & OPERATING EXPERIENCE HAS DEMONSTRATED VIABILITY OF OPERATION IN EXTENDED MODES

--ANALYSIS VERIFIED VIA TESTING/OPERATIONS

--EXPECTED MARGINS DEMONSTRATED

PLANT OPERATIONS HAVE BEEN SIGNIFICANTLY ENHANCED

--OPERATING FLEXIBILITY/SIMPLICITY

--CAPACITY FACTORS

SIGNIFICANT LICENSING EXPERIENCES HAVE BEEN OBTAINED

SUMMARY

MANY BWR PLANTS HAVE BEEN SUCCESSFULLY LICENSED WITH THESE REACTOR PERFORMANCE IMPROVEMENT PROGRAMS

MOST OF THESE PLANTS HAVE OPERATED WITH THEM RESULTING IN SIGNIFICANT ECONOMIC AND OPERATING FLEXIBILITY BENEFITS

PLANT OPERATION PERFORMANCE IMPROVEMENTS OF THE

GENERAL ELECTRIC (GE) BOILING WATER REACTORS (BWR'S)

T. C. LEE, General Electric Company
175 Curtner Avenue, M/C 740
San Jose, California 95125
(408) 925-6136

R. C. STIRN, General Electric Company
175 Curtner Avenue, M/C 740
San Jose, California 95125
(408) 925-6139

ABSTRACT

This paper summarizes some of the plant operation performance improvement techniques developed by the General Electric Company Nuclear Energy Business Operation for the General Electric Boiling Water Reactors (GE BWR's). Through the use of both thermal and plant hardware operating margins, substantial additional flexibility in plant operation can be achieved resulting in significant improvements in plant capacity and availability factor and potential fuel cycle economics for the currently operating or requisition GE BWR plants. This list of techniques includes expanding the BWR thermal power/moderator flow operating domain to the maximum achievable region, operation with a single recirculation loop out of service and operation at rated thermal power with reduced feedwater temperatures. These plant improvements and operating techniques can potentially increase plant capacity factor by 1% to 2% and provide additional fuel cycle economics savings to the GE BWR's owners.

INTRODUCTION -

The design of the current operating General Electric Boiling Water Reactors (GE BWRs) is based on the proper combination of many design variables and operating experience. These factors contribute to the achievement of reliability, performance and fuel cycle economy. Some of the examples that accounted for the significant flexibility that existed in the operation of a GE BWR to date are given below:

1. Load Following Capability

As the difference between peak and minimum load demand increases and the percentage of nuclear generation capacity in utility system grows, it becomes more important to perform load following operation and to make the grid system more flexible. Load following is a term describing a power plant whose power is raised and lowered to meet the day-to-day demand of its electrical grid. GE BWR power plants can basically change power easily and rapidly by controlling recirculation flow and control rods in the allowable licensed thermal power/moderator flow operation region. The BWR's load following potential and capability have been successfully demonstrated. Examples of these discussions and demonstrations are presented in References 1 and 2.

2. Reactor Water Recirculation System:

The GE BWRs are designed with the reactor water recirculation system to circulate the required coolant through the reactor core. For the GE BWR 5 and 6 product lines, the system consists of two loops external to the reactor vessel each containing a pump with a directly coupled water-cooled (air-water) motor, a flow control valve and two shutoff valves. High performance jet pumps located within the reactor vessel are used in the BWR recirculation system. The jet pumps, which have no moving parts, provide a continuous internal circulation path for a major portion of the core coolant flow.

3. Thermal Power Coastdown Capability:

Cycle extension initiated at the end of a normal fuel cycle could be obtained by permitting the reactor thermal power level to decrease gradually after all control rods have been fully withdrawn at the end of an operating cycle. As the core reactivity decreases due to fuel depletion, positive reactivity is introduced into the core by reducing the core thermal power level. Thermal power coastdown utilizes the power coefficient to obtain the reactivity gain to extend the fuel cycle. The end-of-cycle coastdown method is being used extensively by BWR owners to stretch out more energy out of the existing fuel to improve the fuel cycle economy.

Due to the flexibility and conservative margin in the GEBWR design, further enhancement in operation performance can be made to achieve higher plant capacity factor. This paper summarizes three plant operation performance improvement techniques that are developed in relationship to the three examples given above to further enhance the performance and flexibility of the GE BWRs.

DESCRIPTION

In addition to the performance and flexibility demonstrated by the existing BWR operating plants, the current lines of the GE BWRs have been designed with significant conservatisms and margins. Based on operating experience established by GEBWR's over the years, the feasibility has been demonstrated that by utilizing some of these design and operating margins (both thermal and plant hardware), further enhancements in BWR plant operation flexibility can be achieved. This is accomplished by defining new modes of operations and performing safety and impact analyses and/or providing hardware modifications to justify that all requirements under the Code of Federal Regulations are met for these operation techniques.

This section summarizes these three special modes of operation developed for the GEBWR's by use of the above discussed operating margins. They are:

1. Operation in a modified reactor operating domain by extending the existing thermal power/moderator flow operation region. This is termed "Maximum Extended Operating Domain" in this paper. This mode of operation enhances the BWR to reach rated power faster, maintain rated power longer and further enhances the load following capability.

2. Operation with only one recirculation system flow loop running. This is called "Single Loop Operation" in this paper. This mode of operation provides insurance to protect against recirculation system equipment malfunction to maintain the GE BWR reliability.

3. Operation with reduced rated feedwater temperature at rated thermal power. This is called "Feedwater Temperature Reduction Operation" in the paper. This mode of operation provides an alternative to thermal power coastdown to extend an operating fuel cycle as well as provides protection against feedwater heating system equipment failures.

A. Maximum Extended Operating Domain

1. Description

The existing BWR Technical Specification allowable thermal power/moderator flow operation region for the operation of the GE BWR is defined by:

a) The 100% rated power condition
b) The 100% rated steam flow power constant control rod line
c) The 100% rated moderator core flow
d) Recirculation system component cavitation at low power
e) Minimum core flow resulting from pump speed or recirculation flow control valve position

The expanded power flow map under the maximum extended operating domain improvement program can be divided into 3 regions.

a) Expanded operation in the lower than 100% rated core flow region which is termed extended load line region.

b) Expanded operation in the higher than 100% core flow region which is termed increased core flow region.

c) Expanded operation above 100% rated power which is termed power uprate region.

A typical GE BWR power/flow operating map with the existing allowable operating region and the typical maximum extended operating domain is presented in Figure 1. The extended load line region is operationally feasible due to the existing operational thermal limit margins at these operation conditions as the result of improved plant hardware protection designs. The increased core flow region is available due to the reactor water recirculation system excess design capacity. The power uprate region is available due to the additional design margin utilized at 100% rated design power.

Safety and impact evaluation consistent with the analyses documented in the existing Standard Final Safety Analysis Report (FSAR) are required to justify operation in this extended domain. Reformatting of a few Technical Specification and minor plant hardware modifications are also necessary.

Several of the currently operating BWRs have demonstrated rated power operation over a core flow range of 87% to 105% of rated and operation under the corresponding constant control rod position load line. For the GEBWR 6 product line, feasibility studies and licensing evaluations using Nuclear Regulatory Commission (NRC) approved methods and codes have shown that rated power operation over a core flow range of 75% to 105% of rated and operation under the corresponding constant control rod position load line are both safe and operationally feasible. Several of the currently operating BWR's have been uprated from their initial thermal power rating to a new rating ranging from 102% to 105% of rated.

2. Benefits

In addition to the added load following capability with the expanded operating region, the primary benefits of the maximum extended operating domain fall into two main areas: Reactivity compensation beyond midpoint of the operating cycle and plant startup with either xenon or xenon free condition.

In order to ensure that 8%[a] or more excess flow capability exists at rated power for exposure compensation, additional operating room is required above the rated rod line in order to compensate for power reduction during plant startups with transient xenon. Gross power reduction prior to re-establishment of equilibrium xenon condition at rated power have been observed, from plant operating experience, to be as great as 10%-12% during startups with peak xenon and 8-10% during xenon free startups. In either case, excess flow capability of approximately 25% would be required in order to ensure that subsequent to attainment of xenon equilibrium, the plant would have an additional 8% of excess flow available for reactivity compensation. Therefore, in order to achieve rated power in the shortest time possible and to maintain 100% power thereafter, it is advantageous to allow for operation above the standard allowed 100% rod line during power ascensions. The extended load line region provides this operational flexibility.

Once rated power is achieved, the additional flow range at rated power as provided by both the extended load line region and the increased core flow region can be used to compensate for reactivity reduction due to fuel burn-up during the operating cycle. Rated power can be maintained longer without control rod movements. This is especially beneficial late in the operating cycle if deep control rod manipulation is not recommended.

[a] 8% is the estimated flow change required to compensate for reactivity reduction for about two weeks of exposure. The time represents the nominal interval between power reductions for surveillance tests and power shaping rod movements.

In addition, the extended load line and increased core flow region provides the required flow range for optimized flow control spectral shift operation. This mode of operation offers fuel cycle advantages over standard operation at rated power and flow.

Flow control spectral shift is a method used to increase fuel utilization thereby reducing fuel cycle costs. The concept of flow control spectral shift is basically to operate at rated core power and the lowest core flow possible throughout the cycle until the "all rods out" condition is reached. Core flow is then gradually increased to maintain rated power until maximum core flow is attained. Operation at rated power and reduced core flow results in a neutron energy spectrum shift due to the higher core average void content. The higher energy spectrum achieved during reduced core flow generates more plutonium and results in a more bottom peaked power distribution. In addition, uranium utilization is improved since a larger fraction of the fission power is generated from U238 fast fissions. As core flow is increased toward end of cycle, the power shifts to the top of the core. This results in an increase in cycle exposure length because power is shifted to a region of lower exposure and higher fissile plutonium content and the void fraction is decreased resulting in increased reactivity. If constant cycle length is desired, initial U235 loading can be decreased.

It must be noted that optimized spectral shift mode of operation is <u>not</u> part of this improvement program. This program merely provides the ground work and necessary flow range required to accomplish spectral-shift operation.

GE estimates a capacity factor improvement ranges from 0.3% to 1% from the extended load line and increased core flow region. Plant benefits vary between GE BWR product lines and existing plant modifications in place (e.g., Pellet-Clad-Interaction PCI resistant fuel). The additional potential fuel cycle economic improvements due to spectral shift operation ranges from 2 to 3% depending on flow range and operating strategy utilized.

The obvious benefit of the power uprate region is increased power production. The actual benefit resulting from this option will depend on the replacement power cost in different parts of the world.

3. Supporting Analysis

In order to implement operation in the

maximum extended operating domain, safety and impact evaluations consistent with the analyses documented in the existing Final Safety Analysis Report (FSAR) are required. The software package includes the analyses to support safe plant operation in the proposed extended region as depicted in Fig. 1 including a licensing report for submittal to the appropriate Nuclear Regulatory Agency/Commission and the recommended changes to plant Technical Specifications. The hardware portion includes the modification to the Neutron Monitoring System to accommodate the extended operating region. The detail scope of the required software and hardware analyses and modifications has been developed by the General Electric Company. In general, the highest feasible rod line (upper bound of the extended load line region), the highest feasible core flow and the highest uprate power level will be determined from thermal, reactivity and stability consideration as well as plant hardware capability, vibration and internal pressure difference evaluations. General Electric Thermal Analysis Basis (GETAB) evaluation, core wide transients, loss of coolant accidents and containment responses are some examples of the required analyses. Detail analysis scope is to be provided to the BWR utility owners interested in this plant improvement program.

B. Single Recirculation Loop Operation

1. Description

The standard US BWR plant Technical Specification specifies a 12 hour operation limit when one recirculation loop is inoperable. This results from the fact that analyses were not originally performed to justify extended single loop operation as part of the GEBWR standard plant offering. For single loop operation, the core pressure-drop is reduced, the total discharge flow from the active bank of jet pumps increases at rated (2 loop) drive flow, flow through the inactive loop reverses direction, and the jet pump flow pattern in the reactor lower plenum becomes highly asymmetric relative to rated conditions with balanced two loop inlet flow. However, the core power distribution remains unchanged because the core coolant flow distribution is not altered during single recirculation loop operation.

Since single loop operation was not a plant design basis operating condition, it is necessary to establish a proper technical basis for operating in this mode. This implies that a broadly based analysis must be performed to establish the limiting factors that still satisfy the many criteria applied in normal design practice. Ultimately, these limits translate to a maximum core thermal power in the range of 65% to 75% of the rated thermal power level dependent on the different reactor water recirculation system designs for various product lines of the GE BWRs.

Almost all currently operating GEBWRs have completed the licensing evaluations for single loop operation. A few of these plants have successfully operated with only one recirculation loop running up to as long as two months to prepare for the maintenance work required to repair the malfunction component at a planned outage.

2. Benefits

The major benefit for single recirculation loop operation is protection against costly plant shutdown due to recirculation system equipment failures resulting in improvements in plant availability and corresponding capacity factor. A recirculation loop can become inoperable due to malfunction of one of several component items (e.g., recirculation pump, pump motors, valves, etc.). The actual capacity factor improvement realized will depend on the achievable power level during operation with single recirculation loop. Based on operating plant experience and data, General Electric estimates single loop operation will result in a capacity factor improvement of approximately 0.1%. This is estimated based on the average number of days downtime due to recirculation loop problems and the probability that it will happen during a BWR plant lifetime. Another benefit of single recirculation loop operation is that it allows for maintenance schedule flexibility. Unplanned recirculation loop maintenance can be postponed until a planned outage or a period when power demands are low.

The ability to operate and produce power with single recirculation loop with no time limitation is a sound insurance policy for any GE BWR owner to have in place.

3. Supporting Analysis

In order to implement single recirculation loop operation, safety and impact evaluations consistent with the analyses documented in the existing Final Safety Analysis Report (FSAR) are required. This software package includes a licensing report for submittal to the appropriate Nuclear Regulatory Agency/Commission and the recommended changes to plant Technical Specifications. The detail scope of the required analyses to establish the bases and limits justifying safe operation in single loop operating mode has been developed by the General Electric Company. In general, a new break spectrum analysis for the loss of coolant accident during single loop operation will be performed. General Electric Thermal Analysis Basis (GETAB) evaluation, core wide transients, containment responses and stability considera-tion are some of the examples of the required analyses. Various other safety and operational impact evaluations which address vessel vibration, reverse flow phenomenon, flow uncertainties, water chemistry, cavitation, recirculation flow control, neutron noise and neutron monitoring system setpoints will be performed. Detail analysis scope is to be pro-vided to the BWR utility owners interested in this program.

C. Feedwater Temperature Reduction Operation

1. Description

The standard US BWR license constrains plant rated thermal power operation at feedwater temperatures corresponding to the rated design values. Operation with reduced feedwater temperature at rated power can be separated into two categories. One is operation during an opera-ting cycle and the other is operation at the end of the fuel cycle.

Operation at reduced feedwater tempera-ture during an operating cycle is termed "Feed-water Heater(s) Out of Service Operation" in this paper. This mode of operation is used in the event that a malfunction of the feedwater heating system occurs reducing feedwater temperature and continued operation of the plant is highly desirable to maintain power output. Operation with reduced feedwater temperature at the end of an operating cycle is termed "Final Feedwater Temperature Reduction" in this paper. This mode of operation is aimed toward extending the opera-ting cycle to improve capacity factor (relative to the thermal power coastdown method) and fuel cycle economics.

Final Feedwater Temperature Reduction is an alternative to the traditional method of end of cycle (EOC) thermal power coast-down used extensively by BWR owners to extend an operating fuel cycle. The coast-down method of extended operation is accomplished at reduced thermal power but with normal feedwater temperature and steam pressure. Continued operation with thermal power coastdown is possible because reduced steam voids, reduced fuel temperature and reduced equilibrium xenon yield reactivity gains which compensate for reactivity losses due to deple-tion of fissionable material. This method of cycle extension results in relatively rapid rate of decrease of electrical output resulting from reduced core thermal power.

Final Feedwater Temperature Reduction extends a fuel cycle by maintaining rated core thermal power. This is accomplished by systematically valving out the feedwater heaters. Reduction of feedwater heating causes colder water to flow into the reactor vessel. Colder feedwater mixes with the reactor water increases core subcooling, thereby decreases steam voids in the core which in turn increases core reactivity as more moderator is available. This mode of operation requires moderator control to maintain rated core thermal power. To optimize the benefit and minimize perturba-tion to plant operation, feedwater temperature should be reduced in several gradual steps. These step changes occur periodically to com-pensate for the continually decreasing core reactivity due to fuel depletion. This gradual reduction in feedwater temperature extends the fuel cycle while maintaining core thermal power at or near rated throughout the entire extended operating time. Electrical power output de-creases during Final Feedwater Temperature Reduction operation, but the rate of decrease is less than that of EOC coastdown. As core inlet subcooling increases, more thermal mega-watts are used to heat up the moderator and therefore less steam is produced. As steam production decreases, electrical power output in turn decreases. The reduction in electrical power output for Final Feedwater Temperature Reduction is approximately 1-1 1/2% per week whereas end of cycle coastdown results in a 2 1/2% reduction per week. This results in a five to ten percent increase in station electrical power output over coastdown capabi-lity through the period of extended operation.

These modes of operation with reduced feedwater temperatures during the operating cycle as well as at the end of the cycle is feasible due to existing design margins in the BWR plant. One example of this is the margins to thermal fatigue usage on the feedwater nozzles and spargers. Although reduced feed-water temperature operation increases fatigue on the feedwater nozzles and spargers, the design margins on both of these components allow the operation in these reduced feedwater temperature modes with only minor time limita-tion at reduced feedwater temperature operation during an operation cycle.

Proper technical bases need to be established for these modes of operation. Safety and equipment impact evaluations are therefore required to justify operation in these feedwater temperature reduction modes. Minor technical specification modifications are also required.

Several of the currently operating BWRs have used Final Feedwater Temperature Reduction to extend their operating cycle. For the GEBWR6 product line, impact and licensing evaluations have shown that end of cycle rated thermal power operation with a feedwater temperature reduction of 170°F (down from 420°F to 250°F rated) is safe and operationally feasible. Evaluations have also been completed for some BWRs for Feedwater Heater(s) out of service operation during an operating cycle at rated thermal power with a feedwater temperature reduction of 100°F (down from 420°F to 320°F of rated for the GEBWR6 product line).

2. Benefit

Feedwater Heater(s) out of service operation during an operating cycle protects against costly plant shutdown due to feedwater heating system equipment malfunctions, thus providing availability and capacity factor improvements as well as operational flexibility by allowing maintenance schedule to be arranged at a planned outage or a period when power demands are low. Final feedwater temperature reduction offers significant fuel cycle economic advantages through lower fuel cost resulting from cycle extension while at the same time reducing the capacity factor losses incurred by the more traditional end of cycle thermal power coastdown method of extending fuel cycles. For some product lines of the GE BWR's, capabilities exist that a reduction in feedwater temperature up to 170° F is feasible. This reduction in feedwater temperature can extend a cycle by 1000 MWD/T exposure or up to 2 months while maintaining 100% thermal power. Although electrical output will fall off during the extended period, the decrease is only about half of what occurs during an end of cycle coastdown.

General Electric estimates these feedwater temperature reduction modes of operation will result in a capacity factor improvement of about 1% per cycle of BWR plant operation over the thermal power coastdown method of cycle extension with additional fuel cycle economic improvements.

3. Supporting Analysis

In order to implement these feedwater temperature reduction modes of operation, safety and impact evaluations consistent with the analyses documented in the Final Safety Analysis Report (FSAR) are required. This software package includes a licensing report for submittal to the appropriate Nuclear Regulatory Agency/Commission and the recommended changes to the plant Technical Specifications. The detail scope of the required analyses to justify operation in these feedwater temperature reduction operation modes has been developed by the General Electric Company. General Electric Thermal Analysis Basis (GETAB) evaluations, core wide transients, loss of coolant accident, containment responses and stability consideration are some examples of the required analyses. Various other safety and operational impact evaluations which address feedwater nozzle and sparger fatigue, reactor protection system setpoints, impact on internals as well as optimized feedwater temperature reduction steps from both heat rate efficiency and compliance with fuel design standpoint are to be performed. Detail analysis scope is to be provided to the BWR utility owners interested in this program.

CONCLUSION

The current lines of operating GE BWRs have demonstrated excellent operation flexibility and performance. In addition, the GE BWRs have also been designed with significant conservatisms and margins. By use of some of these thermal and plant hardware operation margins, flexibility in GE BWR plant operation can be further enhanced resulting in substantial capacity factor improvements, protection against costly plant shutdown due to equipment failures and subsequently economic benefits to the GE BWR owners. These plant operation improvement techniques can potentially increase plant capacity factor by 1% to 2% and provide additional fuel cycle economy savings. The capacity factor improvements could roughly be translated to a cost saving of about 2 to 4 million US dollars per year (1984 value) to a GE BWR owner utility based on an approximate average equivalence of 1% capacity factor equals to 2 million US dollars per year to a GE BWR owner utility.

Plant safety and impact evaluations are required to justify these special modes of operation. Reformatting a few Technical Specifications and minor plant hardware modifications are also required. The Plant Operation Performance Improvement Programs described in this paper have been proven to be operationally feasible and have been successfully demonstrated by several operating GE BWR owners throughout the United States and the rest of the world.

REFERENCES

1. P. N. Ontko, General Electric Company, "Load Following at Brunswick" presented at the American Nuclear Society Winter Meeting, November 12-16, 1978, Washington, D.C.

2. D. G. Carroll, R. G. Serenka, H. R. Propst, General Electric Company, "BWR Maneuvering Capability", presented at the American Power Conference, April 23-25, 1979, Chicago, Illinois.

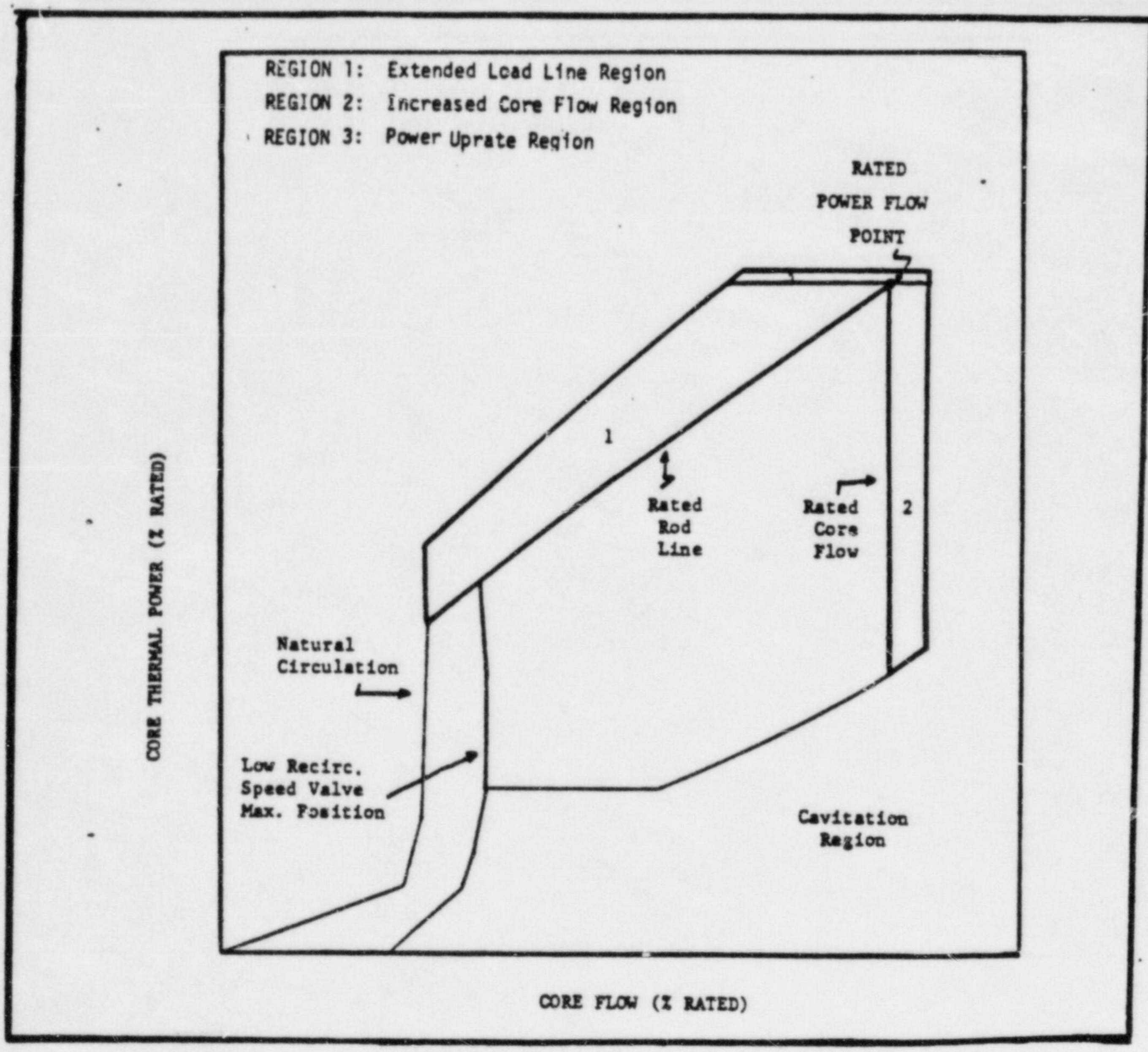


FIGURE 1 TYPICAL GE BWR POWER/FLOW MAP

GENERAL ELECTRIC BWR/6 THERMAL

POWER UPRATE CAPABILITY ASSESSMENT

T. C. Lee, General Electric Company
175 Curtner Avenue, M/C 740
San Jose, California, 95125
(408) 925-6519

ABSTRACT

An assessment study was performed to determine the feasibility of increasing the core thermal power to approximately 104.2% of the original licensed rated value in order to obtain an increase in the core steam flow rate of 5% for a typical General Electric BWR/6 power plant. Areas which are expected to be affected by the increase in the rated thermal power were evaluated in the study. Results of this assessment study show that, overall, the 5% increase in steam flow and the associated 4.2% increase in core thermal power are feasible for a typical BWR/6 plant. However, reduced fuel thermal margins may constrain operation at the 104.2% uprated power level during part of the operating cycle with current GE6/7 fuel designs. Improved GE8B fuel design with a 14.4 kw/ft linear heat generation rate limit, coupled with recently USNRC approved more realistic loss-of-coolant accident licensing models and assumptions, should allow operation at the uprated power level during the entire operating cycle. This assessment conclusion for a typical BWR/6 plant is derived mainly based on the reactor core performance capability of the plant. For specific plant capability assessment application, detailed systems and equipment impact, as uniquely applied to each individual plant, are to be evaluated. It is also felt that, based on assessment results of this study, significant major power uprates beyond the 4% level is potentially feasible for BWRs in the future with fuel and plant modifications and testing verifications.

INTRODUCTION

The General Electric (GE) Boiling Water Reactors (BWRs) have been designed with significant conservatism and engineering design margins. Based on operating experience established by the BWRs over the years, it has been demonstrated that some of the existing analytical, systems and equipment margins that are a part of the GE BWR plant design can be utilized to increase the maximum operating power above the current commercial basis rated reactor power rating.

Most of the GE BWR/2 and 3 product line plants have already uprated their initial power ratings to their design rating of approximately 104%. The GE BWR/4 product line plants have also been coming on-line in the past ten years and their performance at the initial power level is also well established. For these plants, efforts have been underway to uprate their thermal power level to the designed 104% rating.

The GE BWR/6 product line is the sixth generation of the design evolution of the GE BWRs. The BWR/6 product line is capable of producing up to 20% more power from the same size pressure vessels as used in the BWR/5 product line without increasing the size of the respective buildings or supporting systems. Therefore, the BWR/6 product line represents the highest core average power density plants of all of the GE BWR product lines.

The design basis thermal level of BWR/6 plants also represents an approximately 4.2% margin over the commercial and licensing basis rated power level. Recently improved calculational and test data reduction/verification methods have demonstrated considerable extra safety margin. These existing analytical, system and equipment margins can be used to increase the operating thermal power above the originally licensed rated power level. With sufficient engineering assessments, licensing analyses and confirmatory testing, it is possible to increase and license the BWR/6 at 4.2% higher thermal power resulting in a 5% increase in core steam flow production and a corresponding increase in electrical power generation. The obvious benefit for the thermal power uprate is increased power proudction, resulting in economic benefits to the BWR owners.

General Electric has developed a two-step Power Uprate Program for the BWR Owners to implement the power uprate for their BWR plants. The analytical assessments and equipment evaluation, including reviewing existing plant and/or startup test data or definition of required testing to confirm the plant is capable of the uprate, constitute Phase 1 of this program. Phase 2 covers the plant-specific safety licensing analyses and documentation necessary to assure long term safe operation of the plant and would normally begin only after the successful completion, with positive findings, of the Phase 1 feasibility study. This two-phase Power Uprate Program is used for all GE BWR product lines. A Phase 1 power uprate assessment study was recently performed for a typical BWR/6 product line plant. This paper summarizes the results of a few major subjects from this capability assessment study for a 5% steam flow (approximately 4.2% thermal power) uprate for this BWR/6 plant.

FEASIBILITY EVALUATION

The areas that are expected to be affected by the 4.2% increase in operating power were assessed in this typical BWR/6 plant feasibility study. The areas assessed include:

1. Nuclear Steam Supply (NSS) and Balance of Plant (BOP) systems and equipment. This assessment also included the Turbine Island Systems.
2. Reactor Core Performance.
3. General Electric Thermal Analysis Basis (GETAB) and Transient Performance.
4. Emergency Core Cooling System (ECCS) Loss-of-Coolant Accident (LOCA) Performance.
5. Containment Capability.
6. Vessel Internals Vibration.
7. Technical Specification Modifications.

These affected areas were assessed both qualitatively and quantitatively. Existing startup test data were used to assess the available margins and the impact of power uprate on these margins for the typical BWR/6 plant.

In addition, the impact of the power uprate on the various plant performance improvement modes of operation[1] available to the BWR/6, such as the Maximum Extended Operating Domain, Single Recirculation Loop Operation and Feedwater Temperature Reduction Operation, was also examined in this study.

The various NSS and BOP systems and equipment, as well as the Turbine Island impact, are plant-specific. Plant-unique heat load information, design practice and startup or operating data are required to evaluate the impact of power uprate on these systems and equipment.

These specific systems and equipment were evaluated in this typical BWR/6 plant feasibility study. The evaluations show that some adjustments to the turbine protection and control system may be required. It may also be necessary to make some adjustments in some of the cooling systems prior to operation at the uprated condition. Some NSS system setpoints modifications will also be necessary to implement the uprate. Overall, the conclusion is that these typical BWR/6 plant systems and equipment have enough design margin to accommodate the 4.2% thermal power uprate. Details of each of the systems evaluated are not presented in this paper (except the Turbine Control Valve) as they are very plant-dependent. A discussion of the Turbine Control Valve performance is provided because this is one of the major components impacted by the uprate.

The impact of the 4.2% power uprate on the GETAB, transient performance and containment capability are also not presented in this paper. The overall conclusion of this assessment is that there is minimal impact on their performance and capability except some system setpoints will need to be revised. Detailed technical specification modifications are not presented due to the plant-specific nature.

The remaining paper summarizes assessment results of the following major performance subjects from this feasibility study. They are:

1) Turbine Control Valve Performance;
2) Reactor Core Operating Limits;
3) ECCS LOCA Performance;
4) Reactor Vessel Internals Vibration.

This typical BWR/6 feasbility study is performed for the entire power flow operating map as defined by the Maximum Extended Operating Domain (MEOD)[1] program available for the BWR/6 product line. A typical BWR/6 power flow operating map with MEOD is illustrated in Figure 1. A typical 104.2% power uprate operating map with MEOD is illustrated in Figure 2.

TURBINE CONTROL VALVE

The Turbine Control Valve (TCV) of a BWR/6 plant is used to control the steam flow generated in the vessel to the turbine. Heat balance data and the valve flow capacity data were used to determine if the valve can accommodate the 5% increase in steam flow for the uprate. When the TCV has enough capacity to handle the additional steam flow, operation at the uprated power will result in the TCV to be operating higher in the flat region of the TCV lift curve. Operation at the upper end of the TCV position will be subjected to a slightly larger travel distance when responding to small variation in steam flow. Therefore, in order to limit duty on the TCV as well as provide good TCV operating characteristics, it is desirable to avoid operating the valves at the upper end of their position by increasing the pressure regulator setpoint.

Increasing the pressure regulator setpoint to lower the TCV operating position will, in turn, increase the core operating pressure of the plant.

The BWR/6 assessment study included a review of a typical BWR/6 plant startup data. This review concluded that the TCVs of this typical BWR/6 plant have enough capacity to accommodate the 5% increase in turbine steam flow, thereby not limiting the uprate capability of the plant. The data indicated that the capacity of the TCV at the valve wide open (VWO) position when operating at the current 975 psia design TCV inlet pressure, is expected to be about 105% of the rated steam flow. The TCVs are expected to be capable of providing responsive pressure control at a position corresponding to about 97% of the VWO steam flow. Therefore, when operating at 105% steam flow condition, it would be necessary to provide a higher steam pressure at the valve in order to pass the higher steam flow at the same valve position. An approximation of this effect is 10 psi for each 1% extra steam flow desired. Thus, a 30 psi increase would be needed to achieve the required 3% flow margin.

A review of startup data of this typical BWR/6 plant indicated that the steam line pressure drop (at rated steam flow) from the vessel dome to the TCV is about 45 psi which is 20 psi less than the rated power design basis of 65 psi (1040 psi-975 psi). The operating pressure drop is expected to increase to 50 psi at the 105% steam flow condition. Since the present BWR/6 design limitation on operating reactor dome pressure is 1060 psia, this means that a maximum of 1010 psia steam pressure will exist at the TCV. This 35 psi increase over the rated turbine inlet pressure of 975 psia should be sufficient to pass the 105% steam flow at a TCV position which is equivalent to about 96.5% of VWO flow.

It is concluded, based on the review of this typical BWR/6 plant data and normal design practice, that an operating pressure increase of somewhere between 10 to 20 psi is expected to be acceptable for the 104.2% power uprate (from 1040 psi to 1060 psi at the dome). TCV valve flow capacity, heat balance and steam line pressure drop data are strongly plant-dependent. The assessment and conclusion presented above are determined by reviewing a single BWR/6 plant. If TCV data are not available, pressure control testing is recommended to determine and confirm the acceptability of operation at this condition, and to determine the desired operating and TCV pressures.

REACTOR CORE OPERATING LIMIT PERFORMANCE

Increasing the core thermal power by 4.2% will affect the operating thermal margins for the reactor core. The BWR/6 assessment study was performed to examine the impact of the uprated power on core thermal limit margin. The thermal limits examined were: Minimum Critical Power Ratio (MCPR), Linear Heat Generation Rate (LHGR), Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), and hot excess reactivity. The assessment was performed on a typical BWR/6 plant for an early fuel cycle and an equilibrium cycle.

For this thermal margin assessment, a control rod pattern study was performed using the GE 3-Dimensional core simulator at intervals of 1000 MWd/t fuel exposure through the projected cycles to assess reactivity and thermal margins. Initial attempts were performed at the uprated conditions and various flow conditions to allow for flow control reactivity compensation utilizing the MEOD[1]. Rated power conditions were utilized for the part of the operating cycle when design thermal margins were less than desired. Hot excess reactivity was determined by back-burning from the appropriate Haling calculation to the desired cycle exposure and an all Rods Out (ARO) 3-D simulation.

The assessment results of the reactor core operating limits are presented in the following subsections.

A. Minimum Critical Power Ratio (MCPR) Operating Limit

A 4.2% power increase would be expected to result in approximately 4% decrease in actual operating MCPR value. Projected and existing BWR/6 operating data indicate that significant margin exists between the MCPR operating limit and the actual operating value at the current rated power level. Therefore, sufficient MCPR margin is anticipated at the 104.2% uprate condition. The results of the rod pattern study conclude that MCPR is not the limiting factor for the uprate capability.

B. Maximum Linear Heat Generation Rate (LHGR) Limit and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Limit

Available and predicted BWR/6 data indicate that the margin to the 13.4 Kw/ft LHGR limit and the MAPLHGR limit for BWR/6 plants may be limiting the ability to uprate power during part of the operating cycle. Currently, the MAPLHGR limit is constrained by the 13.4 Kw/ft LHGR limit but is also very close to being determined by the 2200°F Peak Cladding Temperature (PCT) limit at some exposures. With a 4.2% power uprate, the MAPLHGR limit may be set by the Emergency Core Cooling System (ECCS) PCT constraints. The subsequent section on ECCS impact provides more discussion on this MAPLHGR limit restriction.

The results of the 3-D simulator rod pattern study show that the limiting parameters for both the early and equilibrium cycles were peak Kw/ft and MAPLHGR. The study indicates that uprated power could be maintained for approximately 75% of the time for early cycles, whereas a larger bottom-peaked Haling power distribution in equilibrium cycle potentially reduced uprated capability to about 50% of the cycle. If the specific plant exhibits significant margin to thermal limits through knowledgement core management, 104% power could potentially be maintained throughout the cycle.

It must be noted that this power uprate study result is based on a typical BWR/6 plant with current GE6 fuel design. With improvement in a fuel design to a peak LHGR limit of 14.4 Kw/ft (i.e., the GE8B fuel), it is expected that LHGR would not limit the power uprate capability.

C. Hot Excess Reactivity

The desired minimum hot excess reactivity at beginning of cycle used for GE core design is 1%. This 1% hot excess reactivity is expected to be achievable with 104.2% power uprate condition with the normal core design and fuel cycle evaluation. The rod pattern study results show that, in general, the uprated power reduces hot excess reactivity by about 0.2% ΔK.

D. Summary Discussion

The core operating limit discussion presented above concluded that the most limiting thermal limit parameters are found to be MAPLHGR and MLHGR limits. The rod pattern study performed indicates that the typical BWR/6 plant has sufficient margin to meet all core thermal limits for a 4.2% power uprate for approximately 75% of an early fuel cycle and 50% of the time for an equilibrium fuel cycle. To increase the likelihood of meeting these core thermal limits and provide operating margins comparable to that of the current power level, GE would recommend the improved GE8B fuel design to increase the LHGR limit to 14.4 Kw/ft. In addition, new ECCS LOCA licensing methodology[2] would also be suggested to restore MAPLHGR margin with the increased LHGR (see later section on ECCS).

It is also important to note the importance of power shaping during uprate operation, Figure 3 shows the predicted typical 104.2% power uprate early cycle and equilibrium cycle end-of-cycle optimum Haling power profiles. The modest increase in the bottom peak of the equilibrium cycle optimum Haling power distribution resulted in a reduction in uprated power availability for that cycle in comparison to that of the early cycle. Since the Haling power profile is dependent upon the power shaping performance in prior cycles, it is imperative that knowledgeable core management be performed in implementing uprated power operation.

EMERGENCY CORE COOLING SYSTEM PERFORMANCE

Two separate Emergency Core Cooling Systems (ECCS) performance analyses can normally be performed for BWR plants at normal operating conditions. The first analysis uses standard legal requirements of 10CFR50 Appendix K model assumptions and assumes one single active component failure (N-1 assumption). This analysis is normally used for most BWR/6 plants. The second analysis assumes one component out-of-service plus one single active component failure (N-2 assumption). This analysis normally takes credit for the Special Emergency Heat Removal (SEHR) system plus various ECCS model improvements.

Whether N-1 assumption or N-2 assumption (or both) is to be used as the licensing basis is dependent on the regulatory requirements set forth by the regulators of each individual country. The calculated PCT of the required ECCS analysis is to be below the legal requirement 2200°F PCT limit of 10CFR50.46.

The PCTs for both current ECCS analyses for the typical BWR/6 plant assessed in this study are below the 2200°F limit. No MAPLHGR restrictions have been placed on the GE6 fuel design.

The ECCS analysis for power uprate must be performed at 102% of the uprated power level in accordance with legal requirement 10CFR50 Appendix K. This assessment study results show that the increase in power at the uprated condition will produce a slightly longer period of core uncovery following a loss-of-coolant accident (LOCA). As a result, the calculated PCT will increase for power uprate. Therefore, it may be necessary to apply slightly reduced MAPLHGR limits in some exposure range to keep the calculated PCT below the limit for the GE6 fuel design. The MAPLHGR multiplier is a plant specific value dependent on the amount of PCT margin existing currently and whether N-1 or N-2 assumption is required for the evaluation. N-2 analysis normally results in a higher PCT value. This typical BWR/6 assessment study showed that a MAPLHGR reduction of 1% is expected in the 10,000 to 20,000 MWD/ST exposure range for an N-2 LOCA calculation.

The MAPLHGR restrictions which may result from the ECCS power uprate analysis can be eliminated by the use of more realistic analysis assumptions. Use of the ANS 5.1-1979 decay heat model is expected to result in a calculated PCT below 2200°F even if a 20% adder is applied to the decay power term. Also, General Electric has obtained USNRC approval for a more realistic set of LOCA licensing methodology models[2]. The application of these models in place of the current models will typically result in reductions of 600 to 1000°F in the calculated PCT for the limiting case.

Furthermore, in order to increase the likelihood of meeting the LHGR thermal limit to provide full uprate capability, GE8B fuel design with 14.4 Kw/ft is recommended. The higher 14.4 Kw/ft LHGR limit will result in exceeding the 2220°F PCT limit with the current ECCS LOCA model. These more realistic LOCA models are essential for meeting the 2200°F PCT limit with no MAPLHGR restriction. Therefore, it is concluded that these realistic LOCA analysis assumptions and models coupled with the GE8B fuel design are expected to provide full 104.2% power uprate capability for the BWR/6 plants.

REACTOR INTERNALS VIBRATION ASSESSMENT

Fatigue strength is the primary material property of concern for reactor pressure vessel components subjected to high-cycle alternating stresses resulting from flow-induced vibration. Vibration measurements are made during plant startup testing by means of instrumentation located on components inside the reactor vessel. The purpose of vibration monitoring is to confirm the structural integrity of major components in the reactor with respect to flow-induced vibration in accordance with requirements of USNRC Regulatory Guide 1.20. This is done by comparing the measured vibration amplitudes (strain or displacement) against a set of acceptance criteria. The acceptance criteria are basically a set of frequencies and corresponding allowable amplitudes (for each sensor) derived from an analytical model. This ensures that stresses everywhere are below the material allowable stress.

An assessment was performed based on the vibration test data collected during startup testing of this typical BWR/6 plant. Reactor internal vibration tests were conducted during preoperational testing, 75% power flow rod line testing, 100% rod line testing and Maximum Extended Operating Domain (MEOD)[1] testing. Operating conditions during each test period include steady-state balanced flow, unbalanced flow, single loop operation (SLO)[1], and transient tests consisting of one pump and two pump trips, and load reject from rated flow conditions.

There are different levels of data reduction techniques. In the first level, a conservative estimate is obtained. The percent criteria for each mode is conservatively estimated from the Root Mean Square (RMS) value from a peak spectrum and all modes are combined by absolute summation. If the sum of the percent criteria exceed 100, a more realistic estimate of the stresses is made by using the second level of data reduction, filtering. Again the contribution of the modes are added by absolute summation. Higher levels of data reduction are used only if necessary.

For this uprate vibration assessment, only balanced flow responses for a few major sensors were evaluated in detail. A qualitative and quantitative review of the other sensors' responses were performed which showed that they are not of concern.

To predict the vibration responses anticipated in the power uprate region of the power flow map, data on the MEOD rod line and the two-loop maximum flow rate line were examined. In addition, the vibration along the constant 80% core flow rate line were also investigated. All vibration responses are estimated from response spectra using the first level data reduction technique (conservative estimates from RMS values).

The vibration responses were examined as a function of core flow on the MEOD rod line, as a function of power at maximum core flow, and as a function of various rod lines at 80% constant flow rate line. Results of the evaluation show that two sensors responded maximum vibration amplitude of close to 100% of acceptance criteria at 100% power at maximum core flow. All other sensors and conditions examined are well below the 100% allowable limits.

Second level data reduction technique (filtering) was then used for these two sensors. Filters are designed to pass signals in some frequency band and reject signals in all others. Band pass filters pass signals that fall between two cutoff frequencies. Hence, the vibration at a specific mode can be separated from the total vibration. This modal response can then be compared to the criteria at that mode. The present criteria for each mode is then added by absolute summation. When filtering, care is taken to assure that filter characteristics and their possible attenuation effects are properly considered. This is achieved by making a spectrum of the filtered signal and superimposing it on the spectrum of the unfiltered signal, and correcting for attenuation effects, if any.

The filtered modal responses for these two sensors are compared to the first level data reduction evaluation results. The comparisons show that filtering eliminated substantial conservatism. Modal responses for these two sensors are plotted as a function of power. The data are then extrapolated to 104.2% power. The extrapolation indicates vibration amplitudes of less than 80% of acceptance criteria using the second level of data reduction.

It is concluded, based on the assessment results from the typical BWR/6 plant startup test data, that the vibration of vessel internals is not expected to exceed allowable limits at the 104.2% uprate power condition.

CONCLUSION

The results of this assessment study show that a typical BWR/6 plant is capable of a 5% increase in steam flow (equivalent to a 4.2% increase in core thermal power). This conclusion is mainly based on the reactor core performance capability of the plant. Detailed system equipment and Balance-of-Plant impact, as applied to each individual plant, are to be evaluated on a plant-by-plant basis for any specific BWR/6 plant application.

While the typical BWR/6 plant may not be able to operate at this uprated power all of the time due to thermal operating limit constraints with current GE6 fuel design and current licensing model assumptions, it is expected that the improved GE8B fuel design, coupled with recently USNRC approved more realistic LOCA licensing models and assumptions, can achieve full uprate power operation during the entire cycle.

In order to implement plant operation at the uprated power level, detailed plant-specific safety engineering licensing analyses and safety analysis report updates are necessary to assure long-term safe operation of the plant. Plant technical specification modifications and plant operation setpoint changes are also required.

A significant major thermal power uprate beyond the 4% level is judged to be technically feasible in the future. This major uprate will probably involve fuel design and plant modifications as well as testing verifications. A 4% uprate is feasible now and is relatively easy to implement. This should pave the way and provide verification experience for an ultimate thermal power uprate for BWR owners in the future.

REFERENCES

1. T. C. Lee, R. C. Stirn, General Electric Company, "Plant Operation Performance Improvements of the General Electric (GE) Boiling Water Reactors (BWRs)", presented in the International Nuclear Power Plant Thermal Hydraulics and Operations Topical Meeting on October 22-24, 1984, Taipei, Taiwan.

2. B. S. Shiralkar, J. G. M. Anderson, A. B. Burgess, S. A. Wilson, General Electric Company, "Evolution of LOCA Analysis at General Electric", presented in the International Nuclear Power Plant Thermal Hydraulics and Operations Topical Meeting on October 22-24, 1984, Taipei, Taiwan.

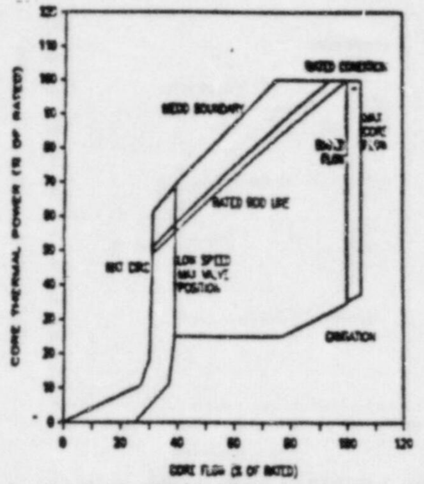


Fig. 1 Typical BWR/6 MEOD Operating Map

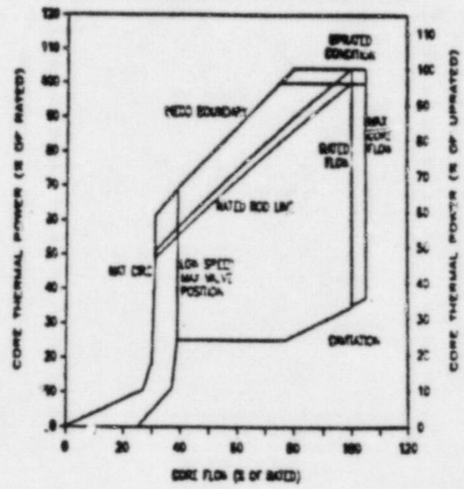


Fig. 2 Typical BWR/6 Power Uprate MEOD Operating Map

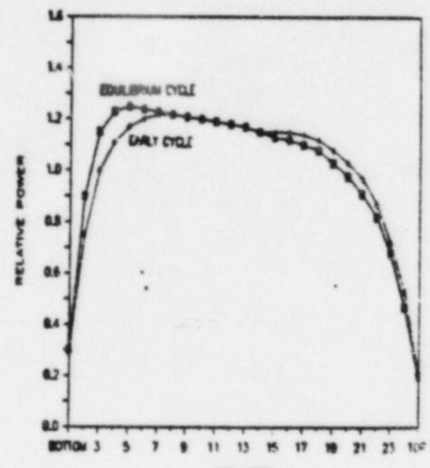


Fig. 3 Typical BWR/6 Power Uprate EOC Power Distribution Comparison

POWER UPRATE

- o OBJECTIVE

  PROVIDE ADDITIONAL POWER FROM OPERATING PLANT AT MINIMUM COST

- o ALTERNATIVES

  - STRETCH TO 105% STEAM FLOW
  - INCREASE BY 10% OR MORE

- o PROGRAM APPROACH

  - TWO PHASES

    - o FEASIBILITY EVALUATION
    - o DETAILED ANALYSES, MODIFICATIONS

  - SCHEDULE

    - o FEASIBILITY -6 MONTHS
    - o DETAILED ANALYSIS - 10 MONTHS
    - o PLANT MODIFICATIONS - DEPENDS UPON POWER LEVEL

- o BENEFITS

  - EXTRA CAPABILITY AT LOW COST
  - REPLACEMENT ENERGY SAVINGS
  - MARGIN TO SCRAM SETPOINTS

FEASIBILITY EVALUATION

- o ESTABLISH UPRATED POWER
  - ASSESS SAFETY AND CAPABILITY
  - REACTOR SYSTEMS
  - TURBINE GENERATOR
  - POWER CONVERSION AND AUXILIARY SYSTEMS
  - DESIGN REVIEW

- o DETERMINE EFFECT ON SYSTEMS
  - REACTOR INTERNALS
  - PRESSURE RELIEF
  - RESIDUAL HEAT REMOVAL
  - PRESSURE REGULATOR
  - TURBINE GENERATOR
  - POWER CONVERSION AND AUX SYSTEMS

- o EVALUATE IMPACT ON FUEL
  - OPERATING MARGINS
  - REACTIVITY
  - STABILITY
  - FUEL CYCLE

- o ACCESS IMPACT WITH OTHER CORE PERFORMANCE IMPROVEMENTS

- o REPORT
  - CONCLUSIONS
  - RECOMMENDATIONS FOR PHASE 2
  - REVIEW

DETAILED ANALYSES

- o HEAT BALANCE AND SETPOINTS
- o REACTOR FLUID CONDITIONS
- o TRANSIENT ANALYSES
  - OVERPRESSURE
  - MCPR
  - STABILITY
- o ACCIDENT EVALUATIONS
  - CONTAINMENT
  - ECCS
  - RADIOLOGICAL
- o REACTOR INTERNALS
  - DIFFERENTIAL PRESSURES
  - FATIGUE USAGE
  - RADIATION
- o REVISED OPERATING PARAMETERS
  - TECH SPECS
  - POWER FLOW MAP
  - INSTRUMENT SETPOINTS
  - PROCESS COMPUTER DATA BANK
- o LICENSING REPORT
  - CONCLUSIONS
  - NRC MEETING SUPPORT

BENEFITS

- o REPLACEMENT ENERGY SAVINGS

- o CAPACITY BENEFITS

- o TOTAL SAVINGS FOR 5% UPRATE $30 - 150M

- o OTHER CONSIDERATIONS

    - PAYS BACK COST IN FIRST YEAR
    - COST OF ADDED CAPACITY MUCH LESS THAN NEW PLANT/KW

EQUIPMENT MODIFICATIONS

(MONTICELLO: 10% UPRATE)

- o MAIN TURBINE
  - NEW NOZZLE PLATE, BUCKETS
  - NEW OR MODIFIED 2ND, 3RD STAGE DIAPHRAGMS
  - NEW PACKING RINGS, THRUST BEARINGS
  - LARGER RELIEF VALVE
- o CONDENSATE DEMINERALIZER
  - NEW BEDS
- o ESTIMATED COST (INCL. DETAILED ANALYSIS) $30M

PROBABILITY OF SUCCESS

- KEY ELEMENTS OF SUCCESS
  - PLANT SAFETY
  - EQUIPMENT CAPABILITY
  - REGULATORY ACCEPTANCE

ENCLOSURE 3

MEETING OF NOVEMBER 21, 1986,
WITH GE

Enclosures

Agenda

NUMAC Log Count Rate Meter

NUMAC Area Radiation Monitor

NUMAC Process Radiation Monitor

NUMAC Log Radiation Monitor

NUMAC Rod Worth Minimizer

Microprocessor Based Nuclear Power Plant Instrumentation

Microprocessor Based Rod Worth Minimizer

NRC NUMAC REVIEW

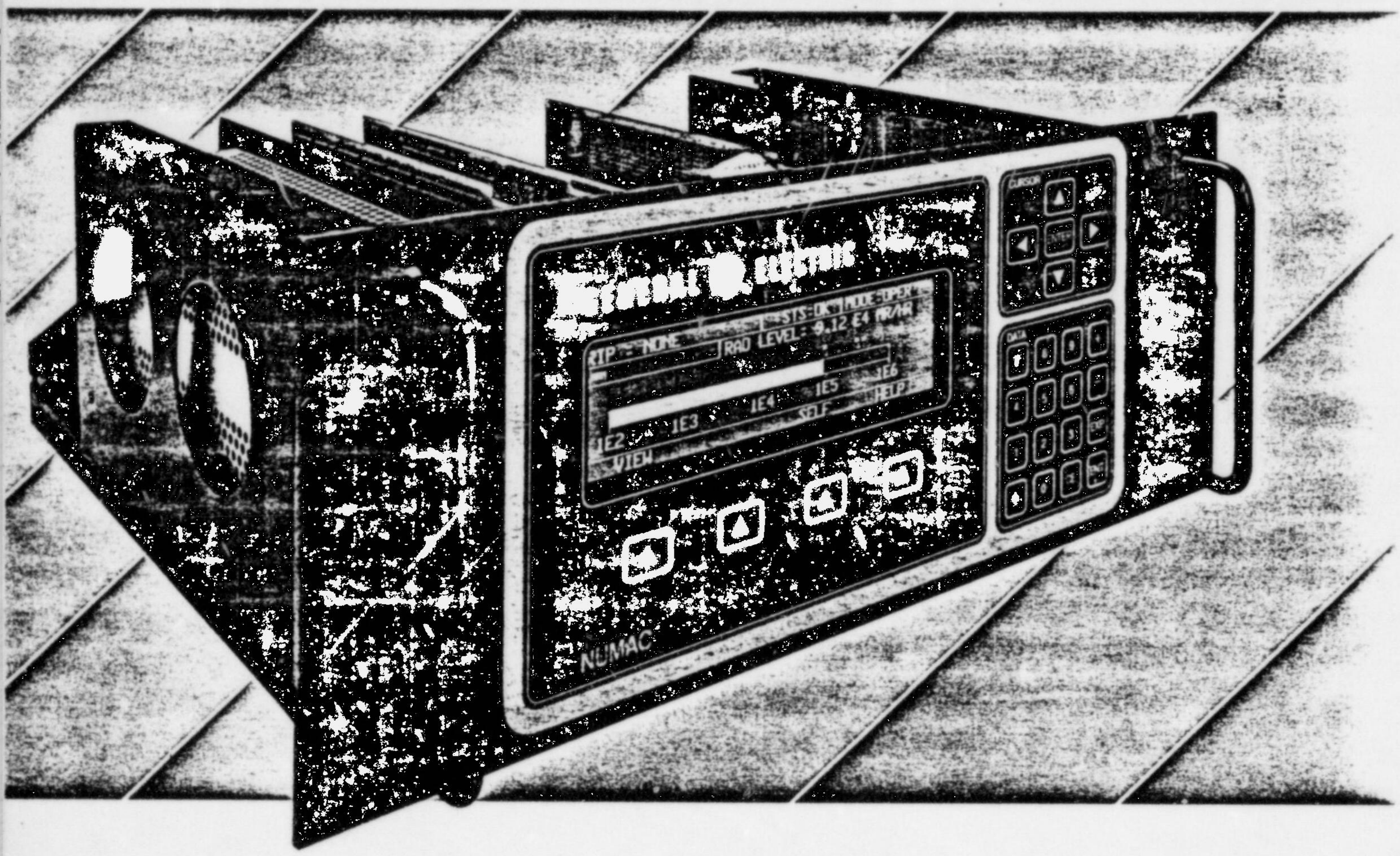
11/21/86

Room 6A, 111 North Market

8:00 - 8:15	AGENDA REVIEW	Dave Reigel
8:15 - 9:00	OVERVIEW OF NUMAC PRODUCT LINE	Dave Reigel
9:00 - 9:30	DAIRYLAND/BIG ROCK POINT FOCUS	Dave Reigel
9:30 - 11:00	LAB TOUR AND HANDS ON DEMO	Eng. Staff
11:00 - 11:45	GETRAM	Bill Rowe Fred Chao
11:45 - 1:00	LUNCH	
1:00 - 2:00	DISCUSSIONS/ADDITIONAL ITEMS	All

NUMAC™

Process Radiation Monitor (304A3703)



The Process Radiation Monitor (PRM), member of the GE Nuclear Measurement Analysis and Control (NUMAC) family of microcomputer based instruments replaces 194X900 which meets the requirements (less the period circuit) of the Source Range Monitor Performance Specification 167A2210. The instrument is used with the NA-07 scintillation detector 117B1681G001 and remote pulse preamplifier 11C2276.

The input signal-conditioning module is the NUMAC discriminator which interfaces to the NUMAC microcomputer. Discriminator adjustment is accomplished through the front panel operational control. The logarithmic count rate function, compensation, calibration, alarm and trip functions are performed by the NUMAC microcomputer.

The standard instrument range is seven decades from 1E-1 counts per second to 1E+6 counts per second.

Four trip circuits are provided: two upscale (Hi and Hi-Hi Radiation), one downscale (Lo Radiation), and one Inop (instrument in calibration, instrument failed, etc.). Each trip circuit has two outputs: one logic (to trip auxiliary units) and one alarm. Since the trip function is handled digitally by the microcomputer, there is no trip circuit drift. Trip accuracy depends only on measurement accuracy.

The instrument contains a polarization power supply for one detector. This supply is under microcomputer control and is adjustable from the operator panel. Power supply gross failures are annunciated via the Inop trips. Each detector power supply is provided with a hardwired, over-voltage protection circuit.

An internal calibration check facility is provided, and automatic calibration of the instrument is a feature of the NUMAC design.

Each PRM channel contains a voltage source to power its associated detector.

Voltage Range: 100 VDC – 1200 VDC
Maximum Current: 3 mA 100 VDC – 333 VDC
1 mA 333 VDC – 1200 VDC
Maximum Voltage
Ripple: 1% RMS
Alarm: When +/− 10% of setpoint is exceeded

Each instrument provides three independent adjustable trip circuits (Hi-Hi, Hi and Downscale). Trip hysteresis is adjustable from the front panel over the range of 0% to 25%.

Logic: 12 VDC, 25 mA
Alarms: 20 VDC, 50 mA

Negative pulses at PRM chassis input connector.

The log integration time constant (0–63% of step change) will not exceed the limits shown below for a factor of two increase in count rate at the discriminator input.

Change in Count Rate (CPS)	Standard Nominal Time Constant(sec)	Optional
0.1 to 0.2	40 ± 10	40 ± 12
1.0 to 2.0	40 ± 10	18 ± 6
10 to 20	35 ± 10	1.8 ± .4
1E+2 to 2E+2	13 ± 3	0.18 ± .03
1E+3 to 2E+3	1.0 ± .3	0.018 ± .002
1E+4 to 2E+4	0.1 ± .03	<.018
1E+5 to 2E+5	0.01 ± .003	<.018
higher	<.01	<.018

The instrument will produce a count at the log count rate output for a negative current pulse equal to or greater than 1.6 mA when the discriminator is set for maximum sensitivity.

Instrument Plus Preamplifier Sensitivity

For the combination of the pulse preamplifier and the PRM, a 2.0 mV negative step through a 10pf capacitor to the pulse preamplifier input will produce a count at the log count rate output. (This is equivalent to an instantaneous charge of 0.02 pico-coulombs.)

The Inop trip monitors gross failure of the detector power supplies, microcomputer "watchdog" timer, card-out-of-file and instrument-out-of-operate mode. Self-test detected failures are also reported via the Inop trips.

Each PRM channel has provisions for driving a recorder (1.0 volt full scale), process computer input (160 mV full scale), and remote meter (1.0 mA full scale). Noise and ripple are less than 0.5% of full scale.

Discriminator Range

The instrument will discriminate negative current pulses with amplitudes between 1.6 and 24 mA (as measured at the instrument input).

All accuracy measurements are referred to the front panel display which is also the basis for trip adjustments. At design center conditions, the observed count rate will not deviate from the true count rate by more than ± 1% of equivalent linear full scale over all seven decades. Coincidence counting losses are excluded from this specification statement.

The coincidence counting losses at 1XE+6 counts per second random input signal are less than 1.2% of equivalent linear full scale for the preamplifier and PRM instrument interconnected with triple-shielded 75 ohm cable (drawing number 167A2510).

At design limit conditions, the observed count rate will not deviate from the true count rate by more than ± 3% of equivalent linear full scale over the top six decades and by more than ± 5% of equivalent linear full scale for the bottom decade. Coincidence counting losses are excluded from this specification statement.

Drift at Design Center

± 1% of equivalent linear full scale for a thirty-day interval after initial warmup of at least two hours.

NUMAC™

Area Radiation Monitor (304A3704)

The Area Radiation Monitor (ARM), member of the GE Nuclear Measurement Analysis and Control (NUMAC) family of microcomputer based instruments, replaces up to eight indicator and trip units (129B2802). Each of the eight detector inputs is polled every 50 ms. The instrument can be used with ionization chambers, Geiger-Mueller (G-M) tubes or scintillation detectors. New auxiliary units are available on request, or the ARM can be interfaced to present 237X892 auxiliary units and 194X927 sensors and converters. The system is available safety related.

Two NUMAC four-channel femtoammeter models serve as input amplifiers when used with ionization chambers in the direct current (dc) mode. An analog-to-digital converter interfaces amplifier output to the NUMAC microcomputer.

When used with ionization chambers, G-M tubes or scintillation detectors in the pulse-counting mode, two NUMAC four-channel discriminators which interface directly to the NUMAC microcomputer act as input signal-conditioning modules. Front panel operational controls accomplish discriminator adjustment.

The maximum instrument range is eight decades from 10 mR/hr to 10,000 R/hr. Any number of decades up to a maximum of eight within this range can be provided for application with a particular detector.

The NUMAC microcomputer performs the logarithmic count rate function, compensations, calibration, alarm and trip functions, detector voltage setting, digital filtering and required mathematical operations.

The standard instrument includes two remote (trip) outputs per chassis: one for Downscale/Hi radiation trip indications and one for self-test detected faults. Each trip output is a 12-volt logic level output. An additional output to the trouble light on the ARM cabinet is also provided. It will light on any Downscale/Hi radiation level trip and on any self-test detected fault. Up to three additional trip outputs per chassis can be supplied to satisfy customer requirements. The microcomputer digitally handles the trip function, eliminating trip circuit drift. Trip accuracy depends only on measurement accuracy.

The instrument contains a redundant (auctioned) polarization power supply. A single voltage, adjustable from the operator panel, is under microcomputer control. The self-test trip output annunciates power supply failures (voltage output errors and current overload). Each detector power supply includes a hardwired overvoltage protection circuit. The NUMAC design features an automatic internal calibration check facility.

Detector Power Supplies

Each of the eight ARM channels contains a voltage source to power its associated detector.

Voltage Range: 100–1250 VDC redundant (auctioned)

Maximum Current: 1 mA maximum

Voltage Ripple: 1% RMS

Self-Test Alarm: When ±10% of setpoint is exceeded or current overload is detected

Trip Outputs

Each of the eight ARM channels has two independently adjustable trip setpoints (Downscale and Hi). Trip hysteresis is adjustable from the front panel over the range of 0 to 25%.

Self-Test Trip

The self-test trip monitors gross failures of the detector power supplies, the microcomputer "watchdog" timer, and the card-out-of-file and instrument-out-of-operate mode.

Self-test failures on the module level are also reported.

Other Outputs

Each channel (up to eight per chassis) furnishes outputs to the auxiliary units which include radiation level indication, visual and sonic alarms and sonic alarm reset. As an option, each channel can provide 4–20 mA, ±2% linear full-scale remote outputs. External supplies to the ARM electronics cabinet provide the power for the 4–20 mA outputs.

Performance Specifications

Input Signal

From either current-type (ion chamber) or pulse-type (G-M tube) detectors.

Channel Accuracy

For ion chamber detectors

Design center conditions are tabulated below:

Input Current Amperes	% Linear Full Scale	% Of Point
1E-13 to 1E-11	±1.2	±25
1E-11	±0.76	±15
All other decades	±0.52	±10

Discriminator and Window Ranges

For G-M detectors

Discriminator: Adjustable from 25 mV–2.5 V

Window: Adjustable from 10 mV–2.5 V

Settings are digital and do not drift.

Response Times

Digitally filtered as a function of either count rate or current level.

NUMAC™

Log Count Rate Meter (304A3701)

The Log Count Rate Meter (LCRM), member of the GE Nuclear Measurement Analysis and Control (NUMAC) family of microcomputer based instruments, replaces up to three existing BWR Log Count Rate Meters (145C3284AA).

Each of the three NUMAC Log Count Rate Meter channels receives input voltage pulses from the existing associated pulse preamplifier. These pulses are routed to the NUMAC input signal-conditioning module three-channel discriminator which interfaces with the instrument's functional microcomputer. The logarithmic count rate function, compensations, calibration, alarm and trip functions, detector voltage settings and required mathematical operations are made by this computer. Discriminator adjustments and calibrations are made through the front panel interface.

The range of the instrument is 10 to 1E6 counts/minute.

Each of the three LCRM channels contains three Form-C contact trip outputs. Since the trip function is handled digitally by the NUMAC microcomputer, there is no trip circuit drift. Trip accuracy depends only on measurement accuracy.

The instrument contains polarization power supplies for three detectors. These supplies are adjustable from the operator panel and are under microcomputer control. Power supply gross failures are annunciated via the Inop trips. Each detector power supply is provided with a hard-wired, over-voltage protection circuit.

An internal calibration check facility is provided, and automatic calibration of the instrument is a feature of the NUMAC design.

GENERAL ELECTRIC

Each of the three LCRM channels contains a voltage source to power its associated detector.

Voltage Range:	450 VDC – 1200 VDC
Maximum Current:	1 mA maximum
Voltage Ripple:	1% RMS
Alarm:	When +/− 10% of setpoint is exceeded

Each of the three Log Count Rate Meters provide three independently adjustable trip circuits (Hi-Hi, Hi and Downscale). Trip hysteresis is adjustable from the front panel over the range of 0% to 25%.

Positive, double differentiated pulses, 25 mV to 4 V, ≤ 1 microsecond.

Response time constants vary with count rate as follows:

Count Rate (CPM)	Time Constant (sec)
10	75
100	75
1E3	75
1E4	12
1E5	2.5
1E6	.2

The Inop trip monitors gross failure of the detector power supplies, microcomputer "watchdog" timer, card-out-of-file and instrument-out-of-operate mode.

Self-test detected failures are also reported via the Inop trips. The Inop trip outputs appear in parallel with each of the three Hi-Hi trip outputs.

Each of the three Log Count Rate Meter channels has provisions for driving a recorder (1.0 V full scale), process computer input (160 mV full scale), and remote meter (1.0 mA full scale). Noise and ripple are less than 0.5% of full scale.

Discriminator:	Adjustable from 25 mV – 2.5 V
Window:	Adjustable from 10 mV – 2.5 V

Settings are digital and do not drift.

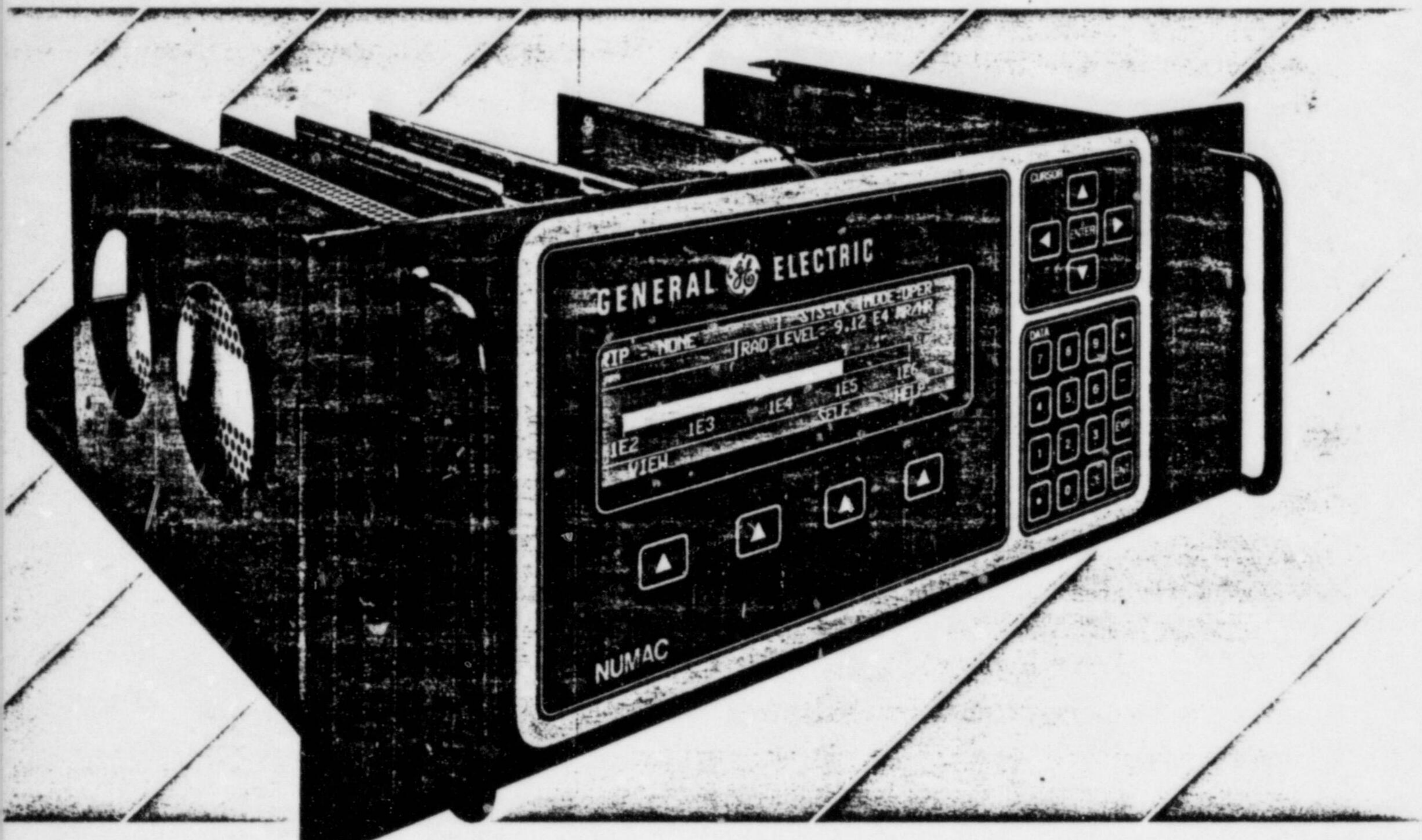
± 2% of linear full scale.

± 3% of linear full scale.

± 0.5% of linear full scale for a thirty-day interval after initial warmup of at least two hours.

NUMAC™

Log Radiation Monitor (304A3700)



The Main Steam Line or Log Radiation Monitor (LRM), member of GE's Nuclear Measurement Analysis and Control (NUMAC) family of microcomputer based instruments, is a direct replacement for existing GE LRMs (194X629, 238X660 or 368X428AA).

Economic analysis of conversion to the NUMAC LRM indicates that savings in 1) instrument technician calibration time, 2) scram avoidance and 3) spare parts inventory reduction can achieve a payback period of three years.

The input amplifier is the standard NUMAC femtoammeter input module. The amplifier has an input span from 10 femtoamperes to 0.3 milli-amperes. The amplifier output is interfaced to the NUMAC microcomputer by an analog-to-digital converter.

Temperature compensation, calibration and mathematical operations are accomplished by the instrument microcomputer.

The standard instrument range is six decades in the range from 3.33XE-13 to 3.33XE-7 ampere for the Main Steam Line Monitor. Alternate instrument ranges are accomplished by firmware change and are available as options.

Trip outputs are configured to be exact replacements for the existing instruments. The standard instrument includes four trips. Each trip output includes a 12-volt logic level output. Additional trip output modules (up to six total) can be provided to satisfy customer requirements and will be quoted on request. The first trip output used is reserved for the chassis Inop function. Since the trip function is handled digitally by the microcomputer, there is no trip circuit drift. Trip accuracy depends only on measurement accuracy.

The instrument contains a polarization power supply for one detector. This supply is adjustable from the operator panel and is under microcomputer control. Power supply gross failures are annunciated via the Inop trips. Each detector power supply is provided with a hard-wired, over-voltage protection circuit.

An internal calibration check facility is provided, and automatic calibration of the instrument is a feature of the NUMAC design.

Each LRM channel contains a voltage source to power its associated detector (5467870G016).

Voltage Range:	100 VDC – 350 VDC
Maximum Current:	3 mA
Maximum Voltage Ripple:	1% RMS
Over-Range Limit:	450 VDC Alarm: When +/− 10% of setpoint is exceeded

Each LRM provides three independent adjustable trip circuits (Hi-Hi, Hi and Downscale). Trip hysteresis is adjustable from the front panel over the range of 0% to 25%.

The Inop trip monitors gross failure of the detector power supplies, microcomputer "watchdog" timer, card-out-of-file and instrument-out-of-operate mode. Self-test detected failures are also reported via the Inop trips.

Other Outputs

Each LRM channel has provisions for ng driving a recorder (1.0 V or 10.0 V full scale) and process computer input (160 mV full scale). Noise and ripple are less than 0.5% of full scale.

Input Signal Polarity

Positive

Response Times

Response time constants vary with input current level as follows:

Change in Current Level (Ampere) From		To	Time Constant (sec)
3.33E-13	–	3.33E-12	6
3.33E-12	–	3.33E-11	1
3.33E-11	–	3.33E-10	0.5
3.33E-10	–	3.33E-9	0.5
3.33E-9	–	3.33E-8	0.5
3.33E-8	–	3.33E-7	0.5
3.33E-13	–	3.33E-7	1

Accuracy at Design Center

Input Current (Ampere)	% of Point	%Linear Full Scale
3.33E-13 – 3.33E-12	±25	±1.6
3.33E-12 – 3.33E-10	±15	±1.0
3.33E-10 – 3.33E-7	±10	±0.7

Accuracy at Design Limits

Input Current (Ampere)	% of Point	%Linear Full Scale
3.33E-13 – 3.33E-10	±50	±2.9
3.33E-10 – 3.33E-7	±20	±1.3

NUMAC™

Rod Worth Minimizer (304A3702)

The NUMAC rod worth minimizer (RWM) is a member of the Nuclear Measurement Analysis and Control (NUMAC) family of microprocessor-based instruments currently being offered for a variety of applications in nuclear power plants. The NUMAC RWM replaces the present BWR rod worth minimizer (RWM) implemented on stand-alone GEPAC computers or as part of GEPAC 4010 or 4020 computers or Honeywell 4400/4500 process computers on most BWRs. It also replaces the hardwired group-notch Rod Sequence Control System (RSCS) and associated RWMs used on most of the BWR/4s and /5s in the U.S.

The NUMAC RWM function monitors and enforces adherence to established low power level control rod insert and withdrawal sequences. This function prevents the operator from establishing control rod patterns that are not consistent with the predescribed sequence by initiating appropriate rod select error warning, rod withdrawal block and rod insert block signals. The instrument enforces control rod sequencing procedures designed to limit (and thereby minimize) individual control rod worths to acceptable levels as determined by the rod drop accident design basis. Its function does not interfere with normal reactor operation, and in the event of a failure, does not itself cause rod patterns to be established which would violate the rod drop accident design basis.

The generic NUMAC RWM functional diagram is shown in the accompanying illustration. It uses the Banked Position Withdrawal Sequence (BPWS) algorithm which is based on stored Boolean logic in read-only firmware and is not cycle dependent. The BPWS logic is described in document NEDO-21231. An option is available which allows the sequence to be loaded, then display the sequence to an operator as requested.

The NUMAC RWM has the following system characteristics:

- Four sequences with operator-selectable names are available.
- Selection of normal/bypass mode. For a dual-channel system, only one channel is bypassable.
- Substitute rod position capability.
- Selection of shutdown margin test mode.
- Rod coordinate indication of insert/withdrawal errors.
- Status indication of select, withdraw and insert blocks.
- Keylock bypass of up to eight inoperable control rods (controlled by Tech Specs.).
- NUMAC RWM failure indication.
- Display of control rod identification and position.
- Identification of rod group from which rod selection is made.
- Error messages.

The NUMAC RWM is designed to interface with the data acquisition equipment at existing plants. The following input interface modules are available:

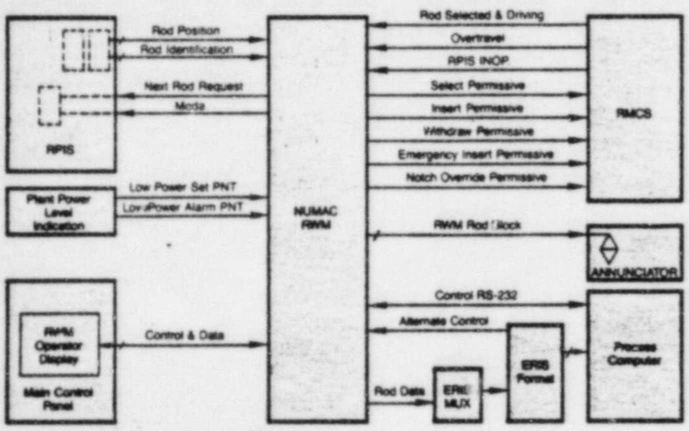
- Module for inputs from an existing relay multiplexer used to provide position data to the instrument.
- Module for inputs from the serial encoded words used in RSCS.
- Module with a fiber optic serial input from a replacement RPIS.

Each NUMAC RWM is constructed of standard plug-in modules in a chassis designed for rack mounting. All components are mounted in functional modules. Maintenance is accomplished by module replacement.

The universal chassis is 16 inches deep × 19 inches wide × 7 inches high. Chassis slides provide access to the top-loaded printed circuit modules for maintenance with the instrument in the extended position. Instrument interfacing connections are mounted on the top rear section of the chassis deck. A complement of mating connections and a retract mechanism for mounting the interfacing cables to the slide-mounted chassis are provided in a package of accessory parts, if required.

The instrument is microcomputer based. This permits the capability for auto-ranging, automatic calibration and on-line computation of operational and maintenance information. All logical functions, arithmetical functions or control algorithms are implemented by the microcomputer module.

The instrument design includes a self-test capability which identifies and annunciates failed modules, if any, to a replaceable module level permitting repair within a one-hour period. The primary function of the self-test feature is to maximize instrument channel availability. With the self-test feature, instrument availability is designed to be greater than 0.9997 for an estimated mean time to repair of 30 minutes. Without self-test, assuming a weekly surveillance interval, instrument availability is designed to be greater than 0.989. The self-test feature verifies instrument repair and minimizes maintenance time.



Single-Channel NUMAC Rod Worth Minimizer

Most RWM modules are common to other NUMAC instruments. This leads to ready availability of spare modules and the potential for substantial spares inventory reductions.

The standard NUMAC RWM instrument has a menu-driven electroluminescent display. The display provides operational and maintenance interface data display information along with operational switches necessary to operate and maintain the instrument.

The instrument uses standard AC power, 120 Vac, 50/60Hz. Other available options are 220 Vac, 125Vdc, 24Vdc and 20Vdc.

It is designed for normal operation in the range of 5°C and 10% to 90% relative humidity noncondensing. It meets GE requirements for EMI-RFI per GE's EMI susceptibility test guide. Dual instrument power supplies are provided to enhance availability.

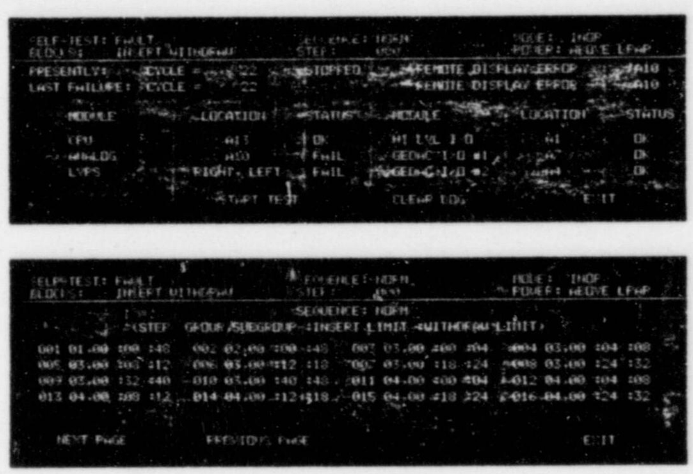
The dual-channel NUMAC RWM is required for all plants with an RSCS and is an option for single-system RWM plants. The dual design supplies direct rod position data and rod motion information to either NUMAC RWM channel. Each channel is powered by separate divisions and electrically isolated from the position information source to minimize the possibility of a failure in one channel causing an interruption of the RPIS signal to the other channel.

System Options

For safety-related applications, the NUMAC RWM is available qualified to the requirements of IEEE-323, 1974 and IEEE-344, 1975. Other options include operation and maintenance training, spare parts and a process computer link wherein one of the RWM's two serial data links can be used to provide data transmission to and from the plant computer. A next rod prompting option will provide rod motion prompting to the operator as prescribed by either utility-loaded startup and shutdown sequences or by the Banked Position Withdrawal Sequence.

Additional options are available such as extended application engineering and site documentation updating, as well as options for power descent and data logging and reporting.

An optional "inoperative" trip signal can be provided which monitors the instrument for a gross failure of the microcomputer, card out of file and the instrument not being in the operational mode. For a dual-channel system, the NUMAC RWM can provide cross-channel checks using fiber optic communication links. Quality checks can be provided on the rod position data to assure reliable position data information. The quality checks include the use of inferred rod positions. The level of quality checking is equivalent to that used in BWR/5s where two channels also share a single source of rod position data.



Typical NUMAC RWM Displays

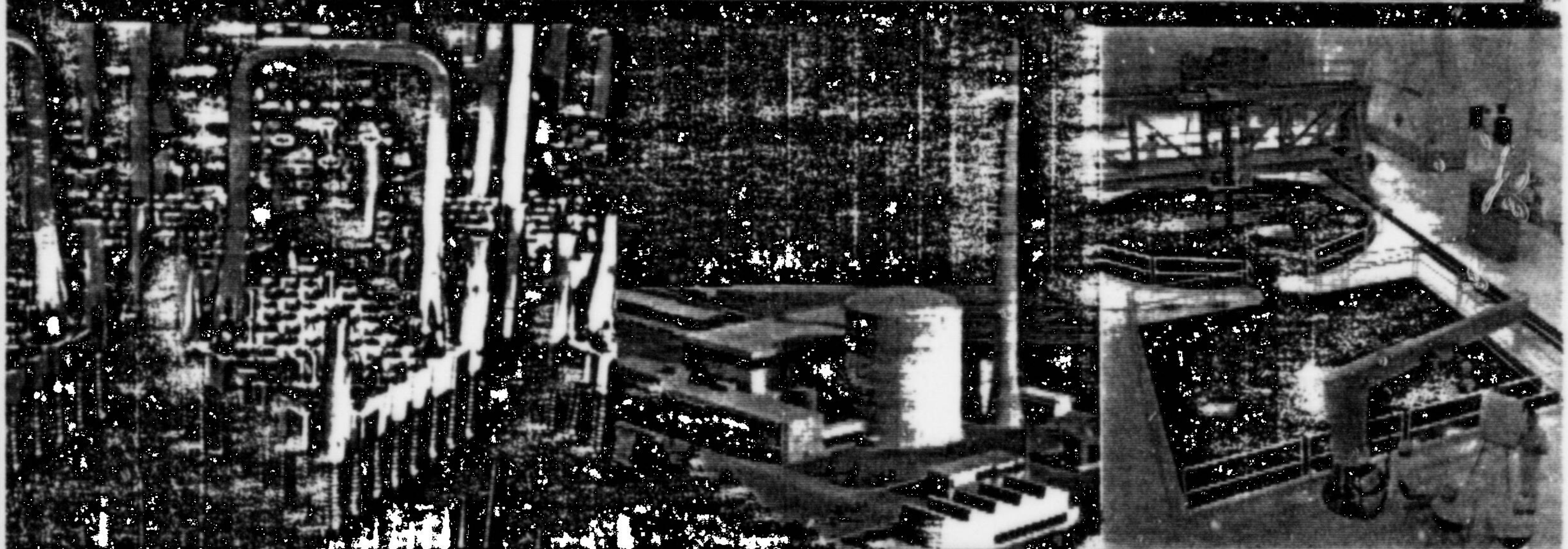
Future Upgrades

The NUMAC RWM is designed to be compatible with future upgrade products such as improved rod position/status display and hardware and analysis to permit relaxation of Banked Position Withdrawal requirements.

Presented at
ENC '86 Conference
June 2-6, 1986
Geneva, Switzerland

MICROPROCESSOR BASED NUCLEAR POWER PLANT INSTRUMENTATION

J. E. Dennis
Nuclear Energy Operations
San Jose, California



GENERAL ELECTRIC

Microprocessor Based Nuclear Power Plant Instrumentation

INTRODUCTION

In the past decade, microprocessors and associated integrated circuits have revolutionized the design of electronic instrumentation. Improved performance, capability and reliability can be achieved while at the same time reducing size and weight, power consumption, and maintenance effort. In the design of instrumentation for both old and new nuclear power plants the incorporation of microprocessors appears attractive.

The replacement of existing instrumentation is usually motivated by one or more of the following: improved performance, better user interface, wearout of installed instruments, and additional requirements. Whatever the reasons, present day digital technology can provide new capabilities and benefits. Since modern electronics is a rapidly evolving field, today's technology is several generations beyond that which was available at the time most original plant equipment was designed.

This paper presents a set of design objectives for microprocessor based instruments and discusses General Electric's response, the NUMAC (NUclear Measurement Analysis and Control) family of instruments.

DESIGN OBJECTIVES

The design of instrumentation for retrofit applications affords many opportunities for providing added value and performance. In order to best obtain these benefits, a set of design objectives should first be formulated. These objectives might vary from plant to plant, but would certainly include many of the following:

- Suitability for both direct mechanical and functional replacement of old instruments and new applications.
- Modularity of design for ease of configuration and adaptability to changes.
- Commonality of component modules and instrument operation to simplify training, maintenance, spares provisioning.
- Ability to communicate with other instruments and systems.
- High performance, reliability and availability.
- Ease of calibration, test and repair.
- Qualifiable designs for safety related applications.

At General Electric's Nuclear Energy Business Operations, the pursuit of the above objectives has lead to the creation of NUMAC, a new, environmentally qualified family of products for the measurement and analysis of radiation and process signals, the interchange of data with host computers, and the initiation of control actions based on these measurements and data exchanges.

HARDWARE DESIGN

Mechanical Configuration

All NUMAC instruments utilize a common chassis designed for slide-mounting in a standard 19″ instrument rack. They occupy 7″ of rack space and vary between 16″ to 20″ in depth. A typical NUMAC instrument, the Log Count Rate Meter, is shown in Figure 1.

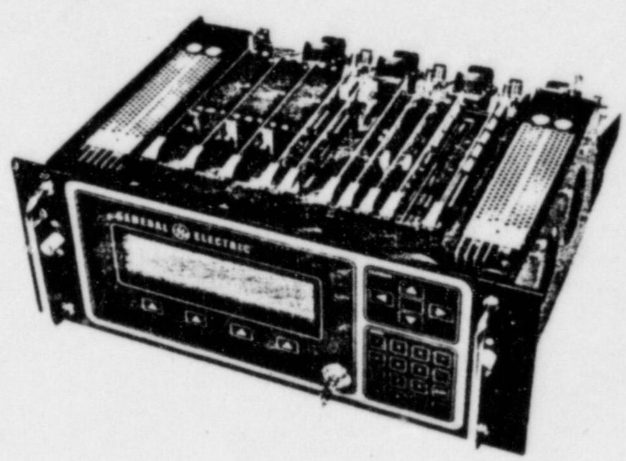


Figure 1. NUMAC Instrument (Log Count Rate Meter)

The standard chassis contains a card file and mother board capable of accommodating a complement of up to 15 printed circuit modules; a front panel with an electroluminescent display, functional softkeys, cursor keys, data entry keys, and a key-lock switch; and redundant

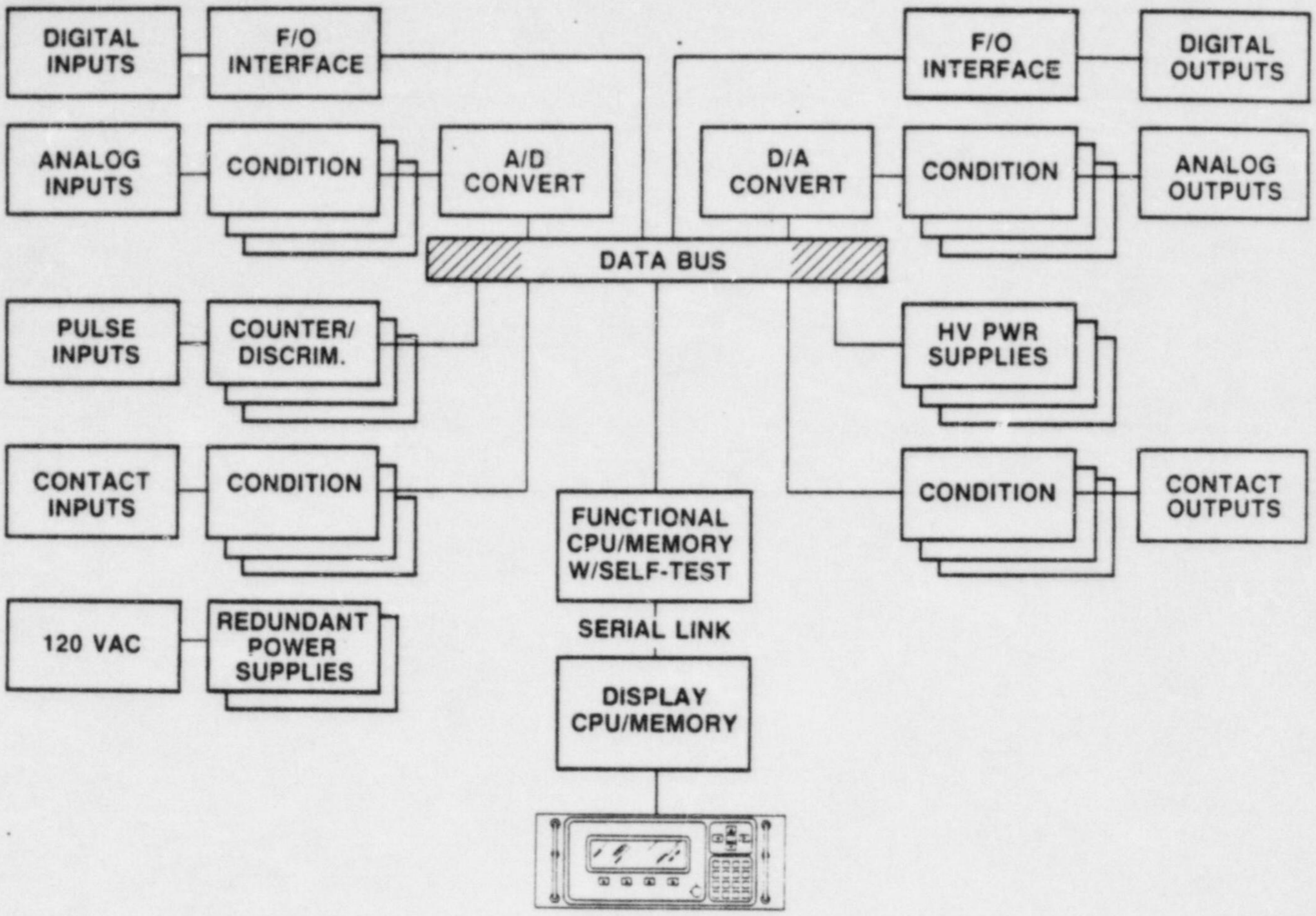


Figure 2. NUMAC Block Diagram

power supplies. Instrument application will determine the number and mix of modules provided.

Most power and signal input/output connections are made at a connector bracket mounted at the rear of the chassis. Its design varies with application. For added reliability, very sensitive signals bypass this bracket and connect directly to appropriate modules.

Electrical Configuration

A NUMAC instrument's electrical circuits are divided into three sections: the functional microprocessor and its related modules which perform the instrument's prime functions; the front panel and its display controller to handle user interface; and the power supplies which provide the operating voltages for both the above. A basic block diagram illustrating this arrangement is provided in Figure 2.

Functional Circuits

Functional circuits are those which perform the instrument's primary measurement, analysis and control functions as well as automatic self-test and calibration. They are based on the use of an 80C86 microprocessor (16 bit) and an associated data bus.

Included in each instrument is a functional computer module containing the microprocessor, random access memory (RAM), a program installed in read only memory (ROM), and electrically alterable read only memory (EAROM). It directs instrument operation and performs such functions as temperature and linearity correction of measured data, parameter and control calculations, trip comparisons and event sequencing. Additionally, it controls self-test and evaluates results, directs autocalibration, and communicates with the front panel controller via a serial data interface.

Process input signals are received by appropriate signal conditioning modules. Analog input modules isolate sensor and transmitter signals and convert them to digital form, contact input modules sense switch and relay signals, and pulse input modules perform pulse height discrimination and counting.

Process output modules convert digital signals to forms usable by devices being driven by the NUMAC instrument. They include analog output modules which perform digital to analog conversion and signal buffering, and contact output modules which provide relay or solid state contact closures.

High voltage power supply modules provide the voltages needed to polarize any associated radiation detectors. These supplies have their own regulating circuits whose setpoints are controlled by the computer and whose output levels are monitored by the self-test system.

External data communications (e.g., high speed serial data messages) are handled by interface modules customized for signal type, speed, level and protocol compatibility. High speed fiber optic as well as RS-232 interfaces can be accommodated.

User Interface

User interface is through a front panel containing keyboard and display units, and a keylock switch to set the instrument in either the Operate or Inoperative (maintenance) mode. The panel is the same for all NUMAC instruments and is customized for different applications by the installation of appropriate software.

An electroluminescent screen with a resolution of 512 x 128 addressable pixels provides both alphanumeric and graphic displays. It is driven by a display control module whose memory contains the logic and formats required for displaying data provided by the functional computer. The module is mounted in the chassis cardfile but does not interface with the functional microprocessor. It contains an NSC800 microprocessor, RAM, ROM, and EAROM.

Four "softkeys" below the display are used to control the instrument's operation. The specific functions associated with these keys vary under software control. In addition to the softkeys, four keys for controlling the display's cursor and a keypad for numeric data entry are provided.

In the "Operate" mode, the computer sends all necessary data to the control module on a periodic basis. Communication in the opposite direction is minimized so that controller failures have little or no effect on operation. With the switch in the "Inoperative" position the instrument can be calibrated, a more thorough self-test can be performed, and, under password control, the more critical operating parameters can be changed.

Displays may contain status highlights, data tabulations, graphs and operating instructions. Typical front panel displays are shown in Figure 3.

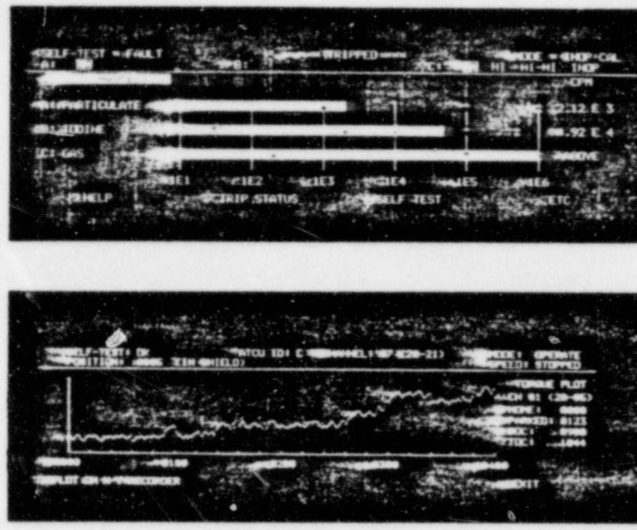


Figure 3. Typical NUMAC Displays

Instrument Power Supplies

Room is provided in each chassis for two redundant low voltage power supply modules. The specific modules used will depend on the power source specified (120VAC, 220VAC, 125VDC, etc.)

Environmental Qualification

NUMAC instruments find use in both safety-related and non-safety-related applications, but are all built in the same manner using the same components. Extensive qualification testing has been performed to date. Additionally, all safety-related modules undergo testing to their individual performance specifications. As a result, all safety-related instruments delivered have been qualified to their specified control room environments in accordance with IEEE Standards 323-1974 and 344-1975. The environmental and seismic profiles used in these tests envelop all known power reactor applications

SOFTWARE DESIGN

A NUMAC instrument includes modularly designed software (firmware) for both its functional computer and its display controller. Such design facilitates the preparation and validation of codes for each specific instrument as well as the addition or deletion of functions for unique applications.

An event-driven, real-time, multitasking operating system kernal is common to all functional computer modules in the NUMAC product line. Its use allows NUMAC software to be partitioned into various functional tasks which execute in pseudo-concurrent fashion. The operating system controls the resources of the functional computer including the microprocessor and certain memory pools. Typical tasks running under its control include those for process I/O and computations, display controller interface, communication with external computers, self-calibration, and self-test. This software is written in PL/M-86 and ASM-86 languages, developed on a VAX 11-780, and tested using Intel in-circuit emulators.

The software design used for the Display Controller utilizes an executive loop which calls on lower level procedures to perform its major functions. These procedures include an executive, a functional computer interface, a display handler, and an instrument status keeper. This software is written in both PASCAL and Z80 assembly language, and developed on an HP 64000 Logic Development System.

MAINTENANCE

Maintenance of a NUMAC Instrument is simplified through use of several built-in features. They include:

- Self-Testing — A built-in self-test system functions continuously while the instrument is operating on line. This testing is done via the functional computer by performing RAM, ROM and internal microprocessor checks, by interrogation of special data resisters on the various circuit modules, and through voltage measurements made by the analog modules. Faults are traced internally to the replaceable module level and announced via trip output circuits. A user can interrogate the instrument via the front panel to determine status.
- Self-Calibration — Each instrument contains appropriate reference standards (e.g., precision resistors and voltage sources) that allow self-calibration. Under direction of the functional processor these standards may be verified against external ones.
- Help messages — Throughout operation and maintenance, the user is guided by explanatory and diagnostic messages on the front panel display. If further assistance is required, a system of explanatory "Help" messages can be called on at any time.

NUMAC INSTRUMENTS

The power and versatility of microprocessor technology in the design of instrumentation for nuclear power reactor application is demonstrated by the list of instruments that are either currently under active design or have already been delivered to the field.

Instruments Delivered

1. *Logarithmic Radiation Monitor (LRM)* — Log Rad Monitors measure the dc current from gamma sensitive ion chambers, typically for 6 to 7 decades in an overall range of 3E-13 to 3E-4 amperes, and provide polarizing voltages for associated detectors.
2. *Logarithmic Count Rate Meter (LCRM)* — Log Count Rate Meters are one to three channel instruments (configurable from the front panel) that discriminate and count output pulses from radiation detectors over 5 to 7 decades in an overall range of 0.1 to 1E6 cps. They also provide independent polarizing voltages for associated detectors.
3. *Rod Worth Minimizer (RWM)* — Rod Worth Minimizers monitor control rod movements and enforce control rod insertion withdrawal sequences when a reactor is below its low power set point. They obtain sequences from and return status to plant computers, provide rod blocks when a sequence is violated, interface with a remote operator's display, and provide alarms via an annunciator on the control benchboard.

To be Delivered in 1986

1. *Neutron Monitoring System (NMS)* — Two instruments make up this version on an out-of-core neutron monitoring system. A Source Range Monitor discriminates and counts pulses from a proportional counter, and calculates count rate and reactor period. A DC Wide Range Monitor measures current from a compensated ion chamber over a ten decade span. The top three decades are measured linearly while the remaining seven (plus overlap) are handled logarithmically. Power level and margin are calculated.
2. *Process Radiation Monitor (PRM)* — The Process Rad Monitor is similar to a single channel Log Count Rate Meter except for the input pulse rate range, the number of trip output contacts, and contact ratings.
3. *Hydrogen Water Chemistry (HWC) Main Control* — This instrument controls both the injection of hydrogen into the recirculation flow to prevent stress corrosion cracking of pipes due to oxygen enriched water and the addition of oxygen elsewhere to allow burning of excess hydrogen in the recombiner.
4. *TIP Control Unit (TCU)* — This instrument controls the insertion and withdrawal of a flux probing monitor within the reactor core. Position and flux data are sent to an external x-y recorder. Discrete control signals to the TIP drive, indexer and ball valve are generated. Data is also sent to a host computer. Probe movement is controlled from the front panel.
5. *Automated TIP Control Unit (ATCU)* — This instrument is the same as the TCU except that its operation is controlled by an external computer.

Proposed Instruments

There are currently several other applications of NUMAC technology under investigation. They include:

1. *MSV Wide Range Monitor* — This instrument combines the functions of a source range neutron monitor (pulse counter) with those of an intermediate range monitor based on mean square voltage technology. The resulting 11-decade monitor covers a neutron range from startup to ten percent power with a single incore or out-of-core detector. Options include continuous ranging rather than manual and the addition of a linear power range.

2. *Neutron Monitoring System* — This incore neutron monitoring system consists of a MSV wide range monitor and an appropriate number of 24-channel power range monitors for each reactor safety division. The power range monitors measure currents from local power range monitoring neutron detectors and provide the Average Power Range Monitoring function.

3. *Area Radiation Monitor (ARM)* — The Area Rad Monitor is an eight-channel instrument that can be configured to accept a combination of dc (picoampere), pulse and mean square analog inputs from various radiation detectors, to furnish polarizing voltages for these detectors, and to provide trip, alarm and recorder/computer outputs.

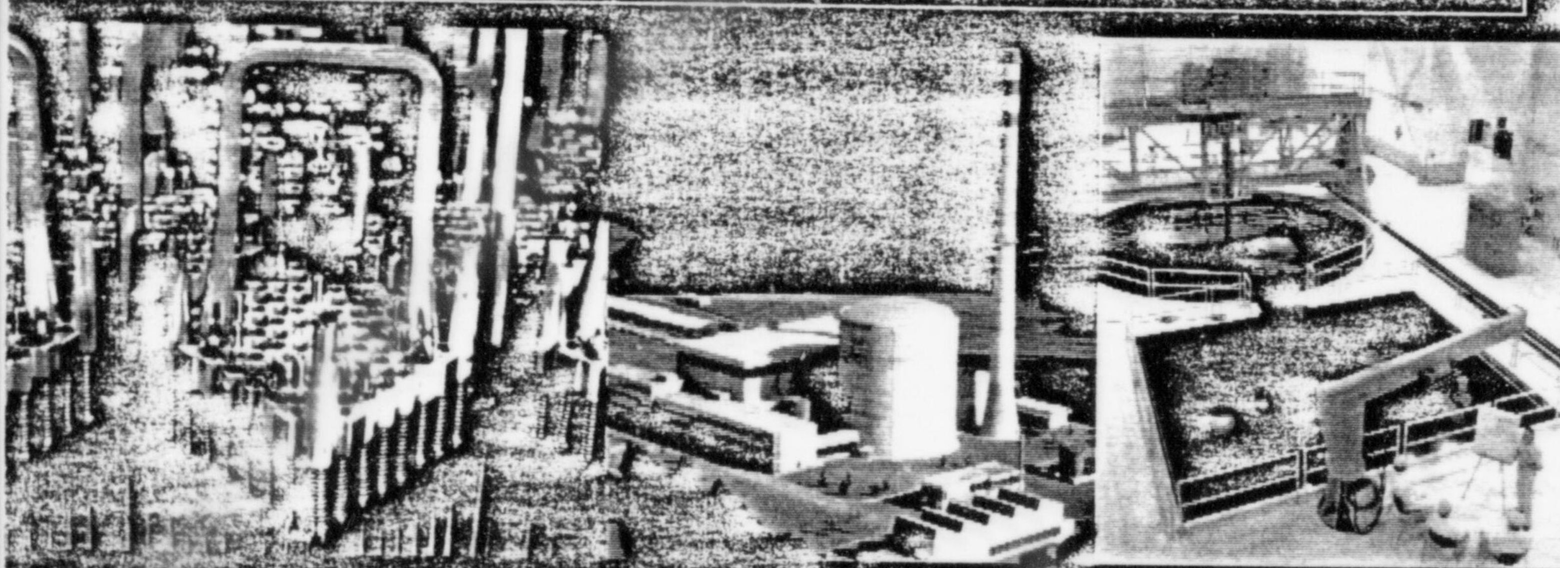
4. *Compensated Water Level Indicator* — This indicator is a safety-related instrument that determines reactor vessel water level by utilizing both vessel pressure and reference leg fluid temperatures to compensate the differential pressure between two reactor water level instrument taps. Flashpoint indication and remote meter and recorder outputs are provided.

5. *Temperature Monitor* — This instrument, used in safety-related applications, monitors the outputs of thermocouples and RTDs, detects open thermocouples, performs compensations and other calculations (e. g., taking temperature differences), provides trip outputs when user-set limits are exceeded and transmits results to external meters, recorders and computers.

Presented at
IEEE Nuclear Power
Systems Symposium
October 23-25, 1985
San Francisco, CA

A MICROPROCESSOR-BASED ROD WORTH MINIMIZER

N. R. Cash
U. E. Dennis
Nuclear Energy Operations
San Jose, California



GENERAL ELECTRIC

A Microprocessor-Based Rod Worth Minimizer

Introduction

In order to prevent undesirable core flux patterns from developing during rod motion, General Electric has always incorporated a Rod Worth Minimizer (RWM) function in its boiling water reactors (BWRs), either in the form of a freestanding computer or as part of a larger process computer system. Maintenance of these RWMs is being increasingly hindered by the reluctance of computer vendors to support old equipment designs. Hence, utilities have begun to replace their near-obsolete computers with new ones, thereby forcing a reimplementation of the RWM function.

Based on reliability, efficiency and economic considerations, General Electric has designed a new Rod Worth Minimizer as part of the company's line of microcomputer-based Nuclear Measurement Analysis and Control (NUMAC™) instrumentation. This RWM can work with new, concurrently installed process computers while using existing rod position indicating and control systems. In addition to providing a direct replacement for existing RWM functions, the new instrument has incorporated capabilities not previously found.

This paper will discuss the functions of a Rod Worth Minimizer, describe the configuration of a new RWM system as well as the NUMAC instrument hardware and software designs, and present the results of the instrument's first application in an operating plant.

System Function

When moving control rods in a BWR during low-power operation it is important to avoid control rod configurations which would give a single control rod a high reactivity worth (ability to affect the power distribution within the reactor core). This rod worth minimization is intended to mitigate the effect of a postulated control rod drop accident. Rod Worth Minimizer instruments are designed to enforce predetermined sequences of control rod motion in order to minimize the worth of any one control rod in the reactor core at all times.

Control rods are withdrawn from the reactor core in a predetermined sequence of steps. For each step, a group of rods and their insert and withdraw limits are identified. The reactor operator selects and drives these rods to their withdraw limits. When all rods in a current step are withdrawn to their limits, the operator proceeds with the next step. This process continues until the Low Power Set Point (LPSP) (about 20% of rated power) is reached. Above the LPSP, no single rod's worth is great enough to cause fuel damage if it were to drop out of the reactor core.

If the reactor operator does not properly follow the rod motion sequence, the Rod Worth Minimizer will remove the rod motion permissive signals which it sends to the Reactor Manual Control System (RMCS) thereby blocking the operator from any further motion of the selected rod. For example, if the operator selects a control rod which is not a member of the group of rods to be moved in the current step, both insert and withdraw permissives are removed, and the operator is blocked from moving the selected rod. Once a rod reaches its step limit, the permissive for that direction of rod motion is removed.

Though rod motion sequences define the movements of all rods from their fully-inserted to their fully-withdrawn positions, the operator need not follow them exactly once power exceeds the LPSP. In that case, the RWM serves only in an advisory capacity and allows for "out of sequence" rod motions. Above the LPSP, the reactor operator may change control rod positions, adjust power level, or modify the power distribution in the core without incurring rod blocks.

When the reactor operator brings the plant down in power, rod worth minimization again becomes important. Prior to reaching the LPSP, the operator must configure the control rod pattern to match some step of the prescribed rod insertion sequence. Once the LPSP has been reached, the Rod Worth Minimizer again enforces the rod motion sequence and any attempt to move control rods out of sequence will result in the removal of rod motion permissives.

System Configuration

The configuration of a given RWM system will depend on both existing and replacement hardware, on features and options desired and on other plant-specific matters. Described below is just one of many possible configurations.

The major elements composing the RWM system are shown in Figure 1. The system includes the RWM instrument chassis (located in a backrow panel), the RWM operator's chassis (located on the reactor operator's main control console) and portions of the plant's process computer data acquisition system, Rod Position and Information System (RPIS) and Reactor Manual Control System (RMCS).

Control rod motion sequences which assure rod worth minimization are normally developed and modified on the process computer and validated by computer checking against various sequence constraints. Validated sequences may then be downloaded from the computer to the RWM instrument chassis. All communication between the RWM and the process computer is performed via a data formatter (i.e., a buffer device) and sent over fiber-optic serial communication lines. Once sequences are loaded into the RWM, the instrument can perform its sequence enforcement function without the aid of the process computer.

Control rod position and other rod status information is obtained from the RPIS and the RMCS by a data acquisition system which may either be incorporated in the RWM or be part of a separate device.

When an operator selects a rod for either insertion or withdrawal, the RWM will perform an evaluation based on the current rod motion sequence, the selected rod's identification and position, and the position of all other rods in the core. If it is determined that the selected rod may be moved, a permissive signal is sent to the RMCS. If not, the RWM will block the rod's motion by removing the permissive and will send an annunciation to the operator's console.

The RWM has four basic modes of operation, OPERATE, BYPASS, TEST, and INOP. When the instrument is in the OPERATE mode, rod position and status information is received and processed continuously. The instrument performs a data integrity check to test

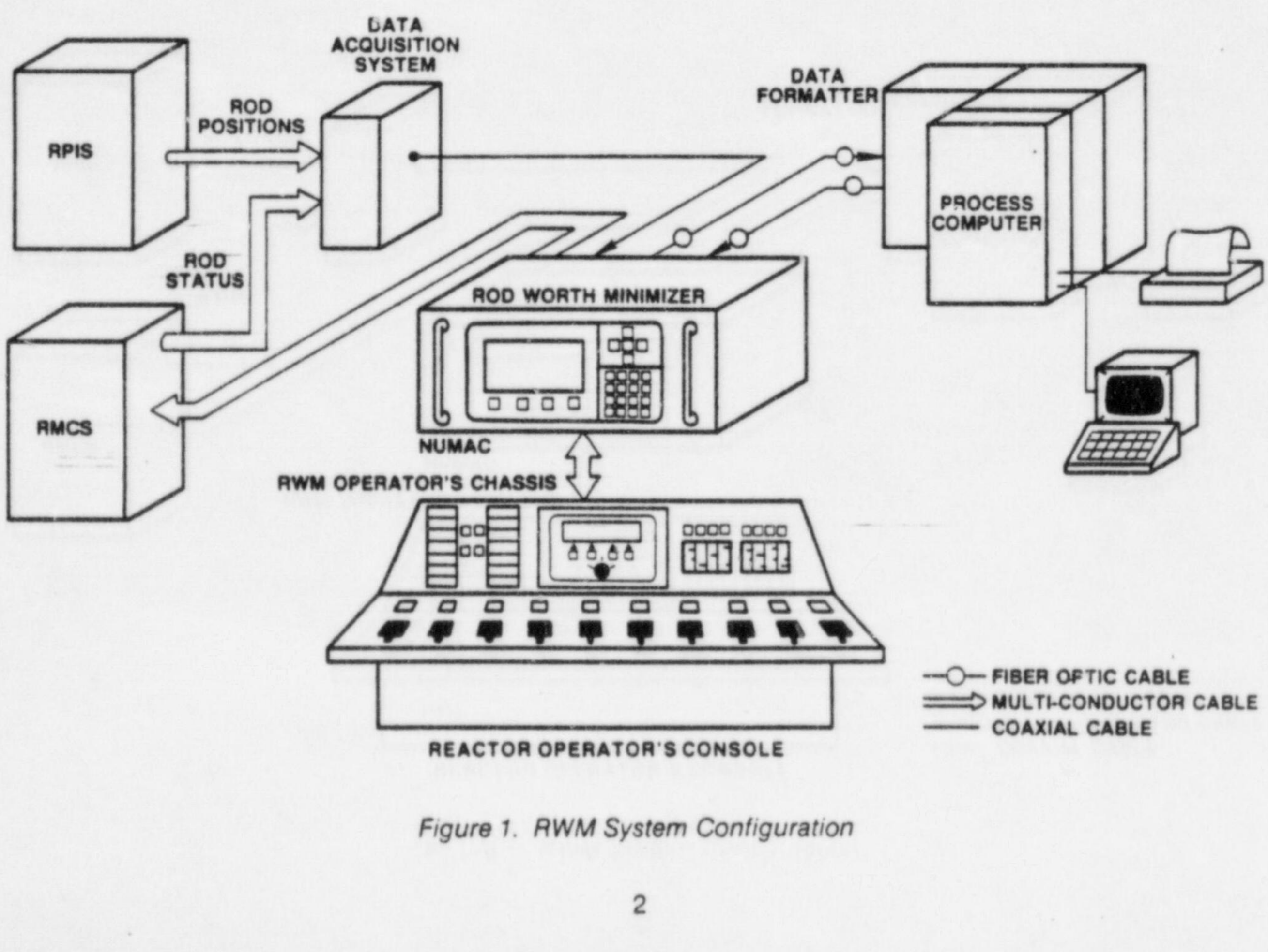


Figure 1. RWM System Configuration

the validity of all input data and other internal data. Sequence enforcement calculations are then performed to determine if rod motion permissives need to be removed and if annunciation is required. The operator's display provides a variety of information such as current RWM mode of operation, sequence identification, step in the sequence, power range, self-test status, status of the permissive outputs, selected rod and its position and more. RWM status information is continuously transmitted to the process computer for logging.

The reactor operator may bypass the RWM with the keylock switch on the operator's chassis. While in the BYPASS mode the RWM will continue to perform all sequence enforcement calculations as in the OPERATE mode, and the operator will still be provided the same RWM status information. However, the output to the RMCS will be locked in the permissive state, and no annunciations to the operator's console will be made.

The keylock switch on the operator's chassis may also be used to place the RWM in the TEST mode. While in the TEST mode the RWM can enforce either a shutdown margin test sequence or a single rod test. The shutdown margin test sequence identifies two control rods which may be withdrawn from the reactor core while all other rods remain fully inserted. During the test, the RWM ensures that only the two identified rods are withdrawn from their fully-inserted positions and that they are not withdrawn past their assigned limits. The single rod test is used to evaluate the motion of a given control rod while all other rods are fully inserted. During this test, the RWM ensures that only the selected rod is moved.

A keylock switch on the instrument's chassis may be used to place the RWM in its INOP mode. When in this mode, the permissive signal to the RMCS is removed, and an annunciation is made at the operator's console. The INOP mode is used to perform sequence downloads from the process computer, place individual rods in bypass (remove from the sequence enforcement logic) and perform more exhaustive self-tests.

The RWM design also provides for the following optional features:

(1) *Bypass of Control Rods* — Under certain conditions, up to eight control rods may be removed from the sequence enforcement logic. This bypass is under keylock control.

(2) *Substitute Rod Position* — If the RPIS indicates an unknown position for a rod, the operator may, (under certain conditions), provide a substituted position for the rod.

(3) *Operating Sequence Alignment* — The RWM identifies the minimum number of rods which must be moved to get the control rod pattern into sequence (for any step).

(4) *Next Rod Prompting* — The RWM identifies the next rod to be moved, its current position and its limits of motion.

(5) *Rod Scram Timing* — The positions of all rods as a function of time are recorded immediately following a scram event.

(6) *Confirmation of Shutdown* — The RWM confirms that, after a shutdown, all rods have been inserted past predetermined limits.

The RWM system configuration described above may, of course, be modified to meet the application requirements of specific power plants. For example, the RWM can be made independent of the process computer by having its interface with a microcomputer instead. The data acquisition function can be incorporated directly into the RWM through the addition of appropriate hardware modules.

Hardware Design

The Rod Worth Minimizer is a member of General Electric's NUMAC (Nuclear Measurement Analysis and Control) line of instrumentation. NUMAC products can replace obsolete or near-obsolete nuclear instruments with an integrated product line offering both better performance and simplicity of installation. NUMAC products benefit from a combination of modern technology and modular design which allows for improved reliability, reduced maintenance and surveillance costs and reduced spares inventory [Ref. 1]. They also utilize a self-test function which continuously checks hardware integrity and alerts operators to any self-test detected failures. This reduction in time needed to detect hardware failures greatly increases instrument availability.

The NUMAC RWM consists of both an operator's chassis and an instrument chassis. The operator's chassis (shown in Figure 2) is a 10" x 8.4" x 10.4" enclosure mounted on the reactor operator's main control console. It contains a 7.3" x 1.7" electroluminescent display having 512 x 128 addressable pixels, four "softkeys" directly below this display whose functions are under software control, a keylock switch to control the operational mode of the instrument and a display control module (printed circuit board) to control the display, scan the soft keys and communicate with its counterpart in the instrument chassis. This module uses an NSC800 microprocessor and onboard random access memory (RAM), electrically alterable read-only memory

Figure 2. RWM Operator's Chassis

Figure 3. RWM Instrument Chassis

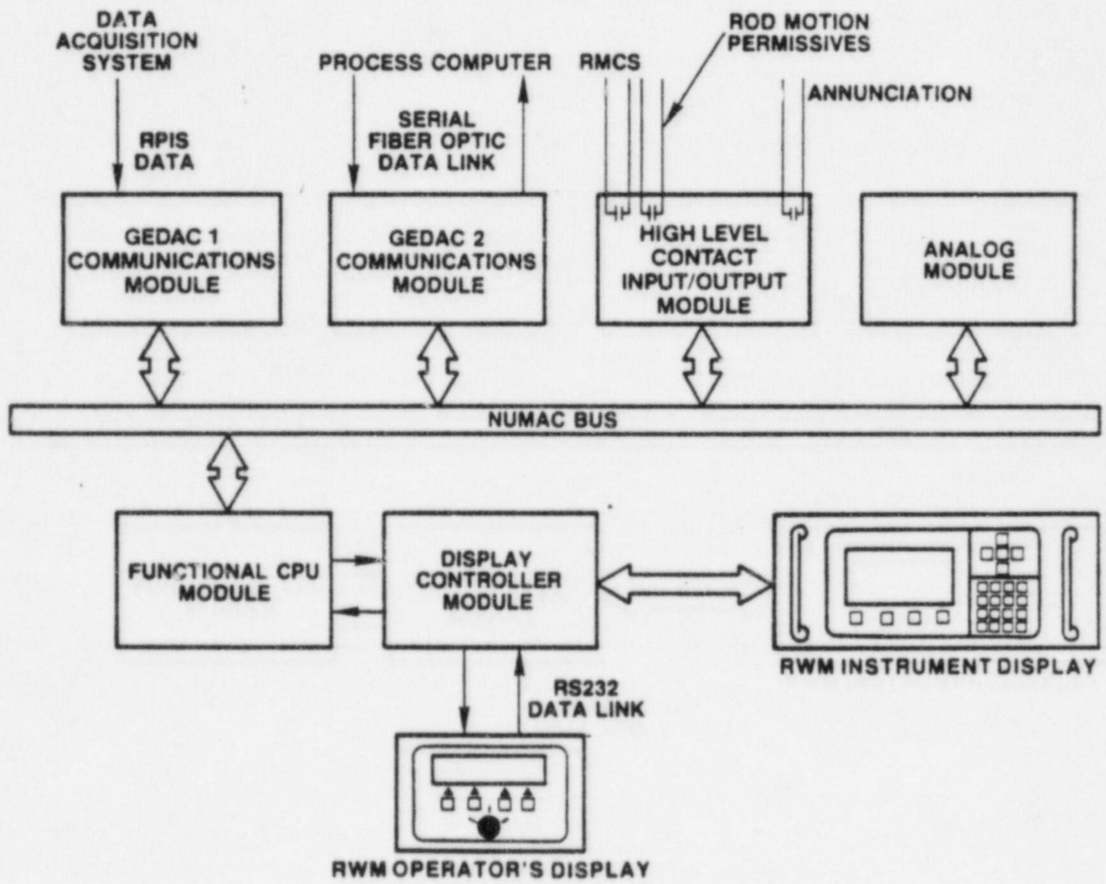


Figure 4. RWM Hardware Block Diagram

(EAROM) and programmable read-only memory (PROM).

The instrument chassis (shown in Figure 3) is a standard 7" x 19" slide-mounted instrument which may be located some distance from the operator's chassis. It contains a card file and mother board capable of accommodating up to 15 printed circuit modules, redundant power supplies for improved availability, a front panel consisting of the same electroluminescent display as the operator's chassis as well as soft keys, cursor keys, data entry keys, a keylock switch and, typically, six electronic modules (see Figure 4). The number and types of these modules may vary in order to accommodate I/O requirements of specific plant system configurations.

The GEDAC™ 1 module is used to receive control rod position and status information from the data acquisition system and is equipped with both electrical and fiber-optic receivers and transmitters. Data reception and transmission is performed under Direct Memory Access (DMA) control to minimize processor overhead. The module also contains on-board, battery-backed RAM which is fully accessible by both the functional processor and the DMA. Positions for all control rods in the core are stored in this card module. In some RWM system configurations, the GEDAC 1 is replaced by modules capable of direct interface with the RPIS and RMCS.

The GEDAC 2 module is used for process computer communication. The rod motion sequences, which are downloaded from the process computer, are stored in this module. Up to four motion sequences and two test sequences may be stored. The module also handles transmission of "alarm" messages to the process computer for logging.

The High Level Contact I/O Module is used to output the rod motion permissives and annunciation signals. This module contains five Form C contacts and is equipped with a hardware failsafe timer which will cause all contacts on the card to go to their normal (power off) condition if the functional CPU fails to periodically reset it.

The analog module reads the operator's keylock switch position. It is also used during self-testing to monitor the redundant power supplies, the instrument bus voltage levels and the status of the output contacts on the I/O module.

The functional CPU module performs the instrument's system and self-test functions. It controls the GEDAC communication modules and the output contacts on the High Level I/O Contact Module. It also communicates with the instrument's display control module. The module contains a CMOS 8086 microprocessor and on-board RAM, EAROM and PROM memory.

The display control module is identical (both in hardware and software) to the one used in the operator's chassis. It is used to control the electroluminescent display, scan the four soft keys, data entry keypad and cursor control keys, communicate with the functional CPU module and communicate with the display control module in the operator's chassis.

Software Design

The RWM design includes software (firmware) for both the functional CPU and display controller(s). Functional CPU module software utilizes a multitasking operating system (NM86) used in the software design of all functional CPU modules in the NUMAC product line. NM86 is an event-driven, real-time, multitasking operating system kernel. Its use allows NUMAC software to be partitioned into various functional tasks which may then execute in pseudo-concurrent fashion. Figure 5 depicts the software design for both the functional CPU and the display controller. The NM86 operating system controls the resources of the functional computer including the CPU and certain memory pools. The tasks running under its control in the RWM design include:

(1) *RWM Coordinator Task* — Performs data integrity checking, the sequence enforcement function, the substitute rod position function, the shutdown margin test function, the rod test function and other essential RWM functions.

(2) *Data Acquisition Receiver Task* — Processes raw control rod data from the data acquisition system and rearms the DMA on the GEDAC 1 module for the next data reception.

(3) *Process Computer I/O Task* — Performs all solicited process computer communications including sequence download and computer requests for information.

(4) *Unsolicited Message Task* — Handles all unsolicited status messages to the process computer.

(5) *Display Message Input Task* — Used for the reception of messages from the display controller.

(6) *Display Message Output Task* — Transmits messages to the display controller.

(7) *Self-Test Task* — Performs the self-test function. This task is the lowest priority task and is always running "in the background" when no other task is ready to run.

The software for the functional CPU module was written in PL/M-86 and ASM-86 languages, developed on a VAX 11-780 and debugged using the Intel 121CE 8086 emulator.

The software design used for the display controller (also shown in Figure 5) utilizes an executive loop which calls on lower-level procedures to perform its major functions. These procedures include:

(1) *Executive* — The display system controller. The executive loops continuously, checking for new messages from the functional CPU, for user inputs from the keypad, for expired timers,

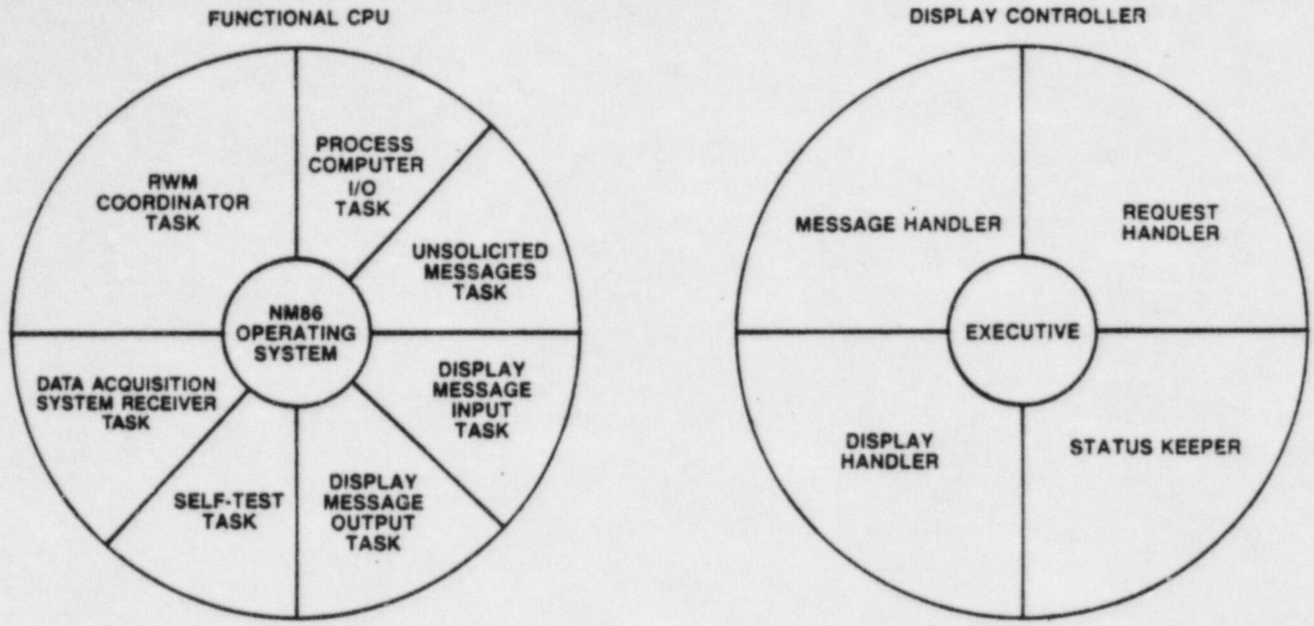


Figure 5. RWM Software Design

and for messages to relay from or to the operator's display chassis.

(2) *Message Handler* — The messages processor for the display system. It includes communication with the functional CPU and the other display controller.

(3) *Request Handler* — Responds to user requests as entered from the data entry keypad or soft keys.

(4) *Status Keeper* — Keeps track of the overall instrument status.

(5) *Display Handler* — Controls the screen seen on the electroluminescent display.

The software developed for the display controller module used in the RWM operator's chassis is the same as that used in the instrument chassis (making the modules completely interchangeable). The software was written in both PASCAL and Z80 assembly language and was developed on an HP 64000 Logic Development System.

The software design for both computers utilizes NUMAC product generic software wherever possible. The modular design facilitates the addition or deletion of functions for unique applications.

Figure 6 shows some typical RWM user displays. All displays are partitioned into an upper, a middle, and a lower portion. The upper section contains general instrument information including instrument mode, self-test status, power level, the present rod sequence and step, and any rod blocks. The middle section displays information appropriate to the operator's current use of the instrument. The lower section labels the functions of the soft keys. These functions (and labels) change during the course of RWM operation, thereby prompting and guiding an operator in the instrument's use.

The following user options are available in the standard version of the RWM:

(1) *Show "Help" Messages*
(2) *Show "Error" Messages*
(3) *Show Self-Test*
(4) *Run Self-Test*
(5) *Show Bypassed Rods*
(6) *Bypass/Unbypass Rod*
(7) *Show Substitute Rods*
(8) *Substitute a Position*
(9) *Explain Blocks*
(10) *Download a Sequence*
(11) *Set Parameters*
(12) *Show Sequence*
(13) *Check Display*
(14) *Check Pushbuttons*
(15) *Select Sequence*
(16) *Turn Display Off*
(17) *Shutdown Margin Test*
(18) *Rod Test*

The RWM has undergone many hours of simulation testing. Its current version of software has been validated both at General Electric's San Jose facility and on site.

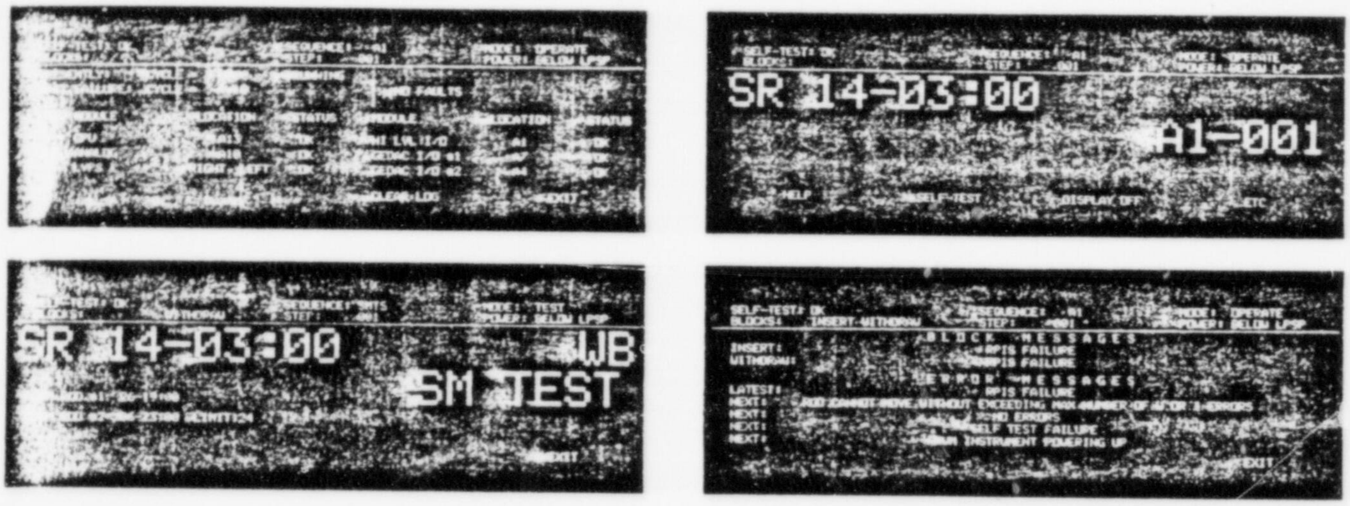


Figure 6. Typical RWM Displays

Field Experience

The RWM hardware was installed in a European operating boiling water reactor in June 1985 (see Figures 7 and 8). Partial software (firmware) installation was made on July 1, 1985. The plant subsequently used the RWM to reduce power during shutdown on August 6, 1985 before scramming the plant in order to obtain scram timing information. Final software installation, allowing process computer interface to be fully operational, was made in September. The plant's rod motion sequences were developed and validated on the process computer and then downloaded to the RWM. The RWM was next used during shutdown margin testing and performed properly.

Finally, on September 11, the plant started up using the RWM which performed flawlessly. It enabled the plant operator to cleanly move through his sequence of steps and to keep control rods within their withdraw limits. Through successful use of the instrument's many displays he quickly developed confidence in the functioning of the NUMAC RWM.

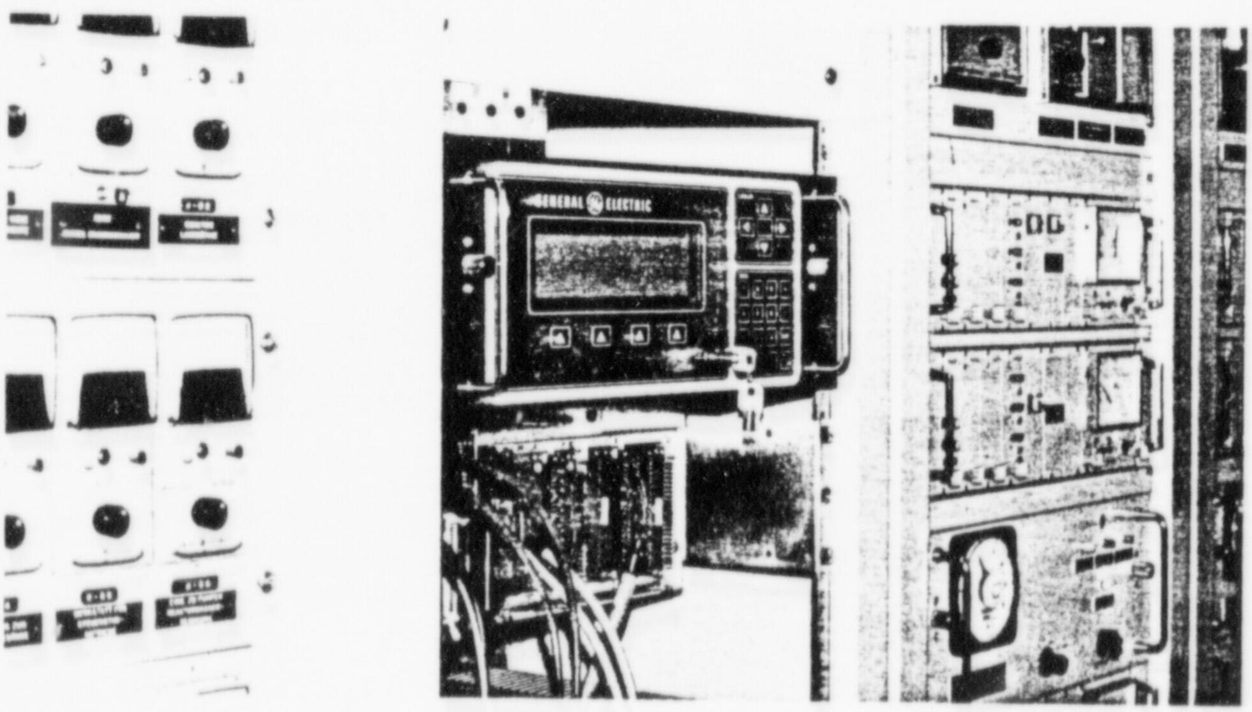


Figure 7. Installation of RWM Instrument

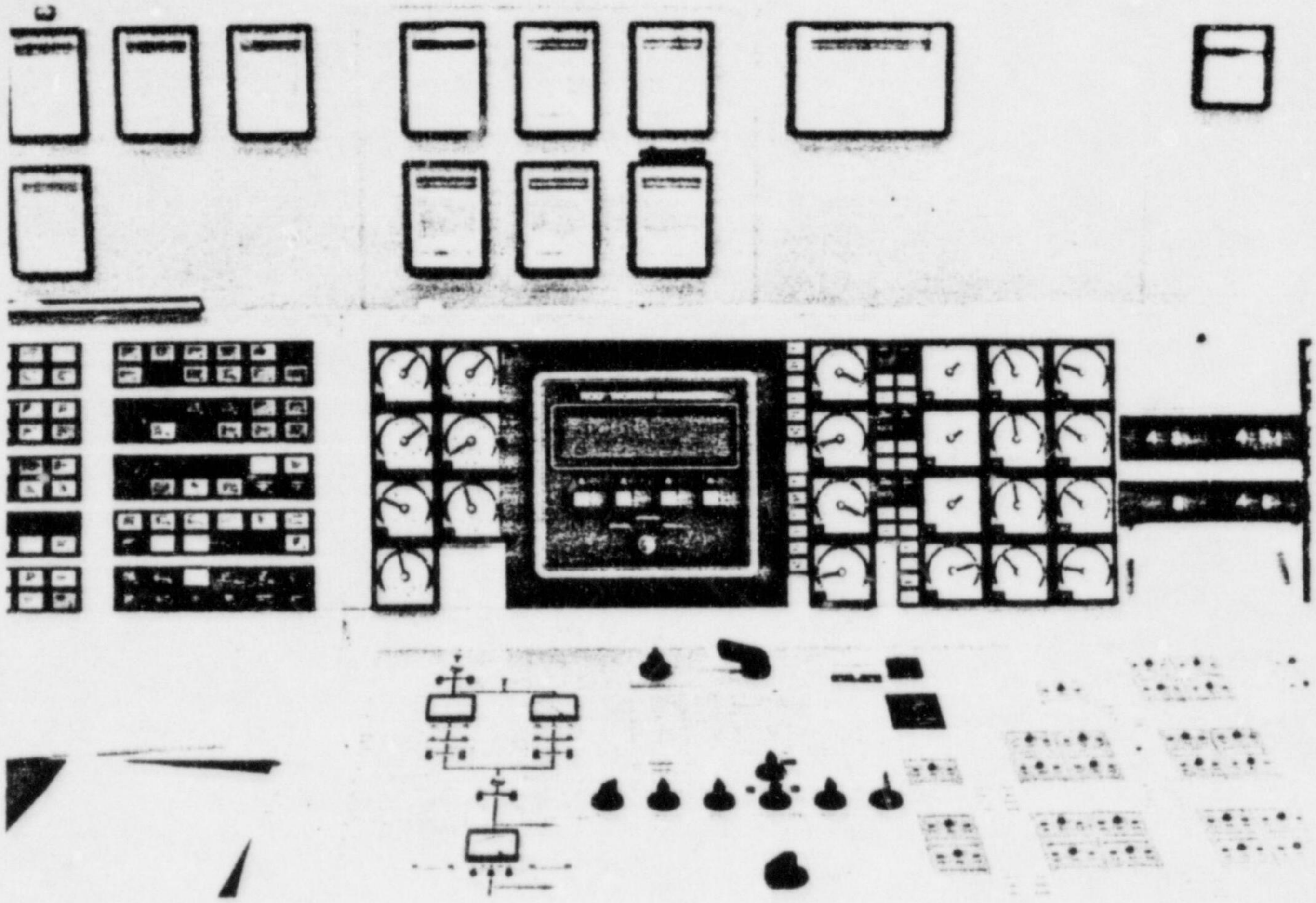


Figure 8. Installation of Operator's Display

Conclusions

The successful design and application of the NUMAC Rod Worth Minimizer has demonstrated that high-quality microprocessor-based hardware can:

(1) Serve as the vehicle for the modular design of an integrated family of nuclear instrumentation and control products.

(2) Be used to design flexible, plant-specific RWM systems.

(3) Be retrofitted to older nuclear power plants to provide enhanced performance.

(4) Provide distributed control functions, including those that depend on the exchange of data with plant process computers.

References

1. M. R. Benson and S. D. Sawyer, "A New Startup Range Neutron Monitor," **IEEE Transactions on Nuclear Science, Vol. NS-31, No. 1, pp. 868-871.** February 1984.

2. U. E. Dennis, "Digital Instrumentation for Retrofit Applications," **EPRI Seminar: Power Plant Digital Control & Fault-Tolerant Microcomputers,** April 9-12, 1985.