ROUTING AND TRANSMITTAL SLIP		ACTION
1 TO (Name, office symbol or location)	INITIALS	CIRCULATE
Z. R. Rosztoczy	DATE	COORDINATION
2 cc: D. Ross	INITIALS	FILE
T. Novak	DATE	INFORMATION
3 L. Beltracchi	INITIALS	NOTE AND RETURN
F. Odar	DATE	PER CONVERSATION
4	INITIALS	SEE ME
	DATE	SIGNATURE

REMARKS

ATWS calculations result in conditions which may be significantly more severe than Ch. 15 transient analyses in terms of BWR stability. These concerns and proposed means to resolve them are discussed in NUREG-0460, App. XVI.

Since Leo is no longer working on the GE evaluation model review, please provide a plan and the schedule for resolution of these concerns.

Incidently, the attached paper may be of help in developing the plan.

Do NOT use this form as a RECORD of approvals, concurrences, disapprovals, clearances, and similar actions

FROM (Name, office symbol or location)	DATE
ACT → Ashok C. Thadani, RSB	6/13/78
	PHONE 27341

OPTIONAL FORM 41
AUGUST 1967
GSA FPMR (41CFR) 100-11.206

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

MAY 30 1978

MEMORANDUM FOR: R. L. Baer, Chief, Reactor Safety Branch, DOR

THRU: F. Coffman, Section B Leader, Reactor Safety Branch, DOR

FROM: M. M. Mendonca, Reactor Safety Branch, DOR

SUBJECT: SUMMARY OF MAY 17, 1978 MEETING ON EXXON NUCLEAR (ENC) BWR THERMAL-HYDRAULIC STABILITY CODE DEVELOPMENT

Find attached a list of attendees and a copy of ENC's presentation. About one year ago ENC started development of a BWR stability code. ENC has a code operational and requested the subject meeting to discuss the code's status with the staff.

The code, COTRAN, is a combination of two existing codes, COBRA IV and XTRAN. COBRA IV supplies the thermal hydraulic portions of COTRAN and is coupled to XTRAN which describes the neutron kinetics of the core. COTRAN iterates between these two codes for a coupled neutronic and thermal hydraulic solution.

COTRAN operates solely in the time domain and, thus, will calculate decay ratios as a standard for stability analysis. Modifications, such as, elimination of cross flow have been made to COTRAN to reduce computer run time. Also, the capability to specify power distribution and key parameter variation with time (forcing functions), and direct neutron heating of the moderator have been added. COTRAN models the core in an R-Z configuration and does not simulate the rest of the primary coolant system or the secondary system.

In the use of COTRAN, the reactor core can be modeled as radial rings of homogeneous characteristics. One core model which has been established consists of 9 radial regions and 12 axial nodes. The radial region widths are kept about the same size, so that the continuity of neutron diffusion characteristics is maintained. The radial regions are assumed to have the nuclear and thermal-hydraulic characteristics of the most abundant type of bundle in that region. ENC has not established the precise modeling which will be used in the BWR stability analysis, but some derivation of the above will be used.

ENC also stated the possibility of stability tests at Oyster Creek and verification to previous BWR stability tests.

After the presentation and discussion, a brief staff caucus was held to summarize the key points. The most important preliminary concern is the validation of COTRAN to appropriate data. Some potential data for this comparison was mentioned and noted. Specifically, the comparison to electrically heated rod thermal hydraulic stability data and to integral reactor core stability tests. The importance of sensitivity studies through the possible range of BWR conditions was also mentioned. Parameters, such as, nodalization, smearing, and power distribution were identified for consideration. Also, any potential uncertainties and errors in the code should be evaluated, so that, appropriate licensings acceptance criteria may be established. Finally, the staff felt that ENC's current plans for review submittal were appropriate.

Marv Mendonca

M. M. Mendonca
Reactor Safety Branch

cc: F. Coffman
S. Rubin
R. Woods
Attendees (see attached list)
Meeting Summary Distribution List (attached)

LIST OF ATTENDEES

ENC'S BWR THERMAL HYDRAULIC STABILITY
CODE DEVELOPMENT

<u>Attendees</u>

Marvin Mendonca	NRC/DOR/RSB
Jerald Holm	ENC
R. B. Stout	ENC
G. E. Owsley	ENC
Wayne Hodges	NRC/DSS/AB
Chuck Leach	ENC
B. Morris	NRC/DOR/RSB
Roger Farr	CAROLINA POWER & LIGHT CO.
Gordon Bond	GPU SERVICE CORP.
Paul Boehnert	NRC/ACRS
Robert B. Lee	GPU SERVICE CORP.
Gary Holahan	NRC/DSS/AB

MEETING SUMMARY DISTRIBUTION

E. Case
V. Stello
D. Eisenhut
K. Goller
L. Shao
R. Bae.:
A. Schwencer
B. Grimes
D. Ziemann
G. Lear
R. Reid
D. Davis
R. Boyd
H. Denton
R. Mattson
D. Skovholt
R. Denise
R. DeYoung
D. Ross
R. Tedesco
V. Moore
R. Vollmer
M. Ernst
W. Gammill
P. Collins
C. Heltemes
R. Houston
T. Speis
R. Clark
J. Stolz
K. Kniel
O. Parr
W. Butler
D. Vassallo
J. Knight
S. Pawlicki
I. Sihweil
P. Check
T. Novak
Z. Rosztoczy
J. McGough
V. Benaroya
G. Lainas
T. Ippolito
G. Knighton
B. Youngblood
W. Regan
D. Bunch
J. Collins
W. Kreger
R. Ballard
M. Spangler
J. Stepp
L. Hulman
OELD
OI&E
R. Fraley, ACRS (16)
T. B. Abernathy, DTIE
J. Miller
H. Thornburg, IE
K. Seyfrit, IE
Docket Files/Central Files
NRR Reading
RSB Reading
Attendees

TOPICS OF PRESENTATION

- O ENC MODEL TO CALCULATE MARGIN OF STABILITY IN BWR'S
- O APPLICATION OF ENC MODEL TO CHANNEL STABILITY, REACTOR CORE STABILITY AND TO TOTAL SYSTEM STABILITY
- O VERIFICATION OF ENC MODEL AGAINST REACTOR DATA

ENC STABILITY ANALYSIS MODEL

- O ENC HAS DEVELOPED A MODEL TO CALCULATE REACTOR STABILITY MARGIN BY COMBINING THE REACTOR KINETICS PROGRAM XTRAN AND THE THERMAL-HYDRAULIC PROGRAM COBRA IV
- O THIS COMBINED PROGRAM IS CALLED COTRAN
- O THE MARGIN OF STABILITY IS ANALYSED IN THE TIME DOMAIN
- O THE CRITERIA FOR MEASURING THE MARGIN OF STABILITY IS THE DECAY RATIO (X_2/X_0). IF THE DECAY RATIO IS BELOW 1.0 THE REACTOR IS CONSIDERED STABLE.

XTRAN

- XTRAN PROVIDES THE NEUTRONIC SOLUTION TO COTRAN.
- THE OUTPUT FROM XTRAN INCLUDES
  - REACTOR REACTIVITY
  - TIME DEPENDENT TOTAL REACTOR POWER
  - TIME AND SPACE DEPENDENT POWERS AND FLUXES
- XTRAN IS A TWO-DIMENSIONAL (R-Z) REACTOR KINETICS CODE WITH
  - TWO GROUP INPUT WHICH IS COLLAPSED TO ONE GROUP FOR DIFF THEORY SOLUTION
  - FUEL TEMPERATURE AND MODERATOR DENISITY REACTIVITY FEEDB
  - SIX DELAYED NEUTRON GROUPS
- XTRAN HAS BEEN CHECKED AGAINST ANALYTICAL SOLUTIONS AND AGA OTHER REACTOR KINETIC COMPUTER PROGRAMS
- XTRAN HAS BEEN REVIEWED AND APPROVED BY THE NRC FOR REACTOR KINETIC ANALYSIS

COBRA IV

- 0 COBRA IV PROVIDES THE THERMAL AND HYDRODYNAMIC IMPUT TO COTRAN

- 0 TO REDUCE COMPUTER RUNNING TIME A STRIPPED VERSION OF COBRA HAS BEEN INCORPORATED INTO COTRAN

- 0 CROSS FLOW HAS BEEN ELIMINATED IN COTRAN VERSION BUT THIS SHOULD HAVE NO EFFECT ON BWR CALCULATIONS

- 0 VARIABLE FUEL TEMPERATURE CONDUCTIVITY HAS BEEN MODIFIED TO GIVE AN IMPROVED FIT TO DATA

- 0 DIRECT MODERATOR HEATING FROM NEUTRON MODERATION HAS BEEN ADDED

COTRAN

- o XTRAN SOLUTION FOR TIME DEPENDENT POWERS AND FLUXES
- o COBRA IV SOLUTION OF THERMAL AND HYDRAULIC FEEDBACK
- o SPECIAL FEATURES
  - SPECIFICATIONS OF INITIAL POWER SHAPE
  - PLOT OF POWER PROFILE VS. TIME FOR EASY VISUALIZATION OF REACTOR RESPONSE
- o APPLICATION OF COTRAN
  - STABILITY ANALYSIS
  - CONTROL ROD WITHDRAWAL
  - SCRAM REACTIVITY CURVES

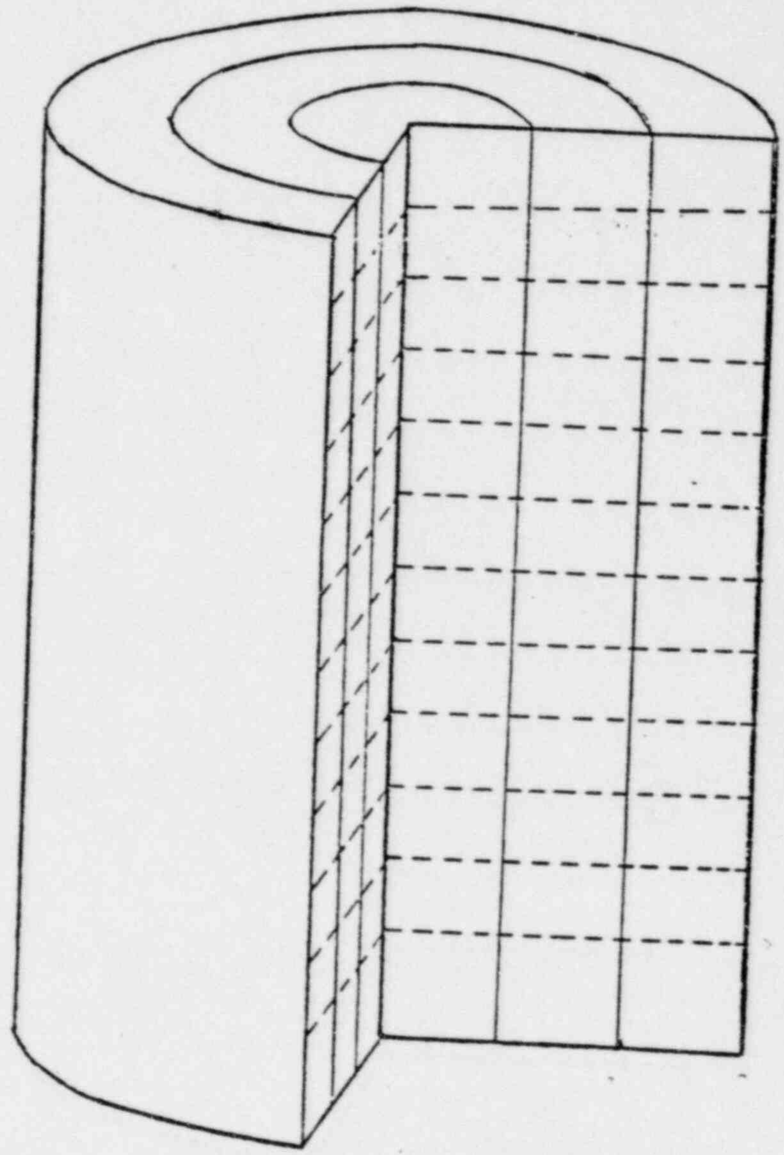
MARGIN OF STABILITY CALCULATION

- MODULE ANALYSIS
  - R-Z SIMULATION OF 4 BUNDLE MODULE.
  - REMAINDER OF REACTOR HOMOGENIZED INTO RADIAL REGIONS SURROUNDING THE MODULE
- REACTOR ANALYSIS USING MULTIPLE FUEL TYPES
- THE TIME RESPONSE OF ALL PARAMETERS WHICH ARE NOT MODELED IN COTRAN WILL BE INPUT AS FORCING FUNCTIONS.
- BOTH SYSTEMS ANALYZED FOR MARGIN OF STABILITY BY
  - INDUCING A PRESSURE, INLET FLOW OR CONTROL ROD PERTURBATION TO CAUSE A SMALL TRANSIENT.
  - CALCULATING A DECAY RATIO FROM THE TIME RESPONSE OF THE BUNDLE OR TOTAL REACTOR POWER
  - MARGIN OF STABILITY CALCULATED FOR A MATRIX OF LIMITING CONDITIONS.

REACTOR MODEL

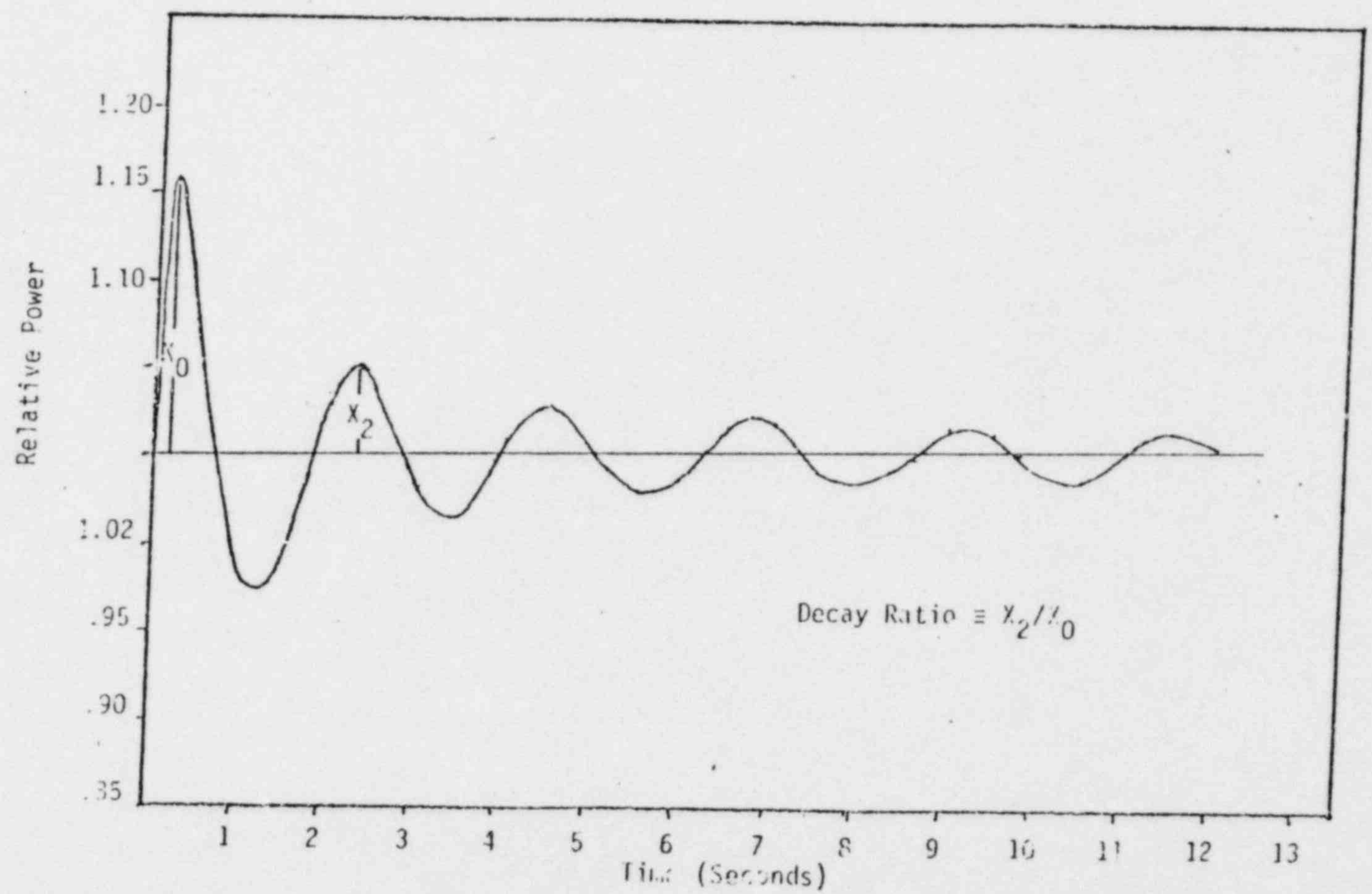
- O NINE RADIAL REGIONS - 12 AXIAL NODES (R-z)
- O REACTOR MODEL SET UP TO MAINTAIN THE RELATIVE PROPORTION OF
  - FUEL TYPES
  - ORFICING REGIONS
  - FLOW AREA
  - HEATED AND WETTED PERIMETERS
  - FUEL RODS
- O RELATIVE LOCATION APPROXIMATED BY LOCATION OF RADIAL REGIONS
- O EACH REGION APPROXIMATELY SAME WIDTH FOR BEST NEUTRONIC SOLUT TO DIFFUSION THEORY
- O LOSS COEFFICIENTS DETERMINED BY MEASUREMENT OF ENC FUEL OR BY MODELING AGAINST STEADY STATE SOLUTION
- O NEUTRONIC CROSS SECTIONS ARE GENERATED WITH A MULTIGROUP TWO-DIMENSIONAL DIFFUSION THEORY CODE USING A MICROSCOPIC DEPLETIC OF A BWR ASSEMBLY.

R-Z MODEL UTILIZED IN COTRAN
ANALYSIS OF REACTOR CORE



VERIFICATION OF STABILITY MODEL

- 0 CHECK AGAINST AVALABLE REACTOR DATA.
  - 0 PEACH BOTTOM DATA (WHEN RELEASED)
  - 0 POSSIBLE TESTS IN OYSTER CREEK
- 0 PARAMETER STUDY TO DEFINE SENSITIVITY OF NEUTRONIC AND THERMAL-HYDRAULIC INPUT PARAMETERS



TYPICAL REACTOR POWER RESPONSE TO A STEP
PERTURBATION IN SYSTEM PRESSURE

SIMPLIFIED FLOW DIAGRAM FOR COTRAN