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Sixteenth Water Reactor Safety Information Meeting

Volume 1
- Plenary Session
- Decontamination and Decommissioning
- License Renewal
- Human Factors
- Generic Issues
- Risk Analysis/PRA Applications
- Innovative Concepts for Increased Safety of Advanced Power Reactors

Held at
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Gaithersburg, Maryland
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ABSTRACT

This five-volume report contains 141 papers out of the 175 that were presented at the Sixteenth Water Reactor Safety Information Meeting held at the National Institute of Standards and Technology, Gaithersburg, Maryland, during the week of October 24-27, 1988. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included twenty different papers presented by researchers from Germany, Italy, Japan, Sweden, Switzerland, Taiwan and the United Kingdom. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

PROCEEDINGS OF THE
16th WATER REACTOR SAFETY INFORMATION MEETING

October 24-27, 1988

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REGISTERED ATTENDEES (NON-NRC)
16th WATER REACTOR SAFETY INFORMATION MEETING

J. A. ADAM
STONE & WEBSTER
245 SUMMER ST
BOSTON MA 02107
USA

S. L. ADDITON
TENERA
2011 EYE STREET N.W., SUITE 300
WASHINGTON DC 20006
USA

M. J. ADES
SAVANNAH RIVER LABORATORY
BLDG. 773-43A
AIKEN SC 29808
USA

D. W. AKERS
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

H. AKIMOTO
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

N. S. AKSAN
PAUL SCHERRER INSTITUTE
WUERENLINGEN CH5303
SWITZERLAND

R. P. ALLEN
PACIFIC NORTHWEST LABORATORY
P.O. BOX 999
RICHLAND WA 99352
USA

R. A. ALLEN
SAVANNAH RIVER LABORATORY
AIKEN SC 29808
USA

K. ALMENAS
UNIVERSITY OF MARYLAND
COLLEGE PARK MD 20742
USA

A. ALONSO
MADRID POLYTECHNIC UNIVERSITY
JOSE GUTIERREZ ABASCAL,2
MADRID 28006
SPAIN

H. ALSMEYER
KERNFORSCHUNGSZENTRUM, IRB/PRS
POSTFACH 3640
7500 KARLSRUHE
FRG

J. G. ANDERSEN
GENERAL ELECTRIC CO.
175 CURTNER AVE
SAN JOSE CA 95125
USA

J. M. ANDERSON
BECHTEL POWER CORP.
15740 SHADY GROVE ROAD
GAITHERSBURG MD 20874
USA

K. B. ANDERSSON
SWEDISH NUCLEAR POWER INSPECTORATE
BOX 27016
STOCKHOLM SW S-10252
SWEDEN

C. E. APPERSON
SAVANNAH RIVER LABORATORY
1635 ALPINE DR.
AIKEN SC 29801
USA

K. J. ARAJ
BDM
7915 JONES BRIDGE DRIVE
MCLEAN VA 22102
USA

W. C. ARCIERI
ENSA
15825 SHADY GROVE ROAD
ROCKVILLE MD 20850
USA

W. W. ASCROFT-HUTTON
HMNI
BAYNARDS HOUSE/WESTBOURNE GROVE
LONDON ENGLAND W2 4TF
UK

B. ATEFI
SCIENCE APPLICATIONS INT'L CORP.
1710 GOODRIDGE DR.
MCLEAN VA 22102
USA

N. I. AVDOSHKIN
USSR EMBASSY
1125 16TH ST NW
WASHINGTON DC 20036

S. M. BAJOREK
WESTINGHOUSE POWER SYSTEMS DIVISION
P.O. BOX 355
PITTSBURGH PA 15230-2728
USA

K. K. BANDYOPADHYAY
BROOKHAVEN NATIONAL LABORATORY
BLDG 129
UPTON NY 11973
USA

R. A. BARI
BROOKHAVEN NATIONAL LABORATORY
BLDG. 197C
UPTON NY 11973
USA

G. P. BAROIS
FRAMATOME
TOUR FIAT - CEDEX 16
PARIS LA DEFENSE FR 92084
FRANCE

J. H. BARON
MADRID POLYTECHNICAL UNIVERSITY
JOSE GUTIERREZ ABASCAL,2
MADRID 28006
SPAIN

B. R. BASS
OAK RIDGE NATIONAL LABORATORY
PO BOX 2003
OAK RIDGE TN 37831
USA

J. A. BAST
GENERAL ELECTRIC CO.
P.O. BOX 1072, BLDG. F3-8
SCHENECTADY NY 12301
USA

P. D. BAYLESS
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

P. BEDNARIK
SCIENCE & TECHNOLOGY
3900 LINNEAN AVE., NW
WASHINGTON DC 20008
CZECHOSLOVAKIA

K. D. BERGERON
SANDIA NATIONAL LABS.
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

R. F. BEYER
WESTINGHOUSE
206 NAVAJO RD.
PITTSBURGH PA 15241
USA

V. M. BHARGAVA
VIRGINIA POWER
5000 DOMINION BLVD.
GLEN ALLEN VA 23060
USA

W. BINNER
AUSTRIAN RESEARCH CENTER
SEIBERSDORF A-2444
AUSTRIA

D. P. BIRMINGHAM
BABCOCK & WILCOX CO.
1562 BEESON STREET
ALLIANCE OH 44601
USA

D. BIZZAK
CARNEGIE-MELLON UNIVERSITY
386 HUNTINGDON AVENUE
N. HUNTINGDON PA 15642
USA

R. J. BOHL
LOS ALAMOS NATIONAL LABORATORY
P.O. BOX 1663, MS K560
LOS ALAMOS NM 87545
USA

T. C. BORDINE
CONSUMERS POWER COMPANY
1945 WEST PARNALL RD.
JACKSON MI 49201
USA

R. B. BORSUM
BABCOCK & WILCOX CO.
1700 ROCKVILLE PIKE, #525
ROCKVILLE MD 20852
USA

B. E. BOYACK
LOS ALAMOS NATIONAL LAB
PO BOX 1663
LOS ALAMOS NM 87545
USA

D. R. BRADLEY
SANDIA NATIONAL LABS.
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

R. J. BRANDON
GE NUCLEAR CORP
6728 LOOKOUT BEND
SAN JOSE CA 95120
USA

P. A. BRATBY
NATIONAL NUCLEAR CORP.
BOOTHS HALL, CHELFORD ROAD
KNUTSFORD
UK

R. J. BREEDING
SANDIA NATIONAL LABS., DIV. 6410
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

S. J. BRETT
CEGB, GENERATION DEV. & CONSTRUCTION
GEN. DEV. & CONST. DIV.,BARNWOOD
GLOUCESTER GL47RS
UK

I. BRITTAIN
UKAEA/AEE WINFRITH
DOERCHESTER
DORSET DT28DH
UK

J. M. BROUGHTON
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

S. BRYANT
CENTRAL ELECTRICITY GENERATING BD.
BOOTHS HALL, CHELFORD RD
KNUTSFORD, CHESHIRE WA168QG
UK

C. E. BUCHHOLZ
GENERAL ELECTRIC CO.
175 CURTNER AVE
SAN JOSE CA 95125
USA

B. BUESCHER
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

L. C. BUFFARDI
GEORGE MASON UNIVERSITY
4400 UNIVERSITY DRIVE
FAIRFAX VA 22030
USA

J. D. BYRON
ELECTRIC POWER RESEARCH INSTITUTE
3412 HILLVIEW AVE.
PALO ALTO CA 94303
USA

J. I. CALVO
CONSEJO SEGURIDAD NUCLEAR
SOR ANGELA DE LA CRUZ 3
MADRID 28020
SPAIN

A. L. CAMP
SANDIA NATIONAL LABS. DIV. 6412
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

D. D. CARLSON
SANDIA NATIONAL LABS, DIV. 6513
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

R. CARO
CONSEJO DE SEGURIDAD NUCLEAR
C/SOR ANGELA DE LA CRUZ, 3
MADRID 28020
SPAIN

D. E. CARROLL
SANDIA NATIONAL LABS.
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

R. CARUSO
NUCLEAR ENERGY AGENCY
38 BLVD. SUCHET
PARIS 75016
FRANCE

D. A. CASADA
OAK RIDGE NATIONAL LABORATORY
PO BOX 2009, BLDG. 9104-1
OAK RIDGE TN 37831
USA

J. H. CHA
KOREA ADVANCED ENERGY RESEARCH INST.
PO BOX 7, DAEDUK-DANJI
TAEJON CHUNG-NAM
KOREA

S. CHAKRABORTY
PAUL SCHERRER INSTITUTE
WUERENLINGEN CH5303
SWITZERLAND

R. CHAMBERS
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

Y. I. CHANG
ARGONNE NATIONAL LABORATORY
9700 S. CASS AVE.
ARGONNE IL 60439
USA

D. CHAPIN
MPR ASSOCIATES, INC.
1050 CONNECTICUT AVENUE, N.W.
WASHINGTON DC 20036
USA

S. C. CHENG
TAIWAN POWER COMPANY
TAIPEI

ROC

L. Y. CHENG
BROOKHAVEN NATIONAL LABORATORY
BLDG 703
UPTON NY 11973
USA

R. D. CHEVERTON
OAK RIDGE NATIONAL LABORATORY
PO BOX 2009
OAK RIDGE TN 37831
USA

W. G. CHOE
TU ELECTRIC
400 N. OLIVE STREET
DALLAS TX 75201
USA

O. K. CHOPRA
ARGONNE NATIONAL LABORATORY
9700 S. CASS AVE., BLDG 335
ARGONNE IL 60439
USA

B. CHUNG
KOREA ADVANCED ENERGY RESEARCH INST.
NSC, DAE DUK DAN-JI, BOX 7
CHOONG-NAM
KOREA

D. T. CHUNG
SCIENTECH
11821 PARKLAWN DR.
ROCKVILLE MD 20878
USA

H. CHUNG
ARGONNE NATIONAL LABORATORY
9700 S. CASS AVE.
ARGONNE IL 60439
USA

J. P. CHURCH
SAVANNAH RIVER LABORATORY
BLDG. 773-41A
AIKEN SC 29808
USA

D. B. CLAUSS
SANDIA NATIONAL LABS. DIV. 6442
P.O. BOX 5800
ALBUQUERQUE NM 87122
USA

M. COLAGROSSI
ENEA/DISP
VIA VITALIANO BRANCATI, 48
ROME 00144
ITALY

M. W. CONEY
CENTRAL ELECTRICITY GENERATING BD.
C.E.R.L., KELVIN AVENUE
LEATHERHEAD SURREY KT227SE
UK

W. R. CORWIN
OAK RIDGE NATIONAL LABORATORY
PO BOX 2009
OAK RIDGE TN 37831
USA

D. CROUCH
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

W. H. CULLEN
MATERIALS ENGINEERING ASSOCIATES
9700-B M. L. KING HIGHWAY
LANHAM MD 20706
USA

G. E. CUMMINGS
LAWRENCE LIVERMORE NATIONAL LAB
P.O. BOX 808, L-198
LIVERMORE CA 94526
USA

H. D. CURET
ADVANCED NUCLEAR FUELS
2101 HORN ROAD
RICHLAND WA 99352
USA

B. D. CURRY
PHILADELPHIA ELECTRIC CO.
PO BOX A SARATOGA BR.
POTTSTOWN PA 19464
USA

R. A. CUSHMAN
NIAGRA MOHAWK POWER CORP.
301 PLAINHELD ROAD
SYRACUSE NY 13212
USA

D. A. DAHLGREN
SANDIA NATIONAL LABS., ORG. 6440
P.O. BOX 5800
ALBUQUERQUE NM 87123
USA

J. DALLMAN
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

R. DAMERON
ANATECH
10975 TORREYANA RD., SUITE 301
SAN DIEGO CA 92121
USA

J. DARLASTON
CENTRAL ELECTRICITY GENERATING BD.
BERKLEY NUCLEAR LABS
GLOUCESTER GL139PB
UK

W. L. DAUGHERTY
SAVANNAH RIVER LABORATORY
PO BOX A
AIKEN SC 29808
USA

N. W. DAVIES
UKAEA, RISLEY LABORATORY
WARRINGTON
CHESHIRE WA3 6AT
UK

M. A. DAYE
BECHTEL POWER CORP.
15740 SHADY GROVE RD.
GAITHERSBURG MD 20877
USA

J. A. DE MASTRY
FLORIDA POWER AND LIGHT CO.
BOX 14000
JUNO BEACH FL 33408
USA

L. O. DelGEORGE
COMMONWEALTH EDISON
P.O. BOX 767
CHICAGO IL 60690
USA

L. V. DeWITT
SAVANNAH RIVER LABORATORY
1313 WILLIAMS DR.
AIKEN SC 29801
USA

M. DIMARZO
UNIVERSITY OF MARYLAND
MECHANICAL ENGINEERING DEPT.
COLLEGE PARK MD 20016
USA

M. R. DINSEL
BECHTEL POWER CORP.
15740 SHADY GROVE ROAD
GAITHERSBURG MD 20874
USA

S. R. DOCTOR
PACIFIC NORTHWEST LABORATORY
P.O. BOX 999
RICHLAND WA 99352
USA

C. V. DODD
OAK RIDGE NATIONAL LABORATORY
PO BOX 2008
OAK RIDGE TN 37831
USA

T. F DORIAN
DOUB, MUNTZING & GLASGOW, CHARTERED
808 17TH STREET, N. W., SUITE 400
WASHINGTON DC 20006
USA

J. DRCEC
NPP KRSKO
PO VRBINA 12
KRSKO 68270
YUGOSLAVIA

S. S. DUA
GENERAL ELECTRIC
M/C 769, 175 CURTNER AVE.
SAN JOSE CA 95125
USA

S. W. DUCE
EG&G IDAHO INC
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

J. J. DUCO
CEA FRENCH ATOMIC ENERGY COMMISSION
DAS/SASC,CEN/FAR, BP NO. 6
FONTENAY-AUX-ROSES 92265
FRANCE

R. B. DUFFEY
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

K. K. DWIVEDY
VIRGINIA POWER
5000 DOMINION BLVD.
GLEN ALLEN VA 23060
USA

J. L. EDSON
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

G. R. EIDAM
BECHTEL/GPUN
P.O. BOX 72
MIDDLETOWN PA 17057
USA

D. M. EISSENBERG
OAK RIDGE NATIONAL LABORATORY
PO BOX 2009, BLDG. 9104-1
OAK RIDGE TN 37831
USA

Z. J. ELAWAR
ARIZONA NUCLEAR POWER PROJECT, 7202
P.O. BOX 52034
PHOENIX AZ 85072
USA

G. ELETTI
ENEA/DISP
VIA VITALIANO BRANCATI, 48
ROME 00144
ITALY

F. A ELIA
STONE AND WEBSTER
PO BOX 2325
BOSTON MA 02014
USA

G. T. EMBLEY
GENERAL ELECTRIC
BOX 1092
SCHENECTADY NY 12301
USA

T. C. ENG
DEPARTMENT OF ENERGY
EH-35, MAIL STOP F-137
WASHINGTON DC 20545
USA

R. G. ESPEFALT
SWEDISH STATE POWER BD.
VATTENFALL
VALLINGBY S-16287
SWEDEN

C. R. FARRAR
LOS ALAMOS NATIONAL LAB.
MS J576
LOS ALAMOS NM 87545
USA

H. E. FILACCHIONE
SCIENTECH
11821 PARKLAWN DR.
ROCKVILLE MD 20878
USA

S. R. FISCHER
MIDDLE SOUTH UTILITIES
188 E. CAPITOL STREET, SUITE 600
JACKSON MS 39201
USA

R. G. FITZPATRICK
BROOKHAVEN NATIONAL LABORATORY
BLDG 130
UPTON NY 11973
USA

C. W. FORSBERG
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2008, BLDG 4500N
OAK RIDGE TN 37831
USA

E. FOX
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2009
OAK RIDGE TN 37831-8063
USA

G. FREI
SIEMENS
HAMMERBACHERSTR. 12
ERLANGEN 852
FRG

B. C. FRYER
ADVANCED NUCLEAR FUELS
1930 CYPRESS
RICHLAND WA 99352
USA

H. FUJIMOTO
COMPUTER SOFTWARE DEVELOPMENT CO.
4-1, SHIBAKOUEN 2-CHOME, MINATO-KU
TOKYO 105
JAPAN

P. J. FULFORD
NUS CORPORATION
910 CLOPPER ROAD
GAITHERSBURG MD 20878
USA

R. R. FULLWOOD
BROOKHAVEN NATIONAL LABORATORY
BLDG 130
UPTON NY 11973
USA

F. M. GANTENBEIN
CEA FRENCH ATOMIC ENERGY COMMISSION
CEN-SACLAY DEMT/SMTS/EMSI
GIF-SUR-YVETTE 91191
FRANCE

F. GELBARD
SANDIA NATIONAL LABS.
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

D. I. GERTMAN
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

T. GINSBERG
BROOKHAVEN NATIONAL LABORATORY
BLDG 820
UPTON NY 11973
USA

T. GINZBURG
APPLIED BIOMATHEMATICS INC.
100 NORTH COUNTRY ROAD
SETAUKET NY 11733
USA

B. GITNICK
ENSA, INC.
15825 SHADY GROVE RD., STE. 120
ROCKVILLE MD 20850
USA

H. G. GLAESER
GESELLSCHAFT FUR REAKTORSICHERHEIT
FORSCHUNGSGELANDE
D-8046 GARCHING
FRG

J. GLEASON
WYLE LABORATORY
7800 GOVERNORS DRIVE WEST
HUNTSVILLE AL 35807
USA

D. W. GOLDEN
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

M. GOMOLINSKI
CEA FRENCH ATOMIC ENERGY COMMISSION
CEN/FAR, BP NO. 6
FONTENAT-AUX-ROSES 92265
FRANCE

C. GONZALEZ
MADRID POLYTECHNICAL UNIVERSITY
JOSE GUTIERREZ ABASCAL,2
MADRID 28006
SPAIN

E. D. GORHAM-BERGERON
SANDIA NATIONAL LABS.
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

T. C. GORRELL
SAVANNAH RIVER LABORATORY
RX TECH, DU PONT (SRP)
AIKEN SC 29808
USA

G. A. GREENE
BROOKHAVEN NATIONAL LABORATORY
BLDG. 820M
UPTON, NY 11973
USA

P. GRIFFITH
MIT
ROOM 7-044
CAMBRIDGE MA 02139
USA

F. GRIMM
SIEMENS
HAMMERBACHERSTR. 12
ERLANGEN 852
FRG

C. GRIMSHAW
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

W. W. GUNTHER
BROOKHAVEN NATIONAL LABORATORY
BLDG 130
UPTON NY 11973
USA

S. HABER
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

D. R. HAFFNER
WESTINGHOUSE HANFORD CO.
P.O. BOX 1970 A3-30
RICHLAND WA 99352
USA

F. M. HAGGAG
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2008, BLDG 4500-S
OAK RIDGE TN 37831-6151
USA

A. N. HALL
HM NUCLEAR INSTALLATIONS INSPECTORATE
ST PETER'S HOUSE, BALLIOL RD
BOOTLE,MERSEYSIDED L203LZ
UK

R. J. HAMMERSLEY
FAUSKE & ASSOCIATES
16W070 WEST 83RD STREET
BURR RIDGE IL 60521
USA

A. M. HASHIMOTO
JAPAN INST. OF NUCLEAR SAFETY
3-17-1 TOYANOMON
MINATOKU, TOKYO 105
JAPAN

Y. A. HASSAN
DEPT OF NUCL ENERGY, TEXAS A & M
COLLEGE STATION TX 77843
USA

J. R. HAWTHORNE
MATERIALS ENGINEERING ASSOCIATES
9700-B M. L. KING HIGHWAY
LANHAM MD 20706
USA

H. D. HAYNES
OAK RIDGE NATIONAL LABORATORY
PO BOX 2009, BLDG. 9201-3
OAK RIDGE TN 37831
USA

J. HAZELTINE
WYLE LABORATORY
7800 GOVERNORS DRIVE WEST
HUNTSVILLE AL 35807
USA

R. E. HENRY
FAI
16W070 WEST 83RD STREET
BURR RIDGE IL 60521
USA

G. F. HEWITT
UKAEA/HARWELL LABORATIRY
THERMAL HYDRAULICS DIV, BLDG 392
OXON OX11 ORA
UK

J. HIBBARD
MPR ASSOCIATES, INC.
1050 CONNECTICUT AVENUE, N.W.
WASHINGTON DC 20036
USA

D. E. HIDINGER
GENERAL ELECTRIC
14 CORONET CT.
SCHENECTADY NY 12309
USA

J. C. HIGGINS
BROOKHAVEN NATIONAL LABORATORY
BLDG 130
UPTON NY 11973
USA

C. J. HIGGINS
APPLIED RESEARCH ASSOCIATES, INC.
4300 SAN MATEO NE, STE B 380
ALBUQUERQUE NM 87111
USA

P. R. HILL
PENNSYLVANIA POWER & LIGHT CO.
2 N. NINTH STREET
ALLENTOWN PA 18101
USA

J. E. HINTON
DU PONT DE NEMOURS
SRP 707-C, RM. 329
AIKEN SC 29808
USA

T. J. HIRONS
LOS ALAMOS NATIONAL LABORATORY
P.O. BOX 1663, MS E561
LOS ALAMOS NM 87545
USA

M. HIROSE
NUCLEAR POWER ENG'G TEST CTR
FUJITA KMKO BLDG., TORANOMON, MINATO-
TOKYO 105
JAPAN

S. A. HODGE
OAK RIDGE NATIONAL LABORATORY
PO BOX 2009, BLDG. 9104-1
OAK RIDGE TN 37831
USA

P. G. HOFMANN
KERNFORSCHUNGSZENTRUM, IMF
POSTFACH 3640
7500 KARLSRUHE
FRG

C. H. HOFMAYER
BROOKHAVEN NATIONAL LABORATORY
BLDG 129
UPTON NY 11973
USA

G. S. HOLMAN
LAWRENCE LIVERMORE NATIONAL LAB
P.O. BOX 808, L-197
LIVERMORE CA 94550
USA

K. R. HOOPINGARNER
PACIFIC NORTHWEST LABORATORY
P.O. BOX 999
RICHLAND WA 99352
USA

R. G. HOPPE
WESTINGHOUSE ELECTRIC CORP.
P.O. BOX 79
W MIFFLIN PA 15122
USA

K. G. HORNBERGER
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2009
OAK RIDGE TN 37831-8056
USA

P. J. HOSEMANN
PAUL SCHERRER INSTITUTE
WUERENLINGEN CH5303
SWITZERLAND

B. J. HSEIH
ARGONNE NATIONAL LABORATORY
9700 SO CASS AVE.
ARGONNE IL 60439
USA

A. H. HSIA
URA
240 MONROE STREET
ROCKVILLE MD 20854
USA

H-N HSIAU
TAIWAN POWER COMPANY
TAIPEI TAIWAN
ROC

K. HU
BROOKHAVEN NATIONAL LABORATORY
BLDG. 475B
UPTON NY 11973
USA

D. S. HUMPHRIES
SCIENTECH
11821 PARKLAWN DR.
ROCKVILLE MD 20852
USA

M. L. HYDER
SAVANNAH RIVER LABORATORY
AIKEN SC 29808
USA

C. R. HYMAN
OAK RIDGE NATIONAL LABORATORY
PO BOX 2009, BLDG. 9104-1
OAK RIDGE TN 37831
USA

T. IGUCHI
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

R. L. IMAN
SANDIA NATIONAL LABS. DIV 6415
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

J. R. IRELAND
LOS ALAMOS NATIONAL LAB
PO BOX 1663, MSK 551
LOS ALAMOS NM 87544
USA

M. ISHII
PURDUE UNIVERSITY
WEST LAFAYETTE ID
USA

S. K. ISKANDER
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2008
OAK RIDGE TN 37831-6151
USA

R. IVANY
COMBUSTION ENGINEERING, INC.
1000 PROSPECT HILL RD.
WINDSOR CT 06095
USA

J. M. IZQUIERDO
CONSEJO SEGURIDAD NUCLEAR
SOR ANGELA DE LA CRUZ 3
MADRID 28020
SPAIN

M. J. JACOBUS
SANDIA NATIONAL LABS., DIV. 6447
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

J. F. JANSKY
BTB-LEONBERG
RILKESTRASSE S
LEONBERG D-7250
FRG

D. B. JARRELL
PACIFIC NORTHWEST LABORATORY
P.O. BOX 999
RICHLAND WA 99352
USA

G. W. JOHNSEN
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

A. B. JOHNSON
PACIFIC NORTHWEST LABORATORY
P.O. BOX 999
RICHLAND WA 99352
USA

E. R. JOHNSON
WESTINGHOUSE ELECTRIC CORP.
P.O. BOX 2728
PITTSBURGH PA 15230-2728
USA

H. V. JULIAN
VOLIAN ENTERPRISES, INC.
PO BOX 410
MURRYSVILLE PA 15668
USA

J. E. KALINOWSKI
SAVANNAH RIVER LABORATORY
AIKEN SC 29808
USA

P. S. KALRA
ELECTRIC POWER RESEARCH INSTITUTE
3412 HILLVIEW AVE.
PALO ALTO CA 94303
USA

F. B. KAM
OAK RIDGE NATIONAL LABORATORY
PO BOX 2008
OAK RIDGE TN 37831
USA

H. KAMATA
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

D. D. KANA
SOUTHWEST RESEARCH INSTITUTE
6220 CULEBRA ROAD
SAN ANTONIO TX 78284
USA

L. D. KANNBERG
PACIFIC NORTHWEST LABORATORY
P.O. BOX 999
RICHLAND WA 99352
USA

S. KARIMIAN
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

W. Y. KATO
BROOKHAVEN NATIONAL LABORATORY
BLDG 197C
UPTON NY 11973
USA

K. R. KATSMA
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

G. KATZENMEIER
KERNFORSCHUNGSZENTRUM, PHDR
POSTFACH 3640
7500 KARLSRUHE
FRG

W. R. KEANEY
GENERAL ASSOCIATES CORP.
1314 OAKVIEW DR.
WORTHINGTON OH 43085
USA

J. E. KELLY
SANDIA NATIONAL LABS. DIV. 6418
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

C. R. KEMPF
BROOKHAVEN NATIONAL LABORATORY
BLDG. 197-C
UPTON NY 11973
USA

W. L. KIRK
LOS ALAMOS NATIONAL LABORATORY
P.O. BOX 1663, N-DO, MS E561
LOS ALAMOS NM 87545
USA

E. KNOGLINGER
PAUL SCHERRER INSTITUTE
WUERENLINGEN CH5303
SWITZERLAND

A. KOHSAKA
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

M. J. KOMSI
IMATRAN VOIMA OY
P.O. BOX 112
VANTAA 01601
FINLAND

C. A. KOT
ARGONNE NATIONAL LABORATORY
9700 S. CASS AVE., BLDG 335
ARGONNE IL 60439
USA

G. S. KRAMER
BATTELLE COLUMBUS
505 KING AVE.
COLUMBUS OH 43201
USA

T. S. KRESS
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2009
OAK RIDGE TN 37831-8063
USA

C. A. KROPP
ENEA/DISP
VIA ANGUILLARESE K 1+300
ROME 00060
ITALY

G. A. KRUEGER
PHILADELPHIA ELECTRIC
2301 MARKET ST., N2-1
PHILADELPHIA PA 19101
USA

R. C. KRYTER
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2008
OAK RIDGE TN 37831-6010
USA

R. KUBOTA
HITACHI
SAIWAICHO 3-1-1
IBARAKI-KEN 317
JAPAN

C. A. KUKIELKA
PENNSYLVANIA POWER & LIGHT CO.
2 N. NINTH STREET
ALLENTOWN PA 18101
USA

D. S. KUPPERMAN
ARGONNE NATIONAL LABORATORY
9700 S. CASS AVE.
ARGONNE IL 60439
USA

K. F. KUSSMAUL
UNIVERSITY OF STUTTGART
PFAFFENWALDRING 32
STUTTGART 80 7000
FRG

P. S. LACY
URA
51 MONROE STREET
ROCKVILLE MD 20854
USA

T. K. LARSON
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

D. LEAVER
TANERA
1340 SARATOGA-SUNNYSIDE ROAD, SUITE 2
SAN JOSE CA 95129
USA

R. E. LECKENBY
UKAEA, RISLEY LABORATORY
WARRINGTON
CHESHIRE WA36AT
UK

M. LEE
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

N. E. LEE
COMBUSTION ENGINEERING
1000 PROSPECT HILL RD
WINDSOR CT 06095
USA

J. R. LEHNER
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

K. M. LEIGH
UKAEA/SRD
WIGSHAW LANE, CULCHETH
WARRINGTON WA34NE
UK

S. J. LEVINSON
BABCOCK & WILCOX CO.
3315 OLD FOREST RD
LYNCHBURG VA 24506
USA

P. M. LEWIS
PACIFIC NORTHWEST LABORATORY
P.O. BOX 999
RICHLAND WA 99352
USA

K. LIESCH
GESELLSCHAFT FÜR REAKTORSICHERHEIT
FORSCHUNGSGELANDE
D-8046 GARCHING
FRG

J. N. LILLINGTON
UKAEA/AEE WINFRITH
DOERCHESTER
DORSET DT28DH
UK

C. L. LIN
ELECTRIC POWER RESEARCH INSTITUTE
3412 HILLVIEW AVE.
PALO ALTO CA 94303
USA

L. LINDSTROM
SWEDISH NUCLEAR POWER INSPECTORATE
BOX 27016
STOCKHOLM SW S-10252
SWEDEN

Y. Y. LIU
ARGONNE NATIONAL LABORATORY
9700 S. CASS AVE.
ARGONNE IL 60439
USA

R. LOFARO
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

J. P. LONGWORTH
CENTRAL ELECTRICITY GENERATING BD.
COURTNEY HSE., WARWICK LA
LONDON
UK

F. J. LOSS
MATERIALS ENGINEERING ASSOCIATES
9700-B M. L. KING HIGHWAY
LANHAM MD 20706
USA

A. L. LOWE, JR.
BABCOCK & WILCOX CO.
PO BOX 10935
LYNCHBURG VA 24506
USA

T. S. LUBNOW
MPR ASSOCIATES
1050 CONNECTICUT AVE., NW
WASHINGTON DC 20036
USA

W. J. LUCKAS, JR.
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

H. L. MAGLEBY
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

A. P. MALINAUSKAS
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2008
OAK RIDGE TN 37831-6135
USA

R. M. MANDL
SIEMENS
HAMMERBACHERSTR. 12
ERLANGEN 852
FRG

C. F. MARKUS
WESTINGHOUSE ELECTRIC CORP.
P.O. BOX 79
W. MIFFLIN PA 15122-0079
USA

C. W. MARSCHALL
BATTELLE COLUMBUS
505 KING AVE.
COLUMBUS OH 43201
USA

P. MARSILI
ENEA/DISP
VIA VITALIANO BRANCATI, 48
ROME 00144
ITALY

B. MAVKO
J. STEFAN INSTITUTE
JAMOVA 39
LJUBLJANA 61000
YUGOSLAVIA

D. E. MC CABE
MATERIALS ENGINEERING ASSOCIATES
9700-B M. L. KING HIGHWAY
LANHAM MD 20706
USA

L. D. McCANN
WESTINGHOUSE ELECTRIC CORP.
P.O. BOX 79
W. MIFFLIN PA 15122-0079
USA

R. K. McCARDELL
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

D. J. McCLOSKEY
SANDIA NATIONAL LABS.
P.O. BOX 5800
ALBUQUERQUE NM 87122
USA

K. P. McKAY
WESTINGHOUSE ELECTRIC CORP.
P.O. BOX 79
W. MIFFLIN PA 15228
USA

N. R.H. McMILLAN
UKAEA/SRD
WIGSHAW LANE, CULCHETH
WARRINGTON WA34NE
UK

C. MEDICH
SIET
VIA NINO BIXIO 27
PIACENZA ITALY 29100
ITALY

H. B. MEIERAN
H B MEIERAN ASSOCIATES
458 SOUTH DALLAS AVENUE
PITTSBURGH PA 15208
USA

M. MERILO
ELECTRIC POWER RESEARCH INSTITUTE
3412 HILLVIEW AVE.
PALO ALTO CA 94303
USA

J. G. MERKLE
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2009
OAK RIDGE TN 37831-8049
USA

J. F. MEYER
SCIENTECH
11821 PARKLAWN DRIVE
ROCKVILLE MD 20852
USA

A. MEYER-HEINE
CEA FRENCH ATOMIC ENERGY COMMISSION
CEN CADARACHE-DERS/SEMAR BP NO. 1
SAINT PAUL LEZ DURANCE 13108
FRANCE

S. M. MIHAIU
YANKEE ATOMIC ELECT. CO.
508 MAIN ST.
BOLTON MA 01740
USA

J. S. MILLER
GULG STATES UTILITIES
P.O. BOX 220
ST. FRANCISVILLE LA 70775
USA

A. MINATO
ENERGY RESEARCH LAB., HITACHI LTD.
1168 MORIYAMA-CHO
HITACHI-SHI IBARAKI-KEN 316
JAPAN

S. M. MODRO
FZS-AUSTRIA C/O EG&G
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

T. MOMMA
JAERI C/O GENERAL ELECTRIC SAPS
RT. 168 S.
BEAVER PA 15077
USA

F. J. MOODY
GE NUCLEAR ENERGY
175 CURTNER AVE. MAIL CODE-769
SAN JOSE CA 95125
USA

R. J. MOORE
SAVANNAH RIVER LABORATORY
BLDG. 707C
AIKEN SC 29808
USA

F. I. MOPSIK
NAT'L INST. OF STDS. & TECH.
GAITHERSBURG MD 20899
USA

M.J. MOREAU
GENERAL ELECTRIC
P.O. BOX 1072
SCHENECTADY NY 12301
USA

V. MUBAYI
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

Y. MURAO
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

S. A. NAFF
SIEMENS AG.UB.KWU.U8S
POSTFACH 3220
ERLANGEN 8520
FRG

C. NAKAMURA
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

R. K. NANSTAD
OAK RIDGE NATIONAL LABORATORY
PO BOX 2008, MS 6151
OAK RIDGE TN 37831
USA

A. NATLIZIO
ATOMIC ENERGY OF CANADA
SHERIDAN PARK RSCH. COMM.
MISSISSAUGA ONTARIO L5K 1B2
CANADA

D. J. NAUS
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2009, BLDG 9204-1
OAK RIDGE TN 37831-8056
USA

E. NEGRENTI
ENEA/DISP
V. ANGUILLARESE, 301
ROME 00060
ITALY

D. B. NEWLAND
NATIONAL NUCLEAR CORP.
BOOTHS HALL, CHELFORD ROAD
KNUTSFORD ENGLAND
UK

L. Y. NEYMOTIN
BROOKHAVEN NATIONAL LABORATORY
BLDG 475B
UPTON NY 11973
USA

Y. NOGUCHI
CHUBU ELECTRIC POWER CO. INC
900 17TH ST. N.W., SUITE 714
WASHINGTON DC 20006
USA

P. NORTH
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

H. NOURBAKHSH
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

S. P. NOWLEN
SANDIA NATIONAL LABS.
PO BOX 5800, DIV. 6447
ALBUQUERQUE NM 87185
USA

E. I. NOWSTRUP
CONSULTANT
17605 PARK MILL DR
ROCKVILLE MD 20855
USA

A. NUHM
TECHNICATOME
CEN CADARACHE
CADARACHE 13115
FRANCE

J. O'HARA
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

K. R. O'KULA
SAVANNAH RIVER LABORATORY
BLDG. 773-41A
AIKEN SC 29808
USA

C. F. OBENCHAIN
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

T. OHNO
NUCLEAR POWER ENG'G TEST CTR
BLDG. 4-3-13, TORANOMON, MINATO-KU
TOKYO 105
JAPAN

M. OHUCHI
JAPAN SYSTEMS CORP.
NOMURA BLDG., 4-8 YOMBANCHO, CHIYODA-
TOKYO 102
JAPAN

T. OKUBO
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

R. C. OLSON
BALTIMORE GAS & ELECTRIC CO.
CCNPP-NGF PO BOX 1535
LUSBY MD 20657
USA

A. OMOTO
TOKYO ELECTRIC POWER
1901 L ST., NW, STE. 720
WASHINGTON DC 20036
USA

N. R. ORTIZ
SANDIA NATIONAL LABS., DIV. 6410
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

J. PAN
UNIVERSITY OF MICHIGAN
2250 G. G. BROWN BLDG., MECH. ENG.
ANN ARBOR MI 48108
USA

R. K. PAPESCH
BECHTEL-KWU ALLIANCE
15740 SHADY GROVE RD.
GAITHERSBURG MD 20877
USA

C. PARK
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

W. R. PEARCE
CONSULTANT
6846 GLENBROOK ROAD
BETHESDA MD 20814
USA

G. A. PERTMER
UNIVERSITY OF MARYLAND
DEPT. OF CHEMISTRY & NUCLEAR ENGR.
COLLEGE PARK MD 20742
USA

G. PETRANGELI
ENEA/DISP
VIA VITALIANO BRANCATI, 48
ROME 00144
ITALY

J. L. PIERREY
CEA FRENCH ATOMIC ENERGY COMMISSION
CEN/FAR, BP NO. 6
FONTENAT-AUX-ROSES 92265
FRANCE

A. PINI
ENEA/DISP
VIA VITALIANO BRANCATI, 48
ROME 00144
ITALY

M. G. PLYS
FAUSKE & ASSOCIATES
16W070 WEST 83RD STREET
BURR RIDGE IL 60521
USA

M. Z. PODOWSKI
RENSSELEAR POLYTECHNIC INSTITUTE
TROY NY 12180-3590
USA

A. Y. PORRACCHIA
CEA FRENCH ATOMIC ENERGY COMMISSION
CEN CADARACHE-DERS/SEMAR BP NO. 1
SAINT PAUL LEZ DURANCE 13108
FRANCE

D. A. POWERS
SANDIA NATIONAL LABS.
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

N. PRASAD
WESTINGHOUSE POWER SYSTEMS DIVISION
P.O. BOX 2728
PITTSBURGH PA 15230-2728
USA

T. PRATT
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

D. A. PRELEWICZ
ENSA, INC.
15825 SHADY GROVE RD. (SUITE 170)
ROCKVILLE MD 20850
USA

J. G. PRUETT
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2008
OAK RIDGE TN 37831-6135
USA

J. PUGA
UNITED ELECTRICIA, S. A. (UNESA)
FRANCISCO GERVAS, 3
MADRID 28020
SPAIN

C. E. PUGH
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2009
OAK RIDGE TN 37831
USA

W. J. QUAPP
WESTINGHOUSE HANFORD CO.
PO BOX 1970
RICHLAND WA 99352
USA

P. J. QUATTRO
MBZ, INC.
1175 HERNDON PKWY., STE. 150
HERNDON VA 22070
USA

Z. H. QURESHI
SAVANNAH RIVER LABORATORY
786-5A
AIKEN SC 29808
USA

H. J. REILLY
IDAHO NATIONAL ENGINEERING LAB
PO BOX 1625
IDAHO FALLS ID 83402
USA

L. RIB
LNR ASSOCIATES
8605 GRIMSBY CT.
POTOMAC MD 20854
USA

B. RIEGEL
GESELLSCHAFT FUR REAKTORSICHERHEIT
FORSCHUNGSGELANDE
D-8046 GARCHING
FRG

D. E. ROBERTSON
PACIFIC NORTHWEST LABORATORY
P.O. BOX 999
RICHLAND WA 99352
USA

S. B. RODRIGUEZ
EG&G IDAHO INC.
1646 GRANDVIEW #1
IDAHO FALLS ID 83402
USA

U. S. ROHATGI
BROOKHAVEN NATIONAL LABORATORY
BLDG 475B
UPTON NY 11973
USA

B. ROSENSTROCH
EBASCO SERVICES INC.
2 WORLD TRADE CENTER 89E
NEW YORK NY 10048
USA

S. T. ROSINSKI
SANDIA NATIONAL LABS, DIV. 6513
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

J. C. ROUSSEAU
CEA FRENCH ATOMIC ENERGY COMMISSION
CEN/GRENOBLE
GRENOBLE 38000
FRANCE

D. RUBIO
ELECTRIC POWER RESEARCH INSTITUTE
3412 HILLVIEW AVE.
PALO ALTO CA 94303
USA

K. A. RUSSELL
EG&G IDAHO INC.
1520 SAWTELLE
IDAHO FALLS ID 83415
USA

J. RUTHERFORD
CENTRAL ELECTRICITY GENERATING BD.
BOOTHS HALL, CHELFORD ROAD
KNUTSFORD CHESHIRE WA16 806
UK

B. F. SAFFELL
BATTELLE COLUMBUS DIVISION
505 KING AVENUE
COLUMBUS OH 43201
USA

R. T. SAIRANEN
TECHNICAL RSCH CTR OF FINLAND
POB 169
HELSINKI SF-00181
FINLAND

K. SAKANA
JAPAN INST. OF NUCLEAR SAFETY
FUJITA KANKOU TORANOMON MINATO
TOKYO 105
JAPAN

K. SAKANO
JAPAN INSTITUTE OF NUCLEAR SAFETY
FUJITA KANKON TOR. BLDG. 3-17-1
TOKYO 105
JAPAN

A. SALA
HIDROELECTRICO ESPANOLA
HERMOSILLA 3
MADRID 28001
SPAIN

J. SALUJA
VIKING SYSTEMS INTERNATIONAL
2070 WM PITT WAY
PITTSBURGH PA 15238
USA

L. SCHOR
YANKEE ATOMIC ELECT. CO.
508 MAIN ST.
BOLTON MA 01740
USA

D. G. SCHRAMMEL
UFK
WEBERSTR. 5
KARLSRUHE
FRG

S. SETH
MITRE CORP.
7525 COLSHIRE DR.
MC LEAN VA 22102
USA

W. J. SHACK
ARGONNE NATIONAL LABORATORY
BLDG. 212
ARGONNE IL 60439
USA

V. N. SHAH
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

R. H. SHANNON
CONSULTING ENGINEER
P.O. BOX 2264
ROCKVILLE MD 20852
USA

R. S. SHARMA
AMERICAN ELECTRIC POWER
ONE RIVERSIDE PLAZA
COLUMBUS OH 43017
USA

D. A. SHARP
SAVANNAH RIVER LABORATORY
AIKEN SC 29801
USA

D. L. SHAW
BALTIMORE GAS & ELECTRIC COMPANY
CALVERT CLIFFS NPP, P.O. BOX 1535
LUSBY MD 20657
USA

L. SHEN
ATOMIC ENERGY COUNCIL, ROC
NO 67, LANE 144, KEELUNG RD. SEC. 4
TAIPEI TAIWAN 107
ROC

G. L. SHERWOOD
U.S. DEPT. OF ENERGY
GERMANTOWN MD 21701
USA

P. SHEWMON
ACRS
2477 LYTHAM ROAD
COLUMBUS OH 43220
USA

K. SHIBATA
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

K. SHIMIZU
HITACHI, LTD.
1-1 SAIWAI-MACHI
HITACHI
JAPAN

A. SHIMIZU
OHBAYASHI CORP.
777 RIVERVIEW DR., APT. 9
ROCHESTER PA 15074
USA

J. J. SHIN
EBASCO SERVICES, INC.
2 WORLD TRADE CENTER
NEW YORK NY 10048
USA

M. S. SHINKO
EMERGENCY RESPONSE TEAM
PO BOX 129
WASHINGTON GROVE MD 20880
USA

B. S. SHIRALKAR
GENERAL ELECTRIC CO.
175 CURTNER AVE (M/C 186)
SAN JOSE CA 95125
USA

D. A. SIEBE
LOS ALAMOS NATIONAL LABORATORY
P.O. BOX 1663, MS K555
LOS ALAMOS NM 87545
USA

E. G. SILVER
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2009, BLDG 9201-3
OAK RIDGE TN 37831-8065
USA

F. A. SIMONEN
PACIFIC NORTHWEST LABORATORY
P.O. BOX 999
RICHLAND WA 99352
USA

F. B. SIMPSON
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

L. SLEGERS
SIEMENS KWU
BERLINER STR 295-303
OFFENBACH 6000
FRG

G. L. SMITH
WESTINGHOUSE HANFORD CO.
P.O. BOX 1970 X0-44
RICHLAND WA 99352
USA

A. W. SNYDER
SANDIA NATIONAL LABS., ORG. 6500
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

P. SOO
BROOKHAVEN NATIONAL LABORATORY
BLDG 830
UPTON NY 11973
USA

H. SPECTER
NEW YORK POWER AUTHORITY
123 MAIN STREET
NEW YORK NY 10601
USA

J. E. SPEELMAN
ECN
3 WESTERDUINWEG, P.O. BOX 1
PETTEN NEW HOLLAND 1755 ZG
THE NETHERLANDS

K. E. ST. JOHN
YANKEE ATOMIC ELECTRIC CO.
1671 WORCESTER RD
FRAMINFHAM MA 01701
USA

H. STADTKE
JOINT RESEARCH CENTRE-ISPRA ESTABLISH
ISPRA 21020
ITALY

D. D. STEPNEWSKI
WESTINGHOUSE HANFORD CO.
P.O. BOX 1970 NI-31
RICHLAND WA 99352
USA

E. J. STUBBE
TRACTABEL
31 RUE DE LA SCIENCE
BRUSSELS 1040
BELGIUM

R. K. SUNDARAM
YANKEE ATOMIC ELECT. CO.
508 MAIN ST.
BOLTON MA 01740
USA

J. D. SUTTON
YANKEE ATOMIC ELECTRIC CO.
580 MAIN ST
BOLTON MA 01740
USA

T. SUZUKI
TOSHIBA
SHINSUGITA ISOGO-KU
YOKOHAMA
JAPAN

I. SZABO
C.E.A.
C.E.N CADARACHE
ST. PAUL LES DURAN 43
FRANCE

A. TAKAGI
TOSHIBA
4921 NORWALK DR., APT V202
SAN JOSE CA 95125
USA

K. TASAKA
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

J. H. TAYLOR
BROOKHAVEN NATIONAL LABORATORY
BLDG 130
UPTON NY 11973
USA

B. J. TOLLEY
COMM. OF THE EUROPEAN COMMUNITIES (CI
200 RUE DE LA LOI
BRUSSELS 1049
BELGIUM

J. S. TONG
ATOMIC ENERGY CONTROL BOARD
PICKERING NGS OPERATIONS
PICKERING ONTARIO L1V2R5
CANADA

L. S. TONG
TAI
9733 LOOKOUT PLACE
GAITHERSBURG MD 20879
USA

F. M. TOUBOUL
CEA FRENCH ATOMIC ENERGY COMMISSION
CEN-SACLAY DEMT/SMTS/RDMS
GIF-SUR-YVETTE 91191
FRANCE

H. E. TRAMMELL
OAK RIDGE NATIONAL LABORATORY
104 OGLETHORPE PL.
OAK RIDGE TN 37830
USA

J. D. TROTTER
GROVE ENGINEERING
15215 SHADY GROVE RD.
ROCKVILLE MD 20878
USA

C-K TSAI
WESTINGHOUSE POWER SYSTEMS DIVISION
P.O. BOX 355
PITTSBURGH PA 15230-2728
USA

T. TSUJINO
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

B. D. TURLAND
UKAEA CULHAM
CULHAM LABORATORY
ABINGDON
UK

G. TYROR, DIRECTOR
UKAEA/SRD
WIGSHAW LANE, KNUTSFORD
WARRINGTON WA34NE
UK

R. E. UHRIG
OAK RIDGE NATIONAL LABORATORY
113 CONNORS DRIVE
OAK RIDGE TN 37830
USA

R. A. VALENTIN
ARGONNE NATIONAL LAB - BLDG. 208
9700 S. CASS AVE
ARGONNE IL 60439
USA

G. L.C.M. VAYSSIER
MINISTRY OF SOCIAL AFFAIRS
P.O. BOX 69
VOORBURG 2270MA
THE NETHERLANDS

O. VESCOVI
SIET
VIA NINO BIXIO 27
PIACENZA 29100
ITALY

G. L. VINE
ELECTRIC POWER RESEARCH INSTITUTE
3412 HILLVIEW AVE.
PALO ALTO CA 94303
USA

D. S. WALLS
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2009, MS-8057
OAK RIDGE TN 37831-8065
USA

S. F. WANG
INST. OF NUCLEAR ENERGY RSCH.
PO BOX 3-3, LUNG TAN
TAIPEI TAIWAN 32500
ROC

O. J. WANG
WESTINGHOUSE HANFORD CO.
P.O. BOX 1970 XO-44
RICHLAND WA 99352
USA

R. WANNER
PAUL SCHERRER INSTITUTE
WUERENLINGEN CH5303
SWITZERLAND

E. A. WARMAN
STONE & WEBSTER
245 SUMMER ST.
BOSTON MA 02107
USA

K. E. WASHINGTON
SANDIA NATIONAL LABS.
P.O. BOX 5800
ALBUQUERQUE NM 87185
USA

W. L. WEAVER
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

E. O. WEINER
WESTINGHOUSE HANFORD CO.
P.O. BOX 1970, L2-57
RICHLAND WA 99352
USA

P. A. WEISS
SIEMENS AG / UB KWU
HAMMERBACHERSTR. 12+14
ERLANGEN 8520
FRG

H. J. WELLAND
EG&G IDAHO INC.
442 JOAN AVENUE
IDAHO FALLS IDAHO 83401
USA

E. T. WESSEL
CONSULTANT
312 WOLVERINE STREET
HAINES CITY FL 33844
USA

H. WESTPHAL
GESELLSCHAFT FUR REAKTORSICHERHEIT
SCHWERTNERGASSE 1
D-5000 COLOGNE 1
FRG

A. M. WHITE
SAVANNAH RIVER LABORATORY
BLDG. 773-11A
AIKEN SC 29808
USA

J. D. WHITE
OAK RIDGE NATIONAL LABORATORY
P.O. BOX 2008
OAK RIDGE TN 37831-6009
USA

G. M. WILKOWSKI
BATTELLE COLUMBUS
505 KING AVE.
COLUMBUS OH 43201
USA

W. K. WINEGARDNER
PACIFIC NORTHWEST LABORATORY
P.O. BOX 999
RICHLAND WA 99352
USA

F. J. WINKLER
SIEMENS
RINGSTRASSE 19
SPARDORF
FRG

L. WOLF
KERNFORSCHUNGSZENTRUM, PHDR
POSTFACH 3640
7500 KARLSRUHE 1
FRG

A. J. WOLFORD
EG&G IDAHO INC.
P.O. BOX 1625
IDAHO FALLS ID 83415
USA

D. N. WOODY
SAVANNAH RIVER LABORATORY
AIKEN SC 29808
USA

R. O. WOOTON
BATTELLE COLUMBUS
505 KING AVE.
COLUMBUS OH 43201
USA

J. WREATHALL
SAIC
2929 KENNY RD.
COLUMBUS OH 43221
USA

D. W. WRIGHT
ANCHOR/DARLING VALVE
701 FIRST ST.
WILLIAMSPORT PA 17701
USA

W. WULFF
BROOKHAVEN NATIONAL LABORATORY
BLDG 475B
UPTON NY 11973
USA

H. XU
BROOKHAVEN NATIONAL LABORATORY
BLDG. 130
UPTON NY 11973
USA

Y. YAMAMOTO
TAKENAKA CO.
21-1 8CHOME CHUO-KU
TOKYO
JAPAN

H. YASOSHIMA
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

M. YOKOTA
JAPAN ATOMIC RESEARCH INST.
TOKAI-MURA, NAKA-GUN
IBARAKI-KEN 319-11
JAPAN

Y. YOSHIMOTO
HITACHI, LTD.
SAIWAI-CHO 3-H
HITACHI-SHI 316
JAPAN

R. W. YOUNGBLOOD
BROOKHAVEN NATIONAL LABORATORY
BLDG 130
UPTON NY 11973
USA

R. ZIPPER
GESELLSCHAFT FUR REAKTORSICHERHEIT
SCHWERTNERGASSE 1
D-5000 COLOGNE 1
FRG

P. C. ZMOLA
C&P ENGINEERING
5409 NEWINGTON RD.
BETHESDA MD 20816
USA

R. ZOGRAN
MPR ASSOCIATES, INC.
1050 CONNECTICUT AVENUE, N.W.
WASHINGTON DC 20036
USA

PROCEEDINGS OF THE
SIXTEENTH WATER REACTOR SAFETY INFORMATION MEETING

October 24-27, 1988

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
Office of Governmental and Public Affairs
Washington, D.C. 20555

Remarks by
Chairman Lando W. Zech, Jr.
U.S. Nuclear Regulatory Commission
at the
16th Water Reactor Safety Information Meeting
National Institute of Standards & Technology
October 24, 1988

NUCLEAR SAFETY RESEARCH - A VITAL ROLE IN ACHIEVING NUCLEAR SAFETY

Good morning ladies and gentlemen. It is my pleasure to have this opportunity to welcome you to the Sixteenth Annual Water Reactor Safety Meeting. I would like to extend a special greeting to our international guests.

As you are all aware, nuclear safety is of international importance. I have recently returned from the Soviet Union, Austria, Germany, and Italy where I spoke with senior nuclear safety officials. The most important issues we discussed were the continued safe operation of existing nuclear power plants, the improvement of nuclear safety throughout the world, and ways that we might share knowledge and experience in the fields of civilian nuclear reactor safety.

NRC is sponsoring the 16th Water Reactor Safety meeting to provide a forum for the exchange of the latest work in Water Reactor Safety Research throughout the world. I believe that this exchange can help to enhance the safe operation of nuclear power plants world-wide.

Today I would like to discuss briefly the importance of the NRC's nuclear safety research program, and the direction of the NRC research program through the mid-1990's. You will be hearing in the next few days -- in considerable detail -- about many important areas of research that are all important to nuclear safety.

The Commission regards the nuclear safety research program as an integral part of our decision-making process. We are constantly faced with making very difficult decisions, decisions which potentially can have profound impacts not only on public health and safety but on the continued viability of our national nuclear energy program.

It is important that we realize the role public confidence plays in our country's use of nuclear power. Through tough decisions we must demonstrate our ability to protect our fellow citizens and the quality of our environment.

Without public confidence, I question the future of nuclear power in our country.

To make these important safety decisions, we on the Commission must have at our disposal the best technical information available. We must be able to understand the magnitude of the risk involved in nuclear power generation, the inherent uncertainties in risk estimates, and the costs and impacts of alternative courses of action.

Over the years, our research program has provided the information necessary to enable us to make many difficult decisions. Specifically, our research program has been vital in the areas of analysis of severe reactor accidents and the formulation of rules to deal effectively with issues such as Anticipated Transients Without Scram (ATWS), Pressurized Thermal Shock, and Station Blackout.

It is the primary role of our research program to characterize the overall level of safety of our operating plants. Research has supported the preparation of the Reactor Risk Reference Document, NUREG-1150, which is intended to update and upgrade the information set forth in the Reactor Safety Study (WASH-1400) published over a decade ago. The management of our forthcoming Individual Plant Examination (IPE) program to assess risk outliers in currently operating plants is another example of our research staff providing substantial support to the mission of NRC. Both of these efforts are providing additional assurance that our operating plants adequately protect the public by identifying the outliers and correcting the vulnerabilities of these plants to severe accidents and transients.

Internally we are able to validate the safety margins provided by our current regulations. This feedback allows us to make appropriate modifications to our regulations when needed. Our recent revisions of our Emergency Core Cooling System Regulations (Appendix K) and General Design Criterion 4 (GDC-4) addressing pipe restraints and jet impingement shields, are examples where research has shown our previous regulatory position to be overly conservative. In the case of GDC-4, such conservatism may even have been counter to reactor safety.

In addition to the internal monitoring capability of our research program, our research program provides this agency with an independent audit capability by which it can objectively evaluate the data and analyses submitted to support license applications and license amendments. This audit capability helps to ensure that the technical information we are using as a basis for our safety decisions is free from any unintentional bias. NRC research verifies basic safety assumptions and provides the scientific foundation for positions taken by the agency.

I believe this capability strengthens the credibility of the agency. The broad-based research programs of the NRC, the Department of Energy and the industry, provide all of us with a longer range perspective by which we can identify and evaluate potential safety questions before they become major issues affecting the safety of the public.

The Commission has carefully considered this longer range perspective in directing the future activities of the reactor safety research program.

Presently, there are two clear agency objectives in reactor regulation which provide the focus for our future research effort. The first objective is ensuring the continued safe operation of existing nuclear power plants. Although we have resolved many of the outstanding safety issues through our licensing, inspection and research programs, we must strive not only to maintain the current level of safety provided by our nuclear power plants, but also to improve plant performance and reliability where that appears achievable.

Our success in achieving this goal of improved plant performance is not related solely to hardware, but to the performance of plant operating and maintenance personnel as well. Meeting this goal is, I believe, essential to the future of nuclear power.

Improved operator training, including training in responding to severe accidents and transients, is an important element in meeting this objective. Accordingly, a large component of our future research effort must be focused on directly supporting this objective.

Research will be initiated to improve our understanding of the human factor in plant operations and to provide the technical basis to support the establishment of accident management programs that will provide validated procedures for dealing with situations that could potentially lead to core damage. These procedures must also include provisions for initiating actions which would effectively mitigate accidents should they occur and thus minimize any adverse impacts on the public.

In addition, we must deal with the problems that will undoubtedly arise as plants get older. We must continue to develop and promulgate effective criteria for plant operations, for maintenance and testing and we must develop improved methods to monitor overall plant performance.

Research programs to characterize and quantify plant aging processes, such as corrosion, erosion, and radiation damage must be strengthened. The development of risk and reliability methods to allow us to monitor and improve upon plant performance will be accelerated.

To support these initiatives, we will clearly need to maintain a base program of more fundamental research on materials and severe accident phenomenology, such as fission product behavior and containment loading. Seismic and structural engineering programs will also be continued. We expect feedback from our Individual Plant Examination program will also be an important element in defining research in our future program for advanced reactor designs.

The second objective of the agency to which the research program must be directed is to provide a regulatory framework which will facilitate the development and licensing of standard LWR plants and the development and deployment of improved LWR concepts. We will be systematically examining the risks and advantages of such concepts and will make modifications to our regulations as appropriate. Closely related to this is the need to establish design and performance criteria for use by those who are designing and testing other reactor concepts, such as gas and sodium cooled reactors. A stable, appropriately conservative regulatory environment that will ensure a comparable level of safety for these plants is essential to the logical development of these concepts, and our programs must reflect this need. Again, our defense-in-depth philosophy will be utilized.

Although I have spoken today primary on reactor safety research, our research program is much broader in scope. Similar programs, although less extensive, are being carried out to support the regulation of radioactive waste disposal, transportation, fuel cycle facility operations, and nuclear material safety and safeguards.

The importance of the NRC's research program cannot be overstated. The Comission regards this program as a vital component to our ongoing decision-making process. We believe that we must maintain and support our research program to fully protect the interest and safety of our country in its use of nuclear energy.

We have taken into account the recommendations of the National Academy of Science review of NRC's research program. In addition, the agency continues to work vigorously to pursue cooperative programs with the international nuclear safety community, the Department of Energy and the nuclear industry. One of our more important actions in the area of international cooperation is the Memorandum of Cooperation with our Soviet counterparts, which will allow us to cooperate to our mutual benefit in the area of reactor safety research.

For the future, we must all continue to work together to make nuclear power safe world-wide. I believe nuclear safety research has an extremely important role to play in ensuring that we meet our common goal of safety of nuclear energy on a world-wide basis. Working co-operatively in research we can achieve more than individual efforts might produce. Our successes in sharing technical information, operating experience, and jointly-funded experiments should inspire us to increase these efforts.

I am convinced that safety research - has in the past - is now - and will remain in the future - an <u>essential</u> element toward ensuring safety in achieving the full benefits of nuclear energy.

OVERVIEW OF THE NUCLEAR REGULATORY COMMISSION'S
SAFETY RESEARCH PROGRAM

Remarks by

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission

16th Annual Water Reactor Safety Meeting
Gaithersburg, MD
October 24-27, 1988

I want to thank Chairman Zech very much for his keynote speech, and I want also to thank Dr. George Tyror of SRD for making a special trip to address our opening session of the meeting. I am also happy to add my welcome to you to the 16th Annual Water Reactor Safety Meeting.

This morning I want to look back over the past year, and then to look forward to the expectations and needs of next year and beyond. 1988 has been a year of great accomplishment in the Office and program of safety research. The short list of accomplishments is more than 6 pages long; I will not take the time to read the entire list, but I will mention several highlights.

1. <u>ECCS Appendix K Revision</u>: The research effort of twenty years duration on emergency core cooling for the operating reactors is now essentially complete. The result is an understanding of the important phenomenon, and validated best estimate computer codes for calculating the key thermal-hydraulic variables of operating LWRs in LOCAs. This is a scientific and engineering achievement of the first rank.

2. <u>Pressurized Thermal Shock</u>: Regulatory Guide 1.99 incorporates the most recent results of research on the effect of copper and nickel on neutron fluence embrittlement of reactor vessels. It provides the best methods of analysis and data base for safety assessment of PTS.

3. <u>NUREG-1150, or the PRA Method Reference Document</u>: NUREG-1150 is nearing completion, and will be discussed at one of this afternoon's sessions. It is the most important advance in PRA methods since the WASH-1400 Reactor Safety Study. It includes level 1, 2, and 3 studies of five plants, and external events, i.e., earthquake, fire, and flood for 2 of them. Three peer review groups, and many individuals have reviewed the draft 1150 published early in 1987, and the final version addresses the comments made. The expert opinion elicitation method that has been developed for levels 1 and 2 is especially noteworthy: I believe this development will gain world class recognition. Also I want to point out that modifications to these five plants have already improved their safety as a consequence of problems disclosed by the study and actions taken.

4. Resolution of Station Blackout: PRAs have shown that station blackout is a dominant accident sequence in a number of plants. Many years of engineering effort produced the station blackout rule, recently approved by the Commission and now enthusiastically endorsed by the utilities. When implementation of reliability measures for diesel and other emergency power systems is completed over the next 3 years, a major improvement in the safety of U.S. nuclear plants will have been accomplished.

5. Severe Accident Integration Plan: This plan points the way to resolution of severe accident issues, and it defines the elements of the plan and their interrelationship. It is a strategic plan, as you can see from its elements, and research is important throughout. The elements are:

   - Individual Plant Examinations (IPE)

     It is self-evident that the extension and application of risk assessment methods to every operating plant in the U.S. will help to disclose hidden severe accident vulnerabilities, and thus make it possible to reduce core damage probability for the entire set of plants.

   - Containment Performance Improvement (CPI)

     This element calls for assessment of capability of the several generic containment types to withstand the loads that severe accidents could cause. As improvements may be needed and are warranted, we will recommend them to the Commission. The MARK I containment assessment will be completed soon and the recommendations made in December.

   - Improved Plant Operations

     The record of the last four years for U.S. plant operations shows clear improvements. There is still room for improvement by a variety of means, including such things as better procedures, diagnostic inspections, SALPs, and the like. Better operation means fewer safety system challenges, and hence enhanced safety.

   - Severe Accident Research provides the knowledge of the phenomena and the ability to define important sequences and calculate the key parameters. Results from research are essential to the success of the plan.

   - Accident Management

     Its purpose is to develop the strategy and measures that, following an initiating event, will prevent core damage, or terminate the progress of damages; if that is not possible, to maintain containment as long as possible and minimize the consequences of offsite release.

- External Events

  An assessment of the risks arising from earthquakes, fire, and flood is a vital element of the plan. Work on definition of these events is underway now, and the assessment at individual plants is expected to begin in early 1990.

6. <u>Nuclear Safety Research Review Committee</u>

   Last year I spoke about the response to the recommendations of the National Research Council report, "Revitalizing Nuclear Safety Research". I am happy to report that the NSRRC is now in operation, chaired by Professor Neil Todreas of MIT. This Committee of 12 distinguished people has written its first letter on the overall research program, and is now reviewing each of the four main research elements. I am confident that their advice will help to focus and improve the quality of safety research.

7. <u>RES Reorganization</u>:

   I announced a change in organization of RES last July, in order to concentrate our people resources most effectively, and to define clearly the responsibilities for resolution of safety issues on the one hand, and for follow-up and exploratory research on the other. Dr. Wayne Houston heads the Division of Safety Issue Resolution (DSIR) focusing on the aspects of problem solving, cost benefit analysis, and responsible for the IPE. Dr. Brian Sheron heads the Division of Systems Research (DSR), responsible for systems and severe accident research, and for accident management and severe accident research, and for accident management and human factors research. There is a strong and symbiotic relationship between DSIR and DSR, especially in the area of severe accidents. Mr. Guy Arlotto's Division of Engineering (DE) continues to perform outstanding research in engineering and materials, and in high level and low level waste. Dr. Bill Morris' Division of Regulatory Application (DRA) continues to carry out its duties with vigor and effectiveness in rulemaking, in radiation and health effects, and in advanced reactors and standardization.

For these and for all the other research accomplishments in the past year, I want to express my sincere appreciation to the many people in RES, at laboratories and contractors, that made them possible. Well done. I hasten to add that there is much to do, and we do not have time to rest on any laurels.

I would like now to mention a few of the important tasks ahead.

1. Pressure Vessel Support Embrittlement

   In addition to the well anticipated and planned research on properties, crack arrest, and fatigue in reactor pressure vessels and piping, there is concern and work to be done on pressure vessel supports. Following the finding of low temperature, low neutron fluence embrittlement of the HFIR reactor at Oak Ridge, we have assessed the possible consequences for pressure vessel supports at operating reactors, and conclude that the supports at some plants could be marginal with respect to NDTT before the end of life. A careful plant-by-plant review is therefore indicated.

2. TMI-2 Reactor Vessel Bottom Head

   We plan to cut samples from the reactor vessel for metallurgical analysis next summer. An international consortium has formed under the OECD/NEA to share sponsorship of this effort with the NRC, at a total expected cost of about $7M. This research is expected to reveal the condition of the bottom head, and to indicate what the margin to failure of the vessel was at the time of the accident. This is a question of great importance to accident management development, because of the fact that the vessel did not fail despite the 20 tons or more of molten material that flowed to the bottom head.

3. Aging

   We have begun to extend aging research beyond the scope of reactor vessels and piping to other vital components and systems. Our objective is to uncover hitherto unknown aging phenomena that could reduce design safety margins or cause unexpected common mode failures in vital safety systems as plants approach the end of their design life. We intend to bring our findings to the attention of the utility industry in the event that action is indicated.

4. Human Factors

   It is well known that human errors are a major contributor to severe accident risks. The human factors research that is getting underway this year is expected to contribute to both accident prevention and mitigation by means of improved procedures, better qualification and training, and by the development of techniques of human performance and reliability measurement relating to plant specific PRAs.

5. High and Low Level Waste Research

   There is a clear need for the research that will enable NRC to judge the adequacy of the license application of DOE for the high level waste repository at Yucca Mountain, and also for the regulatory needs of low level waste storage sites. This is true because of the complexities of hydrologic flow, and because of the long geologic time scales involved. I believe that public interest in these questions will intensify the needs for research.

6. Severe Accident Research

Of all the challenges ahead in the research 5 year plan, this is the most difficult. The severe accident closure plan that I mentioned is a major step forward, but the plan does not take the place of data and quantitative models that describe the loads from severe accidents that bear first on the primary system, and in the event of its failure, on containment. In particular, I see an urgency to resolving the pressurized melt ejection/direct containment heating question in the case of PWRs, and the dry well liner melt-through question in the case of MARK I BWRs. These are questions of great importance to safety, worthy of your attention. Let the intellects of this group latch on to them and help lead the way to a resolution.

To conclude my thoughts to you this morning, I wish to say that the Office of Research undertook greatly expanded responsibilities a year and a half ago, adding generic issue resolution and rulemaking for the Agency to the already assigned research role. I say to you that this Office, its laboratories and contractors - and I mean the many people who are putting their effort to work on all the tasks - have come through the breaking-in phase, and are producing now at a very effective level. You have done so during a time of budget difficulty, and of changing priorities. I salute you. And I look for even greater returns in the next year and beyond. The very best job that all of us can do in safety research and issue resolution is a necessity.

SEVERE ACCIDENT RESEARCH IN THE UNITED KINGDOM

J G Tyror
Director, Safety and Reliability Directorate, UKAEA

R Garnsey
Head of the Safety and Technology Branch,
CEGB PWR Project Management Team

D Hicks
Director, Water Reactors Programme Directorate, UKAEA

1. INTRODUCTION

Nuclear reactors and other major hazard plant have traditionally been designed in a 'deterministic' way - that is, provision is made to deal satisfactorily with all accidents that could arise within the design basis of the plant. As safety assessment technology has developed, and particularly as a result of accidents such as those at Three Mile Island and Chernobyl, there has been an increasing awareness of the contribution to risk and to public concern of events beyond this design basis. By 'severe accidents' we mean major system failures outside the design basis which give rise to highly unsatisfactory consequences such as severe damage to the reactor, high risks to site personnel, and in the worst case, large releases of radioactivity to the environment.

The UK position on severe accidents reflects two essential facts. First, the large potential social impact of such events makes their avoidance and control a matter of special concern. Second, their rarity means that world-wide experience of them is very limited. Our understanding of severe accidents must therefore be based on predictive models, validated against a wide variety of experiments, and of course, against such accident experience as we have.

In this paper, in order to provide a background to the UK research programme, we explain the general principles on which control and avoidance of severe accidents are based in the UK. How these principles have developed in practice in the UK PWR programme is then discussed, using as an example the Sizewell 'B' Public Inquiry. Finally two examples of the UK severe accident research are described: the work on steam explosions and on fission product chemistry.

2. PRINCIPLES FOR CONTROL AND REGULATION

The Nuclear Installation Acts of 1965 and 1969 impose an absolute liability upon the operators of commercial reactors for any injury or damage caused by the release of radioactive material from their installations. However, these Acts also provide for a regulatory body with responsibility for issuing licences to the operators of a nuclear plant and for attaching appropriate conditions to those licences. The regulatory authority for all industrial plant in the UK is the Health and Safety Executive (HSE), which delegates the licensing function in respect of nuclear plant to the Head of the Nuclear Installations Inspectorate (NII).

The fundamental principle applied in the UK to the regulation of industrial risks is the so-called ALARP (As Low As Reasonably Practicable) principle. This requires that the operators do whatever is reasonably practicable to reduce risk, bearing in mind the costs of further reduction. Detailed guidance on how this principle is to be implemented for nuclear power reactors is provided by both regulators and designers. The NII has published Safety Assessment Principles [1], to be used by their inspectors as a guide to whether all reasonably practicable steps have been taken to prevent accidents and, should they occur, to minimise their radiological consequences. The Principles distinguish between accidents which could give rise to dose equivalents received by the public of no more than one ERL[1] and those accidents which might give rise to larger doses.

Detailed assessment levels of frequency and radiation dose are set out for the former group. This quantitative guidance is to indicate when a design has reached the point where the principle of ALARP has been fulfilled. For the latter group the Principles state that the frequency should be made as low as reasonably practicable. The relations between dose and frequency in the Principles are illustrated schematically in figure 1.

The principal electrical utility in England and Wales, the Central Electricity Generating Board (CEGB), has published Design Safety Criteria [2], which cover all nuclear reactors, and a set of Design Safety Guidelines [3] which expand on and interpret the Criteria for the PWR. Like the NII's Principles, the Criteria include detailed targets for accidents giving doses of less than one ERL (figure 2 gives a schematic representation of these targets). Accidents which would lead to higher off-site doses are covered by a target total frequency of 10^{-6} per year for all accidents giving a 'large uncontrolled release', with a maximum

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1.

An Emergency Reference Level is the radiation dose below which countermeasures are unlikely to be justified.

contribution of 10^{-7} from any single such accident sequence. Some latitude is allowed for 'uncontrolled releases' at levels between 1 ERL and 10 ERL to have probabilities somewhat higher than 10^{-6}.

3. THE APPLICATION OF THE PRINCIPLES AT THE SIZEWELL B INQUIRY

Although the Public Inquiry into the building of the Sizewell 'B' PWR was not part of the licensing process, it gave an opportunity for the NII and the CEGB to explain their Principles and Criteria, and therefore their attitude to severe accidents, in greater depth. At the Inquiry, the NII indicated the levels of individual and societal risk which might arise from a reactor which just satisfied their Principles [4]. The derivation of these risk levels involved many assumptions, and the NII were careful to explain that these were illustrative levels which might arise from the application of their Principles to a reactor at Sizewell, and had no status as safety goals. What the NII seeks from a licence applicant as regards severe accidents is a satisfactory demonstration that in general risks are as low as reasonably practicable, and in particular there is no sharp increase in risk associated with accidents bigger than those within the design basis, for which specific frequency/dose guidance is provided by their Principles.

The CEGB also presented their Design Safety Criteria to the Inquiry. They stated that their Criteria reflected their considered view that a risk of death to an individual of 10^{-6} per year was of little concern to most people. In the CEGB's pre-construction safety report the greatest emphasis was, of necessity, placed on accidents within the design basis. The aim was to demonstrate that the operation of safety systems could reduce the consequences of such accidents to within the design basis limit on consequences, and that the systems provide the necessary reliability so that the probability of exceeding the design basis limits was remote. This approach has been described as aiming to ensure that accidents which in practical terms could conceivably happen have consequences which can be accepted, and to ensure that accidents which have unacceptable consequences will in practical terms not occur. The reactor design evolved in a number of ways to ensure that the CEGB's design Criteria were met.

Although the main emphasis in the Sizewell 'B' safety case was on design basis accidents, a preliminary Probabilistic Risk Analysis (PRA) dealing with the frequency and consequences of internally initiated severe accidents was presented to the Inquiry [5]. This was not part of the CEGB's licensing case except insofar as it helped to demonstrate that there is no sharp increase in consequences just beyond the 10^{-6} per year frequency. During the course of this work it was shown that the risk associated with severe accidents was dominated by the interfacing systems LOCA (the "V sequence") and, even though the event was beyond the design basis, design modifications were introduced to further reduce risk (an additional valve added to the hot leg suction lines). The frequency of a degraded core accident arising from an internal event was assessed by the PRA to be 1.24×10^{-6} per year. Containment failure, and therefore a large release was predicted in only 6% of these cases.

The source term analysis in the Sizewell 'B' PRA was very similar to that in WASH-1400 [8]; some of the WASH-1400 source terms were used directly and the rest were calculated using the same models for and treatment of fission product transport. It was explained at the inquiry that these methods were intended as being conservative (i.e. pessimistic) and that research had begun in the UK and elsewhere to produce more realistic methods. To give some idea of what changes in source terms might result from these new methods, "second estimates" were produced for some of the source terms [9], using the understanding of the mechanisms available at that time. The second estimates were expressed in terms of subjective probabilities for different magnitudes of source term reduction relative to WASH-1400 values. Typical average reductions in source term values of between one half and one tenth were predicted.

The report of the Inspector at the Public Inquiry, Sir Frank Layfield, to the Secretary of State, ran to 109 chapters [6]. The topic of safety occupied 44 of these chapters, 2 of which dealt specifically with severe accidents. Whilst recommending acceptance of the case for proceeding with Sizewell 'B', the report contained a large number of recommendations and other observations. Two particular issues raised are relevant to the present paper.

The first issue concerned the definition of tolerable levels of risk. The Inspector was concerned that it was difficult to understand directly the means used to express what risk the reactor posed, and that there were no nationally accepted guidelines on what constituted an acceptable risk. He proposed that the HSE should formulate and publish guidelines on the tolerable levels of individual and societal risk to workers and to the public from nuclear power stations. As a first step he recommended that the HSE publish a document on the basis of which public, expert and parliamentary opinion could be expressed. In response to these recommendations, the HSE has recently published such a document [7]. It discusses tolerable levels of individual and societal risk. With regard to individual risk it concludes that, for the public, risks of death higher than 10^{-4} per year are intolerable, whereas risks less than 10^{-6} per year might be broadly acceptable, provided there is a benefit to be gained and proper precautions are taken. Furthermore, it concludes that the risk of any individual in the UK dying from a nuclear accident is on any reckoning very substantially less than 10^{-6} per year if reactors are designed to meet the Safety Assessment Principles and the Design Safety Criteria. For societal risk the document addresses the question of tolerability by comparisons with risks from petrochemical complexes, freak flooding of the River Thames or large aircraft crashes. It suggests that a significant nuclear accident anywhere in the UK might be accepted as just tolerable at a frequency of 10^{-4} per year. Such an accident might involve doses of more than 100 mSv to anyone within 3 km and might, on pessimistic assumptions, lead to the possible eventual death from cancer of up to 100 people.

The second issue raised in the report of the Sizewell 'B' public inquiry relevant to this paper was that the empirical basis for the modelling of degraded core phenomena was limited. On this subject, Layfield commented,

> "The research programmes now being carried out in the UK and overseas to try to provide a better empirical basis for containment and source term analysis are potentially a most valuable addition to knowledge on a subject of great importance. The two most significant aspects are the performance of the containment and the assumptions made about source terms. The NII recognises the importance of those topics and, if consent is given, will wish to ensure that the best practicable estimates are made of source terms."

4. OBJECTIVES OF SEVERE ACCIDENT RESEARCH IN THE UK

It is clear from the above discussion that severe accident analysis has a significant but not dominant part to play in the overall safety case for UK reactors. A research programme aimed at developing an understanding of the phenomenology and consequences of severe accidents and at establishing techniques for associated risk evaluations is however of great importance. It is increasingly necessary in licensing and public debate to demonstrate such understanding and to establish confidence in the low levels of risk quoted for severe accidents.

Severe accident studies are themselves a manifestation of the ALARP principle and provide an opportunity to introduce reasonably practicable modifications in design or operation, aimed at risk reduction. There is also increasing awareness of the potential scope for operator action to mitigate the most severe consequences of severe accidents. Such accident management provisions can be based only on an understanding of the alternative progression pathways which an accident might take, and this in turn requires an understanding of basic accident phenomenology.

Of course severe accident research by its very nature tends to be expensive and the international dimension of severe accident consequences is well understood. There are therefore strong incentives for international collaboration in severe accident research. This is recognised in the UK approach, and we take part in many of the international initiatives in this field. An important objective of the UK severe accident research programme is to maintain a body of expertise capable of assessing the significance for the UK PWR programme of the results of research in other countries.

5. THE UK SEVERE ACCIDENT RESEARCH PROGRAMME - AN OVERVIEW

In many countries, safety research is principally in the hands of utilities and regulatory bodies (and their respective contractors). In the UK we have a third body, the UK Atomic Energy Authority (UKAEA), which provides facilities and staff to perform research on behalf of both Government and industry. The majority of the severe accident research in the UK is carried out in the UKAEA, funded by Government and the electrical utilities.

The funding for severe accident research in the UK is modest, and is focussed in ways which both reflect our appreciation of those issues of particular concern within the UK and contribute to international efforts in this area. There is a considerable reliance on bilateral links, such as that with the USNRC, and on multilateral international activities such as the DEMONA, LACE and ACE experimental programmes. The UK programme should therefore not be seen in isolation, but as a contribution to a world-wide effort.

This having been said, the UK PWR programme does attempt to cover, at some level of detail, all the parts of the study of severe accident phenomenology and source terms, namely:

(A) degraded core behaviour - degradation of overheated fuel, fuel melting and relocation, lower head phenomena and vessel penetration, high pressure melt ejection and ex-vessel steam explosions, core debris interactions with concrete;

(B) thermal hydraulics - heat and mass transfers in the pressure vessel, primary circuit and containment, hydrogen production and combustion, direct containment heating;

(C) fission product transport - in-vessel release from degrading fuel, chemical speciation, primary circuit transport, release in core-concrete interactions, transport within the containment, release to the environment;

(D) containment response - response to slow overpressure, response to rapid overpressure, enhanced leakage versus catastrophic failure.

In addition to these topics the UK does research into the environmental consequences of releases of radioactivity, and into PRA topics such as source term uncertainties, human factors and external hazards. Rather than attempt to describe the whole programme in detail, in the remainder of the paper we describe two areas in which the UK has been particularly active: steam explosions and fission product chemistry.

6. STEAM EXPLOSION RESEARCH

The probability that a core melt accident would lead to a steam explosion of sufficient violence to cause a failure of the containment is generally believed to be low. The discussions at the Sizewell 'B' public inquiry focussed on the question of how low this probability is. The original PRA assigned a value of 10^{-4} to the conditional probability for this mode of containment failure, given core degradation. A UK reassessment of the probability arrived at a value of 10^{-2} as a likely upper bound. Sensitivity studies showed that the effect on risk of increasing the conditional probability to the higher value was not large. However the issue of steam explosions is of particular concern within the UK because it is one of the conceivable ways in which the containment could fail at an early stage of the accident. Work to confirm the low probability of such an event was therefore seen to be desirable. This section describes both the experimental work carried out at Winfrith on steam explosions, and the corresponding theoretical work carried out at Culham. The aim of this work is to develop a basic understanding of the complex phenomena involved and to be able to argue for the low probability of damaging steam explosions from a position of knowledge.

6.1 EXPERIMENTAL STUDIES ON STEAM EXPLOSIONS

The first studies at Winfrith of thermal interactions between molten uranium dioxide and water were started in the late 1970s and used 0.5 kg thermite-generated melts, released under the surface of a pool of water in a 55 litre reaction vessel [10]. In some of the experiments the release was restricted only by system pressure, whereas in others a catchpot was used to restrict the spread of the melt within the water pool. A number of the free-release tests led to molten fuel coolant interactions (MFCIs) triggered by a mechanism associated with the impact of an ejected section of the charge container on the base of the reaction vessel. The maximum energy yield observed corresponded to 1.4% of the available thermal energy in that part of the melt which participated in the interaction. In the restricted geometry releases both spontaneous and externally triggered MFCIs were observed, the maximum energy being 5.9% of the available thermal energy in the participating melt. The mass of interacting uranium dioxide was quite small, typically around 6% of the total material.

It was realised that it was not possible to draw firm conclusions about steam explosions at the reactor scale from these small-scale experiments. To examine the phenomena at a larger scale, the Molten Fuel Test Facility (MFTF) was constructed (see figure 3). This enabled melts up to 24 kg to be released into a vessel containing 1.5 te of water. The small-scale tests described above were replicated at larger scales in the Scaling Urania Water (SUW) tests [11]. A total of eleven tests were carried out in this series, three with a free and the remainder with a restricted geometry. All but two of these tests exhibited one or more MFCIs. Pressure plots from one of these tests, showing two MFCIs are shown in figure 4. These interactions had similar characteristics to those seen in the smaller scale tests. The mechanical energy yield, expressed as a fraction of the thermal energy in that part of the melt which participated in the interaction, had a maximum value of 4.3% for saturated pools, and decreased with increasing subcooling. These tests showed that the fractional yields, expressed in terms of the amount of melt participating in the interaction, showed a simple scaling behaviour over the range of melt sizes studied. A major uncertainty was then the amount of melt which would mix with the coolant and participate in an interaction. The theoretical work described below focusses on the mixing issue, and plans are being drawn up for experiments which will test the models.

Another issue which has been examined experimentally is the effect of pressure on the ability to trigger steam explosions. This has been studied in the High Pressure Thermite Rig (HPTR), which enables 5 kg quantities of melt at a temperature of 3600 K to be poured under gravity into a small, saturated water pool under ambient pressures of up to 25 MPa, and a well characterised shock to be injected. At this time, steam explosions have been triggered at ambient pressures of up to 5.6 MPa.

As well as the experiments described above using water as the coolant, there has also been an extensive programme of experiments in which the melt is poured into liquid sodium. Having data from two such dissimilar liquids furthers the underlying aim of gaining an understanding of the mechanisms responsible for MFCIs (and the mechanisms responsible for their suppression).

The CEGB also has a programme of experimental work looking at aspects of MFCI phenomenology. This involves small-scale experiments under well-controlled conditions using simulant materials. The phenomenology of mixing is studied in a thin, "two-dimensional" tank which allows flow visualisation of the mixing process within a large region and measurements of the local mixing ratio. It is hoped to extend the current isothermal experiments (Mercury/water) to high temperature materials using either solid particles or molten metals. Both pouring geometry and explosion-driven mixing will be investigated, the latter representing the situation where the materials are initially separated, but may be driven together and mixed, either by a small prior explosion or possibly by the main explosion itself.

6.2 THEORETICAL MODELLING OF STEAM EXPLOSIONS

A complete theoretical treatment of an MFCI would have to model the four stages essential to such an event:

A. the initial slow coarse mixing of melt with coolant;

B. the trigger event which initiates rapid fine-scale fragmentation within a small region;

C. the escalation of the interaction and its propagation through an extended region of the mixture;

D. the expansion of the system.

The Culham programme includes consideration of all four stages, but work to date has concentrated on stages A and C.

The initial, coarse mixing stage is modelled by the computer code CHYMES, developed at Culham [12,13,14]. The model is two-dimensional and time-dependent, allowing all three fluids (melt, coolant liquid and coolant vapour) to slip relative to one another with prescribed drag. Additional models, describing the convection and evolution of measures of the length-scales of the melt and coolant have been included in the code. As well as being preferable on theoretical grounds, these augmented code gives improved agreement with the small-scale mixing experiment performed at Argonne National Laboratory. There are grounds for expecting that only sufficiently intimately mixed configurations can support the escalation and propagation stage, and that there are limits to the extent to which large volumes of melt and coolant can intimately mix. Preliminary calculations with CHYMES suggest that intimate mixing is indeed more difficult on a large scale, and at high melt temperatures.

A one-dimensional, compressible, transient, multi-component fluid dynamics model of the escalation and propagation stage is under development at Culham. The model, known as CULDESAC [15,16,17], comprises four components (water, steam, larger melt fragments, and smaller melt fragments). Calculations have been performed for gas-dynamic shocks and simplified detonations for which exact solutions are available, and very good agreement has been obtained. CULDESAC has also been used to study time-dependent strong, weak and Chapman-Jouquet detonations.

The code CSQII [18], developed at Sandia National Laboratories, is used to model the expansion stage. It comprises a two-dimensional thermo-hydrodynamic model, with provision for parametric modelling of detonations, coupled to models of elastic and plastic deformation and fracture of solid materials. The parametric specification of detonations will be set so that CSQII reproduces approximately the output of CULDESAC in the same local circumstances, thus merging modelling of escalation and propagation into that of expansion.

Because the modelling of steam explosions involves considerable areas of uncertainty, the modelling strategy described above has been designed to make uncertainty analyses both efficient and physically meaningful. There has been a clear separation of the models into macro-physics (conservation equations and equilibrium constitutive relations) and micro-physics (disequilibrium constitutive relations, such as the drag between components in multi-component flow). Deficiencies in knowledge are to be found largely in the micro-physics; probability distributions for values of the uncertain parameters are assessed and the resulting uncertainties are propagated through the calculation using the method of Latin Hypercube Sampling [19], specially adapted for the accurate estimation of low probabilities of extreme outcomes [20]. This approach has the advantage that even though the values may be uncertain, the parameters themselves are physically well defined.

6.3 STEAM EXPLOSION RESEARCH - CONCLUSIONS

The UK severe accident research programme is particularly concerned with those events which could lead to a failure of the containment early in the accident sequence. In-vessel steam explosions constitute one of these types of events. It is believed that they have a low probability; the objective of the work is to develop a mechanistic understanding of the processes involved to be used as the basis of a demonstration of this conclusion. The programme has both an experimental and a theoretical component.

Experiments with urania melts in water at different scales have shown that, over the range studied, the efficiency of interactions, expressed in terms of the amount of melt participating, shows a simple scaling behaviour. This leaves the extent of mixing, as a function of melt size and pressure, as an important uncertainty, and current model development and validation work is focussing on this question.

7. FISSION PRODUCT CHEMISTRY RESEARCH

The methodology of WASH-1400 treated each fission product element as a single chemical entity with a single, unvarying set of transport properties (with the exception of iodine, where inorganic and organic forms were distinguished). By the time of the Sizewell 'B' Public Inquiry there was a growing awareness that this was a serious over-simplification. Fission product elements could, particularly in the high temperatures of the primary circuit, change their chemical form, and indeed react with one another. The chemical form would then strongly influence the transport properties of the radionuclides. Such questions as when they would condense, either onto surfaces or as aerosol particles, and how they could react with surface or aerosol materials cannot be answered without a knowledge of chemical form. Some of these issues were discussed at the Inquiry, and qualitative use was made of early results coming from a programme of fission product chemistry studies which had already been started at the UKAEA's Winfrith laboratories. A brief description of this programme and some of the results constitute the subject matter of the present section.

The main issues addressed by fission product chemistry can be divided up into the following five headings.

(i) Chemical Speciation. The behaviour of fission products released from a damaged core depends on which chemical species they form. The chemical forms of high temperature simulant fission product vapours have been determined directly using a combination of matrix isolation-infrared spectroscopy and mass spectrometry. These techniques have, for example been applied to a study of the caesium iodide-boric acid system [21,22,23,24]. The results of this study are described in more detail below.

(ii) Vapour-Surface Interactions. Once the chemical forms of fission product vapours have been established, one can address the question of vapour interactions with surfaces. If the walls are at a sufficiently low temperature the vapours may simply condense there. Alternatively there may be a chemical reaction between the vapour species and the surface materials. One such reaction studied is that between caesium hydroxide vapour and 304 stainless steel [23,25]. Deposition velocities were determined as functions of temperature and the values were incorporated into appropriate modelling codes.

(iii) Aerosol Formation. Materials other than fission products in the core region can form aerosols, and these aerosols can play an important role in fission product transport. The formation of aerosols from overheated silver-indium-cadmium control rods has been studied [26,27]. In an inert atmosphere aerosol production was dominated by cadmium, but in steam indium also contributed, in the form of its oxide.

(iv) Vapour-Aerosol Interactions. Once one knows what vapours and aerosols could be present in the primary circuit, the next question is how they can interact. This work has begun at Winfrith; in one of the early studies molecular iodine vapour was passed over deposits of cadmium aerosol [28]. The rate for reaction with the aerosol was found to be an order of magnitude larger than that for reaction with the stainless steel substrate.

(v) Containment Chemistry. The studies described above are aimed at defining the nature of the source from the primary circuit to the containment. Studies are also required to establish the behaviour of the fission products within the containment. To this end a glovebox facility (FALCON) has been developed at Winfrith, in which the behaviour of material released by heating lightly irradiated fuel pellets can be studied.

Programmes of work in all the above areas are currently in progress, using a wide range of techniques and facilities. There are strong links with international activities, particularly involving the detailed chemical analysis and physical characterisation of samples taken in various large-scale experiments. The studies show that the transport of fission products, particularly within the primary circuit, is strongly dependent on which species are formed both in condensed and vapour phases, and how the vapour species react with surfaces and aerosol materials.

The relevance of these detailed chemical investigations to fission product source terms can be illustrated by studies on the caesium iodide-boric acid system. It was believed that fission product iodine would initially be stabilised in the primary circuit in the form of caesium iodide. There could however be a significant concentration of boric acid also available within the primary circuit at the time of the release. The matrix isolation-infrared spectroscopic and mass spectrometric studies demonstrated conclusively that caesium iodide reacted with boric acid to produce volatile hydrogen iodide and relatively low-volatility caesium borate. In addition to identifying the species present, matrix isolation infrared spectroscopy can also be used to measure the vibrational frequencies of the molecules. This information can be used in statistical mechanical calculations of the thermodynamic properties of the vapour-phase species. This has been done for boric acid and the results obtained were significantly different from previously tabulated values [24]. Application of these data to its reaction with caesium iodide led to predictions for hydrogen iodide production an order of magnitude higher than the previous data, and in good agreement with the experimental results. (The change caused by using the new figures in the calculations is shown on figure 5, which is a plot of a measure of hydrogen iodide production against temperature, as calculated first with the earlier values and then with the new values.)

These studies show that although fission product iodine may start out as caesium iodide, if this vapour meets boric acid, the iodine is likely to be transformed into hydrogen iodide. This represents a considerable change in the transport properties of the iodine. Hydrogen iodide is more volatile than caesium iodide, but it is also much more reactive with steel surfaces. Treating the iodine as caesium iodide in the source term models could lead one to predict its presence in the containment as an aerosol, whereas in fact it has been retained on primary circuit surfaces.

The pieces of information obtained from chemical studies of individual systems have to be integrated in a single calculational framework before the effect of the new information on the source term can be appreciated. At present it is too early to make definitive statements on whether source terms will be significantly increased or decreased. The chemistry of severe accidents, especially as concerns the primary circuit, is intrinsically complex, and research often increases this complexity by discovering new compounds and reactions which have to be taken into consideration. This having been said, it must be an important objective of the research to rule out compounds and reactions which are of little importance, and to focus the attention of the modelling on a minimal set which dominate the transport of important fission products.

8. CONCLUSIONS

The fundamental principle applied in the UK to risks from nuclear accidents is the ALARP principle, which specifies that the risks should be made as low as reasonably practicable. This principle and the way in which it has been applied to PWR safety were scrutinised at the Sizewell 'B' public inquiry. One of the consequences of the ALARP principle is that consideration of accidents does not stop at the design basis boundary. Severe accident analysis plays a small but significant part in the UK research into PWR safety issues.

The UK maintains an expertise in all aspects of PWR severe accident analysis. This enables it to contribute to and obtain information from international programmes of severe accident research. In addition to this general coverage there are specialised areas, reflecting specific UK concerns and capabilities, in which effort is particularly concentrated. Two of these: steam explosions and fission product chemistry, have been described in more detail in the present paper.

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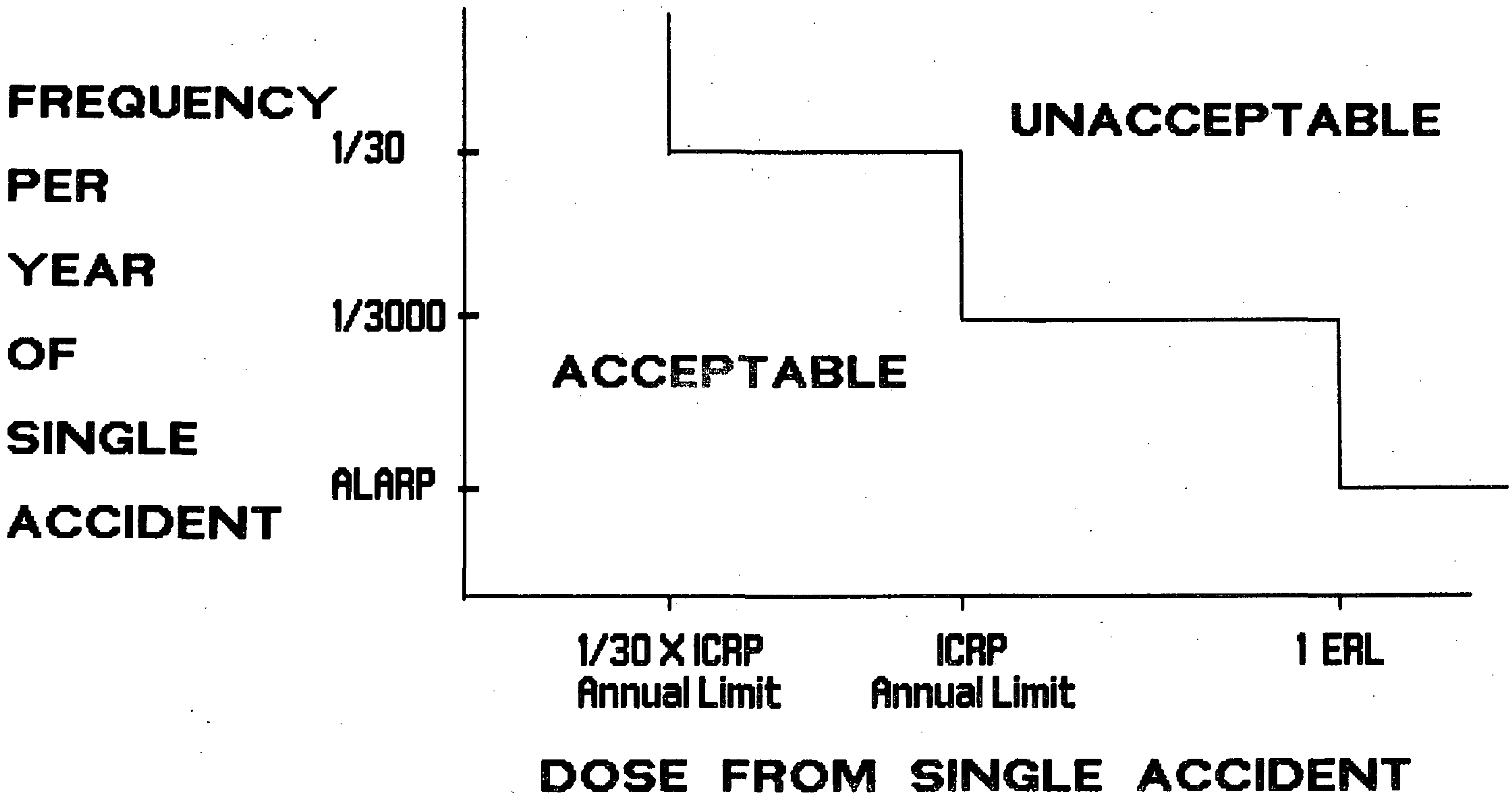

Figure 1 NII Safety Assessment Principles

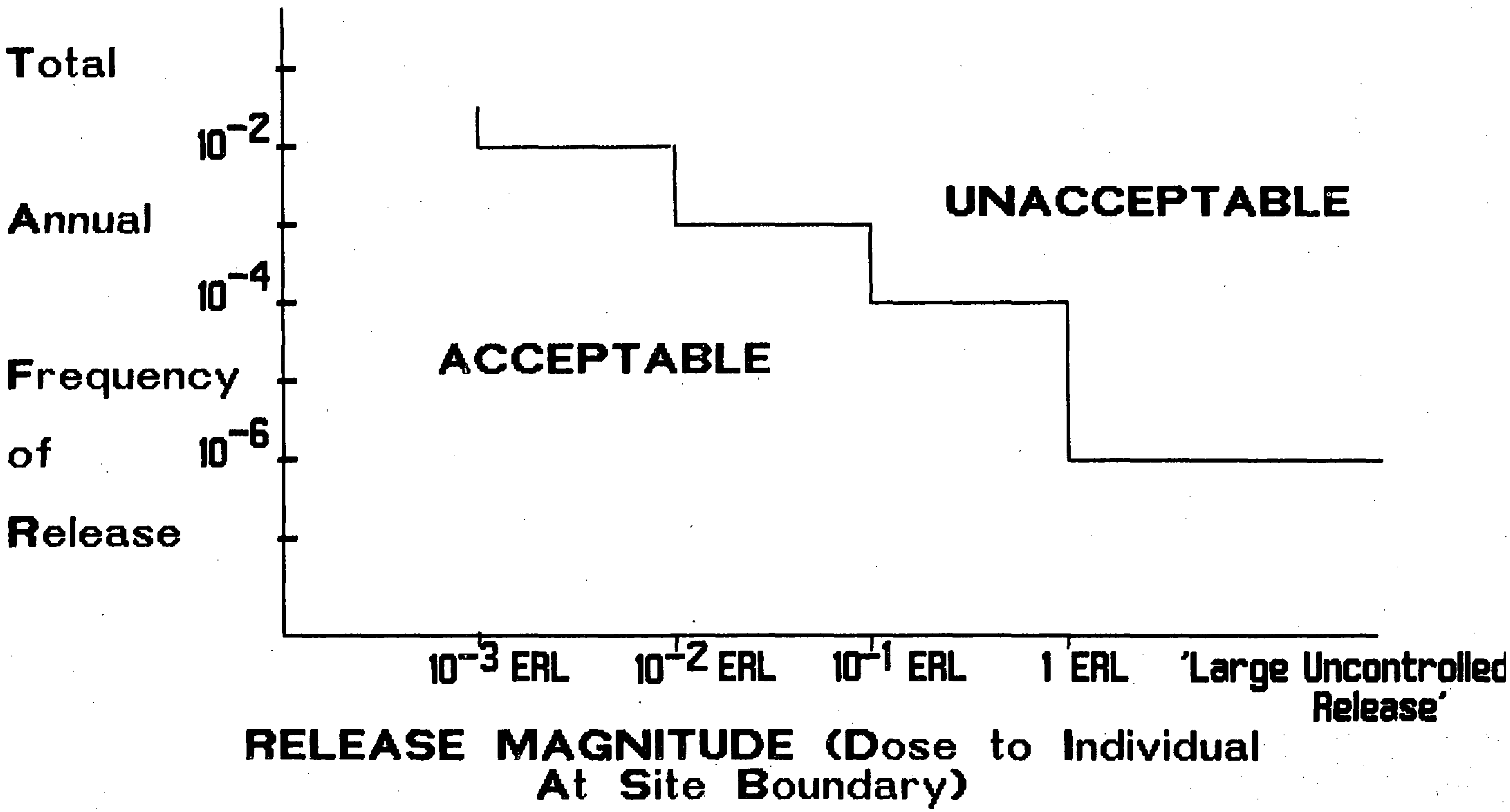

Figure 2 CEGB Design Safety Criteria

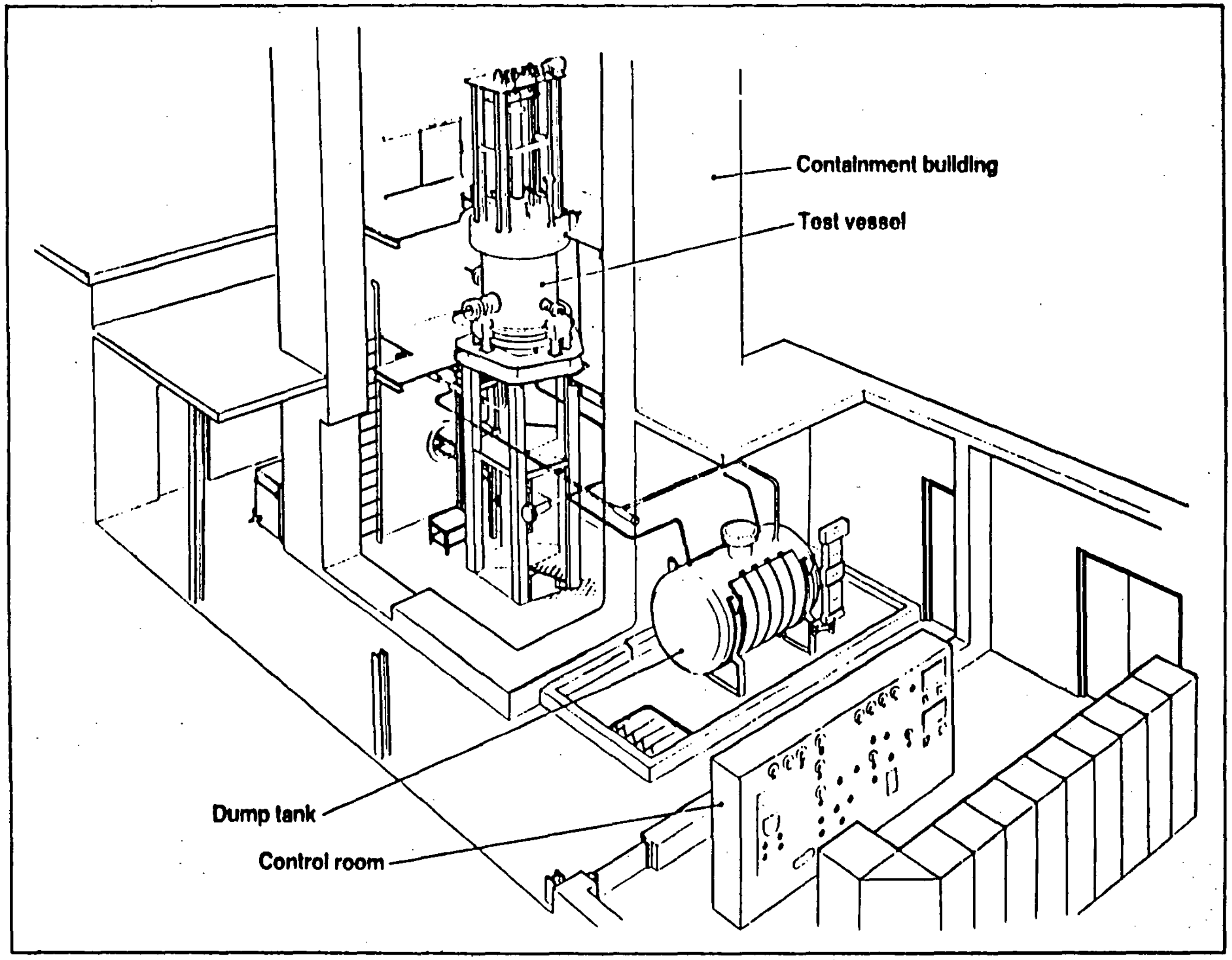

FIGURE 3   THE MOLTEN FUEL TEST FACILITY

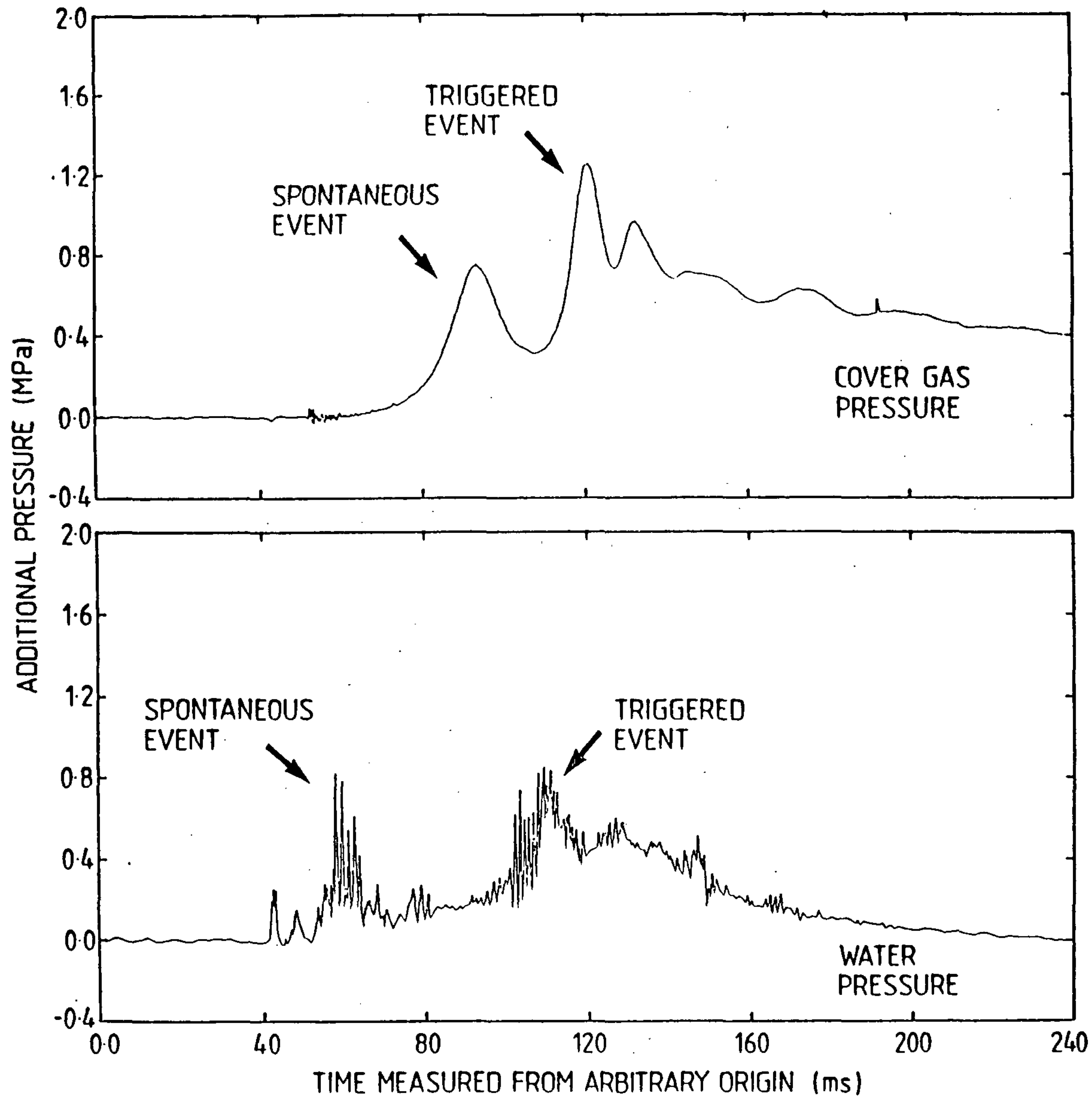

FIGURE 4 PRESSURE HISTORIES FOR SUW 05

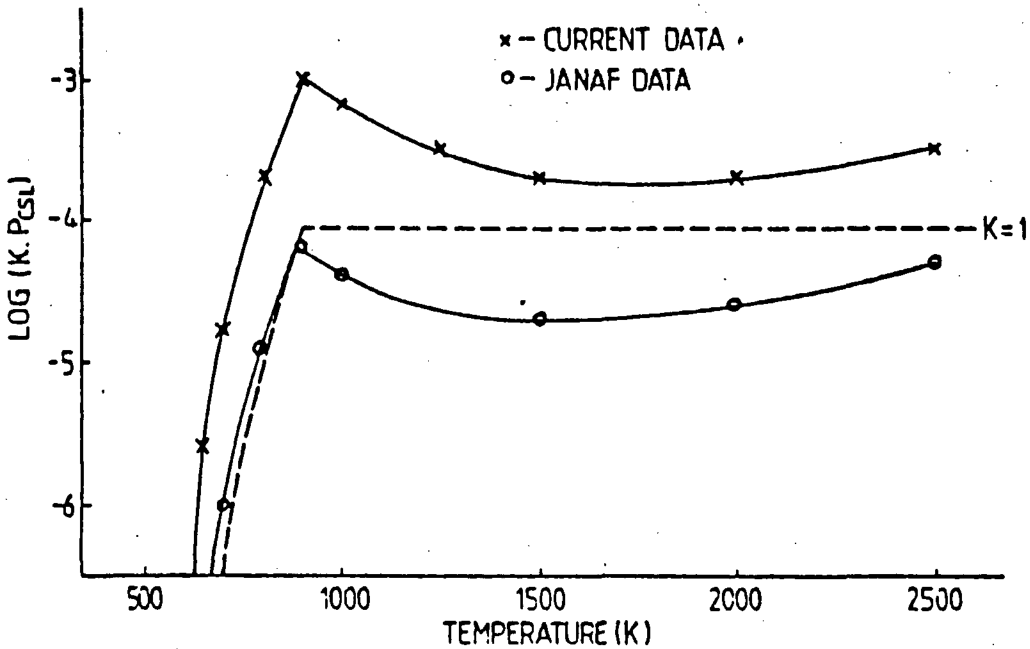

FIGURE 5 EXTENT OF PRODUCTION OF HI FROM CsI AND BORIC ACID

EFFECTIVENESS AND SAFETY ASPECTS OF SELECTED DECONTAMINATION METHODS FOR LWRs "RECONTAMINATION EXPERIENCE 1988"*

S. W. Duce
Idaho National Engineering Laboratory, EG&G Idaho Inc.
Idaho Falls, Idaho 83415-7113

This paper presents information on the recontamination of recirculation piping in commercial boiling water reactors following successive chemical decontaminations or pipe replacement. Several types of pipe pre-treatments have been used at different facilities where the recirculation pipe were replaced to reduce the rate at which radionuclides were incorporated into the oxide films on the inner pipe surfaces. These pipe treatments are briefly discussed and net contamination control effects of the treatments are compared. Net contamination control effects of successive chemical decontaminations on non-replaced pipe is also discussed.

1. INTRODUCTION

Since 1983 licensees of older operating boiling water reactors have been faced with the possibility of cracks appearing in Type 304 stainless steel recirculation system piping. These cracks were formed through a mechanism termed "inter-granular stress corrosion cracking". Crack formation was precipitated by depletion of chromium, a corrosion inhibitor, in the grain-boundary areas as it reacted with carbon to form a chromium carbide. These chromium depleted grain boundary areas were then sensitized for corrosion.[1] Two options were available to licensees for correction of the recirculation system cracks: complete recirculation system pipe replacement or checking for cracks at each refueling outage using ultrasonic testing and applying weld overlays on cracks found. Most licensees chose the complete replacement option with a few choosing to use weld overlays.

Concurrent with pipe cracking discoveries was the development and application of various chemical processes that allowed for in-situ removal of oxide films in piping systems. These processes were a boon to the industry in the area of worker dose reduction offering the possibility of removing highly contaminated oxide films that had developed on the inside of the recirculation system piping. Workers would then be working in lower general area and contact exposure rates. For complete pipe replacement, man-rem savings in the thousands of man-rem was now a reality.

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In October of 1983 the Nuclear Regulatory Commission Office of Nuclear Regulatory Research, funded this research program with the objective of obtaining information on chemical decontamination processes that they might be expected to review. During 1984 and 1985 the focus was on methods, effectiveness, and safety aspects of the chemical decontamination processes as they were applied in primary coolant recirculation systems (PCRSs) and steam generators at commercial nuclear boiling water reactors (BWRs) and pressurized water reactors (PWRs), respectively. During 1986 the emphasis was modified to also include acquiring information on changes in the oxide film radionuclide concentrations of previously decontaminated or replaced PCRSs. In 1987 the emphasis of the program focused on obtaining information on recontamination or contamination of previously decontaminated or replaced PCRSs. In 1988 the emphasis reverted back to that of 1986.

Since the institution of this program, recontamination/contamination data have been measured or otherwise obtained from ten operating BWRs. These ten facilities represent nearly every combination of conditions for which recontamination has been considered:

- Old pipe that were chemically decontaminated,
- New pipe that received no pre-treatment prior to installation,
- New pipe that were electropolished prior to installation,
- New pipe that were electropolished and pre-oxidized prior to installation,
- New pipe that were installed in facilities which have measurable elemental zinc concentrations in reactor coolant.

Table 1 lists these facilities names, pipe treatment, and other pertinent information.

2. METHODS

Recontamination data were acquired using a gamma spectral measurement system, thermoluminescent dosimeters (TLDs), and a dose rate measurement instrument. Figures 1 and 2 show the gamma spectral measurement system components. The system is comprised of an Ortec CPD-1 intrinsic germanium detector, a Davidson multichannel analyzer (Figure 1), and a tungsten shield with interchangeable collimators (Figure 2). With this system gamma spectral measurements were made on piping where dose rates were as high as 500 to 600 mR/h. Dose rates were measured with TLDs made of two lithium fluoride chips encased in a small (1.3 cm diameter by 0.5 cm thick) thin wall aluminum case, and an Eberline E-530 N with a "peanut" GM tube encased in a tungsten hemisphere. Use of the TLDs allowed for a time-weighted average dose rate to be measured while the GM tube provided a backup value in those instances where TLD data were missing.

Measurement locations were standardized as much as possible to facilitate comparison of the data. Gamma spectral measurements were made on each of four vertical sections of the twenty-eight inch diameter suction and discharge pipes. Dose rate measurements were made on each of the ten risers just below the elbows which penetrate the reactor vessel wall, the gamma spectral measurement locations on the suction and discharge piping, and on the outer radius of the elbows going into and

TABLE 1. FACILITY DATA

Reactor PCRS Summaries	
FACILITY	REPLACED PCRS PIPE PRETREATMENT
Pilgrim	Pipes were electropolished.
Monticello	Pipe fabrication was cold rolled and seam welded plate steel. Pipes were electropolished.
Peach Bottom	Pipes were electropolished.
Cooper	Pipes were electropolished and then pre-oxidized using hot moist air at approximately 560°F for 150 hours.
Hatch U2	No pipe pretreatment.
Dresden U3	Pipes were electropolished and pre-oxidized using hot moist air at approximately 560°F for 150 hours.
	NONREPLACED PCRS PIPE PRETREATMENT
Quad Cities U1	Piping was chemically decontaminated using CAN-DECON in 1984 and LOMI in 1986.
Quad Cities U2	Piping was chemically decontaminated using CAN-DECON in 1983 and LOMI in 1986.
Millstone U1	Piping was chemically decontaminated using CAN-DECON.
	NEW FACILITY
Limerick	New facility piping, only finished first fuel cycle.

TYPE OF CONDENSER MATERIAL	
FACILITY	TYPE OF METAL
Pilgrim	Titanium
Millstone U1	Copper/Nickel 70% Cu 30% Ni
Monticello	Admiralty Brass until 1984 and then replaced with stainless steel
Peach Bottom	Admiralty Brass
Hatch U2	Admiralty Brass
Limerick	Admiralty Brass
Cooper	Stainless Steel
Quad Cities U1	Stainless Steel
Quad Cities U2	Stainless Steel
Dresden U3	Stainless Steel

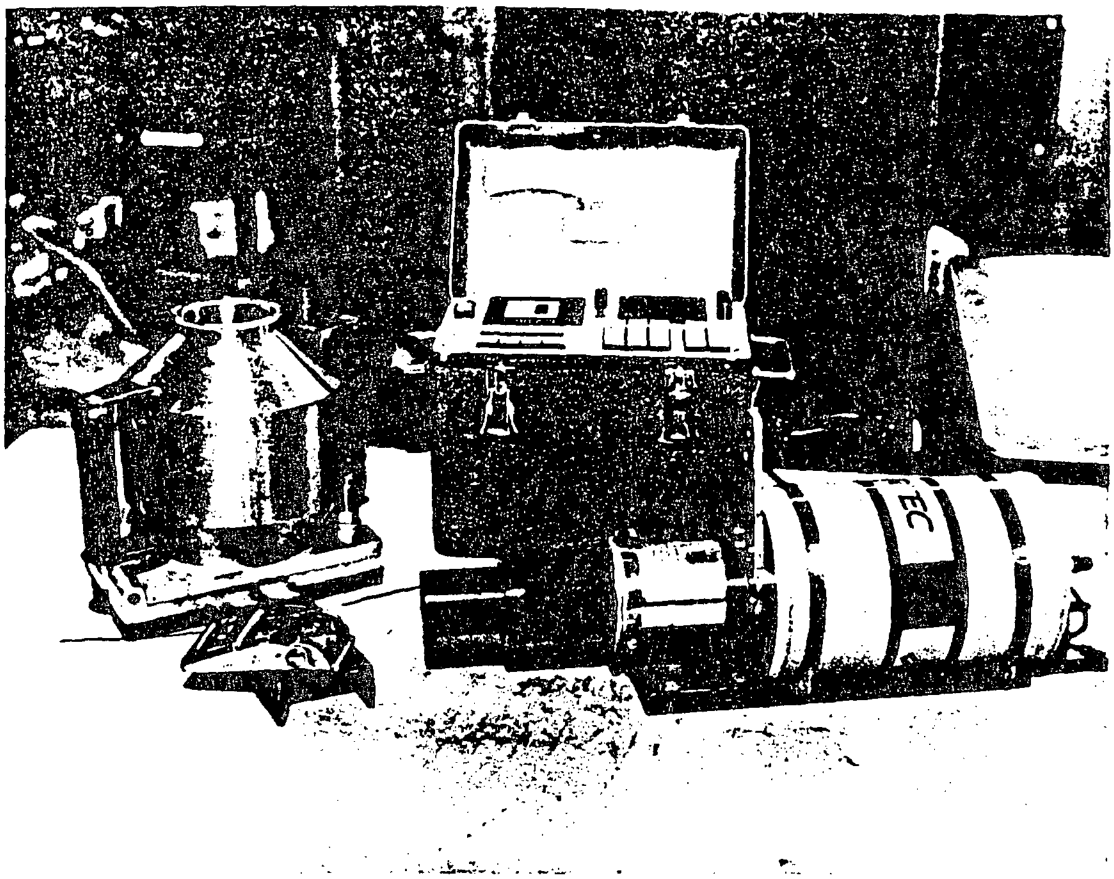

Figure 1. Gamma Spectral Equipment

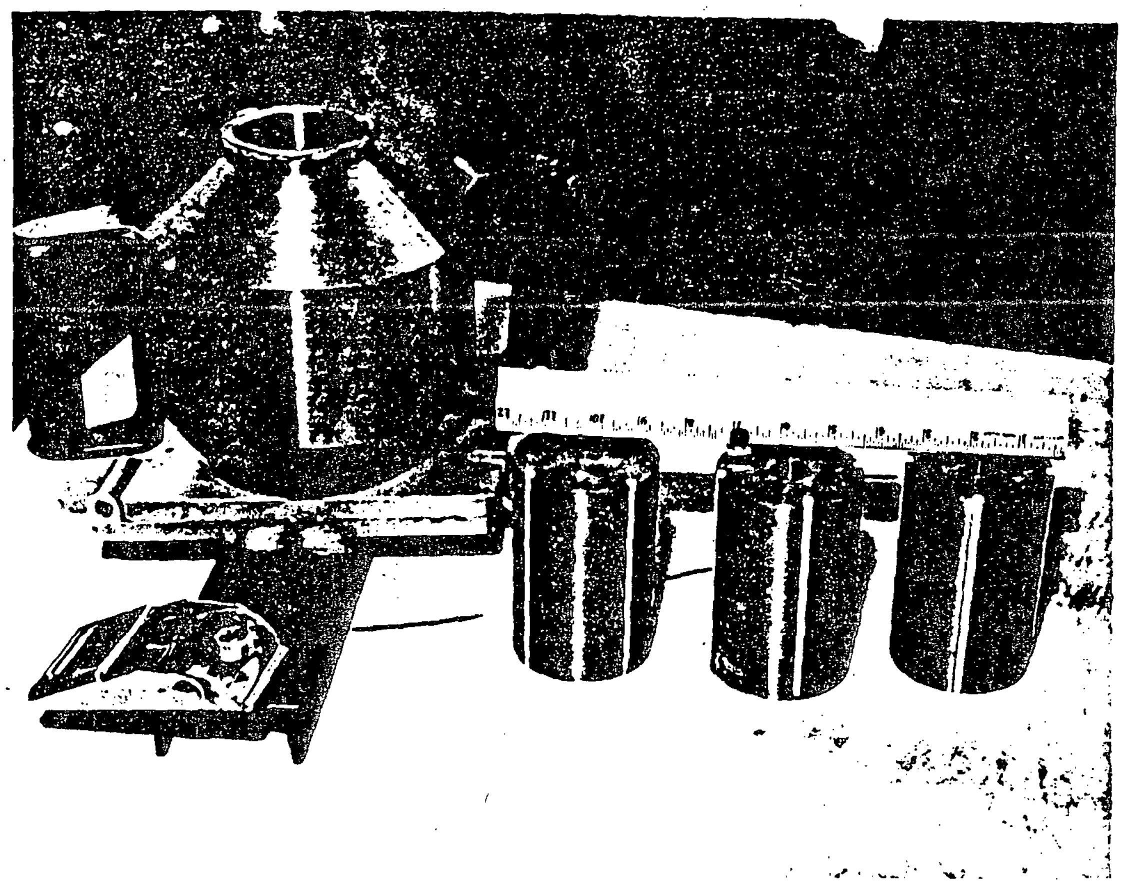

Figure 2. Sheild and Collimaters

from the two recirculation pumps. Figure 3 shows schematically the general location of all measurement points.

Plant specific fuel cycle data were also obtained at each facility. These data were: reactor power history, reactor coolant activated metals analyses, and reactor coolant conductivity analyses. These data were used in the interpretation of the measurement data.

3. DISCUSSION OF RESULTS AND CONCLUSIONS

For purposes of this presentation only dose rate and gamma spectral data for the vertical runs of the suction discharge pipes will be discussed. Limiting the discussion to these data will allow incorporation of General Electric's measured dose rate data. Prior to discussing data results and conclusions, it would be informative to include in this report two different processes that have been developed by General Electric (GE) which claim to effectively reduce the rate of recontamination in BWR recirculation piping and two pipe pre-treatments used on some facility replacement piping.

General Electric's two processes are the GE-ZIP[2] and the Hydrogen Water Chemistry.[3] Both of these processes function by the chemical addition of either zinc or hydrogen to the reactor coolant. In the GE-ZIP process zinc is continuously injected to maintain a coolant concentration of 15 ppb. Two different mechanisms have been postulated for zinc effectiveness at retarding the incorporation of Co-60 into an oxide film.[2] The first mechanism suggests that metallic ions at concentrations of $\geq$ 10 ppb would saturate or block most of the available adsorption sites for Co-60. Although Co-60 ions would still compete for these adsorption sites, the net effect would be to effectively dilute the Co-60 in the oxide film.[2] The second postulated mechanism suggests that metallic ions, like Zn^{+2} create a special oxide film structure which could significantly reduce the film growth. Regardless of exact mechanism the net effect is that the zinc ions limit the incorporation of the high dose equivalent Co-60 atoms. With hydrogen addition the corrosion rate of cobalt-bearing metals in the total reactor-turbine-condenser-feedwater system is reduced. Feedwater has been shown to be the principal source of corrosion products in BWR primary reactor coolant.[4] Therefore, hydrogen addition should limit the source of Co-59 which is activated to Co-60 in the reactor core and is available for incorporation into the oxide film.

Two different pipe pretreatments that have been used to inhibit the buildup of oxide film are moist air pre-oxidation[5] and electropolishing.[6] In the moist air pre-oxidation process a limited tightly bound oxide film is created on the inner surface of the pipe by exposure to a warm moist air environment for approximately 150 hours. This oxide film is not significantly altered following reactor operation, thereby limiting migration of ionic cobalt into the tightly bound oxide layer. This process does not affect the behavior of the particulate Co-60 which is only incorporated into the loosely bound oxide film. However, as most of the Co-60 is associated with the inner oxide layer,[7] this process appears to offer a positive mechanism for reducing of Co-60 uptake in oxide films in BWR piping. Electropolishing of pipe reduces the

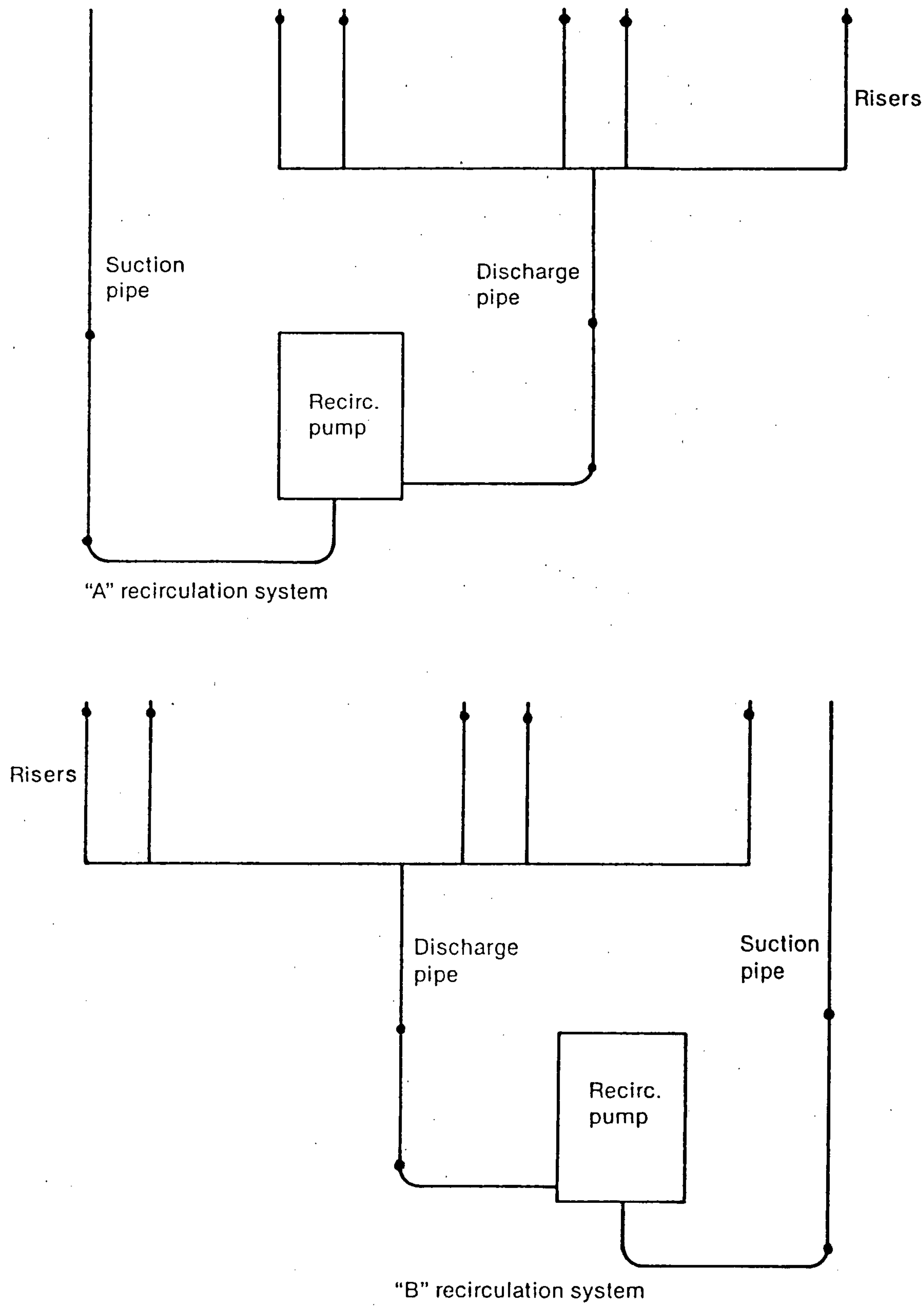

Figure 3. Recirculation System Measurement Locations

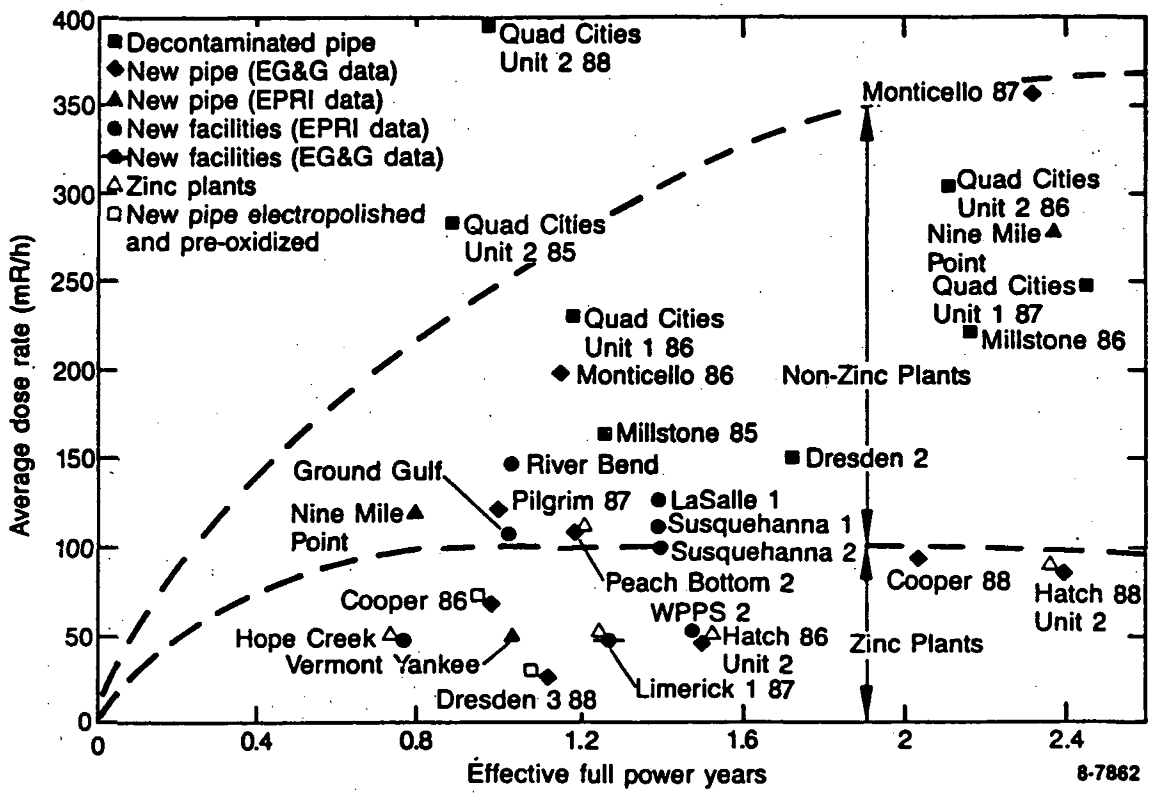

Figure 4. Average Suction/Discharge Dose Rates

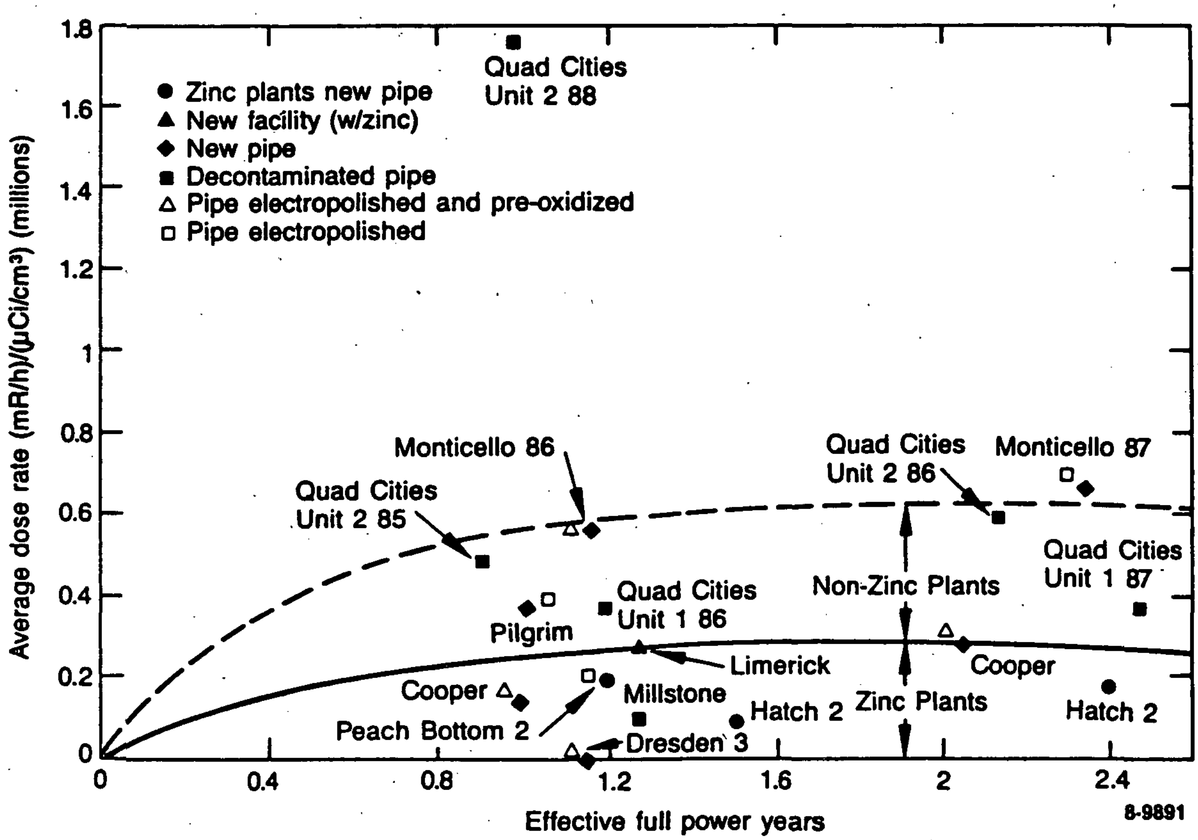

Figure 5. Normalized Average Suction/Discharge Dose Rates

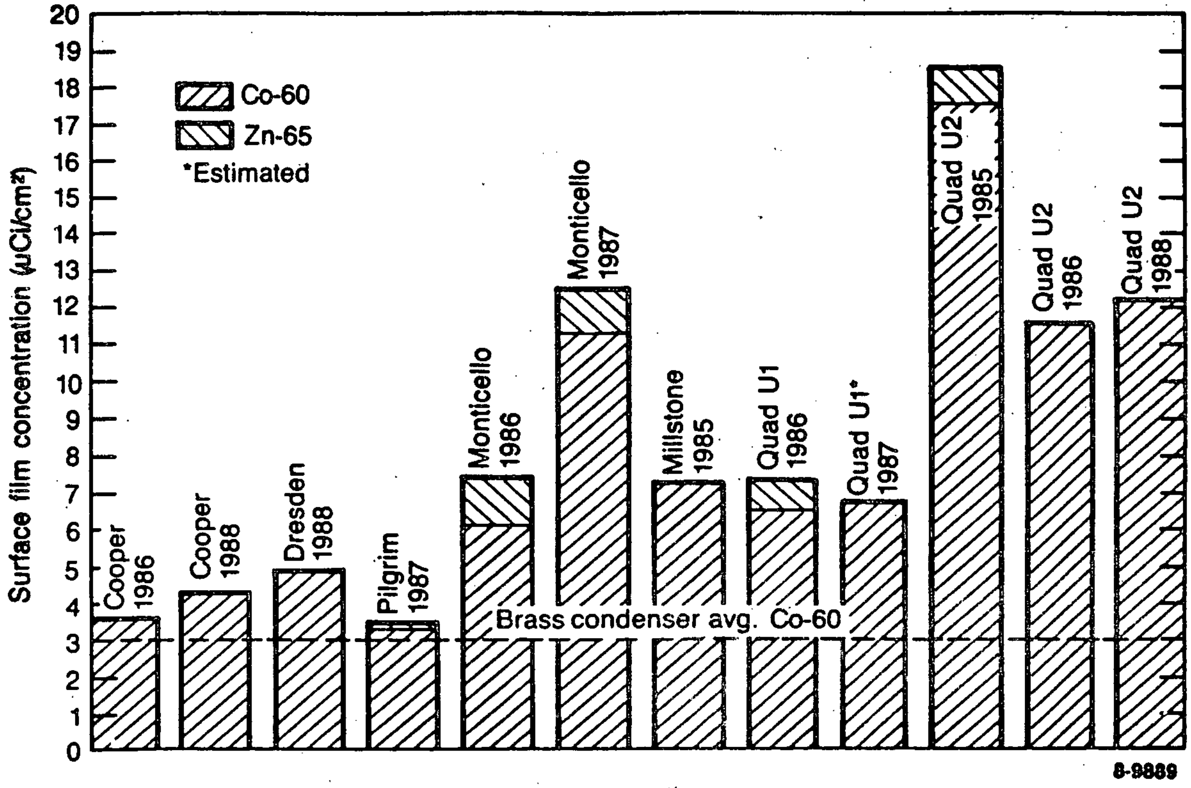

Figure 6. Co-60 and Zn-65 Surface Film Concentrations Non-Brass

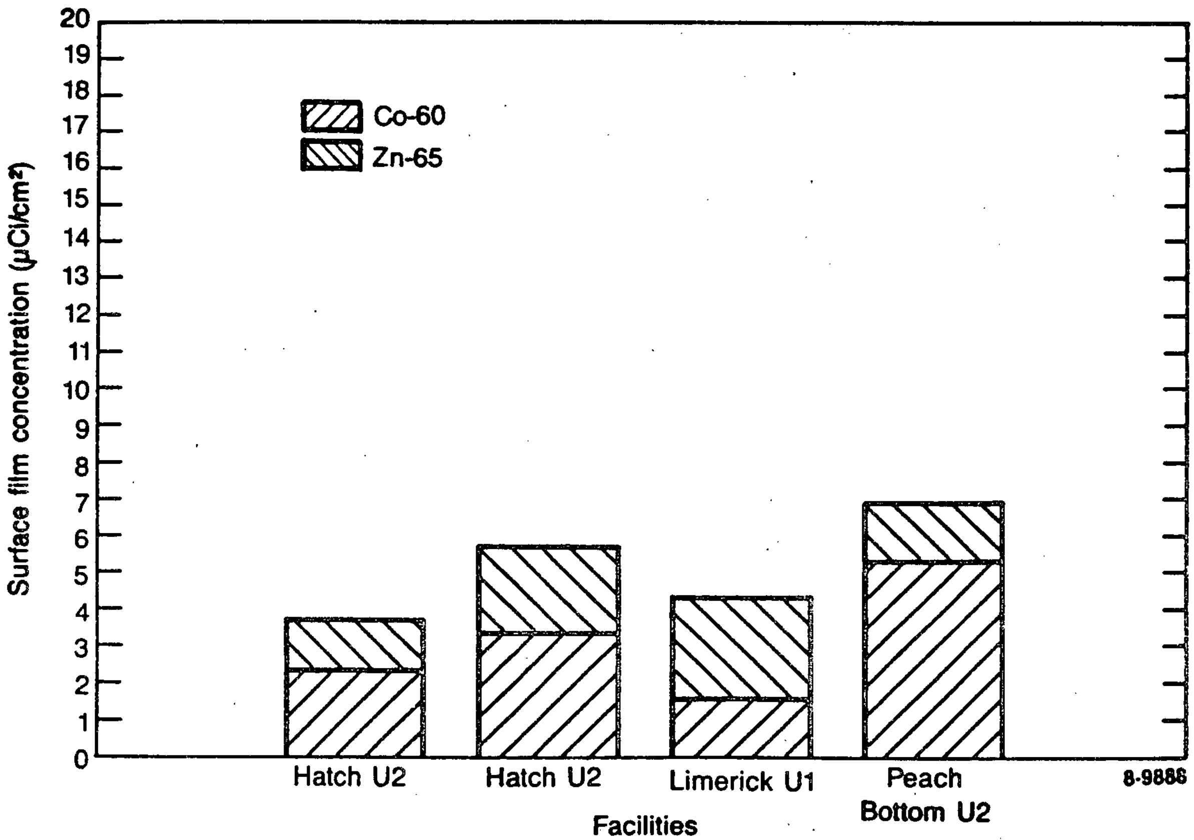

Figure 7. Co-60 and Zn-65 Surface Concentrations Brass

roughness of the pipe surface, effectively minimizing the true surface area available for formation of an oxide film and incorporation of corrosion products (Co-60).

Since electropolishing and pre-oxidation for pipe passivation have both been used and several studied facilities have "natural zinc" (refer to Table 1), it is apparent that the measured results have direct bearing on the previously discussed methods. "Natural" or elemental zinc is provided through the feedwater by erosion of admiralty brass condenser tubing, which contains zinc in trace quantities as a hardening agent, by the condensing steam.

Average dose rate data for the suction discharge pipe are shown in Figure 4. This figure also includes data from the Electric Power Research Institute (EPRI). The two dashed lines are used to indicate in general those plants that had "natural" zinc in the reactor coolant and those who did not. This figure shows that those plants which had zinc in the reactor coolant or those plants where the replacement pipe were electropolished and pre-oxidized experienced lower dose rates on their piping than the other facilities. It also shows wide variation in dose rates for new facilities following their first fuel cycle, indicating that projecting contamination performance for a new facility would be difficult. This finding agrees with C. C. Lin where he has stated, "It has become clearer that the cobalt transport process is a complex chemical reaction which can be affected by many parameters,..."[3]

Figure 5 shows only EG&G data which have been normalized to a Co-60 equivalent reactor coolant. Normalizing data in this fashion removes differences in dose rates due to the source (reactor coolant activated metals). In this figure many of the data points have changed relative positions from those in Figure 4, indicating sensitivity to the reactor coolant activated metals' concentration. Figure 5 also shows that electropolished new pipe did not perform significantly different from non-replaced pipe that were only chemically decontaminated, (see the data points for Monticello, Pilgrim, and the two Quad Cities units). Only Peach Bottom Unit 2, which was electropolished and has "natural" zinc in the coolant, showed a lower dose rate. Therefore, for these data electropolished pipe required a second passivating factor (i.e., either zinc in the coolant or pre-oxidation) to perform as well as or better than those plants which had only zinc in the coolant. These findings agree well with GE's, EPRI's, and vendors' data. In an EPRI study[6] the researchers found that electropolishing in combination with a passivation step provided an effective barrier for Co-60 incorporation into an oxide film.

Figures 6 and 7 present the net Co-60 and Zn-65 surface film concentrations for plants without brass condensers and plants with brass condensers. In Figure 6 the dashed line indicates the average brass condenser plant Co-60 surface film concentration. With the exception of Cooper, Dresden, and Pilgrim, all facilities experienced much higher concentrations of Co-60 than those facilities that had "natural" zinc in the reactor coolant. Cooper and Dresden Unit 3 electropolished and pre-oxidized the piping and Pilgrim electropolished the piping. Figure 8 shows that there may be an inverse relationship between Zn-65, which is an indicator of zinc in the coolant, and Co-60. In Figure 8 the larger the ratio of Zn-65 to Co-60 in the oxide film the lower the Co-60 surface film concentration. Those facilities in this figure were selected because no pipe surface treatments were used which would cloud the interpretation. However, if Peach Bottom Unit 2 data, where the pipes were electropolished, were included they would fit perfectly between the second Hatch Unit 2 and Quad Cities Unit 1 data with values of 3.1 and 5.3 for the ratio and Co-60 concentration respectively.

In Figure 9 the Co-60 surface film concentrations have been normalized to their respective reactor coolant concentrations. Normalizing again eliminates the effect of the reactor coolant concentration on the surface film concentration. There are four features that are of interest in this figure when compared to Figures 6 and 7: (a) the change between the Cooper 1986 and 1988 data is larger than seen in Figure 7, (b) Dresden data show a much lower surface film concentration than seen in Figure 7, (c) the relation of the two Monticello data points is reversed in Figure 9 with the 1987 dat being lower, and (d) Quad Cities Unit 2 1988 surface film concentrations are higher in Figure 9.

Figure 10 shows the Co-60 normalized recontamination rates for all facilities. Quad Cities Unit 2 shows the highest recontamination rate for all non-replaced pipe facilities. It is interesting to note the difference between its sister plant. C. C. Lin's statement is again verified; it is difficult to understand the Co-60 deposition factors. Those plants having either "natural" zinc or electropolished and pre-oxidized pipe experienced lower recontamination rates during the first fuel cycle data and the second fuel cycle data. For those facilities that replaced their piping the recontamination rates for the second fuel cycle is lower than the first fuel cycle by 25.5%, 50.8%, and 15.9% for Hatch Unit 2, Cooper, and Monticello, respectively. This indicates that the Co-60 film is rapidly coming to an equilibrium.

4. SUMMARY

In summary these data indicate that Co-60 incorporation into an oxide film is:

- o Influenced negatively by the presence of "natural" zinc in the reactor coolant.
- o Effectively reduced by electropolishing and pre-oxidizing the exposed metal surface.
- o Reduced by electropolishing the exposed metal surface in conjunction with replacement of the pipe in a "natural" zinc facility.
- o Not effectively reduced by only electropolishing the exposed metal surface.

These findings agree well with the current thinking within the industry concerning minimization of Co-60 incorporation into oxide films in operating BWRs.

In the future more recontamination data/measurements will be obtained following second and third fuel cycles at many of the same facilities studied. Recontamination measurements will be made at Millstone Unit 1 which has been using GE's GE-ZIP process for one fuel cycle. Recontamination measurements will be made at Dresden Unit 2 which has been operating with hydrogen water chemistry for one fuel cycle. The recontamination database will be expanded to include all techniques currently marketed to reduce incorporation of Co-60 into the oxide films.

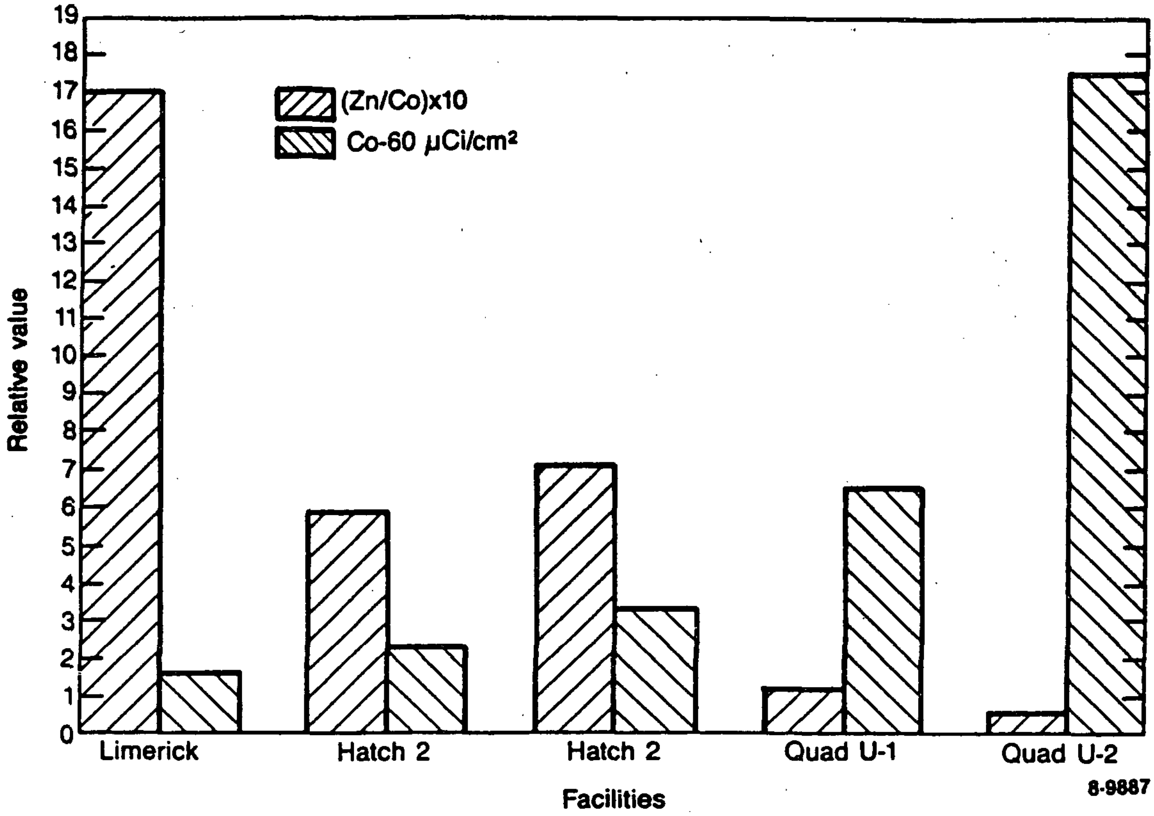

Figure 8. Ratio of Zn to Co -vs- Co-60 Surface Concentration

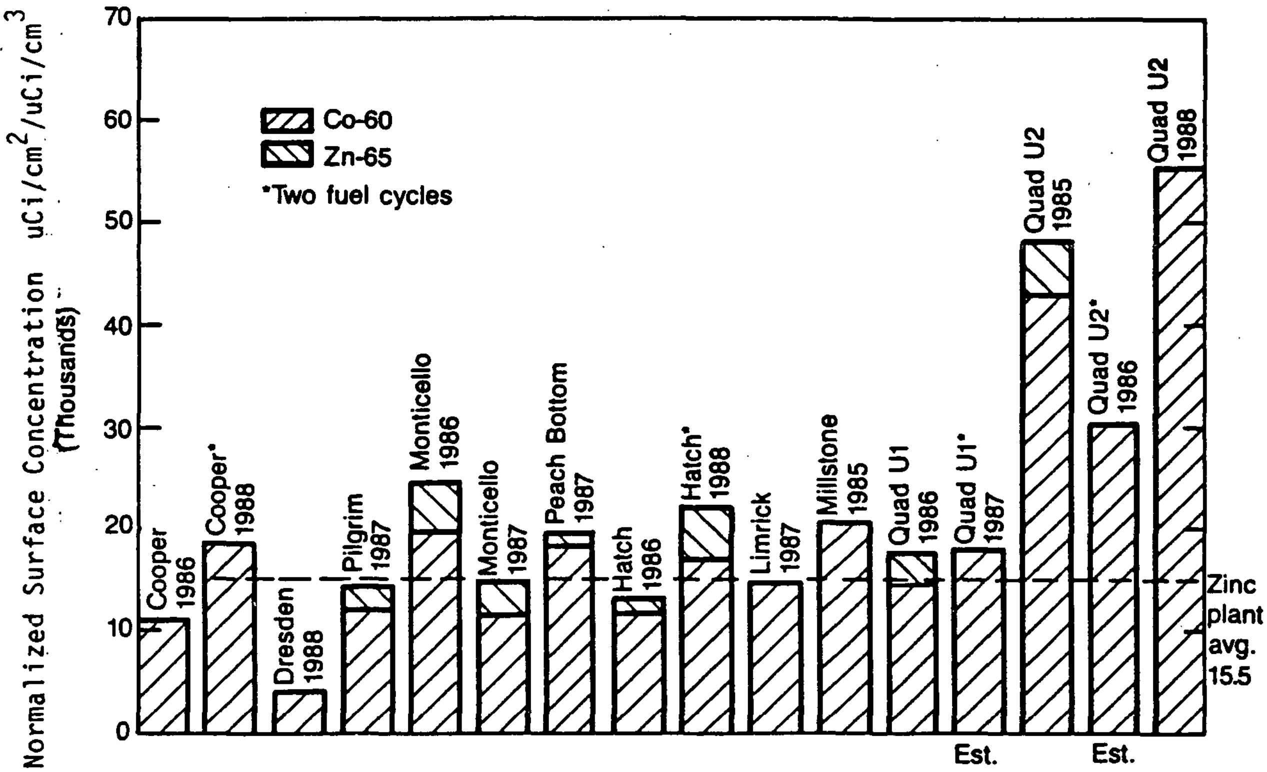

Figure 9. Normalized Co-60 Surface Concentrations

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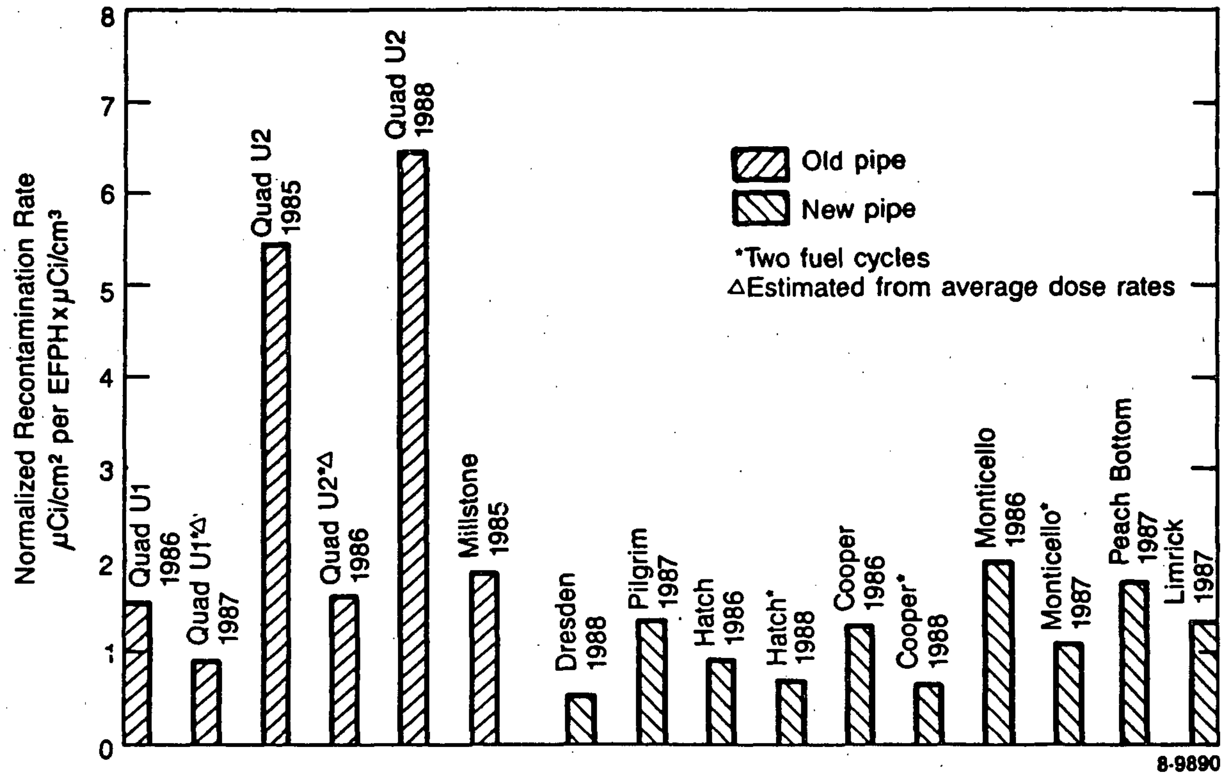

Figure 10. Normalized Recontamination Rates For All Facilities

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Decontamination Impacts on Solidification and Waste Disposal

C. R. KEMPF AND P. SOO
BROOKHAVEN NATIONAL LABORATORY
NUCLEAR WASTE AND MATERIALS TECHNOLOGY DIVISION
UPTON, NEW YORK, 11973

ABSTRACT

Research to determine chemical and physical conditions which could lead to thermal excursions, gas generation, and/or general degradation of decontamination-reagent-loaded resins has shown that IRN-78, IONAC A-365, and IRN-77 organic ion exchange resin moisture contents vary significantly depending on the counter ion "loading." For these resins the EDTA, picolinic acid and Fe^{+2} "loaded" forms, respectively, had moisture contents lower than the regenerated, OH^- and H^+ "loaded" forms. Heat- and gas-generating reactions have occurred with two anion resins used, IRN-78 and IONAC A-365; color changes and precipitates were also observed. The resins were originally in the OH^- form and potassium permanganate and nitric acid were oxidizing solutions used to produce the reactions. The extent/vigor of the reaction is very highly dependent on the degree of dewatering of the resins and (probably linked to this) on the method of solution addition (dropwise or in bulk). The heat generation may be due, in part, to the heat of neutralization (acid addition to hydroxide-form resins) [Brumfield and Kempf, 1988]. Ferrous ion loaded cation resins (IRN-77) showed little reactivity toward nitric acid and potassium permanganate. In studies of the long-term compatibility effects of decontamination waste resins in contact with waste package container materials in the presence of decontamination reagents, radiolysis products and gamma irradiation, it has been found that the corrosion of carbon steel and austenitic stainless steel in mixed bed resins is enhanced by gamma irradiation. However, cracking in high density polyethylene is essentially eliminated because of the rapid removal of oxygen from the environment by gamma-induced oxidation of the large resin mass. Ferralium-255 and TiCode-12 are not attacked even for gamma doses up to 10^8 rad.

INTRODUCTION AND BACKGROUND

During operation of light-water reactors (LWRs), corrosion of metallic components in the primary system occurs. Corrosion products are circulated through the system by the coolant, and some become radioactive as a result of neutron activation in the core. After years of operation, deposition of the corrosion products within the primary system leads to a steady increase in radiation levels. This, in turn, causes increasing difficulty during routine maintenance of the plant because of worker exposure to radiation.

Some reduction of exposure to plant personnel has been or may be achieved through the use of conventional radiation protection measures (exposure time reduction, shielding, or employment of remote-operation methods). However, the application of these traditional methods has not halted a trend of generally increasing radiation exposure to plant personnel. Accumulating evidence indicates that the design lifetime of LWRs may not be obtained if additional action is not taken to minimize the radiation exposure incurred during operation, inspection, and maintenance. Chemical decontamination of the complete primary systems in LWRs has been implemented as a practical and effective means of reducing the radioactivity levels in the system. The Advisory Committee on Reactor Safeguards has noted that, "...continued aging of U.S. nuclear power plants makes it likely that the volumes of LLW from decontamination and decommissioning activities will increase..." (letter from W. Kerr to L. Zech, November 10, 1987).

Decontamination reagent-protocols have been developed for application to corrosion products/oxides in both the oxidizing-chemistry environment of boiling water reactors (BWRs) and the reducing-chemistry environment of pressurized water reactors (PWRs). In general, oxides in BWRs tend to be low in chromium and can be dissolved in organic acid/chelating agent mixtures such as citric acid, oxalic acid (citrox) and EDTA (ethylenediaminetetraacetic acid). These mixtures can be mildly reducing. A more strongly reducing decontamination process is the LOMI (low oxidation state metal ion) system. The LOMI reagents act first to "soften" the oxide coating [through electron transfer from vanadium (II) ion complexed with picolinate/formate to iron (III) in the oxide, reducing it to iron (II)] and then to dissolve it (through the complexing action of picolinic acid) [Wood, 1985].

For PWRs in which the corrosion products tend to be chromium-rich oxides, the best decontamination results have been obtained in those processes which are preceded by an oxidizing stage to convert Cr(III) to Cr(VI), thereby inducing its release to solution. The alkaline permanganate followed by ammonium citrate (APAC) procedure is one such pre-oxidation stage process. Another process of this type involves potassium permanganate in nitric acid solution. This process involves an added step of controlled destruction of surplus reagent (permanganate) by a reducing agent (oxalic acid) [Pick, 1982].

These different processes will generate characteristically different waste types and volumes of radioactive wastes. All of these processes involve the use of complexing agents because they form selective and strong water-soluble complexes with corrosion products.

The decontamination solutions are flushed through anion and cation exchange resin beds after application in the reactor coolant system. This process is carried out to remove excess decontamination reagents (chelating/complexing agents) as well as non-radioactive and radioactive ions/complexes. Species that could be expected in spent decontamination solutions include the cations Mn^{+2}, K^{+}, Cr^{+3}, Cr^{+6}, Fe^{+2}, Fe^{+3}, Ni^{+2}, Co^{+2}; and the anions NO_3^{-}, citrate, oxalate, picolinate, formate, EDTA. Depending on pH conditions, the metal complexes (metal ion plus complexing/chelating agent) could be cationic, anionic or neutral. This is a consequence of the multiple electron-donating groups on various complexing agents. These factors make decontamination waste a unique and complicated type of low-level waste.

The overview of nuclear plant decontaminations written by Wood [Wood, 1986] shows that the majority were on BWR systems. The trend observable from that report was seen to be dominated by London Nuclear/CAN-DECON processes in 1983, followed by a transition period in 1984 in which NS-1, LOMI, CITROX and CAN-DECON were all used. The most recent trend has been toward decontamination with dilute chemicals, CITROX and LOMI processes being dominant.

Several incidents in the recent past have called into question the safety and acceptability of decontamination/resin waste processing. The implications of the problems manifested in these incidents extend beyond just the in-plant environment to the performance of such wastes at the disposal site. Three events involved exothermic reaction and/or significant pressurization of resin or filter media wastes during dewatering or after placement in waste containers/liners. Work has been initiated at BNL (Task 1) to determine chemical and physical conditions which could lead to thermal excursions, gas generation, and/or general degradation of decontamination-reagent-loaded resins. Further, studies are ongoing to study the long-term compatibility effects of simulated decontamination waste resins in contact with waste package container materials in the presence of decontamination reagents, radiolysis products and gamma irradiation (Task 2).

RESULTS AND DISCUSSION

Task 1: Evaluation of Chemical and Physical Degradation in Decontamination Wastes

The purpose of this task is to determine chemical and physical conditions which could lead to thermal excursions, gas generation, and/or general degradation of waste ion-exchange resins used for clean-up at nuclear power plants. This task was initiated as a consequence of concern about three anomalous incidents. In particular these were: a thermal excursion in resins undergoing dewatering at Arkansas Nuclear One (sufficient heat was produced to bring the temperature of the wastes to at least 365°F); and two gas generation/pressurization events in resin wastes undergoing transportation from Millstone Nuclear Station and from the James A. Fitzpatrick Nuclear Power Plant (gas pressures in the wastes were sufficient to result in the lifting of the lid of the high integrity container shipping cask in both cases). In all three cases, resin wastes were involved and the dewatering process had been (or, in the case of the thermal excursion wastes, was in the process of being) performed. The resin wastes were quite heterogeneous and had not been thoroughly characterized. The specific causes of these events were not identified.

A review of the events was performed (Bowerman and Piciulo, 1986) and several possible contributing processes and/or factors were suggested based on the minimal analytical information available from the waste generators and on a literature review of chemical and physical reactions or changes which ion-exchange resins may be subject to, which could lead to heat and/or gas generation. This information forms the basis of the current research effort.

In particular, for the right conditions, radiation-induced reactions, biodegradation processes, and oxidation reactions may lead to heat and gas generation as well as to nongaseous chemical products. Oxidation reactions can occur between resin materials and a number of other chemicals including halogens, dichromate, permanganate, or nitric acid (vendors of these resins specify that exposure to these chemicals should not occur). These chemicals or others potentially reactive with resin materials may be present in resin wastes either as components on the resin or as products of radiolysis, biodegradation, or other chemical reaction occurring at some stage in the waste resin generation lifetime. Explosive oxidation reactions have occurred between resin materials and concentrated nitrates/nitrites (Miles, 1968).

This work is being carried out to provide information to allow determination of whether such events could happen in the future, either during storage or processing at the plant, during transportation or at the final disposal site. The plan for this task has involved setting up a simplified experimental system in which heat and/or gas generation as well as color changes, precipitates or other signs of chemical reaction can be observed.

Specifically, IRN-78 and IONAC A-365 anion and IRN-77 cation resin batches were regenerated and their moisture contents in regenerated form were determined. Then, these resins were "loaded" with typical reagents or species that would be expected to be caught on the resins from a decontamination campaign; for anion resins, picolinic acid and ethylenediaminetetraacetic acid (EDTA) were used, while for cation resins, ferrous ions were used. The equilibrium moisture contents of these loaded resin forms were also determined.

Once batches of regenerated and decontamination reagent-loaded (or, in the case of the cation resins, metal ion-loaded) resins had been prepared, they were subjected to addition of oxidizing chemicals, in particular nitric acid and potassium permanganate solution. These additions were carried out in several ways: (1) in small increments coupled with monitoring of changes in pH of the resin slurry to allow observation of the exchange with nitrate and with permanganate for the regenerated form of the anion resins; (2) dropwise and in bulk to allow observation of the effect of oxidizing agent amount and also of the heat generation and absorbance taking place in the resin slurry; and (3) with intermittent dewatering by vacuum aspiration between additions of nitric acid and potassium permanganate to simulate the dewatering which was known to have occurred in the heat and gas generating incidents described earlier. The results of these procedures are given in the following sections.

Regeneration, Reagent Loading, and Moisture Content Determinations [Brumfield and Kempf, 1988; Kempf, et al, 1988]

The moisture content of the regenerated resins (hydroxide and hydrogen ion forms for anion and cation resins, respectively) and the "loaded" resins was taken for each resin type as the difference in weight between the de-watered (vacuum-aspirated) state and the oven-dried state. Figure 1 shows the results of this determination for the regenerated and "loaded" resins. The average moisture content for IRN-78/OH^- resins was 69.9%; for IRN-77/H^+, 55.9%, and for IONAC A-365/OH^- resins, 46.5%. These results are in agreement with those reported by Rohm and Haas (Siskind, 1987). The average moisture content for IRN-78 resins loaded with EDTA was 47.0%; for IRN-77 resins loaded with Fe^{+2}, 47.7%, and for IONAC A-365 resins loaded with picolinic acid, 32.8%.

A comparison of values in Figure 1 shows that, for IRN-77, the moisture content of the H^+ form is ~8% higher than that for the Fe^{+2}-loaded form. The +2 charge on the iron means that only one-half as many ions (Fe^{+2}) may occupy the fixed ionic sites of the resin as compared to the +1 charge on the hydrogen ion. This may lead to a decrease in total associated "hydration" moisture attached to the Fe^{+2} versus that attached to H^+.

The anion resin, IRN-78, exhibited a moisture content of 69.9% for the OH^- form versus 47% for the EDTA form. The EDTA molecule is considerably larger than the hydroxide ion. It is also capable of existing in a number of ionic states, +2 to -4, depending on the pH (Peters, et al, 1974). Around neutral pH, the principal forms of EDTA are the -2 and -3 states. Compared to hydroxide ion (whose charge is -1), two or three times as many fixed ion sites could be occupied by EDTA as by hydroxide ion. There would thus be expected to be less total associated "hydration" water with a lower net counter ion population.

Similar results occurred for the IONAC A-365 resins loaded with picolinic acid. The picolinic acid group is expected to have a -1 charge identical to hydroxide ion, however, it is a much larger molecule and may therefore allow accommodation of less associated water in the resin structure than hydroxide ion.

Picolinic Acid Loading of Resins

The picolinic acid decontamination reagent loading of the IRN-78 and IONAC A-365 resins used for this task has been studied in detail because picolinic acid itself has a very low dissociation constant, ~10^{-6}. Under conditions such as these, achieving even a 50% resin loading would require a tremendous amount of picolinic acid solution unless, as is theoretically expected, the uptake of picolinate by the resins drives the picolinic acid equilibrium toward dissociation. The process, as it is thought to occur, is summarized below:

(A) $\text{Picolinic Acid}_{(aq)} \longleftrightarrow \text{Picolinate}_{(aq)} + H^+_{(aq)} \quad K \sim 10^{-6}$

(B) $\text{Resin-[OH]} + \text{Picolinate}_{(aq)} \longleftrightarrow \text{Resin [Picolinate]} + OH^- \quad K=?$

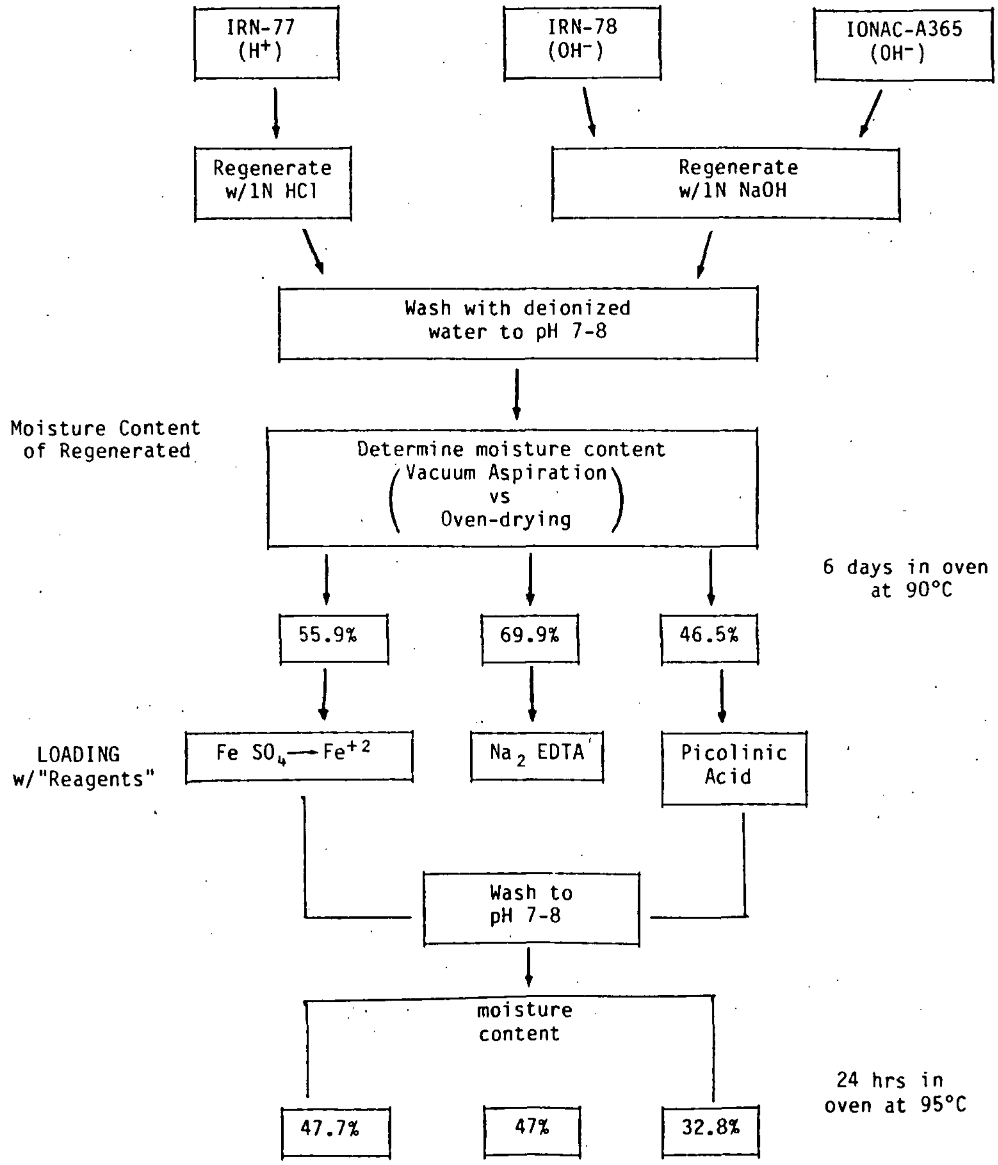


Figure 1 Flowchart of Resin Regeneration, Moisture Content Determinations and EDTA, Picolinic Acid and Fe^{+2} Loading.

The cycle (A)(B)(A)(B) proceeds until the resins have taken up as much picolinate as they can; in the process, the picolinic acid dissociates significantly.

Ten-gram regenerated, vacuum-aspirated samples were taken of IRN-78 and IONAC A-365 resins. These were equilibrated with two different concentrations of picolinic acid, one corresponding to a theoretical 100% loading of the 10-gram sample and the other corresponding to a theoretical 50% loading of the 10-gram sample used. The theoretical loadings were calculated based on reported exchange capacities of 1.76 meq/gram of IRN-78 and 5.28 meq/gram of IONAC A-365. Extents of picolinate loading were determined through (spectroscopic) measurement of picolinate remaining in the supernatant above the 10-gram resin samples after equilibration for 1, 2, and 9 days with and without stirring. The longer the equilibration time, the more picolinate was loaded on the resins and the lower the moles of picolinic acid remaining in the supernatant.

Table 1 provides a summary of the extent of loading of picolinate that can be achieved on IRN-78 and IONAC A-365 resins.

Table 1 Uptake of Picolinate for IRN-78 and IONAC A-365 Resins Equilibrated with Picolinic Acid Solution

Type Resin	Theoretical Loading %	Average Supernatant Picolinic Acid (Moles)	Total Picolinic Acid Added (Moles)	% PA Loaded on Resins Following a 9 day Equil. Period
IRN-78	50	3.16×10^{-5}	8.77×10^{-3}	99.6
	100	2.96×10^{-3}	1.76×10^{-2}	83.2
IONAC A-365	50	1.41×10^{-3}	2.64×10^{-2}	94.7
	100	1.67×10^{-2}	5.28×10^{-2}	68.4

These results show that 50% loading may be accomplished to 99.6% and 94.7% completion and 100% loading can only be achieved to 83.2% and 68.4% for IRN-78 and IONAC A-365 resins, respectively.

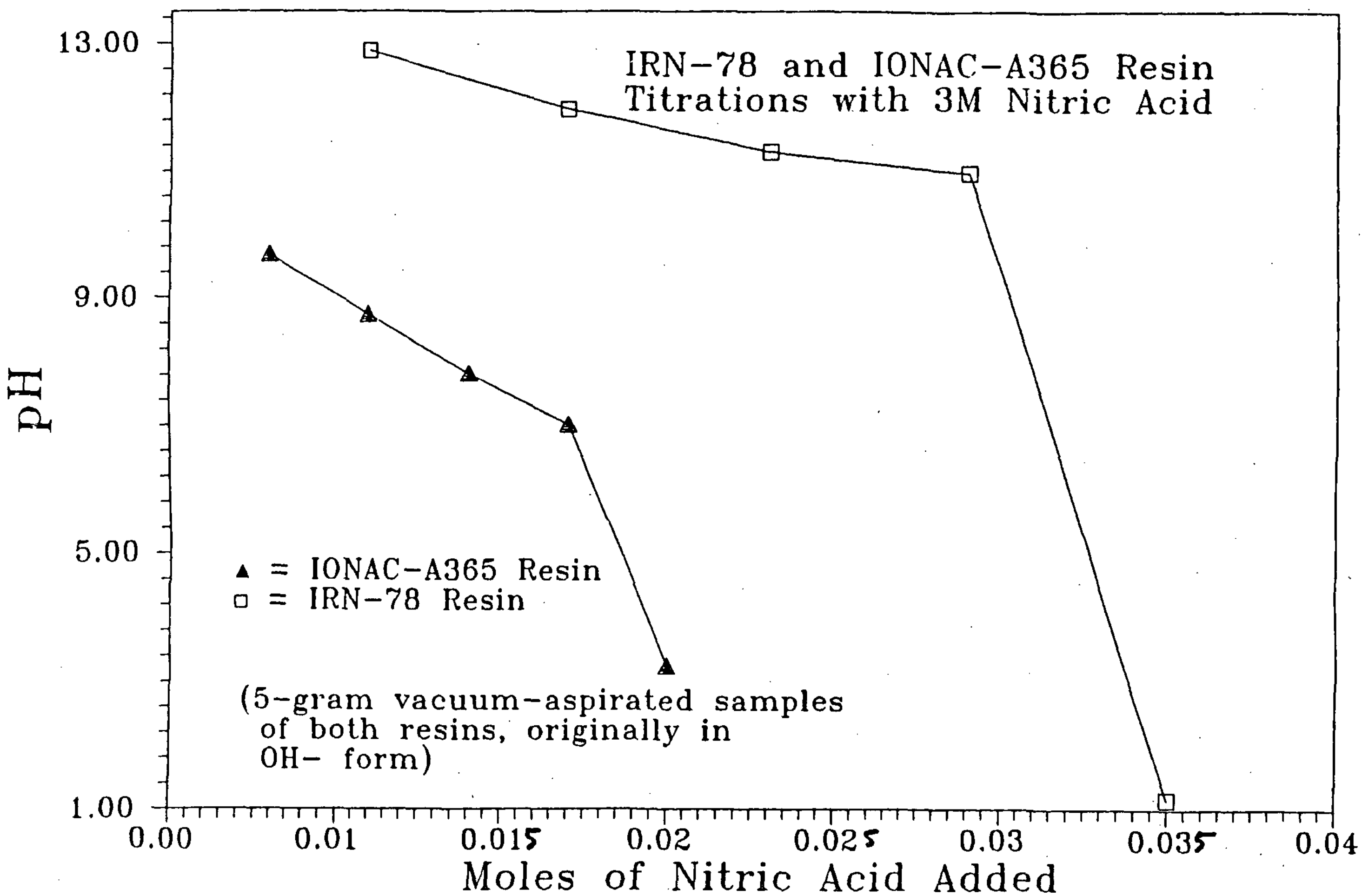


Figure 2 IRN-78 and IONAC A-365 Resin Titrations with Nitric Acid.

Exchange with Nitric Acid and with Potassium Permanganate

IRN-78 and IONAC A-365 anion resins were equilibrated with small increments of nitric acid while the pH was being monitored. The resins were originally in the OH^- form. When nitric acid was added, they exchanged their OH^- ions for NO_3^- ions. The liberated OH^- ions were neutralized by the H^+ from the nitric acid. These reactions may be described by the following three equations:

$$HNO_{3(aq)} \leftrightarrow H^+_{(aq)} + NO_3^-{}_{(aq)} \tag{1}$$

$$OH^-{}_{(resin)} + NO_3^-{}_{(aq)} \leftrightarrow NO_3^-{}_{(resin)} + OH^-{}_{(aq)} \tag{2}$$

$$OH^-{}_{(aq)} + H^+{}_{(aq)} \leftrightarrow H_2O \tag{3}$$

IONAC A-365 and IRN-78 resins titrated with 3M HNO_3 produced the titration curves given in Figure 2. These are similar in shape but different in relative position. The NO_3^- ions were taken up by both the IRN-78 and the IONAC A-365 resins, while the OH^- ions were being released. At the same time, the OH^- ions were being neutralized by the H^+ ion of the nitric acid, thus decreasing the pH of both resins. The shift of the titration curve of the IRN-78 resin to the right indicates that the IRN-78 resins are capable of taking on nitrate ion more readily than the IONAC A-365 resins. The initial pH of the IRN-78 resins was ~13. This would indicate that compared to the IRN-78, the IONAC A-365 resins were somewhat hesitant about giving up their OH^- ions; the IONAC A-365 initial pH was about 9.

IRN-78 and IONAC A-365 resins were also titrated with 0.04M potassium permanganate (the initial pH of the permanganate solution was 6.6).

The results of this experiment are given in Figure 3. The resins were originally in the OH^- form. When permanganate ions were added, the resins exchanged their OH^- ions for MnO_4^- ions. The characteristic purple color of permanganate disappeared as the resins exchanged OH^- for MnO_4^-. When the purple color of MnO_4^- persisted for several minutes, it was assumed that the maximum amount of permanganate had been taken up by the resins and thus, the permanganate remained in the supernatant layer. These reactions may be described by the following expressions:

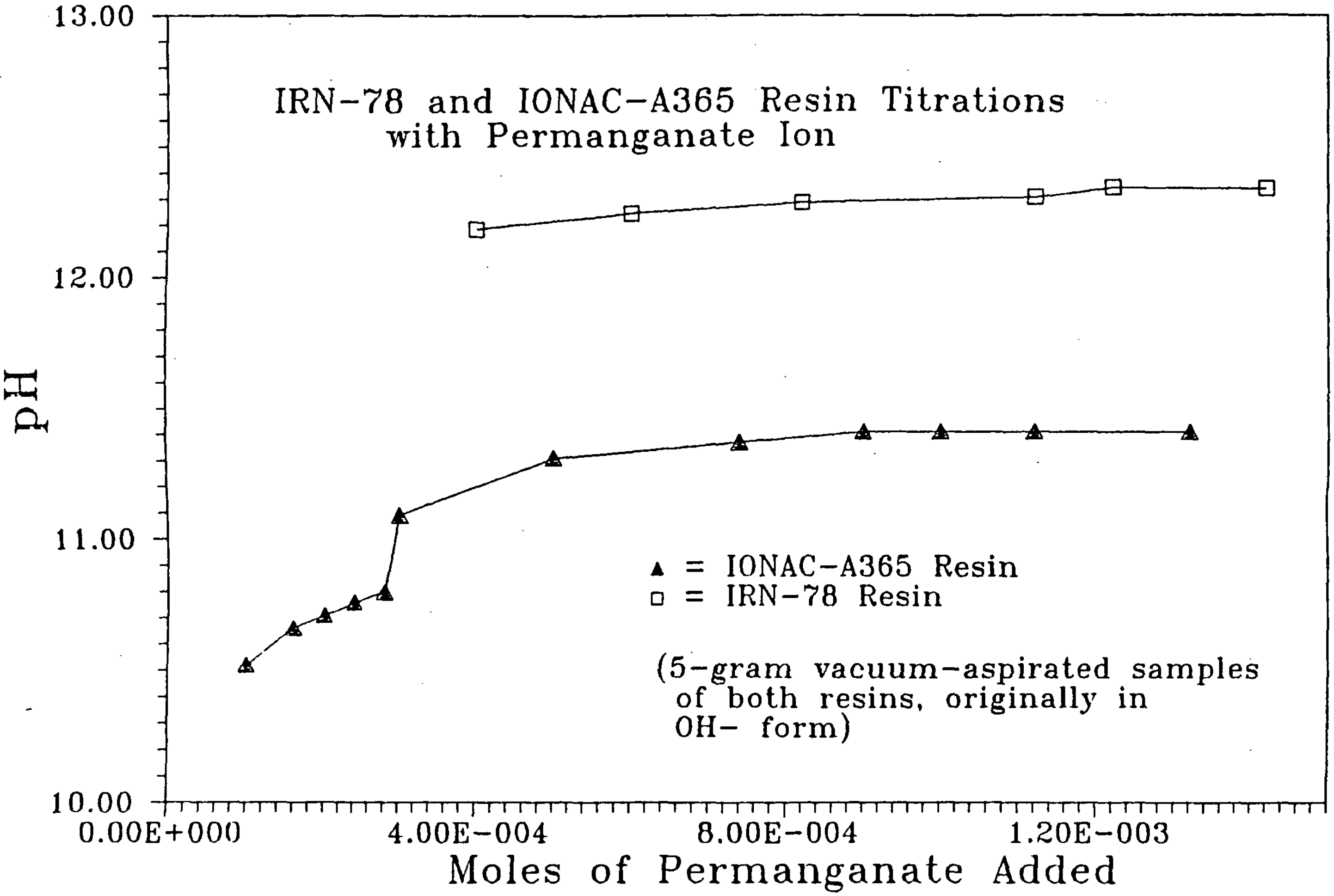


Figure 3 IRN-78 and IONAC A-365 Resin Titrations with Permanganate Ion.

$$KMnO_4 \rightarrow K^+_{(aq)} + MnO_4^-{}_{(aq)} \quad (4)$$

$$OH^-_{(resin)} + MnO_4^-{}_{(aq)} \leftrightarrow OH^-_{(aq)} + MnO_4^-{}_{(resin)} \quad (5)$$

The liberated hydroxide ions were not neutralized in this titration. As a result, the pH of both the IRN-78 and IONAC A-365 resins increased.

The IRN-78 resins remained at high pH for extended periods during these studies. Sorption phenomena are enhanced on these types of resins under these conditions [Moody and Thomas, 1972]. It is believed that some sorption of both permanganate and nitrate ions occurred during these studies, since exchange capacities indicated from the titration curves are considerably larger than expected.

Results of Nitric Acid and Potassium Permanganate Addition to Regenerated and to Reagent-Loaded Resins [Kempf, et al, 1988]

In preliminary reaction studies, oven-dried and vacuum-aspiration dewatered samples of regenerated (OH^- form) IRN-78 and IONAC A-365 resins were subjected to bulk (5 ml) and dropwise additions of potassium permanganate (~0.04M) and nitric acid (3M). This was carried out in a test tube. The results of these experiments are summarized in Figure 4.

A comparison of the reaction results for oven-dried resins versus dewatered (vacuum-aspirated) resins upon nitric acid addition shows that the drying of the resin does have an effect on the extent of the reaction of the resins with HNO_3.

A comparison of the reaction results for dropwise addition of nitric acid versus bulk addition of nitric acid shows that the way in which the nitric acid is added also has a strong bearing on the extent of the reaction.

Resins treated with potassium permanganate reacted to a lesser extent than those treated with nitric acid. Both chemicals are oxidizing agents but a comparison of their relative oxidizing "strength" is not appropriate since the concentrations varied by nearly two orders of magnitude, i.e., the nitric acid was much more concentrated than the permanganate.

The next round of experiments involved two new parameters compared to these preliminary studies, namely: resins were "loaded" with picolinic acid, EDTA or Fe^{+2}; and dewatering by vacuum aspiration was performed intermittently during the study. For example, in a test of bulk addition of nitric acid to IRN-78 resins loaded with picolinic acid, the resin batch

Reactions with $KMnO_4$ and HNO_3

Resin		Reagent		Observation
IONAC-A365/OH^- (oven-dried)	+	$KMnO_4$ (bulk)	$\longrightarrow$	small amount of heat water layer and black MnO_2 present upper beads swelled
IRN-78/OH^- (oven-dried)	+	$KMnO_4$ (bulk)	$\longrightarrow$	small amount of heat water layer and black MnO_2 present
IONAC-A365/OH^- (oven-dried)	+	HNO_3 (bulk)	$\longrightarrow$	steam-like gas upper beads swelled
IRN-78/OH^- (oven-dried)	+	HNO_3 (bulk)	$\longrightarrow$	bubbled and foamed slight smoke
IONAC-A365/OH^- (dewatered)	+	HNO_3 (dropwise)	$\longrightarrow$	no reaction
IRN-78/OH^- (dewatered)	+	HNO_3 (dropwise)	$\longrightarrow$	distinct color change (brownish - yellow)
IONAC-A365/OH^- (oven-dried)	+	HNO_3 (dropwise)	$\longrightarrow$	small amount of heat
IRN-78/OH^- (oven-dried)	+	HNO_3 (dropwise)	$\longrightarrow$	traces of smoke

Figure 4 Summary of Reactions of Resins with Potassium Permanganate and Nitric Acid Solutions.

would be dewatered after each of three 3ml acid additions. In the experimental set-up for these studies, a Buchner funnel was set on a side-arm Ehrlenmeyer flask attached to a pump. This arrangement facilitated addition of the nitric acid and potassium permanganate solutions to the resin batches, measurement of temperature changes in the resin during, and it allowed color change or precipitates in the resins and in the eluates to be seen easily.

A very large number of individual experiments have been carried out, sixty-two to date. These correspond to:

- anion resin types IRN-78 and IONAC A-365, each "loaded" with 50% and 100% theoretical full loadings of picolinic acid and also each loaded to 100% theoretical full loadings of EDTA; the cation resin IRN-77 was given a 100% theoretical full loading of Fe^{+2} from ferrous sulfate solution.

- dropwise (total of 150 to 200 drops in three stages) and bulkwise (three separate 3ml) additions of 3M nitric acid and ~0.4M potassium permanganate individually and then sequentially, i.e., nitric acid followed by potassium permanganate.

- original oven-drying or vacuum-aspiration dewatering of the loaded resins; and

- vacuum-aspiration between additions of nitric acid and potassium permanganate.

Observations made in each of these experiments included: (1) resin bed temperature; (2) resin slurry and eluate pH; (3) resin slurry and eluate color; (4) precipitate color and quantity, and (5) presence of vapors or fumes. An abbreviated table of results is given for 100% picolinic acid-loaded resins, Table 2. The first column of the table gives the resin type and whether it was oven-dried or vacuum-aspirated initially. The second column contains the oxidizing chemical added (nitric acid or potassium permanganate). The third column is broken into four parts corresponding to: (a) whether vapors or fumes were observed, (b) maximum change in temperature, (c) color changes in the resin bed, and (d) presence of precipitates. The largest temperature changes (most heat generation) were observed for nitric acid addition to initially oven-dried IRN-78 and IONAC A-365 resins. Under these conditions, a white precipitate was also observed in the resin batch. Smaller temperature changes, less precipitate and more resin bed color changes were observed for initially vacuum-aspirated resins. Potassium permanganate addition lead, in general, to resin color changes, to small amounts of vapor/fumes, and to little heat generation.

Similar results were obtained for the EDTA-Loaded resins and for Fe^{+2} loaded IRN-77 resins, i.e., the largest temperature changes occurred with initially oven-dried resins and when nitric acid was added.

Table 2 Nitric Acid and Potassium Permanganate Additions to Oven-Dried and Vacuum-Aspiration Dewatered IRN-78 and IONAC A-365 Resins Loaded with Picolinic Acid

Resin	Added Chemical	"Effects" Observed			
		Vapor/ Fumes	ΔT (°C)	Color Change	Precipitate(s)
Oven-Dried					
IRN-78	Nitric Acid (NA)	None	15	None	White
IRN-78	Potassium Permanganate (PP)	Some	0	None	White
IONAC A-365	NA	None	9	Cream-light brown	White
IONAC A-365	PP	Some	0	None	Reddish
Vacuum-Aspirated					
IRN-78	NA	Some	3	Orange-light yellow	None
IRN-78	PP	Little	0	Light-dark	None
IONAC A-365	NA	Little	5	None	White
IONAC A-365	PP	Little	0	Yellow-brown	None

White precipitates were observed for all initially oven-dried, picolinic acid-loaded (50% and 100% theoretical loadings) IRN-78 and IONAC A-365 resins. For the initially vacuum-aspirated batches, only the IONAC A-365 picolinic acid-loaded resins gave a precipitate when nitric acid was added. For EDTA-loaded resins, a different effect was observed: all of the initially vacuum-aspirated resins showed white precipitates while only one type of the oven-dried samples did, namely: IONAC A-365 (50% and 100% theoretical EDTA loadings) when both nitric acid and potassium permanganate had been added. No precipitates were observed under any conditions for the Fe^{+2} loaded IRN-77 resins.

Slight resin color changes were observed in a number of cases across the whole spectrum of sample types. The eluate and resin slurry pH values were, as expected (given the addition of nitric acid), quite low: eluates, pH 2.8 to <1; and resin slurries, pH 3.3 to <1.

Control tests were run on regenerated resins (OH^- form for anion resins IRN-78 and IONAC A-365, H^+ form for cation resin IRN-77). It was hoped that the magnitude of the heat generation contribution could be found from resin hydration and/or neutralization. The results of these tests are given in Table 3. Oven-dried and vacuum-aspirated IRN-78, IONAC A-365 and IRN-77 regenerated resins were separately subjected to addition of: (1) deionized water; (2) nitric acid; and (3) potassium permanganate. The results indicated that little heat of hydration is involved while neutralization heat may be significant. No precipitates were observed, however, on nitric acid addition. This is to be expected since the "product" of the exchange (and the neutralization products, simultaneously) is H_2O.

From this, it is believed that the precipitates observed on addition of nitric acid to picolinic acid-loaded (or EDTA-loaded) resins were solid picolinic acid (or solid EDTA, respectively). These precipitates will be analyzed to cofirm this. Some of the heat evolved in these systems would have been neutralization heat. Potassium permanganate addition to the regenerated control resin batches had very little effect.

SUMMARY

IRN-78, IONAC A-365, and IRN-77 organic ion exchange resin moisture contents vary significantly depending on the counter ion "loading." For these resins the EDTA, picolinic acid and Fe^{+2} "loaded" forms, respectively, had moisture contents lower than the regenerated, OH^- and H^+ "loaded" forms. Heat- and gas- generating reactions have occurred with two anion resins used, IRN-78 and IONAC A-365; color changes and precipitates were also observed. The resins were originally in the OH^- form and potassium permanganate and nitric acid were oxidizing solutions used to produce the reactions. The extent/vigor of the reaction is very highly dependent on the degree of dewatering of the resins and (probably linked to this) on the method of solution addition (dropwise or in bulk). The heat generation may be due, in part, to the heat of neutralization (acid addition to hydroxide-form resins) [Brumfield and Kempf, 1988]. Ferrous ion loaded cation resins (IRN-77) showed little reactivity toward nitric acid and potassium permanganate.

Table 3 Control Test Results

Resin	Deionized Water	Nitric Acid	Potassium Permanganate
IRN-78 (OD)[a]	None	Heat, Fumes	Murky Eluate
IRN-78 (VA)[b]	None	Heat, Color Change	Precipitate (brown-black)
IONAC A-365 (OD)	Heat	Heat	"
IONAC A-365 (VA)	None	Heat	"
IRN-77 (OD)		Heat	"
IRN-77 (VA)	None		"

[a] OD = Oven-dried
[b] VA = Vacuum-aspiration dewatered

Task 2: Compatibility of Container Materials with Decontamination Wastes

This task was initiated to evaluate the compatibility of a range of container materials with a simulated decontamination resin waste. The materials include Ferralium 255 (a duplex stainless steel), TiCode-12 (a dilute titanium alloy), Types 304 and 316 stainless steel, carbon steel, and high-density polyethylene. The carbon steel coupons were added after the first irradiation cycle when some of the original specimens were deemed surplus and removed to provide space. Thus, the carbon steel specimens were exposed to resins which had been pre-irradiated to approximately 5×10^7 rad.

The resin decontamination waste chosen for this task simulates a potential Low Oxidation-State Metal Ion (LOMI) process waste. The reagents used in this process promote rapid dissolution of surface oxides by changing the oxidation state of the metal ions, e.g., Fe(III) to Fe(II). By definition, LOMI reagents contain 1) a reducing metal ion and 2) a chelating ligand (Bradbury, 1982). The vanadous picolinate/formate system is one such reagent which has been successfully applied to full scale reactor decontamination. Because of its superior decontamination capability and the relative non-aggressiveness of the medium, it is one of the most important reagents for present decontaminations. The simulated LOMI resin waste used in this study consists of two volumes of IONAC A-365 anion resin to one volume of IRN-77 cation resin. The IONAC A-365 is loaded with both picolinate and formate ions whereas the IRN-77 is always in the as-received H^+ form. The initial moisture content of the mixed bed resin was 47.3 percent by weight. Full details of the resin preparation procedure are given elsewhere (Adams and others, 1988).

To check how corrosion is influenced by gamma irradiation (which is present in most types of low level waste) and by the presence of organic reagents on the resin, four types of corrosion test were initiated:

a) corrosion in mixed-bed resins with the anion component loaded with picolinate/formate cation resin in the H^+ form;

b) corrosion in as-received mixed-bed resins (i.e., anion resin in the hydroxide cation resin in the H^+ form;

c) similar to (a) but in the presence of a gamma field of about 1×10^4 rad/h; and

d) similar to (b) but in the presence of a gamma field of about 2×10^4 rad/h.

The four resin beds were contained in glass vessels measuring 7.0 cm ID x 30.5 cm in height. Metallic specimens were placed horizontally in two layers, one resting on the flat base of the vessel and covered by resin, and another near the middle of the resin bed where specimens were contacted on both sides by resin.

The high density polyethylene (Marlex CL-100) specimens were made from strips measuring 10.2 x 1.25 x 0.32 cm. They were bent into a "U-bend" configuration by bending them and fastening the two ends with steel nuts and bolts. In the molding of the drum from which the specimens were cut, one side of the drum becomes oxidized by air. When the oxidized material is on the outer surface of a U-bend specimen, cracks are formed because of the lower ductility. When the non-oxidized material is on the outer bend surface, no cracking is present. Additional crack propagation during testing was studied for samples with both oxidized (cracked) surfaces and non-oxidized (uncracked) surfaces on the U-bend specimens. The polyethylene specimens were placed between the two metallic specimen layers with the apex of each U-bend facing upward.

The resin/container material irradiation systems were mechanically sealed so that gas generation could be monitored continuously. In the case of the unirradiated controls, the glass vessels were sealed with a "Parafilm" plastic sheet.

Gas Generation During Irradiation

During the first week of irradiation, the pressures in the irradiated systems containing simulated LOMI resin wastes and the as-received unloaded resins showed a pressure drop of about 20 percent, after which the pressure began to increase at a linear rate. The initial pressure drop is caused by the scavenging of oxygen in the original air environment by the resin beads. It is well-known that gamma-induced oxidation of polymeric materials can reduce oxygen levels to low values. Analysis of gases throughout the irradiation cycles shows that oxygen levels drop to less than 1 percent of the total pressure. The pressure increase is caused by hydrogen and carbon monoxide generation. For the simulated LOMI resin wastes, the H_2/CO ratio is much larger than that for the control resins.

Corrosion Analysis

At the end of the second examination of the irradiated container materials, a dose of 1×10^8 rad had been accumulated. No corrosion was noted for the Ferralium and the TiCode-12 which remained bright and shiny. The relevant unirradiated controls were similarly unaffected. Only the austenitic stainless steel and carbon steel showed evidence of attack.

Figure 5 shows Type 304 stainless steel coupons which have been irradiated to 1×10^8 rad in the presence of simulated LOMI decontamination waste resins. Two of the specimens were from the top layer in the resin column and the other two were in the lower layer. The spot-type localized attack is usually more pronounced for specimens in the bottom layer (see Specimen 13). This is likely to be connected with the higher levels of moisture on resins near the bottom of the column. It was found that additional corrosion spots had been initiated since the last examination which was for an accumulated dose of 5×10^7 rad. Each spot was caused by contact with an individual cation resin bead. Since the resins in contact with the steel were an orange-brown color, it is speculated that the corrosion mechanism involves the replacement of H^+ on the IRN-77 resin by Fe^{2+} from the stainless steel. A film of moisture at the contact point between the resin and the stainless steel facilitates ion exchange. This liquid is typically very acidic based on work by [Swyler and Weiss, 1981] who found that the pH approached 1.0 for a gamma dose of 10^9 rad.

Figure 6 shows Type 304 stainless steel control specimens which were exposed for 412 d to LOMI-resins in the absence of irradiation. The specimens were shiny, with little evidence of corrosion.

Type 304 stainless steel irradiated to 1 x 10^8 rad in the presence of as-received mixed-bed resins (i.e., non-LOMI waste) also showed spot corrosion similar to that described above. This would be expected since the resin component causing attack appears to be the IRN-77 cation resin which is in the H^+ form for all four test conditions.

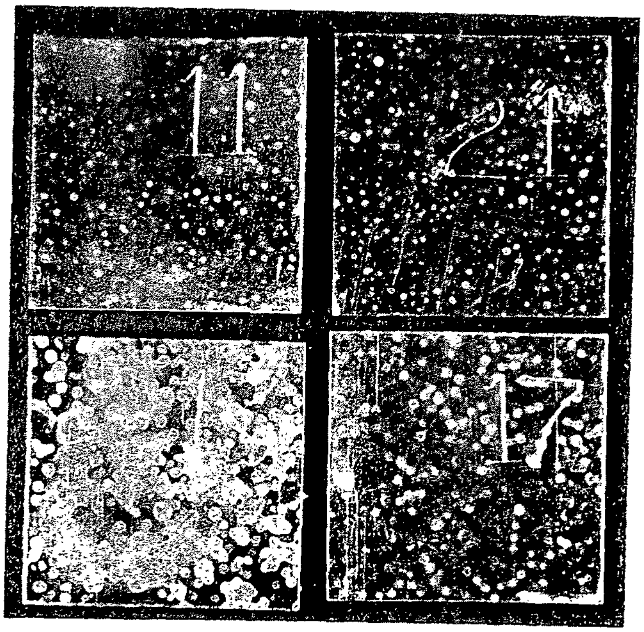
Data for Type 316 stainless steel show basically similar corrosion effects to Type 304. However, the amount of corrosion in the former is significantly less as would be expected based on its higher nickel and chromium contents.

Carbon steel showed very marked attack for all test conditions. Figures 7 and 8, for example, show specimens exposed to LOMI-type resins for irradiated (5 x 10^7 rad) and unirradiated conditions, respectively. The depth of attack at the resin contact points is far greater than for Type 304 stainless steel. The most severe attack was for a specimen on the bottom layer of samples which had been irradiated. Apparently, extra moisture and irradiation enhance attack.

Attack was also noted for carbon steel exposed to non-LOMI (as-received) mixed bed resins and irradiated to 5 x 10^7 rad. The attack appeared to be less regular than that shown in Figures 7 and 8, but it was severe in regions of the specimen surface where it was present.

High-density polyethylene U-bend specimens were examined after irradiation doses of about 5 x 10^7 and 1 x 10^8 rad. The appropriate unirradiated controls were examined also. The examination involved carefully studying changes in crack patterns at the stressed apexes of the bent specimens to check for crack initiation and propagation of cracks that were initially present as a result of the bending. Figures 9 and 10 show crack patterns sketched from the apexes of polyethylene specimens which were in contact with LOMI-type resin waste. Only specimens which had oxidized material on the apex showed cracking effects; non-oxidized material remained uncracked.

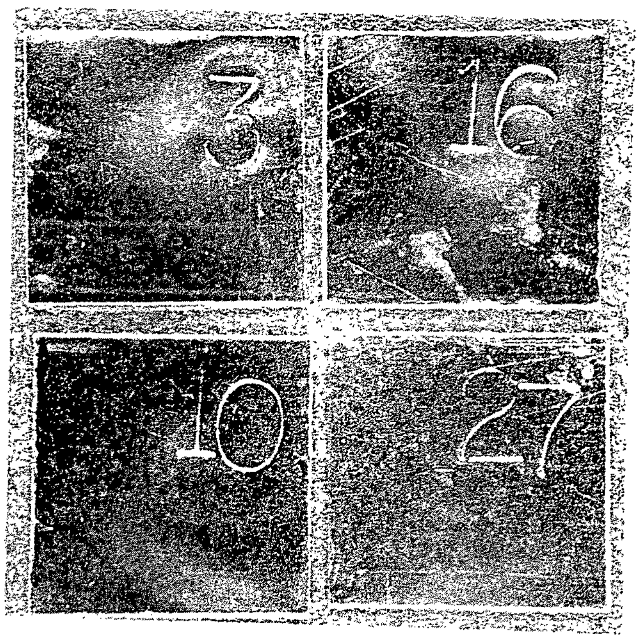
In Figure 9, it may be seen that no crack propagation occurred after an irradiation to 1 x 10^8 rad over a period of 412 d. This is in contrast to non-irradiated polyethylene which, over the same period, showed both new crack initiation and the growth of existing cracks. This lack of crack propagation in irradiated systems is associated with the rapid loss of



TOP LAYER

BOTTOM LAYER

Figure 5 Effect of Gamma Irradiation (10^8 rad) on the Corrosion of Type 304 Stainless Steel in the Presence of Mixed-Bed Ion-Exchange Resins. Mag. 2.5X.



TOP LAYER

BOTTOM LAYER

Figure 6 Corrosion of Type 304 Stainless Steel After 412 Days Exposure to Mixed-Bed Ion-Exchange Resins Loaded with Simulated LOMI Reagent. Mag. 2.5X.

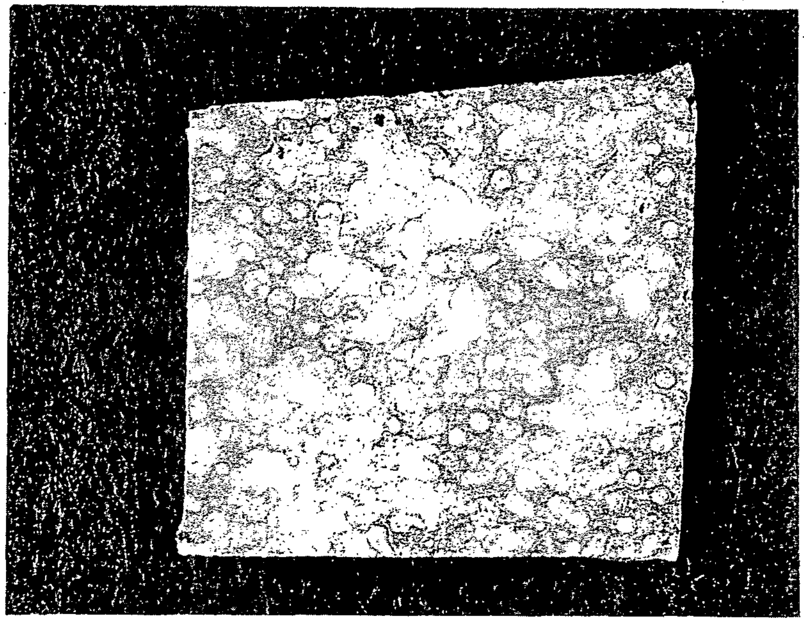


Figure 7 Severe Local Corrosion on Carbon Steel After Irradiation to 5 x 10^7 rad in the Presence of Mixed-Bed Ion-Exchange Resins Loaded with Simulated LOMI Reagent. Specimen was in the Lower Layer of Samples. Mag. 4X.

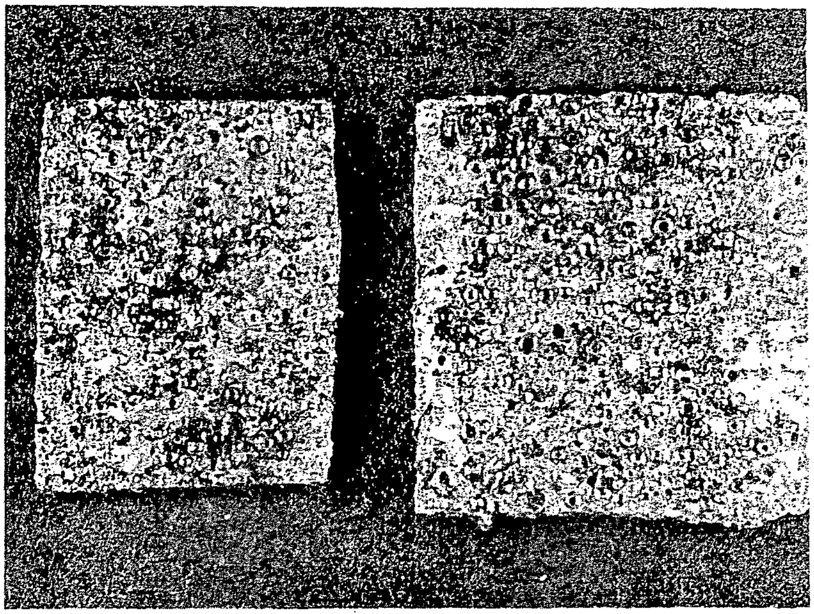


Figure 8 Local Corrosion of Carbon Steel After 208 days' Exposure to Mixed-Bed Ion-Exchange Resins Loaded with Simulated LOMI Reagent. Mag. 4X.

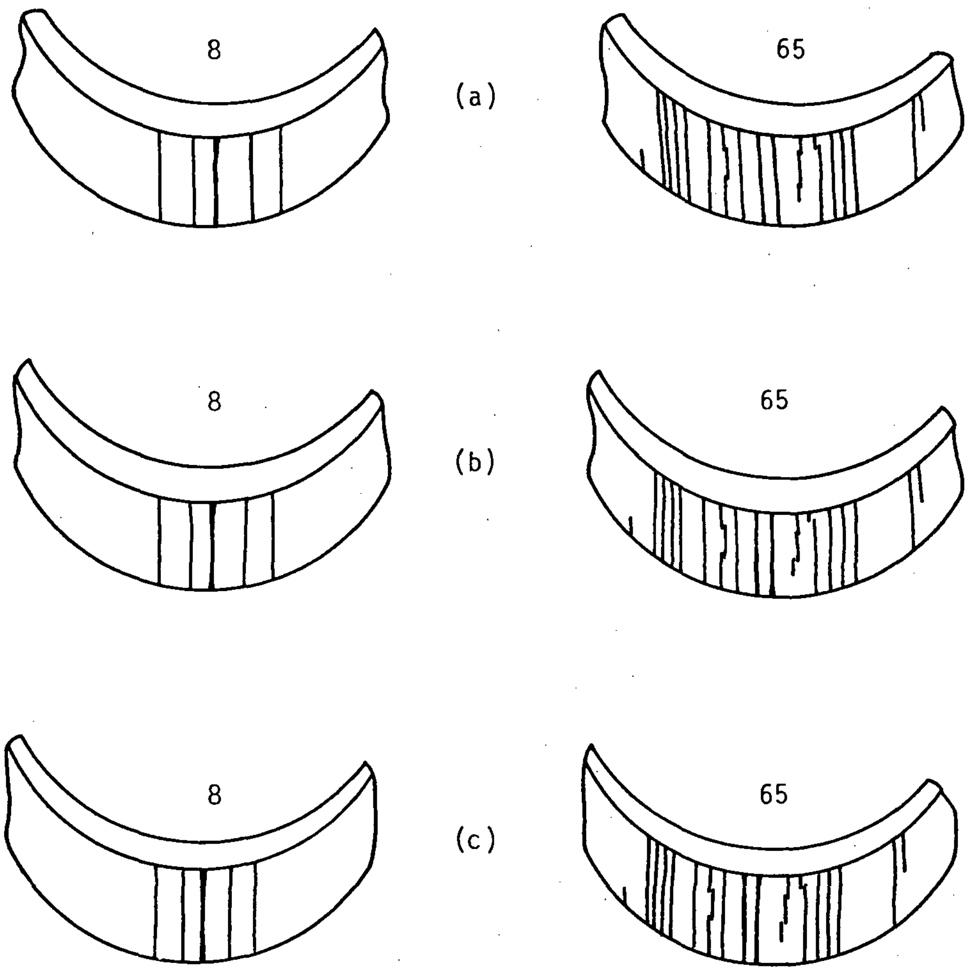


Figure 9 Cracking in the Oxidized Surfaces of Marlex CL-100 HDPE U-Bend Samples Placed in Contact with LOMI-Loaded Mixed-Bed Resins; (a) Crack Patterns at Start of Testing, (b) Crack Patterns After Irradiating for 204 d to 5.0×10^7 rad, (c) Crack Patterns After Irradiating for 412 d to 1×10^8 rad. Specimen Numbers Given Above Each Sketch.

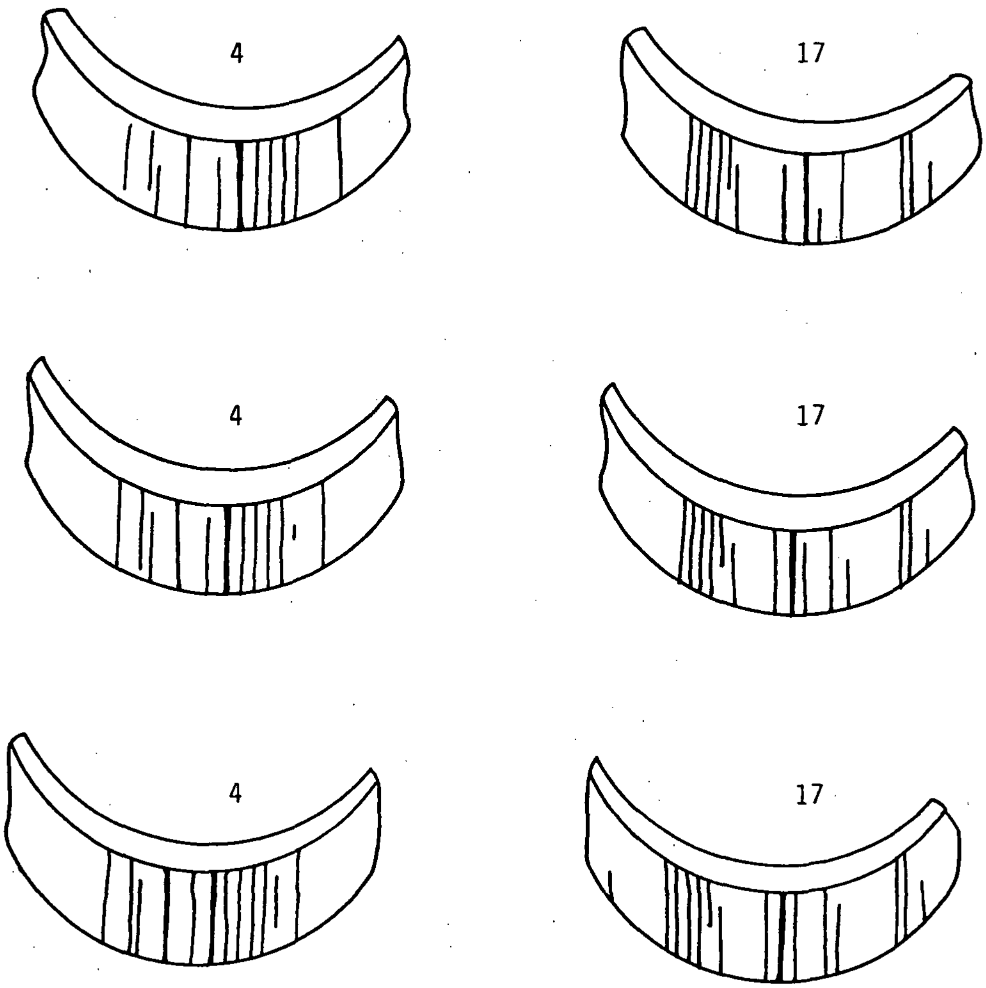


Figure 10 Cracking in the Oxidized Surfaces of Marlex CL-100 HDPE U-Bend Samples Placed in Contact with LOMI-Loaded Mixed-Bed Resins; (a) Crack Patterns at Start of Testing, (b) Crack Patterns After 204 d Without Irradiation, (c) Crack Patterns After 412 d Without Irradiation. Specimen Numbers Given Above Each Sketch.

oxygen noted earlier. Degradation of many polymers is associated with a synergistic effect between irradiation and oxygen (Gillan and Clough, 1981). When oxygen is unable to react with polymer chains which have undergone irradiation-induced scission, the ductility losses normally expected in oxygen-containing environments become small or negligible.

<u>Summary</u>

The corrosion of carbon steel and austenitic stainless steel in mixed bed resins is enhanced by gamma irradiation. However, cracking in high density polyethylene is essentially eliminated because of the rapid removal of oxygen from the environment by gamma-induced oxidation of the large resin mass. Ferralium-255 and TiCode-12 are not attacked even for gamma doses up to 10^8 rad.

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RADIONUCLIDE CHARACTERIZATION OF REACTOR DECOMMISSIONING WASTE AND SPENT FUEL ASSEMBLY HARDWARE

D. E. Robertson, C. W. Thomas, N. L. Wynhoff and D. C. Hetzer
Pacific Northwest Laboratory

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) has recently enacted rules setting forth technical, safety, and financial criteria for decommissioning of licensed nuclear facilities, including commercial nuclear power stations.(1) These rules have addressed six major issues, including decommissioning alternatives, timing, planning, financial assurance, residual radioactivity, and environmental review. Also, the rules governing disposal of low-level radioactive wastes in commercial shallow land burial facilities will be applicable to most of the wastes generated during reactor decommissioning.(2) The appropriate response to each of these issues by the licensee and the NRC depends greatly on an accurate and reliable assessment of the residual radiological conditions existing at the nuclear power stations at the time of decommissioning. Large volumes of data exist which describe the radionuclide concentrations associated with active waste streams generated at nuclear power stations. However, comparatively little information is available that documents the residual radionuclide concentrations, distributions, and inventories residing in contaminated piping, components, and materials of nuclear plant systems and in neutron-activated materials associated with the reactor pressure vessel and biological shield. Especially lacking is a detailed radiological characterization of the numerous types of wastes encountered during an actual reactor decommissioning and a characterization of the highly neutron activated metal components associated with pressure vessel components and spent fuel assembly hardware.

This study has been implemented to provide the NRC and licensees with a more comprehensive and defensible data base and regulatory assessment of the radiological factors associated with reactor decommissioning and disposal of wastes generated during these activities. The objectives of this study are being accomplished during a two-phase sampling, measurement, and appraisal program utilizing: 1) the decommissioning of Shippingport Atomic Power Station, and 2) neutron activated materials from commercial reactors. Radioactive materials obtained from Shippingport Station and from a number of commercial stations for comprehensive radionuclide and stable element analyses are being utilized to assess the following important aspects of reactor decommissioning and radioactive waste characterization:

- radiological safety and technology assessment from an actual reactor decommissioning (Shippingport)
- radiological characterization of intensely radioactive materials (greater than Class-C) associated with the reactor pressure vessel and spent fuel assembly hardware from commercial nuclear power plants
- evaluation of the accuracy of computer codes for predicting radionuclide inventories in retired reactors and neutron activated components
- assessment of waste disposal options associated with reactor decommissioning.

2.0 Radiological Characterization of Shippingport Station Decommissioning Wastes

Specimens of surface-contaminated and neutron activated components from Shippingport Station have been obtained for detailed radiochemical analyses. These materials are providing the basis for evaluating the radiological safety and waste disposal options associated with reactor decommissioning.

2.1 Residual Radionuclides Associated with Primary, Secondary, and Auxiliary Systems

During the dismantlement of the Shippingport Station, numerous components from the primary and secondary coolant loops and the auxiliary systems were made available for sampling and subsequent detailed radionuclide characterization for the 10CFR61 radionuclides. Specimens of primary coolant piping, primary coolant check valves, main steam piping, feedwater piping, coolant purification system piping, monitoring/instrumentation system piping, and fuel pool recirculation system piping were obtained for residual radionuclide characterization of contaminated surfaces. In addition, a 208-liter drum of concrete chips spalled from the surface of the fuel canal was obtained for assessing the radionuclide contamination of the concrete surface of the fuel pool.

2.1.1 Primary Coolant Piping

The majority of the residual radioactive material residing within a retired nuclear plant (excluding the neutron activated pressure vessel and internals) is located within the primary coolant loop, being attached to the surface corrosion film. Five excellent cores of the primary coolant piping were provided for analysis at PNL by the Shippingport Station Decommissioning Project Office for characterization of the contaminated corrosion layer in the primary system. These cores were 7 cm in diameter by 4 cm thick and contained a thin, black, radioactive corrosion product layer on the inside surface which was very hard and retentive. Cores were taken from the "A," "B," and "C" loop primary coolant piping, at the entrance to (cold side) and exit from (hot side) the reactor pressure vessel at the outer surface of the neutron shield tank. The radioactive corrosion film was removed by immersing the contaminated side in hot 6N hydrochloric acid for several minutes and brushing the surface with a stiff nylon brush. The stripped corrosion film was then completely solubilized by heating in a mixture of hydrochloric and nitric acids. The acid solutions were used for direct gamma spectrometric and radiochemical analyses. One of the core specimens ("A" loop-hot side) was saved for special testing and was cut into four equal wedge-shaped pieces for conducting a series of special form tests for shipment of radioactive materials described in Section 2.1.3 of this report.

The primary coolant piping core specimens were analyzed for the long-lived radionuclides of a safety and waste disposal concern. The results are given in Table 2.1. It is immediately obvious that the residual radioactivity at Shippingport Station is somewhat atypical of that observed in a number of commercial nuclear power stations[3,4]. First, the gamma-ray

TABLE 2.1

Residual Radionuclide Concentrations Associated With the Corrosion Layer on Shippingport Primary Coolant Piping

Radionuclide Concentration ($\mu Ci/cm^2$) as of Feb., 1987

Radionuclide	Half-Life(yr)	B-Loop, Cold Side	B-Loop, Hot Side	C-Loop, Cold Side	C-Loop, Hot Side
^{60}Co	5.27	0.38 ± 0.011	0.88 ± 0.029	0.57 ± 0.017	0.88 ± 0.029
^{55}Fe	2.7	0.050 ± 0.0002	1.13 ± 0.034	0.100 ± 0.003	0.62 ± 0.019
^{63}Ni	100	0.035 ± 0.0018	0.53 ± 0.029	0.069 ± 0.006	0.74 ± 0.037
^{59}Ni	8.0×10^4	(2.25 ± 0.113)E-4	(4.04 ± 0.121)E-3	(4.40 ± 0.132)E-4	(3.20 ± 0.096)E-3
^{94}Nb	2.0×10^4	(2.40 ± 0.44)E-6	(1.13 ± 0.07)E-5	(6.09 ± 0.53)E-6	(7.85 ± 0.43)E-6
^{14}C	5730	(5.6 ± 7.7)E-5	(4.9 ± 8.8)E-5	(8.1 ± 7.8)E-5	(6.9 ± 6.1)E-5
^{99}Tc	2.13×10^5	(3.4 ± 2.4)E-6	(2.8 ± 0.24)E-5	(8.1 ± 2.2)E-6	(1.29) ± 0.27)E-5
^{3}H	12.33	(1.4 ± 1.6)E-6	(1.7 ± 1.6)E-6	(1.2 ± 1.3)E-6	(1.6 ± 1.8)E-6
$^{239\text{-}240}Pu$	2.44×10^4	(1.26 ± 0.06)E-7	(1.88 ± 0.10)E-7	(3.09 ± 0.04)E-6	(2.79 ± 0.09)E-7
^{238}Pu	87.8	(7.51 ± 0.43)E-8	(1.16 ± 0.08)E-7	(5.56 ± 0.18)E-7	(1.31 ± 0.07)E-7
^{241}Am	433	(1.10 ± 0.16)E-7	(1.36 ± 0.14)E-7	(1.16 ± 0.04)E-6	(1.67 ± 0.16)E-7
^{244}Cm	18.1	(9.0 ± 7.9)E-9	(5.9 ± 5.9)E-9	(8.7 ± 3.8)E-9	(5.8 ± 5.8)E-9
^{137}Cs	30.2	<3E-4	<5E-4	<4E-4	<5E-4
Dose Rate @ 1 cm					
w/beta shield (mR/h)		10	32	15	22
w/out beta shield (mRad/h)		230	1000	350	800

spectra of the stripped corrosion layer resembled a pure ^{60}Co spectrum. A careful examination of the spectra could not identify any other gamma-emitting radionuclides. Although the samples contained ^{55}Fe and ^{63}Ni concentrations that were sometimes comparable to the ^{60}Co levels, these radionuclides emit only low-energy x-rays and beta particles and cannot be detected by direct gamma-ray spectrometry. The second unusual feature of the residual radioactivity is the almost complete absence of any fission products or transuranic radionuclides. Although trace amounts of Pu, Am, and Cm isotopes were detectible in the corrosion film samples, their concentrations were so low that their origin appears to have been from traces of tramp uranium on the outer surfaces of the fuel elements, and not due to leakage from failed fuel. These measurements confirm the fact that no measurable fuel failures occurred at Shippingport Station during the entire operating history of the plant - a truly noteworthy operational record.

A comparison of the residual radionuclide concentrations associated with the contaminated surfaces of primary coolant piping at Shippingport Station with that observed at seven commercial nuclear power stations is shown in Figure 2.1. The data from the seven commercial units were taken from References 3 and 4. Shown in Figure 2.1 are the range and average concentrations of ^{60}Co, ^{63}Ni, ^{55}Fe, ^{94}Nb, ^{137}Cs, and $^{239-240}Pu$ associated with the residual radioactivity at these stations. The average concentrations of the activation product radionuclides ^{60}Co, ^{63}Ni, ^{55}Fe, and ^{94}Nb are lower in the Shippingport samples by factors of about 10, 2.7, 60, and 40, respectively. The $^{239-240}Pu$ and ^{137}Cs are 1000 and greater than 200 times lower, respectively, than the average concentrations for the commercial units.

2.1.2 Secondary Coolant Piping and Auxiliary System Components

Samples of piping from the 13 in. O.D. main steam line, the 6 5/8 in. O.D. feedwater piping, and the 6 1/2 in. O.D. fuel pool recirculation system piping were obtained for residual radionuclide analyses which are presently in progress. The corrosion film on the inner pipe surface was scraped from areas of 995 cm^2 and 280 cm^2 from the main steam piping specimen and the feedwater piping specimen, respectively. The fuel pool recirculation system piping has yet to be sampled.

Another important sample of opportunity obtained during the decommissioning was a drum of slightly contaminated surface concrete from the fuel canal. The top 0.6 cm of this concrete was mechanically spalled from the walls and floor of the fuel canal to remove surface-absorbed, non-smearable radionuclide contamination.

The entire 208-liter drum of concrete chips weighing 248 kg was directly assayed by gamma-ray spectrometry using a special barrel counting system developed by PNL. This counting system consists of a collimated intrinsic germanium detector which scans the barrel, top to bottom, in eleven 7.6 cm wide vertical segments as the barrel rotates on a turntable at 30 rpm. This method in effect "homogenizes" the sample in the barrel during the counting period. The 208-liter barrel geometry has been calibrated by preparing standardized radionuclide mixtures in various density materials ranging from 0.1 to 1.4 g/cm^3. The density of the concrete chips was 1.2 g/cm^3.

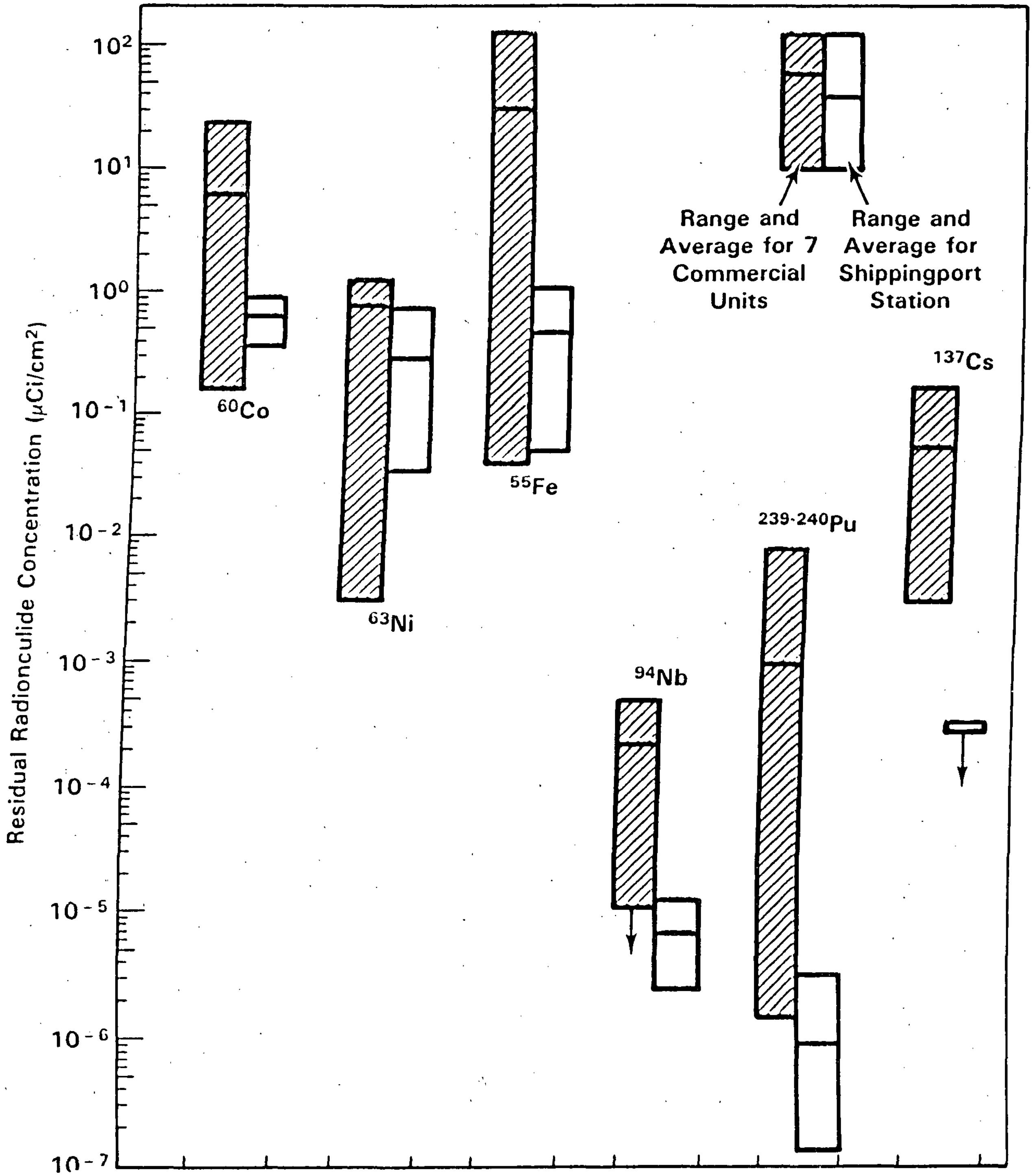


FIGURE 2.1 Comparison of Residual Radionuclide Concentrations on Primary Coolant Piping from Shippingport Station with Seven Commercial Stations

The gamma ray spectrometry of the drum of concrete chips indicated that ^{60}Co was the only gamma-emitting radionuclide detectible, being present at an average concentration of 2.14 ± 0.03 µCi/kg, or 2.14 nCi/g. This concentration of ^{60}Co is just slightly higher than the specific activity of 2 nCi/g considered in 49CFR173.389(e) to be radioactive for transportation purposes. An aliquot of the concrete chips is presently being radiochemically analyzed to determine the concentrations of alpha, beta, and low-energy photon emitting radionuclides, which are expected to be very low.

2.1.3 Radionuclide Characterization for DOT Requirements for Transportation of the Shippingport Station Pressure Vessel as a Type B, LSA Package

One of the important lessons learned from the Shippingport Station decommissioning that is directly applicable to the commercial nuclear power industry is the methodology for characterizing, preparing, packaging, and transporting the reactor pressure vessel for disposal. This information is contained in the "Safety Analysis Report for Packaging - Shippingport Reactor Pressure Vessel and Neutron Shield Tank Assembly"[5]. One important aspect of the radionuclide characterization of the pressure vessel package is an assessment of the dispersivity, or conversely, the retentiveness of the radioactive corrosion film on the inside surfaces of the reactor pressure vessel and internal components under a variety of hypothetical accident conditions during transport to the disposal facility. This assessment is required under 49CFR173.467, "Tests for demonstrating the ability of Type B and fissile radioactive materials packaging to withstand accident conditions in transportation"[6], which details a series of tests to determine the dispersivity of the inner radioactive corrosion film. These tests were conducted at PNL using specimens of the stainless steel primary coolant piping cores taken at the outlet of the Shippingport pressure vessel.

During the decommissioning of Shippingport Atomic Power Station, the pressure vessel will be removed as an intact unit and could be shipped as an LSA package to the Hanford reservation in Washington for burial[7]. However, DOE has decided also to qualify this package as Type B to further demonstrate its integrity. The neutron shield tank surrounding the pressure vessel has been filled with concrete, as in the pressure vessel, and the combination of the pressure vessel with its concrete-filled shield tank will serve as the actual shipping container. To ensure that no hazardous releases of dispersable radionuclides would occur in the event of an accident during transportation, the tests for special form radioactive materials described in 49CFR173.469 were conducted to demonstrate that the pressure vessel package would also comply with Type B packaging requirements. These tests included: 1) an impact test, 2) a percussion test, 3) a heat test, 4) a modified bend test, and 5) a leaching test.

A 7-cm diameter by 4 cm thick stainless steel core taken from the "A Loop - Hot Side" primary coolant piping section at the outlet of the Shippingport reactor pressure vessel was used to simulate the radioactively contaminated inner surfaces of the pressure vessel and internals. This specimen was cut into four wedge-shaped quarters. One-quarter sections, hereafter referred to as a specimen, were used in the impact test, the percussion test, and the heat test.

The special form testing has indicated that the radionuclides associated with the corrosion layer on the primary coolant outlet piping from the SAPS pressure vessel are both very tightly bound to the underlying metal and very insoluble in both high purity water and seawater. These results are summarized in Table 2.2.

The data obtained during this testing have helped provide an assessment of the shipping requirements of the Shippingport pressure vessel package. This assessment has been made in Ref. 5, and indicates that any releases of dispersable radionuclides to the environment in the case of a hypothetical accident are within regulatory tolerances.

TABLE 2.2

Summary of Radioactive Material Releases from Test Specimens During Special Form Testing (49CFR173.469)

Test	Fraction of Radioactivity Released During Test
1. Impact	2.48×10^{-4}
2. Percussion	2.54×10^{-4}
3. Heat	2.56×10^{-5}
4. Modified Bend	$<1.1 \times 10^{-4}$
5. Leaching (1st phase, high purity water)	
-Impact specimen	2.00×10^{-4}
-Percussion specimen	2.28×10^{-4}
-Heat specimen	1.19×10^{-3}
-Modified bend specimen	2.34×10^{-2}
6. Leaching (2nd phase, high purity water)	
-Impact specimen	7.36×10^{-5}
-Percussion specimen	1.01×10^{-4}
-Heat specimen	3.23×10^{-4}
-Modified bend specimen	3.10×10^{-4}
7. Leaching (1st phase, seawater)	
-Impact specimen	6.59×10^{-5}

2.2 Neutron Activated Shippingport Core-3 Fuel Assembly Hardware

Two sets of neutron activated metal specimens were obtained from the Shippingport Core-3 fuel assembly hardware. These samples have been received at PNL and will be radiochemically analyzed in FY 1989 to determine radionuclide classification and to evaluate the accuracy of computer codes for predicting neutron activation product concentrations by comparisons with empirical measurements. The first was a set of three SII-3 stainless steel grid bolt specimens and one SII-3 stainless steel grid bolt locknut. These specimens were collected from a moveable seed module element. Core-3 at Shippingport had no control rods. Power levels were controlled by moving seed modules up and further into concentric blanket element modules. Because of their movement it was difficult to accurately position the location of the grid bolts and locknut in the cores neutron flux. However, sophisticated computer codes have accurately characterized the neutron fluence levels in various areas of interest in the fuel modules. The closest estimates obtainable by this method are fluence levels experienced by fuel rods located directly adjacent to the seed modules. The actual fluence experienced by the grid bolts and nut is less than that of the adjacent fuel rods, but difficult to determine, so the upper bound of neutron fluence experienced by the grid bolt is that of adjacent fuel rod 7Q6 or 7Q7. These calculated fluences were as follows:

Sample No.	Sample Type	Estimated Neutron Fluence
M9971	SS Grid Bolt	5.46 E20 n/cm^2
M9972	SS Grid Bolt	6.52 E20 n/cm^2
M9973	SS Grid Bolt	5.38 E20 n/cm^2
M9974	SS Lock Nut	5.38 E20 n/cm^2

The second set of neutron activated metal specimens from the Shippingport Core 3 were samples of Type 348 stainless steel, Inconel-X750, and Zircaloy-4 removed from various locations from three different types of fuel assemblies: 1) a blanket rod, 2) a reflector rod, and 3) a seed rod. The activated metal specimens from each rod included one piece of Inconel-X750 plenum spring, one piece of Type 348 stainless steel support sleeve, and two pieces of Zircaloy-4 cladding from the midplane and upper end of the rod.

These samples will be extremely valuable for characterizing the long-lived radionuclides produced in fuel assembly hardware and adjacent pressure vessel components, and for assessing the accuracy of predictive neutron activation codes.

2.3 Classification of Shippingport Decommissioning Wastes with Respect to 10CFR61

Although the decommissioning wastes generated at Shippingport Station are not subject to the regulations governing shallow land disposal of commercial low-level wastes (10CRF61), an assessment of the radionuclide contamination associated with the various decommissioning wastes is of interest. Based upon the comprehensive radiochemical analyses of the corrosion film associated with the primary coolant piping (Table 2.1), and assuming that the average concentration and observed range are representative of the contamination level of all plant systems exposed to primary coolant, e.g. steam generators, pressurizer, coolant pumps, primary purification systems, etc., it is possible to classify the waste with respect to the regulations in 10CFR61. Previous related studies have shown that for commercial power reactor stations having 5 to 50 times higher residual radioactivity levels in the primary systems, all components (excluding the pressure vessel) could be disposed of as Class "A" low-level waste (the least restrictive classification) in shallow land burial facilities. All radioactively contaminated concrete spalled from the fuel pool walls would also be Class A waste. It, therefore, becomes obvious that all primary systems removed during the decommissioning would be well below Class "A" radionuclide concentrations and, therefore, eligible for disposal as Class "A" waste if it were to be disposed of in a commercial facility. These results confirm that for well-maintained power reactors, the residual radionuclide levels associated with the most contaminated systems outside of the pressure vessel can be readily disposed of as Class "A" waste during commercial reactor decommissioning.

3.0 Radionuclide Characterization of Spent Fuel Assembly Hardware from Commercial Nuclear Power Stations

Because little information currently exists describing measurements of long-lived radionuclides in activated metal components from within reactor pressure vessels, it is imperative that empirical analyses of such components be conducted. These measurements will be utilized to assess the radionuclide concentrations, waste classification, and disposal options associated with reactor decommissioning activated metal wastes.

A number of well-characterized spent fuel assemblies from commercial nuclear power stations have become available at PNL for obtaining samples of the various metals of construction. These specimens are being radiochemically analyzed for the long-lived activation products of waste disposal concern to determine their 10CFR61 waste classification. The empirical measurements will then be compared with calculated activation product concentrations using existing codes (e.g., ORIGEN, ANISN, etc.) to determine the accuracy with which calculated estimates can be made. This comparison will lend confidence to calculational methods and/or identify shortcomings in these methods.

3.1 Sample Description

Three high-burnup commercial fuel assemblies, listed in Table 3.1, are currently being characterized. The following materials have been obtained for analysis:

TABLE 3.1

Spent Fuel Assembly Hardware Samples

Assembly Type	Reactor Station	Materials Sampled
General Electric (7 X 7)	Cooper	Stainless steel bottom end fittings and upper tie plate, Inconel expansion springs, Zircaloy grid spacers
Combustion Engineering (14 X 14)	Calvert Cliffs	Stainless steel bottom end fittings and flow/hold-down plates, Zircaloy and inconel grid spacers, Inconel hold-down springs
Westinghouse (14 X 14)	Point Beach	Stainless steel bottom and upper end fittings, Inconel hold-down springs, Zircaloy guide tube and grid spacers

These fuel assemblies are representative (both in their irradiation history and material composition) of the type of spent fuel assembly hardware that must be accommodated by the federal waste management system and many utilities.

Tables 3.2, 3.3 and 3.4 list important information for each fuel assembly. Their irradiation histories were obtained from information supplied by the utilities to the Department of Energy.

Metal specimens were taken from each grid spacer in each of the fuel assemblies, and from both the bottom and top end fittings (see Figures 3.1, 3.2 and 3.3). The main casting of the bottom and top end fittings was manufactured of stainless steel. The top end fitting however, had several additional pieces that were composed of various grades of Inconel. Samples were obtained to represent each of the materials of construction, in each possible location, as well as each of the main components.

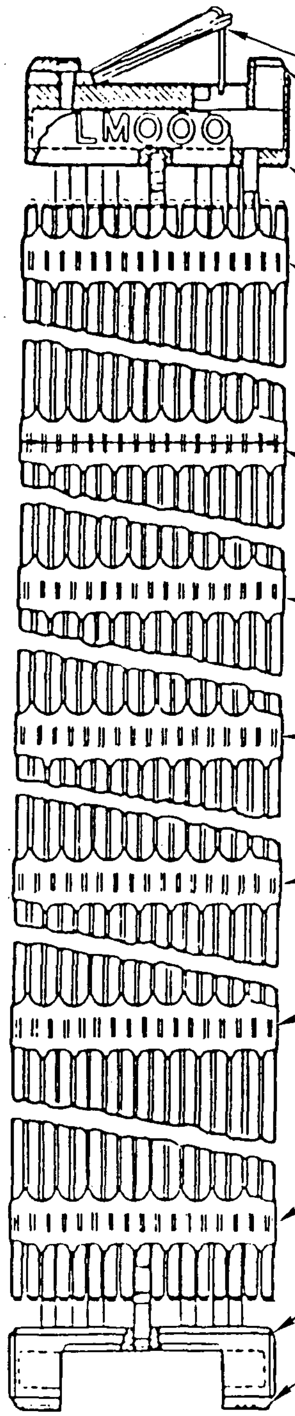
Thirty-eight samples of activated metal were obtained from the three spent fuel assemblies by mechanical means (i.e. by cutting and snipping). The sample sizes were on the order of 0.1 to 5 g. These were latter sub-samples, as described in Section 3.2. The remaining sample was retained in the event that further analysis would be required. The sample locations are shown in Figures 3.1, 3.2 and 3.3. These locations were selected to represent all the different materials available on each fuel assembly in as many different regions as practicable. Samples were taken from each grid spacer to provide as much data as possible regarding the shape of the neutron flux. The grid spacer sample also provided a good indication of the variance in the elemental composition, in particular for the trace elements.

3.2 Laboratory Analyses

3.2.1 Radiochemical Measurements

The 0.1 to 5 gram specimens of neutron activated stainless steel, Inconel, and Zircaloy, cut from the fuel assemblies were transferred from the original hot cell to a sample-preparation hot cell where the metal specimens were initially surface-decontaminated by acid etching. This cleaning consisted of immersing each specimen in hot (80-90°C) 6N hydrochloric acid for 60 seconds, followed by rinsing with fresh 6N hydrochloric acid. This etching was repeated three times, and was followed by a final acid etching by immersing each specimen for 60 seconds in hot (80-90°) 8N nitric acid. The specimens were then immediately rinsed with distilled water, dried on a paper towel, and placed in clean polyethylene vials. The vials were then transported to a radiochemistry laboratory for final decontamination of the metal specimens prior to initiating the radiochemical analyses.

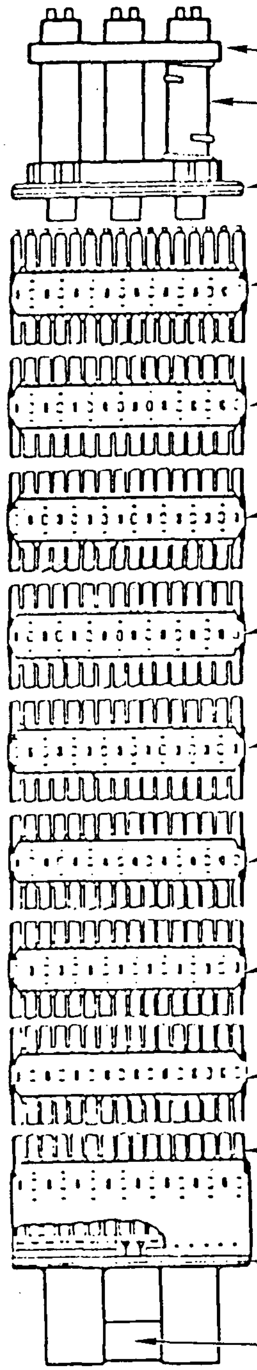
Westinghouse 14x14 Fuel Assembly



Sample Number		Material
W10	–	Holddown Plate (Inconel)
W12	–	Upper Nozzle (Stainless Steel)
W9	–	Upper Nozzle (Stainless Steel)
W8	–	Grid Spacer (Inconel)
W7	–	Grid Spacer (Inconel)
W6	–	Grid Spacer (Inconel)
W5	–	Grid Spacer (Inconel)
W4	–	Grid Spacer (Inconel)
W3	–	Grid Spacer (Inconel)
W2	–	Grid Spacer (Inconel)
W11	–	Lower Nozzle (Stainless Steel)
W1	–	Lower Nozzle (Stainless Steel)

FIGURE 3.1 Sample Locations

Combustion Engineering 14x14 Fuel Assembly



Sample Number		Material
CE25	–	Holddown Plate (Stainless Steel)
CE26	–	Holddown Spring (Inconel)
CE24	–	Flow Plate (Stainless Steel)
CE10	–	Grid Spacer (Zircaloy)
CE9	–	Grid Spacer (Zircaloy)
CE8	–	Grid Spacer (Zircaloy)
CE7	–	Grid Spacer (Zircaloy)
CE6	–	Grid Spacer (Zircaloy)
CE5	–	Grid Spacer (Zircaloy)
CE4	–	Grid Spacer (Zircaloy)
CE3	–	Grid Spacer (Zircaloy)
CE2	–	Grid Spacer (Inconel)
CE14	–	Lower Nozzle (Stainless Steel)
CE1	–	Lower Nozzle (Stainless Steel)

FIGURE 3.2 Sample Locations

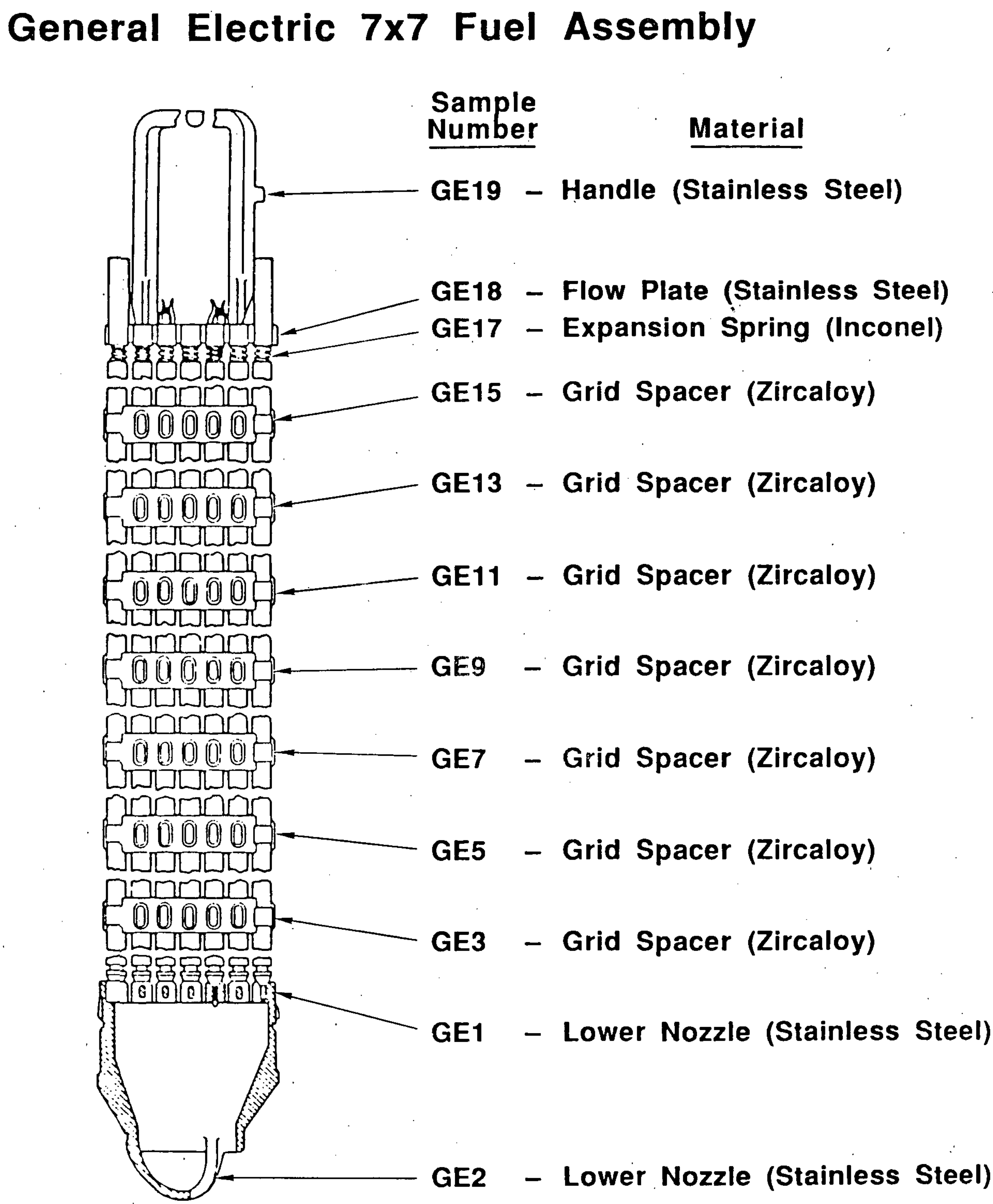


FIGURE 3.3 Sample Locations

TABLE 3.2

Westinghouse 14x14

Peach Bottom 400.7 kg U 3.192 w/o U-235

Cycle #	End of Cycle	Burnup @ EOC [MWD/MTU]
4 (out)	1 OCT 76	0
5	10 OCT 77	6,147
6	20 SEP 78	16,784
7	5 OCT 79	26,195
8	26 NOV 80	29,621
9	8 OCT 81	32,729

TABLE 3.3

Combustion Engineering 14x14

Calvert Cliffs 3.88.6 kg U 3.068 w/o U-235

Cycle #	End of Cycle	Burnup @ EOC [MWD/MTU]
1 (out)	1 JAN 77	0
2	23 JAN 78	9,466
3	21 APR 79	20,895
4	18 OCT 80	32,317
5	17 APR 82	41,781

TABLE 3.4

General Electric 8x8

Cooper Nuclear Station 190.4 kg U 2.506 w/o U-235

Cycle #	End of Cycle	Burnup @ EOC [MWD/MTU]
Begin Commercial Operation	JUL 74	0
1	SEP 76	13,046
2	SEP 77	18,910
3	APR 78	22,098
4 (out)	APR 79	22,098
5 (out)	MAR 80	22,098
6	APR 81	24,974
7	MAY 82	27,480

The final acid etching was conducted in a clean, shielded laboratory fume hood and consisted of repeating the immersion and rinsing steps conducted in the hot cell. This repeated etching and rinsing was necessary to completely remove traces of fission product and transuranic radionuclide contamination picked up during the cutting operations in the original hot cell, as well as removing remnants of contaminated corrosion films formed on the metal surface during its exposure to the reactor primary coolant.

Following the cleaning operation, the metal specimens were initially weighed and then partially dissolved in high purity acid (Ultrex) for radiochemical analyses. The stainless steel samples were immersed in hot (80-90°), 6N Ultrex hydrochloric acid for 10-20 minutes. The samples were then rinsed with doubly-distilled-deionized water, dried, and re-weighed to determine the amount of metal dissolved in the acid solution. The acid was then diluted with high purity water to give a final stock solution of exactly 100 ml in 3N HCl, and the samples stored in cleaned polyethylene bottles. Aliquots of this solution were then taken for gamma spectrometric analysis and destructive radiochemical analyses. The partial dissolution of the Inconel specimens was identical to that for the stainless steel, except that several drops of hydrofluoric acid were added during the acid leaching to aid in the dissolution of niobium constituents, and for preservation of these solutions during storage. The Zircaloy specimens were partially dissolved as described for the Inconel samples, except that a total of 3 to 5 ml of hydrofluoric acid was gradually added during the acid leaching to aid in the sample dissolution and preservation of the zirconium solutions. The HCl/HF acid dissolution of the Inconel and Zircaloy specimens was conducted in cleaned teflon beakers to avoid etching of glass containers by the HF.

The following radiochemical analyses were performed on aliquots of the stock solutions as described below for measurement of ^{54}Mn, ^{55}Fe, ^{60}Co, ^{59}Ni, ^{63}Ni, ^{94}Nb, and ^{125}Sb. Eventually, ^{14}C, ^{90}Sr, ^{99}Tc, ^{129}I, ^{237}Np, ^{238,239,240}Pu, ^{242,244}Cm and ^{241}Am will also be determined.

3.2.2 Stable Element Measurements

Elemental analyses of the 3N hydrochloric acid stock solution of activated metals was accomplished by inductively coupled argon plasma atomic emission spectrometry (ICAP/AES). Appropriate dilutions (10 or 100-fold) of the original stock solutions and reagent blanks were analyzed in a shielded ICAP system.

The ICP is an argon plasma formed by the interaction of an RF field and an inert argon gas stream. This spatially stable plasma is reported to reach a temperature as high as 10,000 °K. This high temperature and inert argon atmosphere minimize chemical interferences such as refractory oxide formations with aluminum and rare earths which are encountered in flame emission methods. The argon carrier gas nebulizes the liquid sample into the spray chamber. It also transports the smaller sample droplets into the center of the plasma. The high temperature in the plasma desolvates the droplets and dissociates the sample material into individual atoms and ions which are excited to emit light at wavelengths characteristic of the elements in the sample. The atomic emission spectrometer (AES) sorts the various wavelengths and measures the intensity of specific spectral lines. The photomultiplier tubes convert the emitted light to an electrical signal which is proportional to the intensity of the spectral lines. The digitized signals are converted by the computer into mg/L units which are printed directly on the input/output terminal.

Three ICP/AES systems are used for various analysis. A Jarrell-Ash Model 95-965 direct reader spectrometer with the capability of determining up to 40 elements simultaneously has the source stand isolated in a hood, and thus allows the analysis of samples containing low levels of radioactivity. An ARL Model 35000 vacuum system for the simultaneous determination of 37 elements is also utilized. A third ARL Model 35800 instrument has the source mounted inside a lead-shielded glovebox. This ICP/AES is used for the analysis of samples containing high levels of radioactive isotopes.

In ICP/AES analyses, spectral interferences from the major elements in the samples (e.g. Fe, Cr, and Ni in SS) is a potential source of error in the determination of trace elements. Correction on the trace elements are performed by analyzing different concentrations of single element standards of the major constituents in the sample at the time of sample analysis and these values are used for spectral corrections of the trace elements.

A fourth plasma system was used to measure niobium at extremely low concentrations in highly diluted samples. This instrument, a VG Plasmaquad inductively coupled plasma mass spectrometer (ICP/MS) is capable of measuring part-per-billion concentrations of niobium, as well as many other elements.

3.3 Results of Radiochemical and Elemental Analyses

The radionuclide concentrations measured in spent fuel assembly hardware materials from the Westinghouse, Combustion Engineering, and General Electric fuel assemblies are presented in Tables 3.5, 3.6, and 3.7, respectively, in units of curies per gram (Ci/g) of metal. The concentrations were decay corrected to the discharge dates when the assemblies were removed from service. The stable element concentrations measured in these same specimens are given in Tables 3.8, 3.9 and 3.10.

The most abundant long-lived radionuclide directly detectable by gamma-ray spectrometry in the stainless steel components after about six years from discharge was ^{60}Co, with much smaller amounts of ^{54}Mn being present. The Inconel components contained about the same concentrations of stable cobalt as the stainless steel in each assembly, and therefore the ^{60}Co levels were also comparable. Following radiochemical separations, the concentrations of ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{93m}Nb, and ^{94}Nb were readily detectable in the stainless steel and Inconel.

The Inconel contained percent levels of stable niobium, and ^{94}Nb could often be detected directly by gamma-ray spectrometry in the presence of the relatively large amounts of ^{60}Co. However, because ^{94}Nb is an important long-lived activation product specified on 10CFR61, selected samples of stainless steel, Inconel, and Zircaloy from each fuel assembly were subjected to special radiochemical and elemental analyses to improve the accuracy and precision of the initial measurements.

TABLE 3.5 Radionuclide Concentrations in Westinghouse Spent Fuel Assembly Hardware Materials (Point Beach Station)

Sample No.	Material	Location	Concentration (Ci/g metal)[a]						
			^{54}Mn[b]	^{55}Fe	^{59}Ni	^{63}Ni	^{60}Co	^{94}Nb	^{93m}Nb[c]
W-10	Inconel	holddown spring @ top end	(5.73±3.08)E-5	(1.84±0.02)E-3	(1.18±0.01)E-5	(2.70±0.03)E-3	(7.34±0.07)E-4	(7.39±1.17)E-7	(5.48±0.55)E-5
W-12	SS	upper end fitting (top)	(1.79±0.48)E-4	(3.67±0.04)E-3	(1.66±0.01)E-6	(3.51±0.04)E-4	(1.63±0.02)E-3	(4.09±0.69)E-10	(7.36±0.74)E-8
W-9	SS	upper end fitting casting (bottom)	(6.39±3.89)E-4	(2.87±0.02)E-2	(1.31±0.01)E-5	(3.05±0.04)E-3	(1.30±0.01)E-2	(1.79±0.37)E-9	(2.39±0.24)E-7
W-8	Inconel	spacer grid #7	<3.5E-4	(1.29±0.01)E-2	(7.52±0.01)E-5	(1.96±0.02)E-2	(1.31±0.01)E-2	(6.98±1.17)E-6	(6.06±0.61)E-4
W-7	Inconel	spacer grid #6	(5.50±1.14)E-3	(5.31±0.05)E-2	(2.55±0.03)E-4	(6.51±0.04)E-2	(4.07±0.04)E-2	--	--
W-6	Inconel	spacer grid #5	(8.52±0.21)E-3	(5.53±0.05)E-2	(3.50±0.04)E-4	(8.49±0.08)E-2	(7.47±0.07)E-2	--	--
W-5	Inconel	spacer grid #4	(3.76±2.33)E-3	(6.27±0.05)E-2	(3.27±0.04)E-4	(8.80±0.08)E-2	(8.82±0.09)E-2	(1.20±0.32)E-4	(2.81±0.28)E-2
W-4	Inconel	spacer grid #3	(3.90±2.36)E-3	(6.45±0.05)E-2	(2.76±0.03)E-4	(7.99±0.08)E-2	(8.01±0.08)E-2	--	--
W-3	Inconel	spacer grid #2	(3.52±1.68)E-3	(6.92±0.05)E-2	(3.35±0.03)E-4	(8.89±0.08)E-2	(8.03±0.08)E-2	--	--
W-2	Inconel	spacer grid #1	<1.1E-3	(3.14±0.02)E-2	(1.35±0.01)E-4	(3.74±0.04)E-2	(3.05±0.03)E-2	(4.14±0.77)E-5	(8.45±0.85)E-3
W-11	SS	bottom end fitting (top)	(3.52±0.33)E-3	(4.75±0.04)E-2	(1.48±0.01)E-5	(3.75±0.04)E-3	(1.28±0.01)E-2	(6.40±0.81)E-9	(8.07±0.81)E-7
W-1	SS	bottom end fitting (bottom)	(2.26±0.69)E-3	(4.21±0.04)E-2	(1.00±0.01)E-5	(2.86±0.03)E-3	(1.83±0.02)E-2	(2.07±0.36)E-8	(2.67±0.27)E-6

(a) Decay corrected to discharge date of 10/8/81.
(b) Parent element is Fe. ^{54}Mn is produced by the fast neutron reaction ^{54}Fe(n,p)^{54}Mn.
(c) Parent element is Nb. ^{93m}Nb is predominantly produced by the reaction ^{93}Nb(n,n')^{93m}Nb.

TABLE 3.6 Radionuclide Concentrations in Combustion Engineering Spent Fuel Assembly Hardware Materials (Calvert Cliffs Station)

Sample No.	Material	Location	Concentration (Ci/g metal)[a]							
			^{54}Mn[b]	^{55}Fe	^{59}Ni	^{63}Ni	^{60}Co	^{94}Nb	^{93m}Nb[c]	^{125}Sb[d]
CE-25	SS	upper holddown plate	(2.92±0.15)E-4	(3.25±0.08)E-3	(1.09±0.02)E-6	(1.64±0.03)E-4	(6.53±0.13)E-4	(4.77±0.81)E-10	(7.70±0.77)E-8	(1.99±0.82)E-6
CE-26	Inconel	upper holddown spring	(4.27±2.37)E-5	(1.71±0.03)E-3	(1.38±0.03)E-5	(4.00±0.08)E-3	(8.79±0.18)E-4	(2.85±0.50)E-7	(3.18±0.32)E-5	(3.70±1.38)E-6
CE-24	SS	upper flow plate	(7.70±1.35)E-4	(3.29±0.08)E-2	(7.12±0.18)E-6	(1.63±0.03)E-3	(7.98±0.33)E-3	(4.35±0.73)E-9	(9.20±0.92)E-7	<7.4E-6
CE-10	Zircaloy	top spacer grid	(6.77±0.89)E-6	(2.83±0.06)E-4	(2.40±0.24)E-9	(6.97±1.10)E-7	(1.32±0.3)E-4	--	--	(1.36±0.14)E-4
CE-9	Zircaloy	spacer grid #7	(1.79±0.06)E-4	(1.17±0.02)E-3	(1.75±0.10)E-8	(5.94±0.49)E-6	(9.03±0.18)E-5	(1.68±0.26)E-8	(1.95±0.20)E-5	(2.46±0.25)E-3
CE-8	Zircaloy	spacer grid #6	(1.80±0.07)E-4	(1.77±0.04)E-3	(1.35±0.12)E-8	(4.50±0.50)E-6	(1.04±0.03)E-4	--	--	(2.49±0.25)E-3
CE-7	Zircaloy	spacer grid #5	(2.02±0.10)E-4	(1.81±0.04)E-3	(3.89±0.18)E-8	(8.80±0.70)E-6	(1.40±0.03)E-4	--	--	(2.68±0.27)E-3
CE-6	Zircaloy	spacer grid #4	(2.06±0.08)E-4	(1.45±0.03)E-3	(1.23±0.10)E-8	(3.59±0.43)E-6	(1.09±0.03)E-4	--	--	(2.29±0.23)E-3
CE-5	Zircaloy	spacer grid #3	(2.65±0.14)E-4	(3.63±0.06)E-3	(3.54±0.22)E-8	(8.38±1.17)E-6	(1.67±0.04)E-4	(2.97±0.58)E-8	(3.64±0.36)E-5	(3.50±0.24)E-3
CE-4	Zircaloy	spacer grid #2	(2.19±0.11)E-4	(2.17±0.06)E-3	(1.77±0.21)E-8	(3.00±0.69)E-6	(1.19±0.03)E-4	--	--	(1.68±0.17)E-3
CE-3	Zircaloy	spacer grid #2	(1.54±0.05)E-4	(1.16±0.02)E-3	(1.08±0.14)E-8	(2.23±0.32)E-6	(8.65±0.19)E-5	(2.09±0.45)E-8	(1.63±0.16)E-5	(1.56±0.16)E-3
CE-2	Inconel	bottom spacer grid	(9.70±0.75)E-4	(1.04±0.03)E-2	(1.63±0.05)E-4	(4.50±0.14)E-2	(4.23±1.07)E-2	(2.26±0.39)E-5	(4.46±0.45)E-3	<4.6E-5
CD-14	SS	bottom retention plate	(8.69±0.12)E-3	(1.18±0.05)E-1	(5.54±0.17)E-5	(1.59±0.05)E-2	(6.18±0.49)E-2	(1.59±0.27)E-8	(2.17±0.22)E-6	<7.0E-5
CE-1	SS	bottom end fitting near axial middle	(2.98±0.48)E-3	(1.03±0.03)E-1	(2.28±0.68)E-5	(6.34±1.83)E-3	(3.10±0.46)E-2	(4.68±0.81)E-9	(1.21±0.12)E-6	<2.9E-5

(a) Decay corrected to discharge data of 4/17/82.
(b) Parent element is iron (Fe). ^{54}Mn is formed by the fast neutron reaction $^{54}Fe(n,p)^{54}Mn$.
(c) Parent element is Nb. ^{93m}Nb is predominantly produced by the reaction $^{93}Nb(n,n)^{93m}Nb$.
(d) Parent element of ^{125}Sb is Sn. ^{125}Sb is formed by the thermal neutron reaction $^{124}Sn(n,\gamma)^{125}Sn$ followed by beta decay to ^{125}Sb.

TABLE 3.7 Radionuclide Concentrations in General Electric Spent Fuel Assembly Hardware Materials (Cooper Station)

Sample	Material	Location	Concentration--Ci/g metal[a]							
			^{54}Mn	^{55}Fe	^{59}Ni	^{63}Ni	^{60}Co	^{94}Nb	^{93m}Nb[c]	^{125}Sb[d]
GE-19	SS	Tab on handle	(1.10±0.30)E-4	(1.43±0.08)E-2	(3.68±0.55)E-6	(8.01±0.95)E-4	(5.95±0.06)E-4	(7.61±1.26)E-10	(2.49±0.26)E-7	(4.23±1.34)E-6
GE-18	SS	Upper tie plate	(1.15±0.32)E-4	(8.79±0.30)E-3	(5.00±0.68)E-6	(8.86±0.90)E-4	(5.41±0.05)E-4	(3.89±0.63)E-9	(9.04±0.95)E-7	<1.9E-6
GE-17	Inconel	Expansion spring (at top of fuel pin)	<9.5E-5	(2.42±0.08)E-3	(4.73±0.68)E-5	(9.48±1.04)E-3	(1.88±0.01)E-3	(5.95±0.99)E-7	(3.98±0.42)E-5	(1.44±0.37)E-5
GE-15	Zircaloy	Spacer grid #7	(3.53±0.95)E-5	(8.41±0.53)E-4	(2.77±0.34)E-7	(3.10±0.27)E-5	(2.42±0.01)E-4	(5.09±1.13)E-9	(2.36±0.25)E-5	(4.16±0.02)E-3
GE-13	Zircaloy	Spacer grid #6	(1.58±0.53)E-4	(2.59±0.08)E-3	(1.07±0.16)E-6	(1.07±0.12)E-4	(1.68±0.01)E-3	--	--	(1.41±0.01)E-3
GE-11	Zircaloy	Spacer grid #5	(1.49±0.49)E-4	(1.87±0.03)E-3	(8.56±1.31)E-7	(8.10±0.19)E-5	(9.20±0.87)E-3	--	--	(1.02±0.01)E-3
GE-9	Zircaloy	Spacer grid #4	(2.14±0.73)E-4	(3.31±0.15)E-3	(9.41±1.31)E-7	(1.39±0.10)E-4	(1.68±0.02)E-3	--	--	(1.81±0.02)E-3
GE-7	Zircaloy	Spacer grid #3	(1.61±0.33)E-4	(3.52±0.21)E-3	(1.35±0.16)E-6	(1.91±0.20)E-4	(7.32±0.07)E-4	(9.14±4.28)E-9	--	(1.72±0.02)E-3
GE-5	Zircaloy	Spacer grid #2	(2.02±0.27)E-4	(2.46±0.08)E-3	(5.45±0.81)E-7	(7.06±1.52)E-5	(4.68±0.02)E-4	--	--	(1.33±0.01)E-3
GE-3	Zircaloy	Spacer grid #1 (starting @ bottom)	(1.01±0.20)E-4	(2.55±0.06)E-3	(2.16±0.28)E-7	(2.63±0.31)E-5	(2.28±0.03)E-4	(9.86±1.89)E-9	(1.50±0.16)E-5	(1.22±0.02)E-3
GE-1	SS	Bottom end fitting (near top of casting)	<4.3E-3	(1.35±0.07)E-1	(8.15±1.13)E-5	(7.54±0.76)E-3	(6.24±0.07)E-2	(7.97±1.35)E-8	(6.86±0.72)E-6	<1.7E-4
GE-2	SS	Bottom end fitting (near nozzle end)	<5.8E-4	(1.27±0.09)E-2	(1.14±0.13)E-5	(1.28±0.13)E-3	(7.03±0.07)E-3	(6.04±0.99)E-9	(4.84±0.51)E-7	<2.3E-5

(a) Decay corrected to discharge date of 5/1/82.
(b) Parent element is Fe. ^{54}Mn is produced by the fast neutron reaction ^{54}Fe(n,p)^{54}Mn.
(c) Parent element is Nb. ^{93m}Nb is produced by the reaction ^{93}Nb(n,n')^{93m}Nb.
(d) Parent element is Sn. ^{125}Sb is produced by the reaction ^{124}Sn(n,γ)^{125}Sn, followed by beta decay to ^{125}Sb.

TABLE 3.8 Elemental Concentrations in Westinghouse Spent Fuel Assembly Hardware Materials (Point Beach Station)

Sample No.	Material	Location	Concentration--Weight Percent							
			Mn	Fe	Cr	Ni	Co	Nb	Cu	Mo
W-10	Inconel	holddown spring @ top end	0.0629±0.004	17.2±0.5	17.6±0.5	50.8±1.5	0.0709±0.009	4.50±0.05	0.062±0.006	2.86±0.08
W-12	SS	upper end fitting (top)	1.52±0.05	67.5±2.0	18.3±0.5	8.27±0.25	0.150±0.015	0.0033±0.0007	0.10±0.010	0.41±0.01
W-9	SS	upper end fitting casting (bottom)	1.44±0.04	65.0±2.0	18.0±0.5	9.10±0.25	0.149±0.015	0.0024±0.0024	0.095±0.009	0.23±0.01
W-8	Inconel	spacer grid #7	0.133±0.008	16.0±0.5	16.6±0.5	53.0±1.6	0.148±0.015	4.40±0.05	0.18±0.02	2.79±0.08
W-7	Inconel	spacer grid #6	0.133±0.008	15.2±0.5	15.7±0.5	46.5±1.4	0.089±0.009	--	0.18±0.02	2.66±0.08
W-6	Inconel	spacer grid #5	0.069±0.002	16.0±0.5	17.0±0.5	48.0±1.4	0.115±0.012	--	0.078±0.008	2.83±0.08
W-5	Inconel	spacer grid #4	0.073±0.002	17.0±0.5	17.8±0.5	50.8±1.5	0.131±0.013	4.60±0.05	0.085±0.008	2.98±0.08
W-4	Inconel	spacer grid #3	0.062±0.002	16.2±0.5	16.2±0.5	47.1±1.4	0.119±0.012	--	0.075±0.008	2.65±0.08
W-3	Inconel	spacer grid #2	0.068±0.006	16.5±0.5	17.4±0.5	49.2±1.5	0.130±0.017	--	0.091±0.009	2.87±0.08
W-2	Inconel	spacer grid #1	0.113±0.006	15.8±0.5	17.4±0.5	49.3±1.5	0.111±0.011	4.60±0.05	0.172±0.017	2.67±0.08
W-11	SS	bottom end fitting (top)	1.71±0.05	65.5±2.0	18.7±0.5	9.36±0.25	0.115±0.035	0.03±0.03	0.094±0.009	0.23±0.01
W-1	SS	bottom end fitting (bottom)	1.54±0.05	65.0±2.0	17.6±0.5	7.96±0.24	0.147±0.015	0.0156±0.0022	0.125±0.013	0.25±0.01

TABLE 3.9 Elemental Concentrations in Combustion Engineering Spent Fuel Assembly Hardware Materials (Calvert Cliffs Station)

Sample No.	Type	Location	Concentration--Weight Percent								
			Mn	Fe	Cr	Ni	Co	Nb	Sn	Mo	Zr
CE-25	SS	upper holddown plate		69.7±2.1	21.2±0.8	9.72±0.39	0.0513±0.004	<0.0015	<0.018		
CE-26	Inconel	upper holddown spring		8.00±0.33	15.5±0.6	72.9±2.2	0.0316±0.003	2.25±0.03	0.27±0.09		
CE-24	SS	upper flow plate		60.5±1.8	16.2±0.7	8.52±0.33	0.0927±0.008	<0.0015	<0.018		
CE-10	Zircaloy	top spacer grid		0.212±0.009	0.11±0.01	<0.005	<0.0027	--	2.2±0.2		91.8±2.9
CE-9	Zircaloy	spacer grid #7		0.213±0.008	0.11±0.01	<0.005	<0.0027	0.0156±0.0009	2.3±0.2		89.0±2.8
CE-8	Zircaloy	spacer grid #6		0.211±0.006	0.11±0.01	<0.005	<0.0027	--	2.4±0.2		91.7±2.7
CE-7	Zircaloy	spacer grid #5		0.229±0.011	0.11±0.01	<0.005	<0.0027	--	2.6±0.3		92.0±2.8
CE-6	Zircaloy	spacer grid #4		0.220±0.018	0.11±0.01	<0.005	<0.0027	--	3.0±0.5		92.5±3.0
CE-5	Zircaloy	spacer grid #3		0.200±0.020	0.13±0.01	<0.005	<0.0027	0.0259±0.0017	3.9±1.0		115±3.5
CE-4	Zircaloy	spacer grid #2		0.260±0.028	0.10±0.004	<0.005	<0.0027	--	3.4±1.0		106±3.9
CE-3	Zircaloy	spacer grid #1		0.182±0.017	0.078±0.023	<0.005	<0.0027	0.0147±0.0009	2.0±0.4		73.5±2.5
CE-2	Inconel	bottom spacer grid	0.093±0.009	2.49±0.07	13.8±0.4	36.6±1.1	0.114±0.029	2.25±0.03	--	14.5±0.4	--
CE-14	SS	bottom retention plate	1.08±0.03	60.8±1.8	18.2±0.5	9.60±0.3	0.128±0.010	<0.0091	--	0.032±0.005	--
CE-1	SS	bottom end fitting near axial middle	1.10±0.03	67.1±2.0	18.8±0.5	9.84±2.95	0.148±0.022	<0.02	--	0.050±	--

TABLE 3.10 Elemental Concentrations in General Electric Spent Fuel Assembly Hardware Materials (Cooper Station)

Sample	Material	Location	Concentration--Weight Percent									
			Mn	Fe	Cr	Ni	Co	Nb	Sn	Cu	Mo	Zr
GE-19	SS	Tab on handle	0.58±0.02	68.3±2.1	19.3±0.1	8.38±0.25	0.026±0.004	0.001±0.00067	<0.22	0.013±0.002	<0.007	<0.004
GE-18	SS	Upper tie plate (at base where fuel pin touches)	0.56±0.02	67.6±2.0	19.0±0.1	8.26±0.25	0.025±0.005	0.00074±0.00016	<0.21	0.009±0.002	<0.006	<0.003
GE-17	Inconel	Expansion spring (at top of fuel pin)	0.070±0.003	6.04±0.18	13.2±0.4	68.2±2.0	0.050±0.008	0.81±0.01	<0.26	0.027±0.003	0.049±0.007	0.094±0.009
GE-15	Zircaloy	Spacer grid No. 7	0.011±0.005	0.47±0.01	0.14±0.02	0.072±0.011	<0.006	0.0051±0.0032	1.09±0.18	<0.004	0.072±0.011	93.1±2.8
GE-13	Zircaloy	Spacer grid No. 6	0.02±0.006	0.43±0.02	0.14±0.02	0.083±0.012	<0.007	--	1.13±0.20	<0.005	0.083±0.012	96.5±2.9
GE-11	Zircaloy	Spacer grid No. 5	0.013±0.005	0.45±0.01	0.10±0.02	0.066±0.010	<0.006	--	0.86±0.17	<0.004	<0.02	72.3±2.2
GE-9	Zircaloy	Spacer grid No. 4	0.02±0.005	0.62±0.02	0.14±0.02	0.088±0.013	<0.007	--	1.09±0.20	<0.005	0.03±0.02	94.9±2.8
GE-7	Zircaloy	Spacer grid No. 3	0.009±0.005	0.47±0.02	0.13±0.02	0.040±0.009	<0.006	0.020±0.003	1.10±0.20	<0.005	<0.02	95.6±2.9
GE-5	Zircaloy	Spacer grid No. 2	0.01±0.005	0.55±0.02	0.13±0.02	0.043±0.009	<0.007	--	1.06±0.20	<0.004	<0.02	94.3±2.8
GE-3	Zircaloy	Spacer grid No. 1 (starting at bottom)	0.008±0.004	0.56±0.02	0.12±0.01	0.030±0.010	<0.007	0.018±0.003	1.10±0.22	<0.003	0.02±0.02	92.4±2.8
GE-1	SS	Bottom end fitting (near top of casting)	1.01±0.03	69.9±2.0	17.5±0.5	8.68±0.26	0.21±0.02	0.037±0.011	<0.3	0.28±0.01	0.37±0.03	0.006±0.003
GE-2	SS	Bottom end fitting (near nozzle end)	1.03±0.03	69.1±2.0	17.4±0.5	8.75±0.26	0.207±0.02	0.018±0.003	<0.2	0.29±0.01	0.87±0.04	<0.003

The improved ^{94}Nb measurements are given in Tables 3.5, 3.6, and 3.7. The highest ^{94}Nb concentrations were associated with the Inconel specimens, particularly the spacer grids from the fueled region of the Westinghouse assembly. These specimens contained up to 1.2E-4 Ci/g of ^{94}Nb in the Inconel, a reflection of the nominal 4% niobium content of this alloy. The stable niobium was measured in diluted aliquots of the acid-digest solutions of the activated metal specimens by extremely sensitive inductively coupled plasma mass spectrometry.

The radiochemically separated niobium was also counted on an intrinsic germanium (IG) detector set up at 0.2 kev/channel to measure ^{93m}Nb, which was present in surprisingly high concentrations. To our knowledge, these are the first measurements of this radionuclide in activated metal components. This radionuclide is produced in the metal specimens primarily by the reaction ^{93}Nb (n,n') ^{93m}Nb. The ^{93m}Nb decays by emission of a 30 kev gamma-ray which is essentially all converted. The predominant external radiation emitted by ^{93m}Nb is, therefore, due to the 16.5 kev Nb x-rays. Previously calculated concentrations of ^{93m}Nb and ^{94}Nb in the neutron activated stainless steel shroud from a reference BWR estimated that the $^{93m}Nb/^{94}Nb$ ratio would be 0.09.[8] For the 13 specimens of stainless steel components from the Westinghouse, Combustion Engineering, and General Electric spent fuel assemblies the average $^{93m}Nb/^{94}Nb$ ratio was 158±74. Thus, the actual measured ^{93m}Nb in neutron activated stainless steel is some 1800 times higher than predicted by calculations. Although this new finding will probably not affect the waste classification or disposal requirements for activated metals, an environmental dose assessment should be conducted to insure that ^{93m}Nb will not be an environmental problem.

The ^{63}Ni and ^{59}Ni concentrations were highest in the Inconel and stainless steel components, where the stable nickel concentrations were usually in the range of 36-72% and 7-9%, respectively. The Westinghouse fuel assembly, which contained Inconel spacer grids (~50% Ni) in the fueled region of the assembly, had the highest observed ^{63}Ni and ^{59}Ni concentrations, averaging (6.63±2.75)E-2 and (2.50±1.06)E-4 Ci/g metal, respectively. The ^{63}Ni concentrations were very similar in magnitude to the observed ^{60}Co concentrations in the spacer grids. The ^{63}Ni and ^{59}Ni concentrations in the Zircaloy spacer girds in the Combustion Engineering and General Electric fuel assemblies were several orders of magnitude lower compared to the Westinghouse assembly.

The ^{55}Fe concentrations were very similar in magnitude to the ^{60}Co concentrations in all materials from each fuel assembly. Iron was a significant constituent of the Inconel components (2.5-17%), and ranged from 61-69% in the stainless steel specimens sampled from all assemblies. However, both the iron and cobalt concentrations in the Zircaloy were very low, resulting in relatively low concentrations of ^{55}Fe and ^{60}Co compared to the Inconel and stainless steel components.

The ^{54}Mn was produced by the fast neutron reaction ^{54}Fe (n,p) ^{54}Mn, and its production is a reflection of the iron content of the parent materials and the fast neutron flux. Generally, the ^{54}Mn concentrations in the various materials were near or slightly below the ^{60}Co concentrations at the time of discharge of the fuel assemblies. Because ^{54}Mn has a relatively short half-life (0.854 yr), it will become a minor constituent in the activated metal specimens after a few years.

Antimony-125 was a major constituent of teh Zircaloy-4 spacer grids used in the Conbustion Engineering and General Electric fuel assemblies. The ^{125}Sb is produced from tin by the reaction ^{124}Sn (n,γ) ^{125}Sn followed by beta decay of the ^{125}Sn to ^{125}Sb. Zircaloy-4 contains about 1-3 tin. Since the ^{125}Sb half-life is only 2.73 years it will decay relatively fast compared to ^{60}Co.

3.4 Radionuclide Scaling Factors for Activated Metal Components

This study has provided one of the few opportunities to systematically measure the long-lived 10CFR61 radionuclides produced in activated metal components from within reactor pressure vessels. Because many of these radionuclides are very difficult to measure, it is desirable to determine if useful correlations exist between the difficult-to-measure radionuclides (^{55}Fe, ^{59}Ni, ^{63}Ni, ^{93m}Nb and ^{94}Nb) and ^{60}Co which is easily measured by gamma-ray spectrometry. If appropriate correlations exist, then scaling factors (relative to ^{60}Co) could be used to estimate their concentrations by multiplying the easily measured ^{60}Co concentrations by the empirically determined scaling factors. Table 3.11 presents the empirical scaling factors determined for stainless steel, Inconel, and Zircaloy components from the three fuel assemblies. In general, the activity correlations are quite good, indicating that the use of scaling factors for estimating radionuclide concentrations of the long-lived, difficult-to-measure 10CFR61 radionuclides in activated metals may be entirely appropriate. The $^{55}Fe/^{60}Co$, $^{59}Ni/^{60}Co$, and $^{63}Ni/^{60}Co$ scaling factors for the fuel assembly hardware components were particularly good. The scaling factors for the BRW assembly hardware generally had larger uncertainties compared to the PWR assemblies. Although the variability of the $^{94}Nb/^{60}Co$ and $^{93m}Nb/^{60}Co$ scaling factors were generally somewhat higher than the other scaling factors, they still appear to be useful for estimating the ^{94}Nb and ^{93m}Nb concentrations in activated metal components.

The $^{59}Ni/^{63}Ni$ was also sufficiently constant that a generic scaling factor for all activated metal components from all assemblies would seem reasonable. An overall $^{59}Ni/^{63}Ni$ scaling factor of 0.00524 ± 0.00227 was obtained for all samples listed in Tables 3.5, 3.6 and 3.7.

3.5 Classification of Spent Fuel Assembly Hardware with Respect to 10CFR61

The licensing requirements for shallow land disposal of radioactive waste, 10CFR61, specifies three classes of waste, A, B and C that are permissible for disposal in commercial low-level waste disposal facilities.[2] Recently, the rule has been amended to require that all waste greater than Class C be disposed in a high-level waste repository.[6] It is therefore critical to carefully assess the radionuclide concentrations in spent fuel assembly hardware and other highly activated internal components of reactor pressure vessels in an effort to seek ways to minimize the volume of greater-than-Class C waste destined for repository burial.

TABLE 3.11 Activity Scaling Factors for Activation Products in Spent Fuel Assembly Hardware Materials

	Average Activity Scaling Factors[a]						
	Westinghouse		Combustion Engineering			General Electric	
Ratio	Stainless Steel	Inconel	Stainless Steel	Inconel	Zircaloy	Stainless Steel	Zircaloy
$^{54}Mn/^{60}Co$	(1.39±0.096)E-1	(7.70±3.90)E-2	(2.14±1.60)E-1	(3.94±1.31)E-2	(1.54±0.62)E0	(2.62±1.37)E0	(4.02±3.17)E0
$^{55}Fe/^{60}Co$	(2.62±0.73)E0	(1.13±0.64)E0	(3.91±0.83)E0	(1.10±1.19)E0	(1.39±0.57)E1	(1.10±1.09)E1	(4.07±3.63)E0
$^{59}Ni/^{60}Co$	(9.30±2.60)E-4	(6.07±4.16)E-3	(1.03±0.44)E-3	(9.68±8.51)E-3	(1.52±0.77)E-4	(4.59±3.82)E-3	(9.11±5.55)E-4
$^{63}Ni/^{60}Co$	(2.25±0.56)E-1	(1.53±0.89)E0	(2.24±0.29)E-1	(2.78±2.50)E0	(3.89±2.06)E-2	(8.23±7.86)E-1	(1.15±0.79)E-1
$^{94}Nb/^{60}Co$	(5.05±4.43)E-7	(1.07±0.39)E-3	(4.16±2.70)E-7	(4.16±1.30)E-4	(2.02±0.34)E-2	(2.76±3.25)E-6	(1.45±2.48)E-3
$^{93m}Nb/^{60}Co$	(6.82±5.50)E-5	(1.79±1.39)E-1	(7.59±4.70)E-5	(6.81±4.51)E-2	(2.07±0.17)E-1	(5.67±7.52)E-4	(8.17±2.24)E-2
$^{59}Ni/^{63}Ni$	(4.12±0.52)E-3	(3.85±0.29)E-3	(4.53±1.47)E-3	(3.54±0.12)E-3	(4.02±1.02)E-3	(7.49±2.87)E-3	(8.47±1.44)E-3
$^{93m}Nb/^{94}Nb$	(1.43±0.25)E2	(1.50±0.81)E2	(1.92±0.54)E2	(1.55±0.60)E2	(1.06±0.24)E3	(1.81±1.19)E2	(5.83±15.2)E2

(a) Activity at time of discharge.

Note: ± values are 1 σ.

The three long-lived radionuclides which control the waste classification of highly activated metals are ^{63}Ni, ^{59}Ni and ^{94}Nb. However, since the ^{59}Ni limit will never be exceeded without first exceeding the ^{63}Ni limit, the classification controlling radionuclides are ^{63}Ni and ^{94}Nb.

Table 3.12 compares the average concentrations of ^{63}Ni, ^{59}Ni, and ^{94}Nb (in units of Ci/m^3) in various components of spent fuel assembly hardware with the 10CFR61 Class C limit, and Table 3.13 gives the ratio of the measured concentrations to the Class C limit. Any materials exceeding the Class C limits for these radionuclides will need to be disposed in a high level waste repository. As shown in Tables 3.12 and 3.13, the ^{63}Ni, ^{59}Ni, and ^{94}Nb concentrations often exceeded the Class C limit for various components of the spent fuel assembly hardware. The Inconel-718 spacer grids on the Westinghouse assembly and the Inconel-625 bottom spacer grid on the Combustion Engineering assembly were the components which exceeded the Class C limit the most. For example, the Inconel-718 spacer grids on the Westinghouse assembly exceeded the Class C limits for ^{63}Ni and ^{94}Nb by average factors of 86 and 2390, respectively. The Inconel-718 and Inconel-625 contained about 4.5% and 2.3% Nb, respectively, and about 50% and 73% Ni, respectively. Thus, from a radiological waste disposal standpoint, these alloys have the highest concentrations of parent elements which produce the activation products exceeding the Class C limit.

The stainless steel end fittings and hold down flow plates in all cases except the bottom end fittings on the General Electric assembly only exceeded the Class C limit for ^{63}Ni. Stainless steel contains about 8 to 9.3% nickel. The ^{94}Nb concentrations in the stainless steel end fittings and hold down/flow plates were often close to the Class C limit and did slightly exceed the limit for the General Electric bottom end fitting.

The Zircaloy-4 spacer grids in the Combustion Engineering and General Electric did not exceed the Class C limits for ^{63}Ni, ^{59}Ni, or ^{94}Nb, although they come close to the ^{94}Nb limit. From a radiological standpoint, nuclear grade Zircaloy-4 is a very desirable material because it contains very low concentrations of the parent elements which produce the long-lived radionuclides of concern.

For future waste disposal considerations, it would be expeditious for fuel assembly vendors to consider alternate materials to replace the Inconel alloys which are high in nickel and niobium concentrations. Although these alloys are used in limited volumes in fuel assembly construction, they have a significant impact on the future radioactive waste disposal options.

4.0 Radionuclide Characterization of Gundremmigen Reactor Pressure Vessel Steel

During the past year it was possible to obtain two specimens of the steel reactor pressure vessel from the decommissioned Gundremmigen KRB-A reactor. The purpose of acquiring these specimens was twofold: 1) to provide real measurements of the concentrations of neutron activation products in a decommissioned reactor pressure vessel, and to provide a comparison with 10CFR61 waste classification levels, and 2) to compare

TABLE 3.12 Average Concentrations of 10CFR61 Radionuclides in Spent Fuel Assembly Hardware Components

	Material	Average Concentration (Ci/m^3) ^{63}Ni	^{59}Ni	^{94}Nb
Westinghouse				
Upper end fittings	SS-304	1.34E4	5.83E1	8.69E-3
Bottom end fittings	SS-304	2.61E4	9.80E1	1.07E-1
Spacer grids	Inconel-718	5.63E5	2.13E3	4.77E2
Upper holddown spring	Inconel-718	2.30E4	1.00E2	6.28E-0
Combustion Engineering				
Upper holddown & flow plates	SS-304	7.09E2	3.24E1	1.90E-2
Bottom end fitting & retention plate	SS-304	8.78E4	3.09E2	8.13E-2
Spacer grids	Zircaloy-4	3.02E1	1.21E-1	1.46E-1
Upper holddown spring	Inconel-625	3.40E4	1.17E2	2.42E1
Bottom spacer grid	Inconel-625	3.83E5	1.39E3	1.92E2
General Electric				
Upper handle & tie plate	SS-304	6.66E3	3.43E1	1.84E-2
Upper expansion spring	Inconel-X750	8.06E4	4.01E2	5.06E-0
Spacer grids	Zircaloy-4	3.45E2	4.87E0	6.83E-2
Bottom end fitting	SS-304	3.48E4	3.67E2	3.39E-1
10CFR61 Class C Limit		7.0E3	2.2E2	2.0E-1

TABLE 3.13 Ratio of Measured Radionuclide Concentrations in Spent Fuel Assembly Hardware to Their 10CFR61 Class C Limit

	Material	Ratio: Measured Concentration/Class C Limit ^{63}Ni	^{59}Ni	^{94}Nb
Westinghouse				
Upper end fittings	SS-304	1.91	0.27	0.043
Bottom end fittings	SS-304	3.73	0.45	0.54
Spacer grids	Inconel-718	80.4	9.68	2390
Upper holddown spring	Inconel-718	3.29	0.45	31.4
Combustion Engineering				
Upper holddown & flow plates	SS-304	0.10	0.15	0.095
Bottom end fitting & retention plate	SS-304	12.5	1.40	0.41
Spacer grids	Zircaloy-4	0.0043	5.5E-4	0.73
Upper holddown spring	Inconel-625	4.9	0.53	121
Bottom spacer grid	Inconel-625	54.7	6.3	960
General Electric				
Upper handle & tie plate	SS-304	0.95	0.16	0.092
Upper expansion spring	Inconel-X750	11.5	1.8	25.3
Spacer grids	Zircaloy-4	0.049	0.022	0.34
Bottom end fitting	SS-304	4.97	1.67	1.7

calculated estimates of the activation product concentrations in the pressure vessel with the empirical measurements to determine the accuracy of the calculational methods. This information is of vital importance in reactor decommissioning because it provides an assessment of disposal options and transportation requirements for decommissioned reactor pressure vessels, and provides confidence (or identifies shortcomings) in calculational methods for estimating radionuclide inventories.

The Boiling Water Reactor KRB-A had a nominal thermal power of 801 MW (250 MW electrical). The reactor was put in operation in November 1966 and, until the last shutdown on January 13, 1977, generated a total of about 16 TWh of electrical power, with an average availability of 75 %.

After decommissioning, 15 cores (trepans) of the reactor pressure vessel were taken at different axial and azimuthal positions within the 90 to 135 degree octant of the reactor. The axial and azimuthal positions of the trepans were named A,B,C,D,E,F,G,K,L,M,N,P,Q,R,T.

The two specimens received at PNL were cut from trepan G (115°). Vessel steel from the 0.41T and 0.67T depths (e.g. 41% and 67%, respectively, through the vessel wall referenced to the steel/cladding interface) were cut from a slab of trepan G. The weights of the 0.41T and 0.67T pieces were 7.8793 g. and 9.4835 g., respectively. Each piece was cut into thirds and subjected to radionuclide and elemental analyses described in Sections 3.2.1 and 3.2.2.

The only long-lived gamma-emitting radionuclide present in the samples was ^{60}Co (see Table 4.1). The most abundant radionuclide was ^{55}Fe, which was almost a factor of ten higher in concentration than ^{60}Co. The ^{63}Ni concentrations averaged about 26 times lower than ^{60}Co, and the ^{94}Nb concentrations were below the limit of detection.

Also shown in Table 4.1 is the ratio of the measured radionuclide concentrations to the 10CFR61 Class A limit for disposal in a low level waste shallow land burial facility. It is obvious that these concentrations are well below the Class A limit and the entire pressure vessel (not including internal components) could be disposed as Class A waste. This is consistent with the classification measurements for the Shippingport reactor pressure vessel. Thus, it appears that disposal of commercial station pressure vessels will not pose a serious problem from a radiological standpoint in future reactor decommissioning.

The elemental concentrations of the Gundremmigen reactor pressure vessel steel are shown in Table 4.2. The material is carbon steel and contains less than one percent of Ni, Mn and Cr. The Co content is slightly lower than that observed for U. S. stainless steels, and the Nb was undetectable at less than 0.001 percent.

TABLE 4.1 Concentrations of Neutron Activation Products in Gundremmigen KRB-A Pressure Vessel Steel

Radionuclide	Radiochemically Measured Concentration (Ci/g steel)(A) Sample #3 (0.41T)	Sample #4 (0.67T)	Ratio: Meas. Conc./ Class A Limit
^{60}Co	2.53E-6	1.32E-6	0.028
^{63}Ni	1.14E-7	4.37E-8	0.025
^{55}Fe	2.91E-5	9.25E-6	0.32
^{94}Nb	<2.8E-12	<3.1E-12	<0.00012

(A) Concentrations decay corrected to time of reactor shutdown (January 13, 1977)

TABLE 4.2 Elemental Concentrations in Gundremmigen KRB-A Pressure Vessel Steel

Element	Concentration - Weight Percent Sample #3 (0.41T)	Sample #4 (0.67T)	Avg. Conc.
Fe	92.7	92.7	92.7
Ni	0.813	0.829	0.821
Mn	0.749	0.757	0.753
Cr	0.409	0.406	0.408
Co	0.0339	0.0338	0.0339
Nb	<0.001	<0.001	<0.001

5.0 Comparison of Calculated Versus Measured Radionuclide Concentrations

This project has provided the opportunity to conduct calculated estimates of the concentrations of neutron activation products in various types of reactor pressure vessel and fuel assembly hardware components, and to compare the calculated values with carefully measured radionuclide concentrations. The empirical measurements involved both radionuclide and stable element analyses in order to obtain specific activities of the radionuclides of interest so that material compositional differences would not obviate the comparisons. Comparisons of calculated versus measured neutron activation product concentrations were made for the three fuel assembly hardware components described in Section 3 and for the Gundremmigen KRB-B reactor pressure vessel steel. These comparisons provide a measure of the degree of accuracy of the calculational methods and identify any shortcomings in the predictive methods such as insufficient cross section information, neutron flux, and energy spectrum measurements, etc. It should be stressed that the calculations were conducted completely independent of the measurements, except that the actual elemental concentrations were supplied for the calculations. Thus, this exercise was a true blind comparison.

5.1 Spent Fuel Assembly Hardware

5.1.1 Measured Specific Activities

The specific activities of the long-lived activation products in the components described in Section 3 are given in Tables 5.1, 5.2 and 5.3. The specific activities were reported in units of Ci/g of parent element by dividing the radionuclide concentrations in units of Ci/g of metal by the elemental concentration in units of g element/g metal. This normalizes the induced activities so that geometrical variations in the radionuclide concentrations can be observed. The sampling locations for the three fuel assemblies are shown in Figures 3.1, 3.2 and 3.3. The highest specific activities observed in each fuel assembly hardware were in the materials adjacent to the fueled region of the assemblies. The activities drop off rapidly at each end of the assemblies. The highest specific activity is due to ^{60}Co, generally followed by ^{55}Fe, ^{63}Ni and ^{93m}Nb. The ^{54}Mn specific activity in the Zircaloy spacer girds in the General Electric assembly were relatively high due to the relatively higher fast neutron flux.

5.1.2 Calculated Specific Activities

The calculated radionuclide concentrations were performed by a nuclear engineering group at PNL and details of the method have been published elsewhere[9]. Briefly, The process of calculating the radionuclide concentrations in the activated metal is two-fold. The first step is to calculate a core average inventory based on the irradiation history of the activated metal. This is performed using the ORIGEN2 code. Since the results of the ORIGEN2 calculation are valid for an average over the cores' fueled region only, adjustments need to be made if the results are to be applicable to metals that are activated outside the fueled region (this is more fully explained below). These adjustment factors were calculated using the one-dimensional neutronics code, ANISN. The factors are then applied to the ORIGEN2 results to obtain the calculated radionuclide concentrations for the regions outside of the fueled sections of the assemblies.

5.1.3 Comparison of Calculated Versus Measured Specific Activities

Figures 5.2 through 5.6 show a comparison of the calculated versus measured long-lived neutron activation products in the Westinghouse fuel assembly hardware. The radionuclides for which direct comparisons were made included ^{60}Co, ^{55}Fe, ^{63}Ni, ^{59}Ni, and ^{94}Nb.

TABLE 5.1 Specific Activities of Long-Lived Radionuclides in Westinghouse Spent Fuel Assembly Hardware Materials (Point Beach Station)

Sample No.	Material	Location	Concentration (Ci/g of parent element)[a] ^{54}Mn[b]	^{55}Fe	^{59}Ni	^{63}Ni	^{60}Co	^{94}Nb	^{93m}Nb[c]
W-10	Inconel	holddown spring @ top end	(3.33±1.79)E-4	(1.07±0.03)E-2	(2.32±0.07)E-5	(5.30±0.16)E-3	(1.04±0.13)E0	(1.64±0.26)E-5	(1.22±0.12)E-3
W-12	SS	upper end fitting (top)	(2.65±0.71)E-4	(5.43±0.16)E-3	(2.01±0.06)E-5	(4.25±0.13)E-3	(1.09±0.11)E0	(1.24±0.26)E-5	(2.21±0.47)E-3
W-9	SS	upper end fitting casting (bottom)	(9.83±5.98)E-4	(4.41±0.13)E-2	(1.44±0.04)E-4	(3.35±0.10)E-2	(8.72±0.87)E0	(7.20±7.20)E-5	(9.9±9.9)E-3
W-8	Inconel	spacer grid #7	<2.2E-3	(8.06±0.24)E-2	(1.42±0.04)E-4	(3.70±0.11)E-2	(8.85±0.89)E0	(1.59±0.27)E-4	(1.37±0.14)E-2
W-7	Inconel	spacer grid #6	(3.62±0.75)E-2	(3.49±0.10)E-1	(5.48±0.16)E-4	(1.40±0.04)E-1	(4.57±0.46)E1	--	--
W-6	Inconel	spacer grid #5	(5.33±0.13)E-2	(3.46±0.10)E-1	(7.29±0.22)E-4	(1.77±0.05)E-1	(6.49±0.65)E1	--	--
W-5	Inconel	spacer grid #4	(2.21±0.62)E-2	(3.69±0.10)E-1	(6.44±0.19)E-4	(1.73±0.05)E-1	(6.73±0.67)E1	(2.61±0.70)E-3	(6.12±0.61)E-1
W-4	Inconel	spacer grid #3	(2.41±1.46)E-2	(3.98±0.11)E-1	(5.86±0.18)E-4	(1.70±0.05)E-1	(6.73±0.67)E1	--	--
W-3	Inconel	spacer grid #2	(2.13±1.02)E-2	(4.19±0.12)E-1	(6.81±0.20)E-4	(1.81±0.05)E-1	(6.18±0.62)E1	--	--
W-2	Inconel	spacer grid #1	<6.9E-3	(1.99±0.06)E-1	(2.74±0.08)E-4	(7.59±0.22)E-2	(2.75±0.28)E1	(9.01±1.67)E-4	(1.83±0.18)E-1
W-11	SS	bottom end fitting (top)	(5.37±0.50)E-3	(7.25±0.22)E-2	(1.58±0.05)E-4	(4.01±0.12)E-2	(1.11±0.34)E1	(2.20±2.20)E-5	(2.7±2.7)E-3
W-1	SS	bottom end fitting (bottom)	(3.48±1.06)E-3	(6.48±0.19)E-2	(1.26±0.04)E-4	(3.59±0.11)E-2	(1.24±0.12)E1	(1.32±0.23)E-4	(1.72±0.24)E-2

(a) Decay corrected to discharge date of 10/8/81.
(b) Parent element is iron (Fe). ^{54}Mn is formed by the fast neutron reaction $^{54}Fe(n,p)^{54}Mn$.
(c) Parent element is Nb. ^{93m}Nb is predominantly produced by the reaction $^{93}Nb(n,n')^{93m}Nb$.

TABLE 5.2 Specific Activities of Long-Lived Radionuclides in Combustion Engineering Spent Fuel Assembly Hardware Materials (Calvert Cliffs Station)

Sample No.	Material	Location	Concentration (Ci/g of parent element)[a]						
			^{54}Mn[b]	^{55}Fe	^{59}Ni	^{63}Ni	^{60}Co	^{94}Nb	^{93m}Nb[c]
CE-25	SS	upper holddown plate	(4.19±0.22)E-4	(4.66±0.14)E-3	(1.12±0.02)E-5	(1.68±0.07)E-3	(1.27±0.09)E0	>3.2E-5	<5.1E-3
CE-26	Inconel	upper holddown spring	(5.34±3.40)E-4	(2.14±0.09)E-2	(1.89±0.05)E-5	(5.49±0.16)E-3	(2.78±0.26)E0	(1.27±0.22)E-5	(1.42±0.14)E-3
CE-24	SS	upper flow plate	(1.27±0.23)E-3	(5.44±0.16)E-2	(8.36±0.32)E-5	(1.91±0.06)E-2	(8.61±0.74)E0	>2.9E-4	>6.1E-2
CE-10	Zircaloy	top spacer grid	(3.19±0.42(E-3	(1.33±0.06)E-1	>4.8E-5	>1.4E-2	>4.9E0	--	--
CE-9	Zircaloy	spacer grid #7	(8.40±0.28)E-2	(5.49±0.21)E-1	>3.5E-4	>1.2E-1	>3.3E0	(1.09±0.17)E-4	(1.24±0.04)E-1
CE-8	Zircaloy	spacer grid #6	(8.53±0.33)E-2	(8.39±0.33)E-1	>2.7E-4	>9.0E-2	>3.9E0	--	--
CE-7	Zircaloy	spacer grid #5	(8.82±0.44)E-2	(7.90±0.32)E-1	>7.8E-4	>1.8E-1	>5.2E0	--	--
CE-6	Zircaloy	spacer grid #4	(9.36±0.36)E-2	(6.59±0.26)E-1	>2.5E-4	>7.2E-2	>4.0E0	--	--
CE-5	Zircaloy	spacer grid #3	(1.33±0.07)E-1	(1.82±0.07)E0	>7.1E-4	>1.7E-1	>6.2E0	(1.19±0.23)E-4	(1.40±0.09)E-1
CE-4	Zricaloy	spacer grid #2	(8.42±0.42)E-2	(8.35±0.33)E-1	>3.5E-4	>6.0E-2	>4.4E0	--	--
CE-3	Zricaloy	spacer grid #1	(8.46±0.27)E-2	(6.37±0.25)E-1	>2.2E-4	>4.5E-2	>3.2E0	(1.42±0.30)E-4	(1.11±0.68)E-1
CE-2	Inconel	bottom spacer grid	(3.90±3.02)E-2	(4.18±0.24)E-1	(4.45±0.13)E-4	(1.35±0.04)E-1	(3.71±0.94)E1	(1.00±0.17)E-3	(1.98±0.19)E-1
CE-14	SS	bottom retention plate	(1.43±0.19)E-2	(2.97±0.08)E-1	(5.77±0.17)E-4	(1.66±0.05)E-1	(4.83±0.38)E1	>1.8E-4	>2.4E-2
CE-1	SS	bottom end fitting near axial middle	(4.44±0.72)E-3	(1.52±0.06)E-1	(2.32±0.07)E-4	(6.44±0.19)E-2	(2.09±0.31)E1	>2.3E-5	>2.8E-3

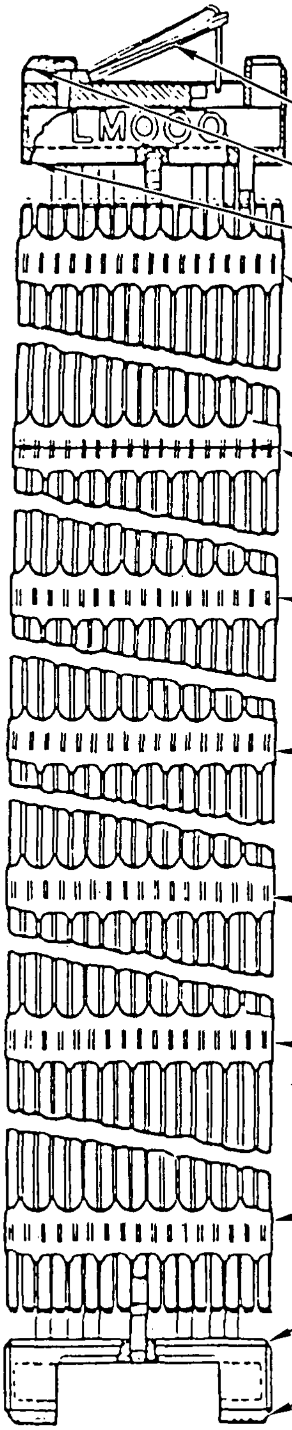
(a) Decay corrected to discharge data of 4/17/82.
(b) Parent element is iron (Fe). ^{54}Mn is formed by the fast neutron reaction ^{54}Fe(n,p)^{54}Nm.
(c) Parent element is Nb. ^{93m}Nb is predominantly produced by the reaction ^{93}Nb(n,N')^{93m}Nb.

TABLE 5.3 Specific Activities of Long-Lived Radionuclides in General Electric Spent Fuel Assembly Hardware Materials (Cooper Station)

Sample	Material	Location	Concentration--Ci/g of parent element)[a] ^{54}Mn[b]	^{55}Fe	^{59}Ni	^{63}Ni	^{60}Co	^{94}Nb	^{93m}Nb[c]	^{125}Sb[d]
GE-19	SS	Tab on handle	(1.61±0.44)E-4	(2.08±0.11)E-2	(4.39±0.66)E-5	(9.55±1.13)E-3	(2.39±0.36)E0	(7.61±5.09)E-5	(2.49±1.66)E-2	--
GE-18	SS	Upper tie plate	(1.70±0.47)E-4	(1.30±0.05)E-2	(6.04±0.82)E-5	(1.07±0.11)E-2	(2.16±0.43)E0	(5.27±1.13)E-4	(1.23±0.27)E-1	--
GE-17	Inconel	Expansion spring	<1.6E-3	(4.00±0.14)E-2	(6.94±0.99)E-5	(1.39±0.15)E-2	(3.77±0.60)E0	(7.34±1.21)E-5	(4.92±0.49)E-3	<5.5E-3
GE-15	Zircaloy	Spacer grid #7	(7.51±2.02)E-3	(1.79±0.04)E-1	(3.85±0.59)E-4	(4.31±0.66)E-2	<3.4E0	(1.00±0.63)E-4	(4.62±2.90)E-1	(3.81±0.63)E-1
GE-13	Zircaloy	Spacer grid #6	(3.67±1.23)E-2	(6.03±0.28)E-1	(1.29±0.19)E-3	(1.29±0.19)E-1	<1.8E1	--	--	(1.24±0.22)E-1
GE-11	Zircaloy	Spacer grid #5	(3.31±1.04)E-2	(4.16±0.09)E-1	(1.30±0.20)E-3	(1.23±0.19)E-1	<1.3E1	--	--	(1.18±0.24)E-1
GE-9	Zircaloy	Spacer grid #4	(3.45±1.18)E-2	(5.34±0.17)E-1	(1.07±0.16)E-3	(1.58±0.23)E-1	<2.4E1	--	--	(1.65±0.30)E-1
GE-7	Zircaloy	Spacer grid #3	(3.43±0.70)E-2	(7.50±0.32)E-1	(3.38±0.71)E-3	(4.78±1.08)E-1	<1.0E1	(4.59±2.16)E-5	--	(1.56±0.28)E-1
GE-5	Zircaloy	Spacer grid #2	(3.67±0.49)E-2	(4.48±0.16)E-1	(1.27±0.27)E-3	(1.64±0.34)E-1	<6.7E0	--	--	(1.26±0.24)E-1
GE-3	Zircaloy	Spacer grid #1	(1.80±0.36)E-2	(4.55±0.16)E-1	(7.20±2.38)E-4	(8.77±2.89)E-2	<3.3E0	(5.45±1.04)E-5	(8.32±1.39)E-2	(1.11±0.22)E-1
GE-1	SS	Bottom end fitting	<6.2E-3	(1.92±0.10)E-1	(9.39±1.30)E-4	(8.69±0.87)E-2	(2.97±0.30)E1	(2.15±0.64)E-4	(1.85±0.55)E-2	<5.8E-2
GE-2	SS	Bottom end fitting	<8.4E-4	(1.83±0.12)E-2	(1.30±0.15)E-4	(1.46±0.15)E-2	(3.50±0.35)E0	(3.35±0.56)E-5	(2.69±0.45)E-3	<1.1E-2

(a) Decay corrected to discharge date from reactor--5/1/82.
(b) Parent element is iron (Fe). ^{54}Mn is formed by the fast neutron reaction $^{54}Fe(n,p)^{54}Mn$.
(c) Parent element is Nb. ^{93m}Nb is produced mainly by the reaction $^{93}Nb(n,n')^{93m}Nb$.
(d) Parent element of ^{125}Sb is Sn. ^{125}Sb is formed by the thermal neutron reaction $^{124}Sn(n,\gamma)^{125}Sn$ followed by ^{125}Sn beta decay to ^{125}Sb.

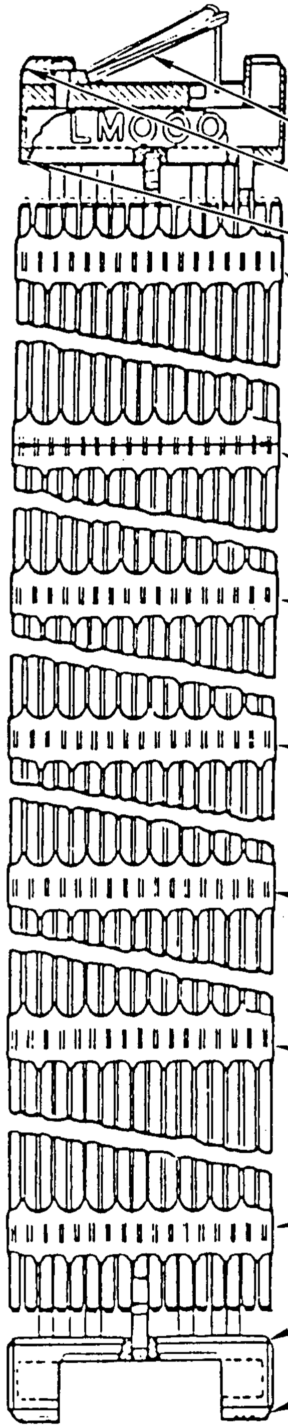
Westinghouse Fuel Assembly, 14x14



Predicted Ci/g-Co	Sample Number	Measured Ci/g-Co	Ratio: Predicted/Measured
8.78E-1	W10	(1.04±0.13)E0	0.844
1.02E0	W12	(1.09±0.11)E0	0.936
3.04E0	W9	(8.72±0.87)E0	0.349
9.99E0	W8	(8.85±0.89)E0	1.13
6.65E1	W7	(4.57±0.46)E1	1.46
8.47E1	W6	(6.49±0.65)E1	1.31
1.01E2	W5	(6.73±0.67)E1	1.50
1.00E2	W4	(6.73±0.67)E1	1.49
9.03E1	W3	(6.18±0.62)E1	1.46
2.57E1	W2	(2.75±0.28)E1	0.935
2.70E1	W11	(1.11±0.34)E1	2.43
1.74E1	W1	(1.24±0.12)E1	1.40

FIGURE 5.2 ^{60}Co Specific Activities in Spent Fuel Assembly Hardware

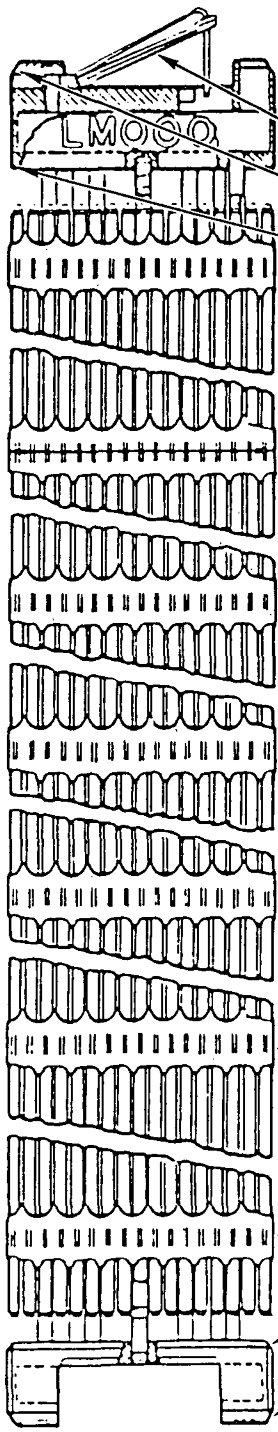
Westinghouse Fuel Assembly, 14x14



Predicted Ci/g-Fe	Sample Number	Measured Ci/g-Fe	Ratio: Predicted/Measured
4.63E-3	W10	(1.07±0.03)E-2	0.433
5.37E-3	W12	(5.43±0.16)E-3	0.989
1.54E-2	W9	(4.41±0.13)E-2	0.349
5.32E-2	W8	(8.06±0.24)E-2	0.660
2.75E-1	W7	(3.49±0.10)E-1	0.777
3.87E-1	W6	(3.46±0.10)E-1	1.12
4.18E-1	W5	(3.69±0.10)E-1	1.13
4.15E-1	W4	(3.98±0.11)E-1	1.04
3.75E-1	W3	(4.19±0.12)E-1	0.895
1.05E-1	W2	(1.99±0.06)E-1	0.533
1.39E-1	W11	(7.25±0.22)E-2	1.92
9.54E-2	W1	(6.48±0.19)E-2	1.47

FIGURE 5.3 ^{55}Fe Specific Activities in Spent Fuel Assembly Hardware

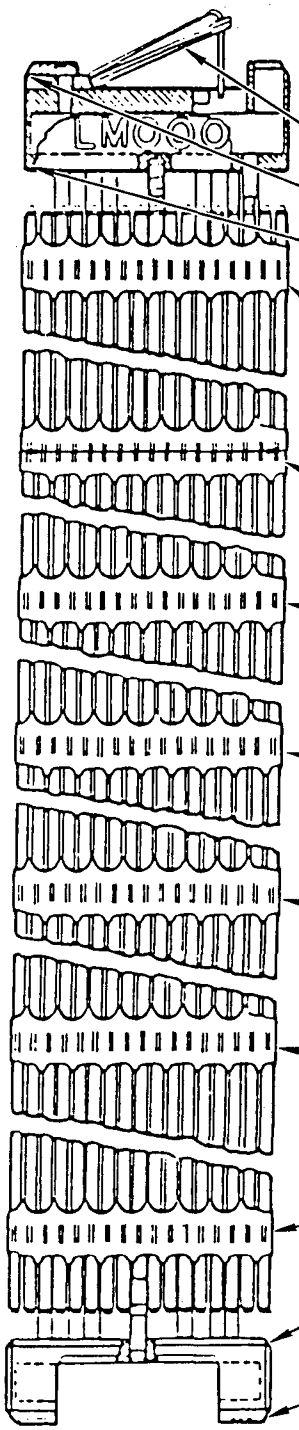
Westinghouse Fuel Assembly, 14x14



Predicted Ci/g-Ni	Sample Number	Measured Ci/g-Ni	Ratio: Predicted/Measured
2.93E-4	W10	(5.30±0.06)E-3	0.0553
3.38E-4	W12	(4.25±0.13)E-3	0.0795
1.19E-3	W9	(3.35±0.10)E-2	0.0355
3.19E-3	W8	(3.70±0.11)E-2	0.0862
5.57E-2	W7	(1.40±0.04)E-1	0.398
7.76E-2	W6	(1.77±0.05)E-1	0.438
8.39E-2	W5	(1.73±0.05)E-1	0.485
8.33E-2	W4	(1.70±0.05)E-1	0.490
7.53E-2	W3	(1.81±0.05)E-1	0.416
2.18E-2	W2	(7.59±0.22)E-2	0.287
9.64E-3	W11	(4.01±0.12)E-2	0.240
4.28E-3	W1	(3.59±0.11)E-2	0.119

FIGURE 5.4 ^{63}Ni Specific Activities in Spent Fuel Assembly Hardware

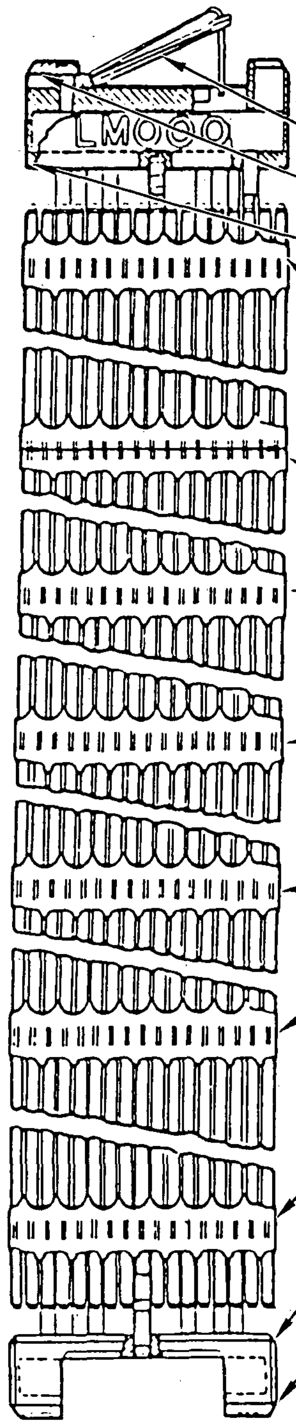
Westinghouse Fuel Assembly, 14x14



Predicted Ci/g-Ni	Sample Number	Measured Ci/g-Ni	Ratio: Predicted / Measured
2.16E-6	W10	(2.32±0.07)E-5	0.0931
2.50E-6	W12	(2.01±0.00)E-5	0.124
8.77E-6	W9	(1.44±0.04)E-4	0.0609
2.36E-5	W8	(1.42±0.04)E-4	0.166
4.12E-4	W7	(5.48±0.16)E-4	0.752
5.74E-4	W6	(7.29±0.22)E-4	0.787
6.21E-4	W5	(6.44±0.19)E-4	0.964
6.16E-4	W4	(5.86±0.18)E-4	1.05
5.57E-4	W3	(6.81±0.20)E-4	0.818
1.62E-4	W2	(2.74±0.08)E-4	0.591
7.13E-5	W11	(1.58±0.05)E-4	0.451
3.17E-5	W1	(1.26±0.04)E-4	0.252

FIGURE 5.5 ^{59}Ni Specific Activities in Spent Fuel Assembly Hardware

Westinghouse Fuel Assembly, 14x14



Predicted Ci/g-Nb	Sample Number	Measured Ci/g-Nb	Ratio: Predicted/Measured
1.11E-5	W10	(1.64±0.26)E-5	0.677
1.29E-5	W12	(1.24±0.26)E-5	1.04
4.19E-5	W9	(7.20±7.20)E-5	0.58
1.43E-4	W8	(1.59±0.27)E-4	0.899
1.37E-3	W7	–	
1.91E-3	W6	–	
2.06E-3	W5	(2.61±0.70)E-3	0.789
2.00E-3	W4	–	
1.85E-3	W3	–	
5.28E-4	W2	(9.01±1.67)E-4	0.586
3.57E-4	W11	(2.20±2.20)E-5	16
1.96E-4	W1	(1.32±0.23)E-4	1.48

FIGURE 5.6 ^{94}Nb Specific Activities in Spent Fuel Assembly Hardware

Figure 5.2 shows the comparison for ^{60}Co in the Westinghouse assembly. The agreement between measured and predicted ^{60}Co for this assembly is quite good. For the fueled region of the assembly the calculated values averaged about 1.3±0.2 times higher than the measured specific activities. Even at the end fittings where the one group neutron cross section for ^{59}Co varies by up to 5-fold over a distance of only a few centimeters, the predicted values were in fairly good agreement.

Figure 5.3 shows the measured versus calculated specific activities of ^{55}Fe in the Westinghouse assembly. The agreement for the fueled region of the assembly is excellent, with the predicted values averaging only about 12% lower than the measured activities over the entire fueled region. Again, larger variability exists at the end fittings.

Figure 5.4 shows the calculated versus measured specific activities for ^{63}Ni in the Westinghouse assembly. Within the fueled region the calculated values averaged about a factor of 2.7 times lower than the measured activities, although in the middle of this region the difference is about 2.0. At the bottom and top end fittings the calculated values underestimated the ^{63}Ni activities by factors of 8 to 28.

Similiar results were obtained for ^{59}Ni as shown in Figure 5.5. These discrepancies may be due to uncertainties in the available cross-section data. There are no available appropriate evaluated cross-sections for the $^{58}Ni(n,\gamma)$ ^{59}Ni or the $^{62}Ni(n,\gamma)$ ^{63}Ni reactions. Therefore, the cross section for natural nickel was used. This certainly affects the potential accuracy of the predictive calculations.

Figure 5.6 shows the measured and predicted ^{94}Nb specific activities for the Westinghouse assembly. Except for sample W11, which appears to be anomalous, possibly due to a very large analytical uncertainty in the elemental niobium measurement, the agreement between calculated and measured activities is quite good. In the fueled region of the assembly, the calculated values are underestimated by an average of only 25%, and at the end fitting they are underestimated by an average of only 6%, although the range is from 4% to 48%.

The general indication of these analyses is that good agreement exists between calculational predictions of radionuclide inventories and measurements within the fueled region of the core. However, the further one goes from the fueled region, the greater the differences become. At this point it is not certain why this discrepancy exists to the extent that it does. There are several areas currently being investigated to determine why the differences exist:

1) The calculation methods were inadequate in some cases. The good agreement in the fueled region indicates that the calculations are sufficiently accurate in that region. However, the neutron flux is dropping off steeply at the end fittings and a small change in the slope would be significant.

2) The samples taken are not reflective of the average over the regions calculated. The calculations assume homogenous regions. Since the samples at the end fittings were primarily surface samples of

castings, the elemental composition at the near-surface may not reflect the average in the component (e.g. the niobium may have precipitated to the surface during the casting process). Determining the curies of radionuclide in the sample per gram of parent nuclide in the sample should have accounted for this, but some other mechanism may be affecting the results.

3) As noted in 1) above, the slope of the flux is very steep outside the fueled region. The change over the upper end fitting alone is over an order of magnitude. A small shift in the sampling location could have a significant effect on the predicted radionuclide concentrations.

4) The uncertainties in the calculations may be greater than we believe. No quantitative estimate of the uncertainty due to the cross-sections used in the ANISN calculations is available. In particular, there are no appropriate evaluated cross-sections for the Ni-58 (n,γ) Ni-59 or the Ni-62 (n,γ) Ni-63 reactions available. Therefore, the cross-section for natural nickel which was used would certainly affect the potential precision of the predictive calculations.

5) The relative location of control rods and burnable poisons with respect to these assemblies was not available. For PWR's, the control rods enter from the top of the assembly, thereby having a significant effect on the local flux. In general, the boron in the water has a more significant effect on the overall flux and reaction rates in the reactor core than do the control rods. However, the effect at a specific sample location may be large enough to account for the differences we are seeing.

All of these possible areas contribute to the overall uncertainty when comparing laboratory results to predictive calculations. At this point, it is not known if one dominates, or if all contribute somewhat equally. These uncertainties are presently being investigated to determine the relative error contributions.

5.2 Gundremmigen Pressure Vessel

The radionuclide measurements of the Gundremmigen reactor pressure vessel steel described in Section 4 were compared with predictive calculations for the concentrations of ^{60}Co, ^{55}Fe, ^{63}Ni and ^{94}Nb. This comparison was made to evaluate the accuracy of predictive methods for calculating neutron activation product concentrations and inventories in decommissioned reactor pressure vessels. The comparison was conducted completely blind. Those conducting the measurements and the calculations were not informed of each others results until after all work was completed and submitted for comparisons. The results of this comparison are shown in Table 5.4.

As shown in Table 5.4 the agreement between measured and calculated activities is quite good considering that the neutron flux varies by over two orders of magnitude through the reactor pressure vessel wall. The calculated ^{60}Co, ^{55}Fe, and ^{94}Nb concentrations were overestimated by an average factor of 1.9, 1.3, and >1.4, respectively. The calculated ^{63}Ni was underestimated

TABLE 5.4 Comparison of Measured vs. Calculated Concentrations of Neutron Activation Products in Gundremmigen Pressure Vessel Steel

Radionuclide	Radiochemically Measured Concentration (Ci/g steel)(A)		Calculated Concentration (Ci/g steel)(A)		Ratio: Calculated/Measured	
	Sample #3 (0.41T)	Sample #4 (0.67T)	Sample #3 (0.41T)	Sample #4 (0.67T)	Sample #3	Sample #4
^{60}Co	2.53E-6	1.32E-6	4.53E-6	2.67E-6	1.79	2.02
^{63}Ni	1.14E-7	4.37E-8	7.53E-8	3.09E-8	0.662	0.709
^{55}Fe	2.91E-5	9.25E-6	3.29E-5	1.42E-5	1.13	1.54
^{94}Nb	<2.8E-12	<3.1E-12	4.4E-12	3.5E-12	>1.6	>1.1

(A) Decay corrected to reactor shutdown date of January 13, 1977.

by an average factor of 1.4. Thus, the agreement is quite good, and utilizing the methods for calculating neutron fluence[9] and vessel activation (this report) provides a reasonably good assurance that the calculational methods are producing reliable estimates of the concentrations of activation products in the reactor pressure vessel. This benchmarking will give confidence to similar methodology which will be used in future decommissioning assessments of commercial nuclear power stations.

6.0 Summary and Conclusions

Although this is an interim program report, there are significant results to date which have enhanced the radiological characterization associated with reactor decommissioning and related radioactive waste disposal.

6.1 Research Findings and Regulatory Implications

The significant research findings can be grouped into two main areas: 1) the radiological assessments conducted during the Shippingport Station decommissioning, and 2) the radiological characterization of activated metal components.

6.1.1 Radiological Assessments During Shippingport Station Decommissioning

From a radiological standpoint, the decommissioning operations at Shippingport Station have been extremely successful and have provided an optimistic and positive projection for the ultimate decommissioning of commercial reactor stations. One of the most significant observations at Shippingport Station was the fact that essentially all of the residual radionuclides were neutron activation products dominated by ^{60}Co. No significant concentrations of fission products or transuranic radionuclides were associated with the residual activity. This would be representative of the commercial nuclear power stations which have experienced little or no fuel cladding failures during their operations. Although the activation products ^{55}Fe, ^{63}Ni, ^{59}Ni, and ^{94}Nb were present with the ^{60}Co, their combined concentrations associated with the radioactive residues in piping, plant components, and other waste materials (excluding the pressure vessel internals) never exceeded the 10CFR61 Class A waste limit. Although the Shippingport Station is a DOE facility and not subject to the regulations contained in 10CFR61, the ramifications of the residual radioactivity levels in decommissioning wastes are of significance. First, it suggests that commercial stations having similar residual radionuclide inventories and distributions can expect to dispose of essentially most radioactive decommissioning materials and components (except reactor pressure vessel internals) as Class A waste. Secondly, this will greatly simplify the disposal methods and the dismantling options during decommissioning.

Another vanguard operation at Shippingport Station was the methodology developed by DOE and its subcontractors for characterization, packaging, shipment, and disposal of the reactor pressure vessel and internal components as an LSA, Type B package conforming to Department of Transportation and Nuclear Regulatory Commission regulations. The physical, chemical, and radiological characterization conducted by PNL of the radioactive corrosion film contained on the inside surfaces of the reactor pressure vessel and internal components showed that this material was extremely cohesive and would not be released under a variety of hypothetical severe accident conditions during transportation to the disposal facility at Hanford, Washington.

Other important radiological "lessons learned" during the decommissioning of Shippingport Station as they apply to commercial stations are being assessed and will be presented in the final report for this project.

6.1.2 Radiological Characterization of Activated Metal Components

During the past year this work has involved the radiological characterization of activated metal components from three commercial fuel assemblies, and characterization of steel specimens from the Gundremmigen reactor pressure vessel. Particular emphasis has been in measuring and assessing the significance of the long-lived radionuclides specified in 10CFR61. This work has shown that the relatively high nickel and niobium content of Inconel, and the nickel content of stainless steel has resulted in ^{63}Ni, ^{59}Ni, and ^{94}Nb concentrations in some fuel assembly hardware components being over the Class C limit. This would require that these components be disposed in a high level waste repository.

It was discovered in this work that the concentrations of ^{93m}Nb, a 13.6 year half-life activation product were present in the activated metal specimens at levels over 1800 times higher than previous calculations. To the best of our knowledge, these are the first actual measurements of ^{93m}Nb in activated metals. This radionuclide decays by emission of a 30-kev gamma-ray which is essentially all converted, and the predominant external radiation is due to the 16-kev Nb x-rays. This radionuclide has not even been considered in 10CFR61, and its long-term environmental significance will need to be assessed.

During the radiological characterization of the fuel assembly hardware it was possible to conduct separate detailed predictive calculations of radionuclide concentrations in the same material. A comparison of the measured versus calculated concentrations of ^{55}Fe, ^{60}Co, ^{59}Ni, ^{63}Ni, and ^{94}Nb in the fuel assembly hardware from Westinghouse and Combustion Engineering PWR fuel assemblies showed quite good agreement in most cases. The agreement between measured versus calculated values for these radionuclides in hardware from the fueled region of the assemblies were generally on the order of 10 to 50 percent, and never exceeded about a factor of two. As the neutron flux and energy spectrum drops rapidly between the fueled region and the end fittings of the assemblies, the uncertainties in the calculational methods become much larger and large differences in measured versus calculated activated were observed. The largest discrepancies were observed for the ^{59}Ni and ^{63}Ni activities at the end fittings of the fuel assemblies. Since no adequate isotopic cross section data exist for the stable parent nickel isotopes, elemental cross section data were used, and this may have introduced relatively large uncertainties in the calculated results.

The radionuclide measurements of the Gundremmigen pressure vessel steel were in very good agreement with a blind comparison of calculated activities. The average calculated-to-measured ratio for ^{55}Fe, ^{60}Co, and ^{63}Ni were 1.3, 1.9, and 0.69, respectively. The concentrations of the radionuclides were all below Class A limits, indicating that the entire pressure vessel (not including internals) could have been disposed as Class A waste in a low level waste shallow land burial facility.

The measurements and calculational methods utilized in this work have lent confidence to calculational methods for predicting radionuclide inventories in activated metals, and have identified certain problem areas where better cross section data or calculational methodology are needed.

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EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS (ENFDP) -PROGRAM STATUS-

D. R. Haffner
Westinghouse Hanford Company

ABSTRACT

The Evaluation of Nuclear Facility Decommissioning Projects (ENFDP) program was undertaken by the U.S. Nuclear Regulatory Commission (NRC) to compile and evaluate the activities of completed and ongoing decommissioning projects. This report discusses briefly, the status of those decommissioning projects selected for this study with emphasis on ongoing projects.

1.0 INTRODUCTION

Major studies have been undertaken in recent years by the U.S. Nuclear Regulatory Commission (NRC) and others, on the technology, safety, and costs associated with the decommissioning of nuclear facilities. The Evaluation of Nuclear Facility Decommissioning Projects (ENFDP) program described in this report is being undertaken by the NRC to compile and evaluate the activities of ongoing decommissioning projects. Assessment and evaluation of the methods, impacts, and costs will provide bases for evaluating licensee's proposed decommissioning plans, and for future decommissioning guidance and regulation.

Program participants include the NRC (through the Office of Nuclear Regulatory Research), Westinghouse Hanford Company (WHC), and nuclear facility licensees of decommissioning projects accepted into the ENFDP program.

A computerized data collection system has been developed for data accumulation and analysis of relevant data from the nuclear facility decommissioning projects in this study. This computerized data base, the Decommissioning Data System (DDS), allows the archiving of large quantities of decommissioning information in a standardized format to ensure a consistent set of data for comparative purposes. Decommissioning information collected for this data base includes:

- Facility description
- Operating history, significant events
- Radionuclide inventory, facility dose rates

- Costs of labor, waste disposal and shipping
- Project manpower requirements
- Radiation exposure to personnel
- Decommissioning techniques, their costs and effectiveness
- Waste disposition, volume and curie content
- Surveillance costs, if applicable
- Lessons learned.

To date, decommissioning data has been collected and stored in the DDS data base for the 17 reactor decommissioning projects as shown in Table 1. Final project summary reports have been prepared for all completed projects and several status reports for ongoing projects listed. Generic conceptual information relevant to decommissioning of four reference reactor types listed was extracted from Pacific Northwest Laboratory NUREG documentation and converted to the standardized DDS data base format for comparison studies.

Table 1. Evaluation of Nuclear Facility Decommissioning Projects--Decommissioning Data System.

Completed projects	Ongoing projects	Reference** reactor studies
1. Elk River (BWR)	1. Three Mile Island-2 (PWR)*	1. PWR
2. Enrico Fermi-1 (FBR)	2. Humboldt Bay-3 (BWR)	2. BWR
3. Ames Laboratory (Research)	3. Shippingport (PWR)	3. Research Reactor
4. North Carolina State University (Research)	4. Northrop TRIGA (Research)***	4. Test Reactor
5. Plum Brook (Test)	5. LaCrosse (BWR)	
	6. Gundremmingen-FRG (BWR)	
	7. Lingen-FRG (BWR)	
	8. Niederaichbach-FRG (HWR)	

*Data and report format modified to fit accident recovery data at TMI-2.
**Generic conceptual decommissioning data was extracted from Pacific Northwest Laboratory NUREG documentation and converted to the standardized DDS data base format for comparative purposes.
***Decommissioning of Northrop has been completed. Data acquisition and analysis scheduled for first quarter of FY 1989 to include final radiological survey data utilizing an improved survey methodology. PST89-1006-1

2.0 STATUS OF ONGOING PROJECTS (DOMESTIC)

2.1 THREE MILE ISLAND-2 (TMI-2)

Cleanup at TMI-2 is scheduled for completion in mid-1989 and is projected to cost approximately $965 million. About 65% of the core debris has been removed from the reactor, and about 60% of the debris removed has been shipped to the Idaho National Engineering Laboratory (INEL).

Cleanup data obtained from the TMI-2 accident recovery project following the March 28, 1979, accident have been divided into the following nine major tasks:

1. Reactor coolant system and systems decontamination
2. Reactor building decontamination
3. Reactor defueling and disassembly
4. Auxiliary and fuel handling building decontamination
5. Common support facilities and systems decontamination
6. Plant stability and safety activities
7. Liquid waste handling
8. Solid waste handling
9. Radioactive waste and laundry shipments.

Since the March 28, 1979 accident data generated for the above nine major tasks cover an estimated 750,000 to 850,000 radiation area entries. The major portion of this data is involved in the first six tasks. To date, about 90% of the data through May 1986 has been collected, analyzed, and entered into the DDS data base. Data covering approximately 575,000 radiation area entries have been analyzed and condensed into about 35,000 lines of data entered into the DDS data base. Waste shipment data through December 1987 have been analyzed and entered into the DDS.

Status reports have been submitted for publication for eight of the nine tasks shown below:

• Reactor coolant system & systems decontamination	Data through December 1984
• Reactor building decontamination	Data through May 1986
• Reactor defueling and disassembly	Data through April 1985

- Auxiliary and fuel handling building decontamination — Data through May 1986
- Plant stability and safety activities — Data through May 1986
- Liquid waste handling — Data through May 1985
- Solid waste handling — Data through May 1985
- Radioactive waste and laundry shipments — Data through April 1987

A summary of TMI-2 labor and exposure data for data processed to date is shown in Table 2. A summary of radioactive waste and laundry shipments is shown in Table 3.

Table 2. Three Mile Island-2 Labor and Exposure Summary.

Major Task	Radiation area entries	Labor (man-hours)	Exposure (man-rems)
Reactor coolant system and systems decontamination	1,075	2,560	14.7
Reactor building decontamination	38,554	110,829	896.1
Reactor defueling and disassembly	4,809	10,502	507.4
Auxiliary and fuel handling building decontamination	46,407	102,519	295.1
Common support facilities and systems decontamination	13,660	8,488	135.3
Plant stability and safety activities	177,132	498,736	589.6
Liquid waste handling	47,614	111,132	100.7
Solid waste handling	20,103	52,432	83.2

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Table 3. Three Mile Island-2 Radioactive Solid Waste and Protective Clothing (Laundry) Shipments.

Type of waste	Containers	Activity (curies)	Volume (ft^3)	Weight (tons)
Solid wastes	7,022	736,969	168,683	2,265
Protective clothing	26,635	6.3	218,838	1,713

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2.2 HUMBOLDT BAY POWER PLANT-3 (HB-3)

Data collection and analysis for the Humboldt Bay decommissioning project is nearing completion as is the facility preparations for SAFSTOR. System layups, permitted under the current Technical Specifications, along with associated surface decontamination, have been completed. Layup of the Gas Treatment System and completion of the electrical systems layup were deferred until after approval of the HB-3 SAFSTOR Decommissioning Plan.

Radioactive waste shipments during SAFSTOR preparations were completed in July 1986 except for two or three additional shipments anticipated before project completion. A total of 156 shipments of radioactive waste consisted of 49,330 ft^3 containing 940.9 curies of activity. It should be noted that these shipments are not indicative of decommissioning waste, since approximately 70% of the waste shipped included wastes generated during HB-3 operation and was not the result of preparations to SAFSTOR decommissioning.

The NRC approval of the Decommissioning Plan on July 19, 1988, has allowed the continuation of final system layups, modifications and decontamination, toward project completion scheduled for the end of 1988.

2.3 SHIPPINGPORT STATION DECOMMISSIONING PROJECT (SSDP)

2.3.1 Major Shippingport Station Decommissioning Project Accomplishments To Date (March 1988)

- Irradiated components transferred to RPV
- Primary and secondary systems drained
- Electrical systems and equipment de-energized
- Piping/equipment and primary component removal completed
- 554,000 gal of radioactive liquid processed
- Asbestos removal completed.

Table 4. Status of Decommissioning Activities at Shippingport Station Decommissioning Project (March 1988).

Work activity	Percent complete March 1988
Preparation for removal of pressure vessel, internals, and neutron shield tank package	89
Removal of piping and equipment	99
Removal of primary system components	100
Removal of power and control systems	91
Liquid waste management	99
Solid waste management	92
Decontamination	85
Removal of structures and containment	33
Reactor pressure vessel transport	5
Overall project	74

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Table 5. Shipments.

Radioactive waste shipments--status (March 1988)			
Shipments	Packaged to date	Total project estimate	Shipped to date
LSA containers	2,664	2,745	2,591
Radioactive mixed waste packages	2	27	2
Type A packages	5	5	5
Number of shipments	204	181	
Total volume (1,000 ft^3)	196.2	181.2	
Scheduled reactor pressure vessel (RPV) shipment			
• RPV package scheduled for lifting and loading on the transporter FY 89-1Q • RPV stored onsite until shipment • RPV barge shipment scheduled for FY89-2Q			

PST89-1006-5

Table 6. Dose Rates (man-rem).

Personnel exposure at Shippingport Station Decommissioning Project	Man-rem
Caretaker phase (prior to September 1985)	4.5
September 1985 to December 1985	10.8
January 1986 to March 1986	16.3
April 1986 to September 1986	19.1
October 1986 to September 1987	90.6
Cumulative exposure through FY 1987	141.3

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Table 7. Cost Data (millions of dollars).

Cumulative expenditure at Shippingport Station Decommissioning Project	Amount ($M)
Through FY 1985	17.68
Through FY 1986	34.79
Through FY 1987	42.46
Through March 1988	52.02
Total estimated cost through end of project	98.30

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2.3.2 Status of Shippingport Station Decommissioning Project Decommissioning Data for Evaluation of Nuclear Facility Decommissioning Projects Program

To date, the majority of the SSDP decommissioning data for use in the ENFDP program consists of estimates and preliminary information. The U.S. Department of Energy (DOE) project management at SSDP has formally agreed to support the ENFDP program; however, data will not be provided until the work for the various Activity Specifications is completed and formal publication is ready for submittal to the technical information centers: Remedial Action Program Information Center (RAPIC) and Technical Information Center (TIC). With most ongoing ENFDP projects, data is collected at regular site visits and processed into the DDS data base as the project work progresses. Actual decommissioning data received to date from SSDP are:

- Radiological characterization data resulting from a DOC survey performed in 1985
- Waste shipment data for shipments conducted from February 1986 to September 1986
- Topical report (DOE/SSDP-0034) on irradiated component transfer to the reactor vessel
- Topical report (DOE/SSDP-0033) on asbestos removal
- Data supporting the two above reports as provided by the respective subcontractors to the Technology Transfer function.

2.4 NORTHROP TRIGA

The ENFDP data requirements for the Northrop TRIGA research reactor decommissioning project include only the final radiological survey data. An improved survey methodology was implemented at the Northrop TRIGA facility which claims a 90% confidence level. Acquisition of the final survey data is planned for October 1988.

2.5 LACROSSE BOILING WATER REACTOR (LACBWR)

The LaCrosse Boiling Water Reactor (LACBWR) is a nuclear power plant of nominal 50 MW(e) capacity utilizing a forced-circulation, direct-cycle boiling water reactor as its heat source. The plant is co-located with a 350 MW(e) coal-fired power plant on the Mississippi River near the town of Genoa, Wisconsin.

The LACBWR was one of a series of power demonstration plants funded in part by the U.S. Atomic Energy Commission (AEC). The plant is owned and was operated by the Dairyland Power Cooperative (DPC). Allis-Chalmers had the responsibility of construction and startup of the plant.

Dairyland Power Cooperative operated the facility as a base-load plant from November 1, 1967, until it was permanently shutdown on April 30, 1987. The decision to shut down the LACBWR was based on projected cost savings associated with operation of the DPC's coal-fired plants.

Final reactor defueling was completed on June 11, 1987. DPC currently plans to store the 333 spent fuel assemblies onsite in the Fuel Element Storage Well until a federal repository is available to receive the spent LACBWR fuel.

The decommissioning mode chosen for the LACBWR was essentially limited to the safe storage (SAFSTOR) option primarily because of the onsite spent fuel storage. Only limited decontamination and dismantling of unused systems can be performed during the planned 30-50 year SAFSTOR period. After completion of the SAFSTOR period, the final decontamination and dismantling of the plant (DECON mode) will be performed to allow release of the site for unrestricted use.

Data collection at the LACBWR for the ENFDP program was initiated in the spring of 1988. Minimal decommissioning activity has occurred at LACBWR because DPC is awaiting NRC approval of their decommissioning plan submitted for review on December 21, 1987. Decommissioning data received to date, consist of the following:

- Special Work Permit (SWP) information (man-hours and personnel exposure) for two tasks completed in October 1987
- Radiation survey data for periodic survey of 100 established locations throughout the plant taken in November 1987, January 1988, and April 1988
- Monthly plant exposures (goals and actual) for 1987
- Sample waste data of actual waste considered to be normal operations and not decommissioning waste
- Annual reports for 1986 (full year of operation) and 1987 (operating through April, shutdown thereafter).

Evaluation of collected data and data entry into the DDS data base is underway.

The LaCrosse project has provided evidence of a well-maintained nuclear power reactor. Decommissioning data from LACBWR should indicate the benefits resulting from the good management practices observed during operation of the plant.

3.0 STATUS OF ONGOING PROJECTS (FOREIGN)

Information on three reactor decommissioning projects in the Federal Republic of Germany was obtained from the DOE as part of DOE's International Technology Exchange Program. Two of the reactors, Kernkraftwerk Lingen (KWL) and Kernkraftwerk Gundremmingen Block A (KRBA) are being decommissioned to the Safe Enclosure condition (equivalent to NRC's SAFSTOR mode). The third, Kernforschungszentrum Karlsruhe Niederaichbach (KKN) is in the final planning stages for dismantlement as a demonstration project.

3.1 KERNKRAFTWERK LINGEN (KWL)

Necessary activities to place the Lingen plant in the Safe Enclosure condition have been completed and the appropriate licenses have been issued. Acquisition of the final data and a copy of the KWL final decommissioning report is scheduled for October 1988.

3.2 KERNKRAFTWERK GUNDREMMINGEN BLOCK A (KRBA)

Cleanup work at the KRBA decommissioning project continues toward the Safe Enclosure condition.

3.3 KERNFORSCHUNGSZENTRUM KARLSRUHE NIEDERAICHBACH (KKN)

The testing of the remotely controlled cutting machine at the Noell Company has been completed. The machine has been moved to the KKN plant with installation and preparations for operation expected near the end of 1988.

4.0 SUMMARY

The ENFDP program is following the decommissioning of reactors to provide the NRC with data to evaluate the methods, impacts, radiation exposure, manpower requirements, and costs of reactor decommissioning. Collecting the experience from these ongoing decommissioning projects and from any future reactor or non-reactor projects added to the program, will aid the NRC in evaluating licensee decommissioning proposals and developing decommissioning direction and guidance.

THE USE OF SOLIDIFICATION SYSTEMS AND HIGH INTEGRITY CONTAINERS FOR DISPOSAL OF LOW-LEVEL WASTE

by

Michael Tokar
U. S. Nuclear Regulatory Commission

Abstract

This paper primarily addresses the 10 CFR Part 61 requirements for waste form stability and the use of solidification media and high integrity containers to provide the required stability for Class B and Class C waste. Tests and criteria that are provided in a Technical Position on Waste Form and the current status of NRC's waste form and HIC reviews are also discussed.

1 INTRODUCTION

NRC regulation 10 CFR Part 61 (Ref. 1) establishes, for land disposal of radioactive waste, the procedures, criteria, and terms and conditions upon which the Commission issues licenses for the disposal of radioactive wastes containing byproduct, source and special nuclear material received from other persons.

Section 61.55 of Part 61 establishes three categories or classes of wastes; viz., Class A, Class B, and Class C, in a generally ascending order with regard to degree of hazard (i.e., type and concentration) of radio-nuclides. Class B & Class C wastes are required to meet both minimum as well as stability requirements that are set forth in 10 CFR 61.56. Class C waste must also be protected (at the disposal facility) against inadvertent intrusion.

As noted in Section 61.56 of Part 61, waste form structural stability can be provided in a variety of ways. Two methods; viz., the use of (a) solidification agents and (b) high integrity containers (HICs) are the subject of this paper. A third approach -- the use of engineered structures has received a lot of attention recently, but a discussion of that alternative is beyond the scope of this paper.

2 PART 61 WASTE FORM REQUIREMENTS

2.1 Minimum

Waste characteristics requirements are established in 10 CFR 61.56. There are two types or categories of requirements: (1) minimum - addressed in Section 61.56(a); (2) stability - addressed in Section 61.56(b). All classes of waste must meet the minimum requirements in 10 CFR 61.56(a). The "minimum" requirements concern: (a) a prohibition against the use of cardboard fiberboard boxes; (b) treatment, packaging and maximum quantities of liquid wastes; (c) restrictions concerning disposal of explosive or detonatable wastes; (d) restrictions against wastes containing, or capable of generating, quantities of toxic gases, vapors, or fumes; (e) a prohibition against pyrophoric wastes; (f) a limit on the maximum pressure and curie content for packaged gaseous wastes; and (g) a general requirement for treatment of hazardous, biological, pathogenic, or infectious material to reduce to the maximum extent practicable the potential hazard from non-radiological materials.

2.2 Structural Stability

Requirements for stability are provided in 10 CFR 61.56(b). Stability is defined in 10 CFR 61.2 as meaning "structural stability." While the term, structural stability, is itself not defined anywhere in Part 61, it is indicated in Section 61.56(b) that "stability" (i.e., structural stability) is intended to ensure that the waste does not structurally degrade and affect overall stability of the site through slumping, collapse, or other failure of the disposal unit and thereby lead to water infiltration. Stability is also stated to be a factor in limiting exposure to an inadvertent intruder, since a stable waste form should be recognizable and nondispersible. Therefore, in addition to recognizability and nondispersibility, the Class B & Class C waste forms are supposed to contribute to the ability of the facility to retain overall stability and to thereby resist water infiltration. Resistance of the disposal facility to water infiltration is thus fundamentally associated with waste form structural stability. Although not explicitly so stated in Part 61, the concern about water infiltration stems from the fact that migration through groundwater is a potentially major pathway for radionuclide release to the offsite environment. The relationship of this concern, which is a thread that runs through Part 61, to the technical criteria and recommendations for immersion and leach testing will be addressed further in detail in this paper.

Further discussion of structural stability is provided in 10 CFR 61.56(b)(1), where it is stated that "a structurally stable waste form will generally maintain its physical dimensions and its form, under expected disposal conditions such as weight of overburden and compaction equipment, the presence of moisture and microbial activity, and internal factors such as radiation effects and chemical changes." This section of Part 61 also indicates that structural stability can be provided in any one of three different ways: (1) by the waste form itself (as in an activated metal component); (2) by processing the waste to a stable waste form (for example, by mixing and solidifying the waste with a cementitious material such as Portland cement); or (3) by placing the waste in a disposal container or structure that provides stability after disposal (such as a HIC).

Section 61.56(b) also provides further requirements concerning waste characteristics with regard to: (a) limitations on the amount of free standing or corrosive liquid (1% by volume of the waste when it is in a disposal container, or 0.5% by volume of the waste when processed to a stable form); and (b) void spaces within the waste and between the waste and its package that must be reduced to the extent practicable.

2.3 Concepts

Though the basic requirements for waste form stability are provided in Section 61.56 of Part 61, the discussion of fundamental concepts or rationale is contained in Section 61.7. In that section is provided a fairly detailed discussion of stability - of the waste as well as of the disposal site. As stated there, "a cornerstone of the system is stability - ...so that...[through stability of the waste and site,]... access of water to the waste can be minimized (emphasis added)." In this way, "migration of radio-nuclides is minimized...." Implicit in these statements is a recognition of the fact that contact of waste with water can lead to extraction (i.e., "leaching") of radionuclides from the waste form. Thus, leaching of radionuclides from the waste form is the first step in subsequent migration of the radionuclides from the waste through the groundwater and off of the site. It is clear, therefore, that, though leaching is not mentioned explicitly in Part 61, it is a phenomenon that is of fundamental concern to low-level waste disposal. Hence, it should come as no surprise that waste form leach testing is recommended in the 1983 "Technical Position on Waste Form" (Ref. 2).

3 TECHNICAL POSITION ON WASTE FORM STABILITY

3.1 Background

Though Part 61 provides the basic licensing requirements for low level waste (LLW) Class B & Class C structural stability, the regulation does not indicate in any detail how those requirements should be demonstrated to be met. That type of detailed guidance is instead provided in a "Technical Position on Waste Form" which was issued in May 1983. For solidified waste forms, the tests (see Table I) essentially involve subjecting the waste specimens to conditions of compression, irradiation, biodegradation, leaching, immersion, and thermal cycling. Most of the tests, which were selected for their relative simplicity and reproducibility, are based on American Society for Testing and Materials (ASTM) or American Nuclear Society (ANS) standard methods of test that were originally developed for specific non-radioactive material applications. Though it is not explicitly so stated in the TP, these methods of test are intended to provide confidence, by means of exposing test specimens to relatively short-term (minutes to weeks) conditions, that low-level radioactive waste forms will have the desired long-term (300-year) structural stability. It is important to remember in this regard that there is a major difference in time scale between the periods of time allotted for the tests and the period of time of concern for LLW disposal. Therefore, the test conditions cannot match, and are not intended to exactly duplicate, the conditions that might actually exist in the disposal facility at the time of disposal or which might exist at some point in time following placement of the waste in the facility. For example, the irradiation test calls for the specimens to be exposed to a minimum of 10E+8 rads, which is the maximum level of exposure for the waste forms expected after (300 years of) disposal; this requires the test specimens to be exposed to a much higher gamma flux than would actually be encountered under real exposure conditions. Thus, in some ways (some of) the TP tests can be considered to be accelerated tests, while in a more fundamental sense they are actually screening tests that are used to weed out material formulations and designs that do not exhibit sufficient assurance of long-term stability.

3.2 Test Parameters

The 1983 "Technical Position on Waste Form" addresses the type of short-term testing that should be performed to demonstrate long-term (300 year) structural stability as well as the acceptance criteria for the tests. As shown in Table 1, there are eight types of tests or test conditions for solidified waste forms called out in the 1983 TP. Five of the tests are patterned after ASTM or ANS Standard Methods of Test. However, the principal acceptance criterion parameter for most of the tests is compressive strength. The compressive strength criterion and the tests are related to Part 61 through the statement (noted above) in 10 CFR 61.56(b)(1), where it is stated that "a structurally stable waste form will generally maintain its physical dimensions and its form, under expected disposal conditions, such as weight of overburden and compaction equipment, the presence of moisture [a rationale for immersion and leaching tests] and microbial activity [a rationale for biodegradation tests], and internal factors such as radiation effects [a rationale for radiation stability tests] and chemical changes." In the 1983 Technical Position, a cover material density of 120 lbs./cu.ft. is assumed, which yields a pressure of approximately 37.5 psi at a burial depth of 45 feet (the then-maximum burial depth at Hanford). Taking into consideration potential additional loads from trench compaction equipment, waste contents, etc., the compressive strength criterion was set at 50 psi, which was raised to 60 psi when Hanford increased the depth of its trenches to 55 feet. Thus, the compressive strength criterion was not established as a result of some direct correlation of an intrinsic material property to long-term structural stability, but was instead intended to accommodate the environmental or in situ loads at the bottom of a disposal trench. For certain types of solidification media, (e.g., Portland cement or vinyl ester styrene), which typically have (in the unadulterated form) compressive strengths on the order of several thousand psi, a 60 psi compressive strength criterion does not appear to have a strong correlation to long-term structural stability. Additionally, for viscoelastic media such as bitumen, which continues to deform under load, measurements of some other property (such as viscosity), in addition to or in place of compressive strength, might be needed to demonstrate long-term structural stability.

Table I

Solidified product guidance

Tests	Methods	Criteria
1. Compressive Strength	ASTM C39 or D1074	60 psi (a)
2. Radiation Stability	(See 1983 TP)	60 psi comp. str. after 10E+8 rads
3. Biodegradation	ASTM G21 & G22	No growth (b) & comp. str._ 60 psi
4. Leachability	ANS 16.1	Leach index of 6
5. Immersion	(See 1983 TP)	60 psi comp. str. after 90 days
6. Thermal Cycling	ASTM B553	60 psi comp. str. after 30 cycles
7. Free liquid	ANS 55.1	0.5 percent
8. Full-scale Tests	(See 1983 TP)	Homogeneous & correlates to lab size test results

(a) The 1983 TP calls for a minimum compressive strength of 50 psi. This has been raised to 60 psi to accommodate an increased maximum burial depth at Hanford of 55 feet (from 45 feet).

(b) The 1983 TP calls for a multi-step procedure for biodegradation testing: if observed culture growth rated "greater than 1" is observed following a repeated ASTM G21 test, or any growth is observed following a repeated ASTM G22 test, longer term testing (for at least 6 months duration) is called for, using the "Bartha-Pramer Method." From this test, a total weight loss extrapolated for full-size waste forms to 300 years should produce less than a 10 percent loss of total carbon in the sample.

3.3 Standard Methods of Test

Though the 1983 TP refers to several ASTM or ANS Standard Methods of Test (see Table 1), none of the listed Standards (Refs. 3 to 9) other than the ANS 16.1 test for leachability were developed specifically for the testing of lowlevel waste forms. All the tests other than the leach test are adaptations of industry standards that were developed originally for specific nonradio active material applications. For example, the ASTM B553 thermal cycling test was developed for metalplated, plastic automobile parts, and the ASTM D1074 compressive strength test (which is used for testing viscoelastic materials) was developed for testing road bitumens. As a result, various details of the test procedures are open to interpretation, as are the results of the tests.

3.4 Waste Solidification and HIC Problem Areas

There has been considerable research and field experience obtained with HICs and waste solidification media since the "Technical Position on Waste Form" was developed in 1983. As a result of knowledge gained through topical Report reviews and the results of tests and/or analytical calculations, the following problem areas have been identified:

- Cement - Test results (Ref. 10) from programs conducted by National Laboratories and the cement solidification vendors, coupled with observed problems with swelling, disintegration, or incomplete solidification of power plant cement waste forms, have led the NRC staff to recommend that waste loading be limited to 18 percent by weight until sufficient data are presented to justify higher loadings.

- Bitumen - There are two primary types of bitumen that are used to solidify low-level radioactive waste: (1) "distilled" and (2) "oxidized." A topical Report has been submitted for review on each of these materials by separate vendors. To this date, the NRC staff has not been presented with any evidence that the distilled bitumen can provide stabilized waste forms that meet the 60 psi compressive strength criterion. Therefore, in February 1988, the technical review of the topical Report on distilled bitumen was discontinued, and the topical Report was returned (Ref. 11) to the vendor (Associated Technologies, Inc.). The topical Report for the oxidized bitumen has been approved (Ref.12).

- High Density Polyethylene Containers (HPDEs) - As a result of an allegation that HDPE HICs do not have sufficient strength to withstand the stresses imposed by the weight of material placed above the HICs in a burial environment, the NRC contracted with Brookhaven National Laboratory (BNL) to analyze existing data on creep of polyethylene and to develop a model and criteria that could be used to evaluate the structural stability of the HICs. BNL recommended (Ref. 13) that the HICs be shown to be able to resist buckling, to not enter tertiary creep, and to not exceed allowable membrane stresses. Preliminary calculations, using the BNL model, indicate that large HDPE HICs may not satisfy the criteria. The HDPE HIC vendors have all been notified (Ref. 14) (along with the Agreement States) and requested to show via (a) analyses, (b) testing, (c) administrative procedures, and/or (d) redesign that their HICs can satisfy the criteria. Each of the HDPE HIC vendors has submitted information that is under review by NRC staff and consultants.

4 THE TOPICAL REPORT REVIEW PROCESS

4.1 Background

As noted earlier, the purpose of the "1983 Technical Position on Waste Form" is to provide guidance on an acceptable approach for demonstrating compliance with 10 CFR Part 61 requirements for LLW structural stability. Under current procedures, the NRC provides a "central" review of topical Reports on waste form solidification media and HICs. The central review is intended to be applicable for all disposal sites. A brief description of the evolution and current status of this review process is provided below.

4.2 Development and Evolution

The current process for NRC's reviews of topical Reports on waste form solidification, HICs and computer codes for classifying waste originated as a result of several actions that occurred primarily during calendar year 1983; (the foundation for these actions and agreements, however, was laid in a series of earlier activities that occurred over several years, but which will not be addressed here in the interest of brevity). The 1983 "Technical Position on Waste Form" was completed in May 1983 and made available to the public in June 1983. The NRC publicized its topical Report review process in September 1983 with a Federal Register Notice that stated that a limited waiver of fees would be granted for Reports submitted before June 30, 1984.

The vendors responded to this by submitting eighteen topical Reports before the expiration of the fee waiver, while twelve Reports have been submitted after the June 30, 1984 expiration date.

In November 1983, NRC's Division of Waste Management (DWM) participated in a review of the South Carolina Agreement State Program. South Carolina (SC) had established acceptance criteria for HICs in 1980 and had issued several Certificates of Compliance (Cs of C) to HIC vendors beginning in May 1981, based on those criteria. The DWM's examination of SC's HIC reviews was limited to a determination that the State had used criteria that appeared to be compatible with the staff's "1983 Technical Position on Waste Form." No determination was made of the adequacy of the reviews with respect to whether reasonable assurance had been provided that the HICs would have 300-year structural stability.

In December 1983, a meeting (Ref. 15) was held in Bethesda to discuss the overall policy for HICs and topical Report reviews. In attendance were representatives from the States of South Carolina, Nevada and Washington, as well as NRC's Office of State Programs and DWM. At this meeting, the topical Report review process and the roles of the NRC and the States were discussed. It was recognized that the Agreement States have the licensing authority for the disposal sites with respect to whether specific HICs or waste forms would be acceptable for disposal at the sites. Before this meeting, the State of South Carolina had issued ten Cs of C and had under review two additional requests for approval of HICs. The State of Washington had two requests for approval. It was at this meeting that an agreement was reached that NRC would provide a "central" review of topical Reports that would be applicable for all the disposal sites.

4.3 Grandfathering

A key outcome of the December 1983 meeting in Bethesda concerned "grandfathering." It was decided that South Carolina (Nevada and Washington had not yet issued any HIC approvals) would continue to accept the use of HICs that had already been issued a C of C.

For such HICs, revocation of a C of C would take place only if a problem were identified or if new information indicated that the HICs would not meet the acceptance criteria. For new HICs that were described in topical Reports submitted to the NRC, the States would not issue Cs of C until the review had been completed by the NRC; (it should be noted, however, that periodic temporary approvals or "variances" for limited quantities of certain types of HICs have been granted by the State of Washington). For solidification processes, those processors who submitted information to NRC in topical Reports submitted before June 30, 1984 would still be acceptable under a grandfathering arrangement.

4.4 NRR Process Control Plans (PCP) Reviews

While NMSS has been reviewing HIC designs and waste solidification media formulations in accordance with 10 CFR Part 61 requirements for structural stability, the office of Nuclear Reactor Regulation (NRR) has been reviewing generic and plant-specific Process Control Plans (PCPs) requirements for reactors. The NRR reviews are intended to be focussed on the systems interactions of the solidification equipment with the plant systems and operation from the standpoint of reactor safety.

4.5 Current Review Status

In general the qualification of HICs appears to be a somewhat simpler process than that for solidification media, in the sense that: the HICs are finished products; they are produced (each HIC by a single vendor) under factory quality assurance procedures; they have material properties that are either well-established or that can be readily measured; and the properties can be used in design calculations. Prototypes can then be built and tested, and the test results can be checked against the calculations. In contrast, waste solidification media interact physically and chemically with the materials comprising the waste stream, and the resultant properties of the waste form are more difficult to predict and reproduce on a routine basis.

Four solidification media topical Reports have been reviewed and approved by NMSS (by the end of October 1988). The staff's evaluation Reports for solidification media topical Reports are carefully written to clearly specify the waste streams and concentrations and the method of preparation of the waste forms so as to ensure that the ensuing waste forms will exhibit characteristics similar to those held by the test specimens used in the qualifying tests.

As of October 1988, a total of 30 topical Reports has been submitted to NRC's Nuclear Material Safeguards for review. Of these, seven have been approved, six have been withdrawn, three have been returned, and fourteen are still under review. A summary of the review status, with a breakdown of the type of product covered by each topical Report, is presented in Table II.

5 SUMMARY DISCUSSION

In summary, 10 CFR Part 61 requires long-term (300 year) structural stability. Assurance of long-term structural stability is provided for the most part by conducting short-term tests and meeting acceptance criteria described in a Technical Position issued in May 1983. The tests called out in the 1983 Technical Position are, in most cases, based on ASTM or ANS Standards that were created for specific non-nuclear applications and materials. These tests, therefore, require some modification for radwaste materials, in either the methods for specimen preparation, the procedures used in the test, the interpretation of the test data, or the acceptance criteria used.

The most widely applied test and criterion identified in the Technical Position is the compressive strength test, which is recommended for virgin (otherwise untested) material as well as waste forms that have been subjected to various conditions of immersion, radiation, biodegradation, and thermal cycling. The current compressive strength criterion is 60 psi (raised from 50 psi in the 1983 Technical Position). The compressive strength test and the 60 psi criterion address the ability of the waste form to withstand the loads placed on the waste form at the bottom of a disposal trench at the time the waste is covered over. The criterion and the test do not address, except in an indirect way, the ability of the waste form to remain integral for 300 years. None of the Technical Position tests result in measurement of some intrinsic property that can be directly correlated with long-term (300-year) structural stability. The tests are simply indirect, short-term indicators of the potential long-term stability of the waste forms. They are intended to be generically applicable, but as evidenced by both field experience as well as laboratory tests, some waste forms have exhibited unstable behavior. In particular, there have been problems with cement-solidified wastes (notably bead resins and sludge), with low-viscosity bituminized waste, and with high-density polyethylene HICs.

There has been a rather complex evolution of the regulatory process for low-level radioactive waste forms, involving NRR, the Office of State Programs, the Agreement States, the vendors, and of NMSS. Under an agreement reached in 1983 with the Agreement States of Nevada, South Carolina, and Washington, the NRC provides centralized review of Topical Reports on waste form solidification media and HICs. Solidification media and HICs accepted by the States before this agreement continue to be accepted. In addition, variances and interim approvals have been granted to certain HICs and waste forms, while Topical Reports on the HICs and waste forms have been under review by NRC. As of October 1988, NRC has reviewed and approved three HIC and four waste solidification media Topical Reports, while six have been withdrawn and three have been discontinued.

Table II Topical report review status summary solidified waste form and high integrity containers (HICs)

October 24, 1988

Vendor	Docket no.	Type	Disposition
Waste Chem	WM-90***	Solidification (bitumen)	Approved.
General Electric	WM-88	Solidification (polymer)	Approved.
U.S. Gypsum	WM-51***	Solidification (gypsum)*	Approved.
Chichibu	WM-81	HIC (poly impreg/concrete)	Approved.
Nuclear Packaging	WM-45	HIC (ferralium/FL-50)	Approved.
Nuclear Packaging	WM-85***	HIC (ferralim/family)	Approved.
DOW	WM-82***	Solidification (polymer)**	Approved.
ATI (U.S. Ecology)	WM-91***	Solidification (bitumen)	Discontinued.
VIKEM	WM-13	Solidification/oil (cement)	Discontinued.
Stock	WM-92***	Solidification (cement)	Discontinued.
Nuclear Packaging	WM-71	Solid/Encap (cement/gypsum)	Withdrawn.
LN Technologies	WM-57	HIC (polyethylene)	Withdrawn.
Chem-Nuclear	WM-47	HIC (fiberglass/poly)	Withdrawn.
Chem-Nuclear	WM-19***	Solidifcation (cement	Withdrawn.
Chem-Nuclear	WM-96***	Solidification (cement)	Withdrawn.
Hittman	WM-79***	Solidification (SG-95)	Withdrawn.
Chem-Nuclear	WM-101	Solidification (cement #1)	Under review.
Chem-Nuclear	WM-97	Solidification (cement #2)	Under review.
Chem-Nuclear	WM-98	Solidification (cement #3)	Under review.
LN Technologies	WM-20	Solidification (cement)	Under review.
LN Technologies	WM-99	Solidification (cement/decon)	Under review.
Hittman	WM-46	Solidification (cement)	Under review.
ATI (U.S. Ecology)	WM-100	Solidification (bitumen)	Under review.
Chem-Nuclear	WM-18	HIC (polyethylene)	Under review.
Hittman	WM-80	HIC (polyethylene)	Under review.
TFC	WM-76	HIC (polyethylene)	Under review.
Nuclear Packaging	WM-87	HIC (316-stainless)	Under review.
LN Technologies	WM-93	HIC (stainless/poly)	Under review.
Bondico	WM-94	HIC (fiberglass/poly)	Under review.
Babcock & Wilcox	WM-95	HIC (coated carbon steel)	Under review.

* Approved for single waste stream for one year.
** Approved pending satisfactory completion of thermal cycling tests.
*** Actions completed in Calendar Year 1988.

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POTENTIAL APPLICATION OF NRC AGING RESEARCH TO LICENSE RENEWAL

J. P. Vora

Office of Nuclear Regulatory Research

U.S. Nuclear Regulatory Commission

ABSTRACT

A strong technical base, when developed and implemented, to manage aging in plant safety related systems, support systems, structures, and components will give confidence to all of us in the nuclear community toward the continuous safe operation of nuclear power plants of all ages. This technical base for managing aging must be built over the foundation of reviews and analyses of original designs, operating experience of over 20 years, experts' opinions, and the utilization of research results. The key elements of managing aging are the understanding of risk significance of aging phenomena and the licensee program(s) for inspection, surveillance, condition monitoring, trending, recordkeeping, and maintenance to mitigate the influence and effects of aging.

The following specific NPAR programs, implemented by the Division of Engineering of the Office of Nuclear Regulatory Research, address the technical safety issues related to aging and will provide the technical bases for the development of regulatory guides and review procedures for license renewal applications. These programs are (1) risk significance of component aging and prioritization, (2) hardware-oriented engineering programs for electrical and mechanical components, safety systems, and support systems; primary system pressure boundary components and materials and nondestructive examination; and civil structures, (3) reviews of technical specifications from the standpoint of aging, (4) development of guidance for recordkeeping to track and evaluate aging impacts, (5) residual lifetime evaluation and development of guidance for aging mitigation of large LWR components and structures, and (6) development of regulatory guides and review procedures for license renewal applications.

This paper provides an overview of the intended application of plant aging research results. In the process it delineates some key steps recommended for the development of appropriate regulatory guides and review procedures involving the technical issues for license renewal considerations. This application of aging research results is based primarily upon the NPAR theme of "Understanding Aging - A Key to Ensuring Safety, and Managing Aging - A Necessity to Ensuring Safety."

I. INTRODUCTION

All of us who are associated with commercial nuclear power, from operation and maintenance to regulation, want safe and reliable operation of nuclear power plants of all ages. That is, safe and reliable power production during the normal license period of 40 years and for extended life of operating nuclear power plants. A discussion, therefore, of the industry perspective for plant life extension and the NRC perspective on potential application of the NRC aging research program to develop technical bases for license renewal would be of interest.

A. Industry Perspective and Needs

In the U.S. today there are over 100 commercial nuclear power plants in operation. Some of these plants have been operating over 25 years, and they have been operating safely and reliably. For a few of these plants, the 40-year operating license will expire in the next 10 to 15 years. Therefore, in order for the associated utilities to meet the power demand of the early years of the next century, they will need to plan now, at least 10 to 15 years prior to license expiration, for various options and alternatives for generation and transmission of electric power to load centers. A viable option that is being considered is "plant life extension or PLEX."

In recent years, the nuclear industry has initiated a significant effort aimed at extending the life of existing reactors beyond their original license term of 40 years. The primary motivation for life extension is economic. It is expected that, by extending the life of a typical nuclear power plant with a generating capacity over 500 MW by 20 years, a net benefit of $800M to $1 billion can be realized. According to a completed Department of Energy study, the projected net benefit to the U.S. economy would be in excess of $100 billion assuming 20-year life extension for current reactors. If a longer (30-40 year) life extension is assumed, the benefits are even larger. The benefits reflect both lower projected fuel costs of nuclear plants relative to fossil plants and reduced outlays for replacement generating capacity if the useful life of current reactors can be extended.

The U.S. Department of Energy and the Electric Power Research Institute (EPRI) are evaluating the utility proposals for LWR lead plant license renewal. Their intent is to select at least two lead plant contracts, preferably one PWR and one BWR; then to submit license renewal applications to the NRC by 1991, expecting license extension by 1993.

To keep pace with these industry plans and prepare for the large number of submittals, the NRC will need to devote substantial efforts over the next several years to license renewal. This will require a firm NRC policy concerning the terms and conditions for license renewal to be in place by 1991.

From a technical perspective, the industry has recognized that some major plant components and structures are not readily refurbished and replaceable and have a propensity for age-related degradation. Therefore, both the industry and NRC need to focus on programs to determine residual life, including aging research. The industry on its part has undertaken a program to develop a screening methodology useful to identify risk significant components and structures that may experience age-related degradation during extended life. It is commonly viewed that reliable and operational readiness of risk significant components and structures subject to age-related degradation must be assured during their entire operating life.

Since the industry perspective on plant life extension has been discussed extensively in various technical conferences, workshops, and symposia over the last 3 years, further discussion is not warranted. But we should discuss in detail what is being done in the NRC Nuclear Plant Aging Research (NPAR) program and the potential application of research results useful to license renewal.

B. NPAR Perspective to Address Technical Safety Issues Related to Plant Aging and License Renewal

Aging, if it is not managed properly, affects the operational safety of all reactor structures, systems, and components, and it has the potential to increase risks to public health and safety. There are significant uncertainties with regard to aging-related degradation processes that affect key components and structures and about the way such degradation can be detected and managed before safety is impaired. Aging is a key concern in the operation of plants and will clearly be a crucial issue in any assessment of the safety implications of license renewal. Specifically, there is concern that simultaneous multiple failures of age-related components could occur during a transient or accident compromising safety system function.

The operating experience indicates that component failures have occurred because of corrosion, radiation, and thermally induced embrittlement of electrical insulation, pitting of electrical contacts, surface erosion, metal fatigue, oxidation, creep, and binding and wear. Therefore, as operating reactors advance in age toward the normal design life of 40 years and as we contemplate extended life, the major regulatory safety issues that exist are: (1) will aging of plant systems and components result in common mode failures that will weaken the defense-in-depth strategy, lead to an accident, or render inoperable the redundant but aged safety equipment needed in accident mitigation and (2) how to ensure operational readiness of aged equipment through the operating years.

II. TECHNICAL SAFETY ISSUES RELATED TO AGING OF COMPONENTS AND STRUCTURES

Aging affects all structures, systems, and components to some degree. If one attempts to define the common issues that apply generically to multiple structures, systems, and components, a large set of issues with fairly broad implications quickly emerges. Table 1, for example, summarizes one set of broad issues. The list is by no means exhaustive.

Table 1. Some Broad Aging-Related Technical Safety Issues for License Renewal

1. What risk significant structures and components are susceptible to aging effects that could adversely affect public health and safety? Which of these structures and components are renewable (by maintenance, refurbishment, replacement, etc.)?

2. What are the degradation processes of materials, components, and structures that could, if unchecked (improperly maintained and/or not replaced), affect safety?

3. Are current requirements for testing, inspection, surveillance, maintenance, and replacement adequate to detect and/or mitigate aging problems before they significantly impact safety? If not, what additional requirements are needed?

4. What criteria are required to evaluate residual life of components and structures? What supporting evidence (data, analyses, inspections, etc.) will be needed? What kind of independent NRC effort is required in consideration of license renewal?

5. How should structures and components be selected for comprehensive aging assessments, and residual life evaluations?

6. What additional changes will be needed in codes and standards to support relicensing, beyond those changes already in progress? What schedule should be followed?

7. What kinds of records and other documentation are needed to support license renewal?

III. NRC AGING RESEARCH PROGRAM

The NRC Nuclear Plant Aging Research (NPAR) program is directed toward gaining knowledge and understanding of degradation processes within nuclear power plants. This hardware-oriented engineering program is a rigorous and systematic investigation into the potentially adverse effects of aging on plant components, systems, and structures during the period of normal licensed plant operation, as well as the period of extended plant life that may be requested in utility applications for license renewals.

Emphasis has been placed on identifying and characterizing the mechanisms of material and component degradation during service and using research results in the regulatory process. The research includes evaluating methods of inspection, surveillance, condition monitoring, and maintenance as a means of managing aging effects that may impact safe plant operation. Specifically, the goals of the program are:

1. Identify and characterize aging effects that, if unchecked, could cause degradation of components, systems, and structures and thereby impair plant safety.

2. Identify methods of inspection, surveillance, and monitoring, and evaluate residual life of components, systems, and structures that will ensure timely detection of significant aging effects before loss of safety function.

3. Evaluate the effectiveness of storage, maintenance, repair, and replacement practices in mitigating the rate and extent of degradation caused by aging.

The entire NPAR program is based upon the theme of "Understanding Aging - A Key to Ensuring Safety, and Managing Aging - A Necessity to Ensuring Safety."

A. Understanding Aging

It is essential to understand the aging processes that occur in a system or a component before the age-related degradation can be effectively managed. To understand these aging processes, one must review the system or component design, fabrication, installation testing, inservice operation, and maintenance cycles. This is due to the fact that all of these elements in the life cycle of a system or component involves the interaction between its operational environment and the stressors associated with its materials.

B. Managing Aging

The NPAR approach to managing aging consists of two major elements. They are (1) to determine where in plant safety-related systems and support systems aging is significant to risk, and then, based upon operating experience and experts' opinions, prioritize components for indepth engineering studies, and (2) recommend effective maintenance to manage aging. A brief description on each of these program elements is provided below:

1. Risk Significance and Component Prioritization

Time-dependent calculations that take into account the effects of aging are necessary to identify and prioritize risk significant components, systems, and structures and then in turn to develop programs to understand and manage aging in those prioritized components, systems, and structures. Techniques for performing time-dependent risk or core-melt probability calculations need to be developed. The task to manage, including the allocation of proper resources, the effects of age degradation in nuclear power plants becomes much easier if the risk significant components and systems that will degrade during a normal operational or extended life are identified and prioritized. To accomplish this, a program that includes a hybrid approach (that is, a deterministic approach in conjunction with a probabilistic approach) must be developed. We must use the knowledge gained from engineering designs, applications, tests, and operating experience. Also, data from in situ assessments, condition monitoring, recordkeeping, and post-service examination and tests are essential for developing suitable deterministic models and for risk assessments and component prioritization. Expert panel workshops also are recommended in the prioritization process. Of particular concern are those components and systems, not easily or routinely inspected and maintained, that may degrade with age and impact upon plant safety.

2. Maintenance to Manage Aging.

Condition monitoring, trending, recordkeeping, and maintenance programs at nuclear power plants are extremely important to manage aging and reliability assurance. Effective maintenance programs will require understanding of what to maintain, when to maintain, and how to maintain plant systems, components, and structures. The earlier discussion on understanding aging degradation processes, risk significance of aging phenomena, and what items we should worry about provides clues on what to maintain. The key steps in determining when to maintain and how to maintain a specific system, a structure, or a component (s/s/c), from the aging perspective, are:

- Identify performance measures/functional indicator(s) for each of the risk significant and prioritized s/s/c, which would give an indication of its health at the time of observation;

- Then, identify methods to detect performance measures/functional indicator(s), in incipient state prior to failures;

- Trend performance measures/functional indicators for each s/s/c under observation and analyze the impact of rate of change;

- Develop a library of data/information/guidelines/criteria;

- Determine minimum functional capability at the end of normal service life;

- Determine minimum functional final capability to mitigate an accident; and

- Interpret, analyze, and make decision for maintenance, replacement.

Maintenance programs are needed, both predictive and preventive, to manage aging. A program to manage aging will aid in making decisions for corrective maintenance, quality assurance and quality control, engineering support, and plant modifications.

IV. NRC AGING-RELATED RESEARCH TASKS AND POTENTIAL APPLICATION OF RESEARCH RESULTS TO LICENSE RENEWAL

The major thrust of the hardware-oriented NRC aging research, considered useful to develop technical bases for license renewal, rests within the Division of Engineering (DE) of the NRC Office of Nuclear Regulatory Research (RES). The major elements/tasks of this DE/RES aging research program are:

- o Determine Risk Significance of Aging Phenomena; Aging-System Interaction Study; Component Prioritization

- o Indepth Aging Assessment of Electrical and Mechanical Components

- o Aging Assessment of Materials in Pressure Boundary Components

- o Aging Assessment of Concrete Structures

- o Residual Life Assessment of Major LWR Components and Structures

- Aging Evaluation at Shippingport Atomic Power Station
- Analysis of Technical Specifications from Aging and License Renewal Perspective
- Develop Recordkeeping Needs from Aging and License Renewal Perspectives
- Develop Technical Bases for Preparing Regulatory Guides and Review Procedures for License Renewal
- Develop Recommendations for Effective Maintenance Program to Manage Aging in Components and Structures
- Interactions with External Institutions and Organizations: DOE, EPRI, ASME-IEEE-ACI, NUMARC/NUPLEX, IAEA

From the review of the aforementioned listing of DE/RES aging related research program, it should be clear that the NPAR program is much more comprehensive than what is widely known as an aging study of some electrical and mechanical components that are routinely maintained and replaced by the licensees.

The following sections of this paper describe the overall scope or objective(s) of each of the major NPAR activities. Also delineated are the potential applications of research results to develop technical bases considered useful for license renewal.

NOTE: It should be noted that the extent of application of NPAR results to license renewal will depend upon (1) the content of the final rule, (2) industry initiatives, (3) involvement of national consensus codes and standards, and (4) availability of resources for the NPAR program.

A. Risk Significance of Aging; Aging-System Interaction Study; Component Prioritization

1. Risk Significance of Aging: Aging models and risk assessment methodologies require development to provide quantitative determination of the effect that aging has on safety. The major activities within this research element are: (a) aging model development, including the treatments of active components, passive structures, and the influence and effects of testing and maintenance; (b) failure data analysis, (c) engineering information analysis, (d) uncertainty analysis, (e) application and demonstration of the risk assessment methodology, and (f) the development of procedures and guidelines for treatment of aging in probabilistic risk assessments (PRAs).

2. Aging-System Interaction Study: An aging system interaction study is essential to determine how aging is affecting component and system unavailability and to establish the relative contribution to risk from age-related component and system failures. This element of the aging research program will facilitate the prioritization of plant safety systems and components for indepth engineering studies and

then gnerate guidelines and recommendations for inspection and maintenance to alleviate aging concerns.

Eight representative safety systems and support systems are being studied in the NPAR program. Phase I assessments, based upon the operating experience reviews, has been completed.

3. Component Prioritization: The risk-based criteria need to be established through the development and application of a state-of-the-art risk-based methodology, the Risk Significance of Component Aging and Aging Management Practices (RSCAAMP) model.

An RSCAAMP model allows the assessment of both the risk significance of component aging and the effectiveness of current practices for maintaining an acceptable plant risk level in the presence of component aging. An RSCAAMP model was developed by enhancing the Risk Significance of Component Aging (RSCA) methodology, which was developed to evaluate a component's contribution to plant risk due to aging. In the basic RSCA model, the change in a component's contribution to risk due to aging is a function of the component's importance to risk, the rate at which the component's failure rate is increasing as a result of aging, and the interval during which the component is aging.

An expert panel workshop was conducted to perform component prioritization. The process used is delineated in Figure 1. Of some 30 components evaluated, the aging of small safety related piping (6-10 in.) ranked at the top in the overall prioritization process.

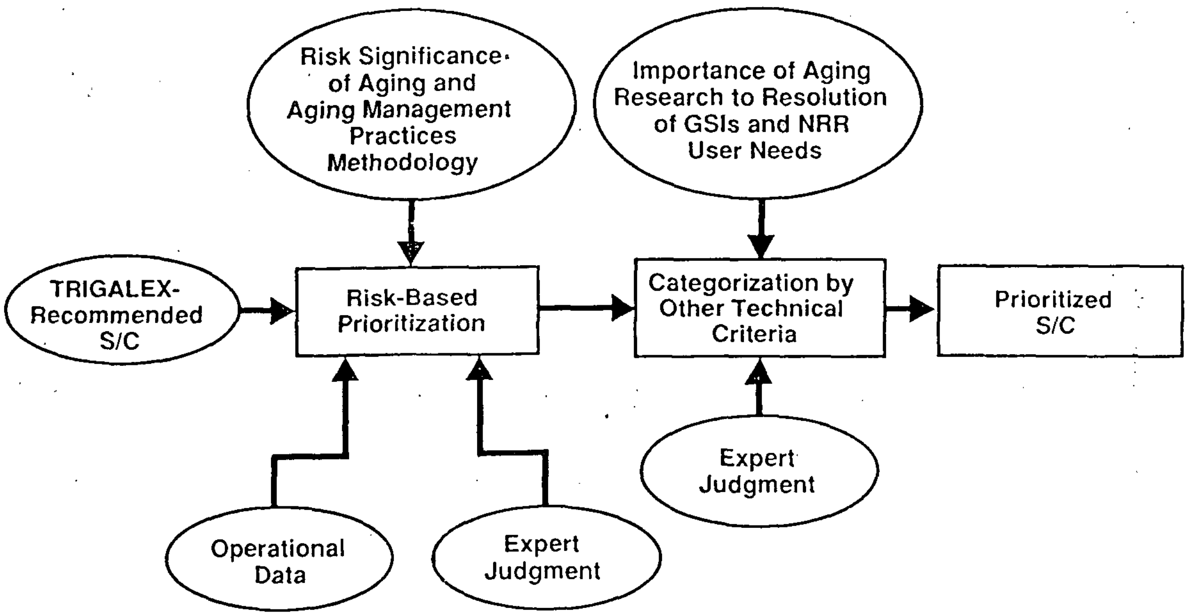


Figure 1. Overview of Workshop Process

TIRGALEX - Technical Integration Review Group for Aging and Life Extension

Potential Applications: Identify where in plant safety-related systems and support systems aging is significant to risk. Prioritize components and structures (c/s) that have a propensity for aging and are risk significant. Also, determine the trend in failure rates of safety-related active components with the age of the plant and evaluate the effectiveness of technical specifications and maintenance in managing aging. It is visualized that all of us who are involved in the license renewal/life extension-related activities, as a minimum, must have respective programs implemented to understand and manage aging in those prioritized and risk significant components and structures. This element of the NPAR effort is intended to develop methods to account for aging in PRA and develop technical bases for the preparation of regulatory guides and review procedures for license renewal.

B. Indepth Aging Assessment of Electrical and Mechanical Components

Search and analysis for aging assessment has been implemented on some 30 categories of components and systems considered risk significant and includes a reasonable cross section of representative electrical and mechanical components and systems. The selection has been based on experts' opinions; systems that are considered important for safety injection function; and systems that have support systems and interfaces of interest to the overall program. This segment of the plant aging research is being performed by five national laboratories and several private institutions and organizations. The research projects use the phased approach for aging assessment and recommendations for the utilization of research results in the regulatory process.

Phase I

Phase I, is based on readily available information from public and private data bases, vendor information, open literature, utility sources, and experts' opinions. In the Phase I component or system analysis, the product of the study is a preliminary identification of the significant modes of degradation and an evaluation of current inspection, surveillance, and monitoring methods. Based on these evaluations, recommendations are developed to identify detailed engineering tests and analyses to be conducted in Phase II. The Phase I evaluation of systems and components is used to decide if a Phase II assessment is warranted.

Phase II

In those cases where Phase II assessments are needed, they generally involve some combination of: (1) tests of naturally aged equipment or of equipment with simulated degradation; (2) laboratory or in-plant verification of methods for inspection, monitoring, and surveillance; (3) development of recommendations for inspection or monitoring techniques in lieu of tests that cause excessive wear; (4) verification of methods for evaluation of residual service lifetime; (5) identification of effective maintenance practices; and (6) in situ examination and data gathering for operating equipment.

With the completion of the aging assessment research on a component or system, a technical basis is available for utilization in the regulatory process. Examples of utilization include implementation of improved inspection, surveillance, maintenance, and monitoring methods; modification of present codes and standards; or development of regulatory guidelines and review procedures for plant license renewal.

The intent of the NRC-sponsored NPAR effort is to study a few selected electrical and mechanical components and a few representative safety systems and support systems; then, to demonstrate how the NPAR strategy can be applied by the industry to components, systems, or structures of interest. It is the industry's responsibility to characterize and evaluate their own plant systems, components, and structures and to ensure their operational safety as the plants advance in age.

Potential Application: It is the industry's general contention that components such as pumps and valves, breakers and relays, motors, and batteries are routinely maintained and periodically replaced and refurbished. Therefore, they pose no technical safety issues related to aging during extended life and they should not be factored into license renewal considerations. The industry wants to consider, for license renewal, only those major components and structures that are expensive to replace and refurbish and in some cases impractical to maintain. The author does not fully agree with this approach. From aging prospective the question we need to ask is, do we know which components to inspect and when? Do we know which components to maintain and replace and when? How do we determine optimized test intervals? Also, are there other components and interfaces, which are the integral parts of safety related systems and support systems but routinely are not inspected, tested, maintained, or replaced and may surprise us during the license renewal period? Therefore, the NPAR research for electrical and mechanical components as delineated in the program plan (NUREG-1144, Rev. 1) is needed. Therefore, regulatory guidelines are needed to manage aging in electrical and mechanical components as well. The potential application of this segment of the plant aging research will be to develop technical bases for extended life of the safety-related systems and components. Also to evaluate licensee programs for effective maintenance to manage aging in risk significant components and systems during extended life.

C. *Aging Assessment of Material Properties in Pressure Boundary Components

This element of the aging related research is focused on aging issues for the materials in components and structures of LWR primary systems. The goal of the program is to provide the confirmatory technical basis for regulatory decisions on the safe operation of reactor vessels, primary system piping, steam generators, and improvements in the techniques and equipment required for nondestructive inservice inspection of these components.

*Research sponsored by the Materials Engineering Branch, Division of Engineering, Office of Nuclear Regulatory Research

Potential Applications: The potential applications of this segment of the plant aging-related research can be described by the following anticipated regulatory products.

1. Reactor Vessels

   a. Basis for revision of ASME B&PV Code Section III, App. G (toughness) and 10 CFR Part 50 Appendices G and H (toughness and surveillance program requirements)

   b. Basis for fracture toughness curves for ASME B&PV Code Section III, Appendix G, and Section XI, Appendix A

   c. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials"

   d. Materials properties basis for 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"

   e. Basis for revision to ASME B&PV Code Section XI Appendix A, Crack Growth Rate Curves, and Section III S-N Fatigue Curves

   f. Basis for resolution of unresolved safety issue (USI) A-11, "Low Upper Shelf Toughness" - modification to 10 CFR Part 50, Appendix G (NUREG-0744)

   g. Recommend revision of standard review plan for license renewal evaluations

2. Piping

   a. Basis for acceptance of ASME XI Rules IWB-3640 and IWB-3650, for evaluation of flaws in stainless and carbon steel pipe

   b. Materials basis for the GDC-4 rule change to allow leak-before-break in specific piping systems and for the replacement of the double-ended, guillotine break criterion

   c. Confirmatory basis for NUREG-0313, "Materials Selection and Procession Guidelines for BWR Coolant Pressure Boundary Piping," which sets forth the acceptability of long-term crack "fixes" and replacement materials and provides the resolution of USI A-42, "Pipe Cracks in Boiling Water Reactors"

   d. Basis for revisions to ASME B&PV Code Section XI, Appendix A, Crack Growth Rate Curves, and Section III, S-N Fatigue Curves

e. Fatigue life data to review Standard Review Plan 3.6.2 on pipe break locations

f. Fracture toughness basis for Standard Review Plan 3.6.3 on leak-before-break evaluation procedures

g. Recommend revision of standard review plan for license renewal evaluations

3. Nondestructive Examination

a. Basis for ASME Code Section XI Case N-409, Rev. 1, "Procedure for Personnel Qualification Requirements for Ultrasonic Detection and Sizing of Flaws in Piping Welds"

b. Mandatory Appendix to ASME Code Section XI Procedures and Equipment for Improved Detection Sensitivity and Reliability through PISC-II Cooperation

c. Basis for update of ASME Code Section XI Procedures and Equipment for Improved Detection Sensitivity and Reliability through PISC-II Cooperation

d. Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds during Preservice and Inservice Examinations"

e. Nonmandatory appendix to ASME Code Section XI, "Continuous Acoustic Emission Monitoring of Nuclear Reactor Pressure Boundaries"

f. Basis for revision to Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes"

g. Basis for revision to Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes"

h. Basis for revision of ASME B&PV Code Section XI, Appendix IV, "Eddy Current Examination of Nonferromagnetic Steam Generator Heat Exchanger Tubing"

i. Recommend revision of standard review plan for license renewal application

4. Steam Generators

a. Basis for upgraded steam generators tube plugging criteria and for improved in-service inspection plans

b. Validate and evaluate methods for detection and sizing of flaws

D. *Aging Assessment of Civil Structures

As a group, concrete structures have a history of reliability and durability. However, there is no standard accepted method for quantifying the aged condition and capacity of an individual structure. Methods are needed to quantify aging if informed licensing decisions are to be made on extension of licensed design life of safety-related nuclear plant civil structures.

This segment of the aging research study involves an indepth assessment of the aging degradation of select concrete civil structures in nuclear plants. It includes identifying the principal structural safety issues; developing material property data for aged civil structures; and evaluating the functional capabilities of aged structures in a postaccident environment.

Potential Applications: Develop technical bases and generate guidelines to evaluate industry program(s) for aging assessment of safety-related structures, including (1) methods to inspect and monitor potential cracking at the prestressed tendon anchorage assemblies, (2) inspection and trending of cracks associated with metal liners, and (3) concrete deterioration due to chemical interaction resulting in aggregate expansion and cracking.

E. Residual Life Assessment of Major LWR Components and Structures

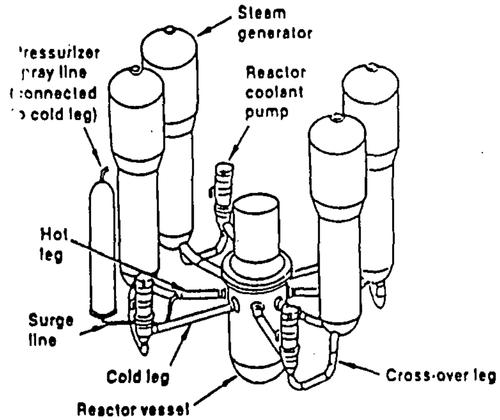
The objective of this element of the NPAR program is to develop technical bases and criteria for NRC to assess methods of mitigating the effects of aging on major components and structures for license renewal consider tions. Emphasis is to address the degradation of the major LWR components and structures caused by the synergistic influences of radiation embrittlement, thermal fatigue, corrosion fatigue, environmental attack, metallurgical changes, microbiologically induced corrosion, moisture intrusion, corrosion/erosion, etc. The major efforts are focused on integrating, updating, and evaluating the research results from related programs, including those sponsored by NRC, DOE, EPRI, NSSS vendors, nuclear utilities, and European and Japanese research institutes. Residual life assessment technologies of other industries are adapted if appropriate.

As of FY 1988, the major components important to plant safety have been identified and prioritized. An initial evaluation has been made of 12 PWR components, including the containment, pressure vessel, primary piping, steam generator, and vessel support; and of seven BWR components, including the pressure vessel, recirculation piping, and vessel supports. In this evaluation, the degradation sites, degradation mechanisms, stressors, and failure modes have been identified. This evaluation also includes a review of the current methods used for inspection and surveillance of these components. The results of this effort have been documented in NUREG/CR-4731, Vols. 1 and 2. Examples in summary form are illustrated in Figures 2, 3 and 4.

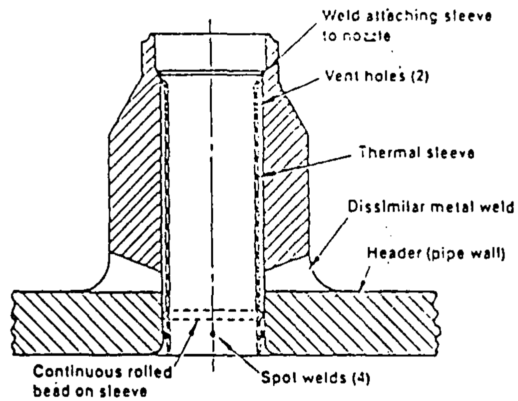
*Research sponsored by the Structural and Seismic Engineering Branch, Division of Engineering, Office of Nuclear Regulatory Research

Understanding and managing aging of PWR RCS piping and nozzles

Materials	Main coolant pipe	• Centrifugally cast SS-Gr. CF8A and CF8M (W), Type 304SS and 316SS (early W plants), SA 516 Gr. 70 (CE), SA 106 Gr. C (B&W)
	Fittings	• Statically cast SS - Gr. CF8A and CF8M (W); SA 516 Gr. 70, Type 309L SS (CE, B&W); Type 308L SS (B&W)
	Cladding	• Type 308L SS (CE), Type 304L SS (B&W)
	Surge line	• Type 316 SS, cast SS - Gr. CF8M (some CE plants)
	Spray line	• Type 316 SS
	Nozzles on main coolant pipe	• SA 105 Gr. 2 (CE), Type 304N SS (W)
	Thermal sleeve	• Inconel SB-168
Stressors and Environment	Operational transients, temperature, flow induced vibrations, stratified flows, thermal striping, and thermal shocks	



Typical RCS piping for W 4-loop plant



Typical nozzle-thermal sleeve assembly for a W plant

<table>
<tr><th colspan="2">UNDERSTANDING AGING
(Materials, Stressors, & Environment Interactions)</th><th colspan="3">MANAGING AGING</th></tr>
<tr><th>Sites</th><th>Aging Concerns</th><th colspan="2">Inservice Inspections, Surveillance, and Monitoring</th><th>Mitigation</th></tr>
<tr><td>Nozzles and thermal sleeves
Charging
Safety injection
Surge
Spray</td><td>Low and high-cycle thermal and mechanical fatigue</td>
<td rowspan="4">NRC requirements

Volumetric and surface examination of 25% of butt welds including the following welds during each inspection interval 10 CFR 50.55a, IWB-2500:
• All dissimilar metal welds
• All welds having cumulative usage factor equal to or greater than 0.4
• All welds having stress intensity range of 2.4 Sm

Same welds are required to be inspected during each inspection interval

Flaw detection and evaluation - 10 CFR 50.55a, IWB-3000

Leakage and hydrostatic pressure tests 10 CFR 50.55a, IWA-5000

Cycle counting of specified design transients Tech. Spec. requirement</td>
<td rowspan="4">Recommendations

Perform more frequent examination of nozzle welds having high cumulative usage factor

Determine fatigue damage by on-line monitoring of coolant and piping temperatures, pressures, and flow rates in nozzles and horizontal portions of piping during operational transients, stratified flows, and thermal shocks

Perform nondestructive examinations and loose parts monitoring to assess status of thermal sleeves

Develop use of acoustic emission method to detect crack growth in the base metal and welds

Develop techniques to monitor actual degree of thermal embrittlement in cast stainless steel piping:
• Analytical modelling of inservice degradation
• Metallurgical evaluation to characterize microstructure
• NDE to establish correlation between ultrasonic attenuation and fracture toughness

Monitor valve leakage in safety injection pipe

Develop UT to detect flaws in cast stainless steel piping</td>
<td rowspan="4">Maintain full flow in spray line and operate it continuously to prevent stratified flow and thermal shock conditions

Replace horizontal section of spray line with sloped section to prevent stratified flow condition

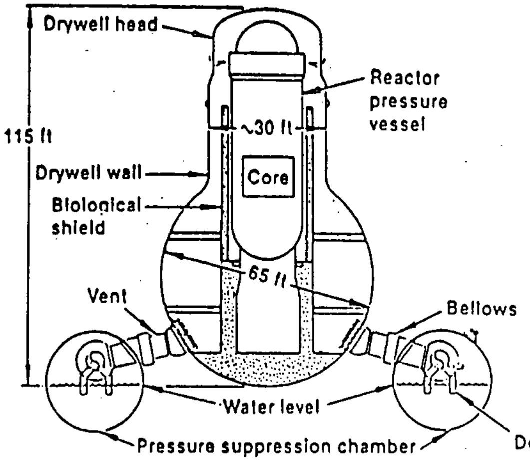
Redesign piping to eliminate valve leakage</td></tr>
<tr><td>Terminal end dissimilar metal welds (between carbon steel components and stainless steel piping)</td><td>Low-cycle thermal and mechanical fatigue</td></tr>
<tr><td>Surge line
Spray line</td><td>Low and high-cycle thermal and mechanical fatigue</td></tr>
<tr><td>Cast stainless steel piping
Hot leg
Cross-over leg
Cold leg
Fittings
Surge line</td><td>Thermal embrittlement</td></tr>
</table>

Figure 2

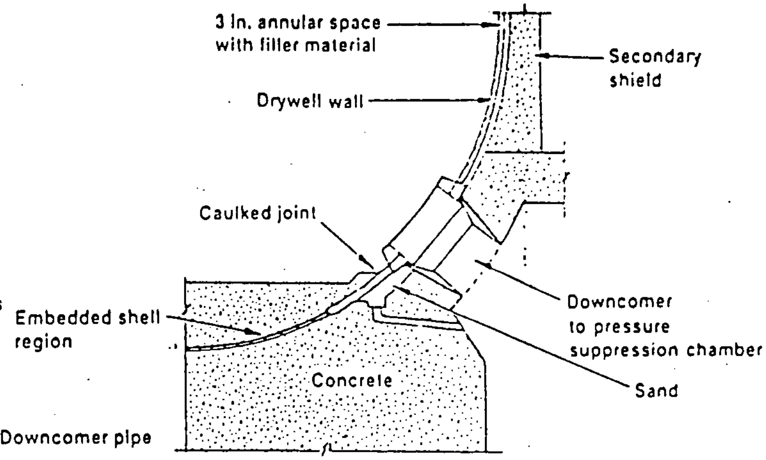
Developed by INEL under NPAR program

Understanding and managing aging of BWR Mark I containments

Materials	Shell	• Carbon steel • SA-516 Gr.70, SA-212 Gr. B
	Bellows	• Type 304 Stainless Steel
	Coatings	• Zinc rich, red lead and epoxy
Stressors and Environment	Corrosive internal environment, temperature, humidity, oxygen content, degraded fill material, moisture, microorganisms, cyclic thermal loading, leak tests	



BWR Mark I type metal containment



BWR Mark I drywell base, concrete shield wall and sand pocket.

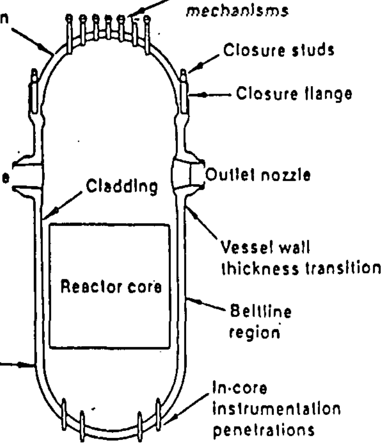
UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)				MANAGING AGING	
Sites		Aging Concerns		Inservice Inspections and Monitoring	Maintenance
Drywell	Exterior surface near sand pocket (unsealed gap)	Aqueous corrosion and microbial influenced corrosion	Thermal, mechanical, and environmental fatigue	NRC Requirements Leak tests • 10 CFR 50 App. J Recommendations ASME subsection IWE • 25% visual examination of pressure retaining welds, and coated and uncoated surfaces during each inspection interval • Being reviewed by NRC to include in federal regulations Visual examination of caulked joints at embedded regions Boroscopic examination of exterior surface near penetration Wall thickness measurements Surface examination of bellows Monitoring coolant leaking from faulty bellows	Recommendations Maintain surface coatings Check bellows alignment Maintain caulked joints at embedment region
	Exterior surface (degraded fill material present)	Pitting and crevice corrosion			
	Embedded shell region (deteriorated caulked joint at concrete-metal interface)	Pitting and crevice corrosion			
	Pipe penetrations Vent pipes	Galvanic corrosion			
	Exterior and interior surfaces (deteriorated coating)	Uniform attack			
Pressure Suppression Chamber	Interior surface (deteriorated coating)	Pitting	Thermal, mechanical, and environmental fatigue		
	Near waterline	Differential aeration			
	Below waterline	Microbial influenced corrosion			
Bellows	Heat-affected zone	Intergranular stress corosion cracking	Thermal and mechanical fatigue		
	Cold-rolled portion	Transgranular stress corrosion cracking			

Figure 3

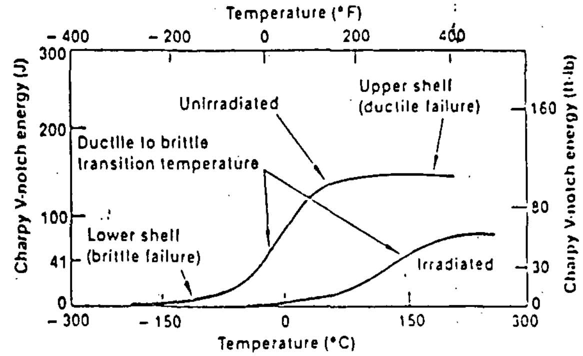
Developed by INEL under NPAR program

Understanding and managing aging of PWR reactor pressure vessels

Materials	Vessel	• Low alloy carbon steel - SA-533B-1, SA-508-2, SA-302B
	Cladding	• Austenitic stainless steel - Type 308 or 309
	Weldments	• Submerged arc (granular flux - linde 80, 91, 124 and 1092 manganese-molybdenum nickel filler wire) narrow gap submerged arc, shielded metal arc, and electroslag
	Closure studs	• SA-540 Gr. B24 Class 3
Stressors and Environment	Neutron flux and fluence, temperature, reactor coolant, cyclic thermal and mechanical loads, boric acid leakage	



Typical PWR vessel showing important degradation sites.



Effect of irradiation on the Charpy impact energy for a nuclear pressure vessel steel.

UNDERSTANDING AGING (Materials, Stressors, & Environment Interactions)		MANAGING AGING		
Sites	**Aging Concerns**	**Inservice Inspections, Surveillance, and Monitoring**		**Mitigation**
		<u>NRC Requirements</u>	<u>Recommendations</u>	
Beltline region	Irradiation embrittlement — Chemical composition of vessel materials (Cu, Ni, P) — Drop in upper shelf energy — Shift in reference nil-ductility Transition temperature Environmental fatigue	Surveillance program to assess irradiation damage, i.e., shift in RT_{NDT} and drop in USE - 10 CFR 50 App. H Reg. Guide 1.99, Rev. 2 - acceptable PTS screening criteria - 10 CFR 50.61 Damage evaluation - 10 CFR 50 App. G Volumetric examination of one weld during each inspection interval - 10 CFR 50.55a, IWB-2500 Flaw detection and evaluation - 10 CFR 50.55a, IWB-3000 Leakage and hydrostatic pressure tests 10 CFR 50.55a, IWA-5000	Include fracture toughness and tensile test specimens in surveillance program Develop use of reconstituted and miniature specimens Accelerated irradiation of reconstituted specimens Revise Reg. Guide 1.99, Rev. 2 to account for phosphorus with low copper Perform volumetric examination of all welds during each inspection interval Use state-of-the-art NDE techniques for improved reliability of defect detection, sizing, and characterization Use fatigue crack growth curves ASME SC XI, Appendix A Develop acoustic emission monitoring to detect crack growth	Flux reduction Inservice annealing ASTM E 509-86 Determine effects of annealing and reembrittlement rate
Outlet/inlet nozzles	Environmental fatigue Irradiation embrittlement Function of nozzle elevation (Potential impact of Reg. Guide 1.99, Rev. 2)	Volumetric examination of 100% nozzle-to-vessel welds and nozzle inside radius section during each inspection interval - IWB-2500 Volumetric and surface examination of all dissimilar metal welds during each inspection interval - IWB-2500	Use on-line fatigue monitoring Evaluate irradiation embrittlement damage	
Instrumentation nozzles CRDM housing nozzles	Environmental fatigue	Visual examination of external weld surface of 25% nozzles during system hydrostatic test - IWB-2500		
Flange closure studs	Environmental fatigue *Boric acid corrosion (if leakage occurs)*	Volumetric and surface examination of all studs and threads in flange stud holes during each inspection interval - IWB-2500		

Figure 4

Developed by INEL under NPAR Program

Potential Applications: For each of the major LWR component and structure, (1) identify and address technical safety issues related to plant license renewal, (2) evaluate and define surveillance and maintenance methods to support license renewal, (3) recommend revisions of codes and standards to support license renewal, (4) identify additional research projects needed to resolve the technical safety issues related to plant license renewal, and (5) develop and/or evaluate age degradation estimate models.

F. Aging Evaluation at Shippingport Atomic Power Station

The examination and testing of naturally aged nuclear power plant components is an important element of the U.S. Nuclear Regulatory Commission NPAR program strategy. The Shippingport Atomic Power Station, now in the latter stages of decommissioning, has been a major source of naturally aged materials and equipment for these NPAR evaluations and for other NRC programs.

More than 140 naturally aged components and samples, ranging in size from small instruments and metallurgical specimens to one of the main coolant pumps, have been removed in conjunction with the decommissioning of the Shippingport Atomic Power Station and shipped to designated NRC contractors. In situ assessments of selected Shippingport station systems and components also have been conducted. Although the detailed evaluation of the naturally aged components and materials from the Shippingport station is just beginning, there are a number of preliminary studies and results that are indicative of the value of the aging information that ultimately will be obtained.

Potential Applications: (1) evaluation of condition monitoring methods for components such as motor-operated valves and cables inside containment, battery chargers and inverters, and electric motors, (2) perform post-service examinations and tests of naturally aged components and materials to determine their reliability and operability, (3) evaluate metallurgical changes due to natural aging process and generate technical basis for evaluating the integrity of pressure boundary structural materials.

G. Analysis of Technical Specifications from Aging and License Renewal Perspectives

From the standpoint of aging and license renewal, technical specifications are important as methods for detecting and managing aging degradation (e.g., surveillance, testing).

As an illustration, surveillance requirements in the technical specifications serve as the major means of detecting problems in dynamic plant equipment (e.g., pumps, valves, circuit breakers, and switches). However, this portion of the technical specifications has not changed to reflect increased understanding of aging. The aging and license renewal program provides an excellent opportunity to further review and revise the surveillance requirements to better reflect the knowledge gained through operating experience, reliability information, insights on the safety significance of particular components gained from PRA studies, and detection of equipment aging. The goal would be to see if better detection of incipient failures could be achieved without imposing a greater total surveillance burden. This might be accomplished by (1) requiring less frequent testing but making the tests more closely approach event or accident conditions, (2) by eliminating tests that provide little or no safety information, and (3) using PRA results that show that the frequency of existing tests may be reduced with only a minimal impact on reliability assurance.

Potential Applications: Generate guidelines, from aging perspective, for improved quality of test data and reduce quantity and number of tests. The potential applications of the research results from this element of the NPAR program will be realized from three perspectives: (1) for their adequacy in considering aging degradation mechanisms, (2) as potential contributors to aging (e.g., technical specification requirements for frequent fast starts of diesel generators or the loads imposed on the auxiliary feedwater pump while testing in a pumping mode), and (3) as methods for detecting and managing aging (e.g., surveillance testing).

H. Recordkeeping Needs from Aging and License Renewal Perspectives

It is anticipated that appropriate recordkeeping will be needed to demonstrate adequate understanding of aging degradation processes and the management of their effects. Adequate recordkeeping is needed on such items as transients, component failures, and repair and replacement of components. The NRC needs to establish a clear regulatory position on this subject as soon as possible since records being generated now will be useful to provide technical bases to support license extension in the future. Guidance under this task will establish needs for recordkeeping on (1) component design specifications (materials, etc.) and applications and performance specifications; (2) stressors (electrical, mechanical, and thermal); (3) environments (chemical, radiation, atmospheric humidity, etc.); (4) root cause data; (5) failure and age tracking data; (6) condition monitoring, including performance indicators, detection and monitoring methods, and trending; and (7) data on component maintenance/refurbishment/replacement activities.

Potential Applications: Develop guidance for recordkeeping to support specific data needs required to track and evaluate aging impacts on component reliability and system unavailability during normal design life and extended life.

I. Develop Technical Bases for Preparation of Regulatory Guide(s) and Review Procedure(s) for License Renewal (Addresses only the technical safety issues related to aging)

The rulemaking process, which is in progress, is scheduled to lead to a license renewal rule in 1991. In addition to a final rule, more detailed regulatory guidance to address the technical safety issues related to aging is needed to implement the rule and to advise licensees on license renewal application requirements. The NPAR program has anticipated the need for a timely strategy and guidance for implementing the license renewal rule by initiating, in 1988, a scoping study aimed at developing regulatory guidance and review procedures for nuclear power plant license renewal. The overall goal of this effort is to provide the technical bases for detailed guidance and requirements needed to implement the rule as it is finally developed in 1991. This approach will complement the rulemaking process and will allow the NRC to prepare for license renewal review in an orderly and timely way. As guidance development proceeds in parallel with rulemaking, each process will generate technical information of potential benefit to the other. This synergism between the two processes will lead to better and more timely products from both processes.

The recommended approach includes (1) the time-phased activities envisioned under the NPAR task; (2) related activities being pursued in the rulemaking process; and (3) opportunities for interactions between the two processes. Both, the NPAR task (regulatory guidance development) and the rulemaking effort are responsive to issues raised by TIRGALEX.

Potential Applications: (1) develop a standard approach and format for nuclear power plant licensees to follow, to address technical issues related to plant aging in applying for an operating license renewal, and (2) develop recommendations for a standard review plan and procedure (limited to technical issues) related to plant aging for NRR review of license renewal applications.

J. Develop Recommendations for Effective Maintenance Program to Manage Aging in Components

Maintenance, in its broadest sense, is one of the keys to managing aging and will play a pivotal role in life extension/license renewal. The Surry feedwater pipe break, the North Anna steam generator tube rupture, and the Aloha 737 accident are some recent events that confirm the premise on which the NPAR program evaluation of component maintenance effectiveness to alleviate aging concerns is based. That premise is that component aging, if not adequately managed, will lead to component degradation and failure, which will result in (1) reduced component reliability, (2) increased system unavailability, and (3) a concomitant increase in overall plant risk.

To identify the factors that can contribute to adequate management of component aging, the NPAR program has focused on resolving three major issues with respect to maintenance:

1. What components, systems, and structures (C/S/Ss) to maintain, which have propensity for aging and are risk significant,

2. When to maintain them, and

3. How to maintain them.

The resolution of these major issues will lead to a description of those factors necessary for an effective maintenance program based on aging research.

Potential Applications: The results of the NPAR evaluations will be component-specific recommendations for "when to" and "how to" maintain to manage aging.

NPAR studies are focused toward demonstrating the importance of maintenance programs in mitigating the impact of component aging on plant risk. Appropriate risk methodologies coupled with proactive measures to understand aging mechanisms and their root causes can identify: the important systems, components, and subcomponents to be evaluated; the aging failure modes that need to be investigated; the optimum inspection practices to be used given the current state of the art in detection and mitigation methodologies; and the need for new detection and mitigation methodologies when the current inspection, monitoring, and maintenance practices cannot manage aging.

K. Interactions with External Institutions and Organizations

An important element of the NPAR program is to ensure that all ongoing research activities are integrated within NRC and coordinated with external institutions and organizations, including ASME, IEEE, DOE, EPRI, NUMARC-NUPLEX, and IAEA.

Potential Applications: Considering the large number of ongoing aging related activities, the potential benefit of program coordination and information exchange is to minimize duplication of efforts and maximize the effectiveness of the NPAR program.

V. SUMMARY AND CONCLUSIONS

- Considerable importance must be placed on understanding and managing aging effects within operating nuclear power plants of all ages. The primary safety concerns are the potential reduction of defense-in-depth and common mode failures attributable to aging and aging-system interactions.

- To understand the significance of plant aging on safety is to understand how plant risk changes because of the aging effects. It must be realized that this risk is time dependent. When the properties of structures and components degrade, their reliabilities can degrade. The reliability is determined by such quantities as the frequency of an initiating event (such as a pipe break), the failure rate, and the unavailability of components and systems. When the reliabilities of structures and components degrade, the safety and risk of the plant can be adversely affected.

- The mitigation of the effects of component and system aging on plant safety and the extension of plant life cannot be achieved only through regulation by the NRC. Ultimately, it is the plant operator's responsibility to ensure continued safe operation of its plants. To do this, one must understand the aging and degradation processes in plant safety-related systems, components, and structures; develop a program of surveillance, monitoring, trending, recordkeeping, and analysis to mitigate the effects of aging; and then commit to implementing a rigorous maintenance program to ensure plant safety throughout its operational life.

- Although specific NRC requirements for a license renewal are not defined, in the author's opinion it is imperative that the "aged" condition of the plant will have to be considered and programs must be implemented <u>now</u> to alleviate aging concerns during extended life.

- The NPAR program is developing and integrating the vast amount of aging-related data so that the technical safety issues related to license renewal are identified and resolved in an effective and timely manner. Program coordination and technical integration are important elements of the NPAR program.

NUMARC View of License Renewal Criteria

Donald W. Edwards

NUMARC NUPLEX Licensing Group Chairman

The Atomic Energy Act and the implementing regulations of the U.S. Nuclear Regulatory Commission (NRC) permit the renewal of nuclear plant operating licenses upon expiration of their 40-year license term. However, the regulatory process by which license renewal may be accomplished and the requirements for the scope and content of renewal applications are yet to be established.

Given that the expiration of existing licenses will not begin until the year 2000, it is reasonable to ask why we should be concerned with this issue today. License renewal is a "today issue" for several important reasons. The year 2000 seems much closer when one considers the numerous technical issues which must be resolved and the fact that a licensing process (which does not now exist) must be put in place. A reasonable and predictable regulatory process is absolutely necessary to enable licensees to make long-range planning decisions regarding life extension investments versus other options. Technical feasibility alone is not sufficient to make license renewal an attractive option.

Utilities inherently require a substantial generation planning horizon. For instance, the typical increment of capacity to be replaced for a retiring nuclear plant is on the order of 600 MW(e) to 1,00 MW(e); thus, major siting considerations could be involved. If comparable new sites and distribution assets are required and current technology is to be used for replacement, up to 12 years will be needed for planning, site approval, designing, licensing, constructing, and testing new base load capacity. Obviously, there are some individual cases where the planning horizon could be much less than 12 years, e.g., reserve margins or purchase power agreements might be available to the system to make up for the retiring plant. However, in the context of the entire national electric grid, i.e., replacing 47,000 MW(e) of electrical capacity over an 11-year period, combined with meeting any load growth at all, a 12-year lead time is optimistic.

Also, within this 12-year lead time for capacity replacement, there is a shorter and perhaps more important threshold for license renewal action involving two intangible but significant issues. First, is the retention of employees - especially the plant staff. Through its years of operation, licensees establish and maintain a skillful and experienced staff. The level of experience is expected to be maintained through the plants entire operating life via careful recruitment and career development. However, as with the end-of-life for any type of facility, approaching licensing expiration with no definitive action to extend operations will likely encourage existing employees to relocate and make it extremely difficult to attract highly qualified replacement employees. Second, capital investment requires an assured and sufficient payback period, which might not be available during the final years of an initial license. Licensees continuously refurbish and upgrade components throughout the life of a plant which, in many instances, required significant capital investments. As the expiration of the license approaches, it may not be possible to justify large capital improvements to the plant that are above and beyond those required to meet regulations.

In short, if the industry is to be allowed appropriate lead time to rationally plan for future generation needs, key details of the license renewal process must be established very soon and probably no later than 1991.

On August 29, 1988, the NRC published an Advanced Notice of Proposed Rulemaking regarding the subject of license renewal. This Advanced Notice and the NUREG which it references, NUREG-1317, "Regulatory Options for Nuclear Plant License Renewal," provide the most recent regulatory thought on this issue.

The basic issue addressed by NUREG-1317 is the definition of an adequate licensing basis for the renewal of a plant license. The report contemplates three alternatives in this regard. First, the renewal could be based on the original licensing basis of the plant, as amended. Second, it could be based on the licensing requirements for a plant that would be initially licensed at the time a renewal application is submitted. Finally, it could be based on a modified version of the original licensing basis supplemented in safety significant areas. This approach could focus on requiring conformance either to standards which are specifically developed to be consistent with the safety goals, or to a subset of current standards that are particularly relevant to the risk-significant aspects of plants requesting license renewal.

Let us discuss each of these three proposals starting in reverse order with Option 3. The NUMARC NUPLEX Working Group strongly opposes this option and has serious concern for the method of implementation suggested by NUREG-1317. Use of standards based on as yet undefined applications of safety goals or some unspecified combination of SRP and risk-significant considerations is impractical and reinforces the concern that NRC staff is using license renewal as an opportunity for arbitrarily raising currently-accepted levels of protection of public health and safety. As described, the method of implementing this option is unacceptable because of its open-endedness and potential for regulatory instability. If selected, this approach could preclude license renewal for many plants, not for technical reasons, but for the perceived regulatory risks and associated cost uncertainties. Therefore, we feel that this approach should be rejected.

A second option discussed in NUREG-1317 is the application of licensing requirements for new plants to renewal applications. This option would involve the issuance of a new license that would be based on the use of "existing procedures for granting an initial operating license." We understand NRC to intend, under this option for license renewal, the complete application of the regulations concerning the issuance of operating licenses for nuclear power reactors. This would include, as we understand it, the application of the provisions of 10CFR, Part 50, including the requirements for the content of operating license applications found in Section 50.34, the notice and hearing procedures for operating licenses found in 10CFR, Part 2, and all other provisions of Title 10 as they apply to initial operating license for nuclear power reactors.

No regulatory reason exists to justify a complete re-examination, at the time of license renewal, of an operating reactor's design and construction. NRC, by the issuance of the operating license and the continued operation of the facility for the initial 40-year term, has determined that operation of the facility is consistent with the protection of the public health and safety. Licensees maintain this protection by constantly implementing a program of inspection and maintenance, and when necessary, replacing or upgrading equipment and systems to meet changing design and operational requirements. NRC assures this result by its continual surveillance of licensees' facilities.

The application of the regulatory guidance for new plants as the licensing basis for the renewal of maximum term operating licenses, as NRC recognizes, would preclude life extension for many utilities. The loss of this future generating capacity would be tragic. The Act does not require that all plants must meet the regulatory guidance established for new plants. Rather, the statute allows NRC to exercise its sound technical judgment to protect, with reasonable assurance, the health and safety of the public. We believe such protection will be achieved after the consideration of pertinent age-related degradation issues based on the current licensing basis.

For these reasons, it is unnecessary and unwarranted to completely re-examine reactor design and construction matters at the time of license renewal. Rather, NRC's review of the renewal application should focus on those safety-significant matters directly related to the renewal request; that is, pertinent age-related degradation issues. In summary, processing a renewal application in the form of a new license that must meet the guidance established for new plants is neither necessary to adequately protect the health and safety of the public nor required by law. This option should be rejected.

Let us now consider NUREG-1317's Option 1. This option is characterized as one where the "as is" condition of the plant at the time of license expiration is accepted as the licensing basis for license renewal purposes. NRC recognizes that some modifications may have been made to the original licensing basis, but the option is criticized as depending on a licensing basis which is "out dated" and "often times poorly recorded." It is strongly implied in NUREG-1317 that this haphazard state of affairs is typical of the licensing basis of many, if not all, operating reactors. This is not the case. The licensing status of the approximately 109 operating reactors in this country is subject to the disciplined regulatory regime discussed above. Moreover, to suggest otherwise, as NUREG-1317 implies, is a disservice to NRC regulators who are discharging responsibly the obligation to assure that power reactor operations are conducted in a manner that protects the public health and safety. The portrayal of the condition of original licensing basis in NUREG-1317 does not depict regulatory reality, and that characterization should be discarded from further rulemaking consideration.

The NUMARC NUPLEX Working Group endorses a license renewal process based on a plant's current licensing basis along with an evaluation of the pertinent components, systems, and structures affected by age-related degradation.

To warrant initial licensing and continued facility operation, a licensee must have demonstrated to the NRC that those conditions listed in Subsection 50.57a of the Commission's regulations are, and will continue to be, met. An appropriate regulatory objective for the NRC is that a facility licensee seeking approval to operate under a renewed license would be required to meet that same standard during its renewal term as during the initial license term. Except where a change is justified by the application of the backfit rule, a facility's compliance with the Subsection 50.57a conditions should be considered established so long as the facility continues to adhere to the regulatory requirements resulting from its "current licensing basis" in effect at the time of submittal for renewal.

The NUMARC NUPLEX Working Group believes that an appropriate scope for NRC review of the license renewal application should focus on those safety-significant structures systems, and components subject to significant age-related degradation that are not subject to existing recognized effective replacement, refurbishment, or inspection programs. Such a review would provide adequate assurance that systems, structures, and components significant to plant safety and subject to age-related degradation will continue to comply with license requirements applicable to the plant during the renewal term. In order to provide such assurance, a renewal application should include the following:

A. A screening of a plant to identify those structures, systems, and components which are safety significant and which have a limited lifetime because of age-related degradation process(es) that could impair functional performance, and that are not mitigated through monitoring and maintenance. Those components within a system which are not important to maintaining the system's safety function(s) or which do not incur significant degradation, would be identified as such and excluded from the detailed age-related degradation analysis.

B. A description of inspection and maintenance programs, intended for use during the term of the renewal license, that ensures consistency with the current licensing basis for those structures, systems, and components that have been identified above in Subparagraph A. This explanation should be sufficient to explain the technical basis for, and demonstrate the adequacy of, such programs. Those components which are shown to be routinely or periodically replaced or refurbished in an adequate manner, or which are subject to detailed inspection as part of the licensee's programmatic maintenance or surveillance program, and which are routinely reviewed by the NRC, would be identified as not requiring further scrutiny in the license renewal context.

C. An analysis to determine whether the identified safety-significant structures and systems will continue to comply with the current licensing basis. This evaluation shall use historical data and/or inspection techniques as appropriate. The relative importance to safety of these structures, systems, and components may be established by deterministic and/or risk assessment methods, and the results of such analyses may be considered in determining the safety significance of any predicted future variance from the current licensing basis.

D. A description of modifications or replacements of facility systems, structures, or components, if any, and of revisions to the current licensing basis, if any, including Technical Specifications, proposed for the facility during the term of the Renewal License to maintain compliance with the current licensing basis. These must be sufficiently complete to explain the technical basis for, and demonstrate the adequacy of (in terms of the Subsection 50.57a conditions) such modifications, replacements, or revisions.

E. A confirmation that the licensee will continue to be technically qualified to engage in the activities authorized by the Renewal License; and

F. An assessment of the environmental impacts associated with the continued operation of the facility for the period of the renewal license.

Let us now briefly discuss NUMARC's view of the role of the "Backfit Rule" in the license renewal process. The "Backfit Rule," as set forth in the Commission regulations (10CFR50.109), provides an appropriate procedural basis for the evaluation of new regulatory requirements potentially applicable to a facility seeking license renewal. The rule provides for backfitting to be carried out in the event that it is necessary to assure adequate protection of the public health and safety or, as determined in a cost-benefit analysis, if it would result in a substantial increase in the level of that protection sufficient to justify associated costs. Accordingly, where new and potentially relevant regulatory requirements have been established during the term of the operating license, the provisions of Subsection 50.109 are applied to determine whether such requirements should be imposed as a condition of continued operation. This should also be the situation for license renewal.

As a matter of policy, Subsection 50.109 should be applied to any staff determination seeking changes of hardware or procedure from those provided for, or permitted by, the existing license. Where the Backfit Rule has been applied in the past to determine that a facility need not comply with new requirements, a further analysis may be appropriate to determine whether the previous conclusions would still be correct under a renewed license because of the additional time involved.

In summary, the NUMARC NUPLEX Working Group believes that it is essential that the NRC's License Renewal Policy include the following basic elements:

Current Licensing Basis. Use of a plant's "Current Licensing Basis" should be the fundamental judgment criterion for determination of adequacy for the renewal license.

Scope of Review. NRC's review should focus on those safety significant components subject to age-related degradation and which are not treated by routine inspection, refurbishment or replacement.

Backfit Rule. The cost-benefit criteria and disciplined process of the Backfit Rule must govern any attempts by the staff to impose changes to the plant licensing basis, plant hardware or procedures.

Industry Initiatives in Support of License Renewal

Louis O. DelGeorge
Assistant Vice President
Commonwealth Edison Company

ABSTRACT

The United States commercial nuclear power industry has made many pioneering contributions since its inception in the 1950's. The world has followed this industry's lead in designing a power source that is safe, reliable, and economical. One of today's great challenges is to successfully blaze new pioneering trails into the frontiers of plant aging. As was the case in the 1950's, and perhaps more so today, the full realization of the economic life of our power production facilities - mindful, of course, of the public safety - is both the challenge and the mandate presented to us.

The stakes are high and objectively understandable. All new things age. If the aging process is not understood and to the extent possible controlled, the benefits of the aged resource are lost. Studies have shown that such losses could amount to more than half of this country's nuclear power capacity between the years 2010 and 2025. The replacement cost of that loss would be staggering -- in the hundreds of billions of dollars.

Action to mitigate these losses is necessary and is being taken by the nuclear industry. Beginning in the mid-1970's and escalating to the present, U.S. electric utilities have considered the affects of aging on their power production assets, both fossil and nuclear. This paper will review the history of the nuclear industry effort and describe its present focus and outlook.

Today, perhaps to a greater extent than was the case in the days of Atoms for Peace, our programs must be scrutable, defensible, and practicable. The public at large, our customers and stockholders, and perhaps most importantly future generations, demand success. For these reasons, our research -- that scholarly and scientific investigation of things heretofore unknown -- must be made clear enough and technically safe enough and with the right price tag so that all would agree that the trail we have chosen is the right one.

INTRODUCTION

Some forty years ago when the first determined thoughts of converting the power of the atom to commercial use saw birth, who amongst the technology's forebearers were thinking about the gerontology of nuclear power plant life. As is the case for most of us in our youth, thoughts of the future are of development, accomplishment and perhaps even eternal life. Aging is a concept foreign to the young, but, inevitable in its progress. Just as we human beings eventually acquire an interest in, "and if we are fortunate," experience the affects of aging, so, too, has nuclear power plant life become a topic of increasing interest to us. In the 1950's, we converted a technology forged in wartime to the wonder of commercial nuclear power.

Today, our challenge is no less demanding and the rewards equivalent in magnitude. Over 100 United States nuclear power production facilities generating over 100,000 megawatts of electric power in the service of the American people are operating and aging. The reality of this aging is no more to be feared than the reality of our own aging. Whereas the challenge to each of us is to make the most productive use of our life, the central objective of nuclear plant license renewal is to sustain the productive and safe use of operating nuclear plants. It would be easy enough to convince even the skeptics among us that the premature extinction of one's life would have a significant affect on our work and our family. It is no less important that those of us here in a position to address the license renewal issue assure that commercial nuclear power not prematurely diminish in its service to the American public.

It is true that license renewal should rest on principles that ensure "the continued adequate protection of the public health and safety."(1) Let us focus our attention, therefore, on those principles and not on unjustified fears of aging as a concept. This paper will discuss the three "R's" of license renewal: <u>Relevance of the inquiry</u> as to the affects of aging, <u>Reliability</u> of the <u>data</u> to support decisionmaking, and, finally, <u>Repeatability</u> of the <u>criteria</u> employed to judge fitness for service. Each of these principles is necessary to the effective implementation of a license renewal process that will, both in concept and in fact, protect the public health and safety while preserving the effective utilization of a developed national energy resource.

What Is Aging

No nuclear license renewal paper can reasonably be considered for publication without an attempt at defining the term aging. In that regard, I will adopt, for purpose of this discussion, the following definition:(2)

- Aging is the net degradation in the physical condition of a component, system or structure due to environment and service. Aging can degrade the capability of a component, system or structure to perform its intended function after being placed in service.

- The environments and service conditions that produce aging degradation are called aging stressors (e.g., heat, radiation, humidity, reactive chemicals, operational cycling, electrical/mechanical loads, vibration, testing).

- Aging degradation is the change in physical properties (such as crack growth, dimensions, ductility fatigue capacity and mechanical or dielectric strength).

- Aging mechanisms are physical or chemical processes (such as wear, erosion, creep, corrosion, and oxidation) that result in aging degradation.

Later, discussion will demonstrate how this definition can be made more useful to the nuclear license renewal process by considering its relevance to safety.(3)

The importance of a consensus definition for "aging" cannot be overstated. Agreement on the meaning of this fundamental term is required to ensure that: our inquiry is relevant to the question of whether there continues to be reasonable assurance that the public health and safety is being protected; our search for reliable data is properly focused; and finally, license renewal review and acceptance criteria are repeatable in their application.

License Renewal - An Industry View

The U.S. nuclear industry license renewal process recommendations have been well documented.(4)(5) That process views license renewal as evolutionary; that is, dependent upon the licensing basis for the plant at the time of license renewal and rigidly focused on the continued reasonable assurance of the protection of the public health and safety. The industry technical program in support of license renewal has similarly depended upon a focused inquiry that hopes to achieve consensus on the data base to be relied upon to make license renewal decisions, common and defensible methods to conduct the necessary license renewal reviews, and the production of consensus industry reports addressing safety significant systems, structures and components relevant to a license renewal decision.

Under the auspices of the industry initiated NUPLEX Program, now functioning as a working group sponsored by the Nuclear Management and Resources Council (NUMARC), a deliberate attempt is being made to seriously review plant aging as it relates to license renewal in order to secure the viability of license renewal as an energy production alternative in the next century.

The focus of the early industry work was in Pilot Plant Studies,(6)(7) the purpose of which was to identify those plant systems, structures and components with the technical or economic potential of limiting plant life. Not surprisingly, the two pilot programs reached the strikingly similar conclusion that our technical attention should be directed at the primary pressure boundary of our plants, and in particular, on the reactor vessels, as well as at the primary containments of our plants, and the sensory infrastructure of our plants comprised of cable.

Those three areas have, as a result, been the focus of industry research attention. In parallel, a methodology has been conceived to screen the entirety of plant systems, components and structures to identify any and all physical plant attributes requiring attention in the license renewal process.

Relevant Scope of Inquiry

This screening methodology logic is shown in Figure 1.[8] The method begins with a systematic review of all plant systems. The review consists of addressing two major questions:

- Is the plant system potentially safety significant?

- Is degradation of the system significant to plant safety?

To ensure a complete list of systems for further consideration, a broad interpretation of safety significant will be used. The existing licensing basis provides a good starting point for this list. Examination of the docket will provide a list of all systems considered as safety significant as part of the existing licensing basis.

Among the systems identified will be those relied upon during design basis events to assure:

1. The integrity of the reactor coolant pressure boundary;

2. The capability to shut down the reactor and maintain safe shutdown; and

3. The capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the guidelines of 10 CFR part 100.

Other systems may have been termed safety significant in docketed information. These will be included as well. Finally, other front-line systems and their support systems known through non-deterministic assessments to be necessary to prevent or mitigate the consequences of severe accidents will be identified and evaluated for further review.

This initial attribute screening is expected to follow the classical boundaries of items agreed to be safety-related. However, because there is neither the intent nor the need to exclude attributes because of a restricted interpretation of what is related to safety, this review step is expected to pass through for finer review all of the plant systems requiring review by the

NRC in the original plant licensing unless it can be demonstrated that the system, if it were to fail, would not unreasonably affect plant safety. Methodologies to implement this decisionmaking process are now being developed and will require close coordination with regulatory authorities to ensure common understanding and acceptance.

The next review step is crucial in that it is here that the methodology establishes relevance to the license renewal process by determining which of the remaining systems, structures or components may be affected by age-related degradation, and wherever appropriate, which degrading elements of those systems will, thereby, affect the safe functioning of the system.

This methodology is intended to incorporate both deterministic and risk-based criteria. Reliant on conventional deterministic licensing review criteria, the current plant licensing basis is expected to identify the systems, structures and components having safety significance and, therefore, requiring further review. Conventional probabilistic techniques, such as the "reliability achievement interval," are expected to provide a defensible and repeatable criteria for identifying those plant elements susceptible to aging degradation which will affect safety. This screening methodology will clearly retain for further review the reactor vessel and containment and large classes of cable. The methodology are expected to exclude from unnecessary consideration for license renewal systems such as make-up demineralizers and components such as reactor dryers and separators.

The review methodology to this point will have identified components which are important to the safety of the plant, which are not routinely replaced or critically inspected during 40-year operation, and which are subject to significant age-related degradation. To justify continued operation of these components, measures must be taken to address potential age-related degradation.

There are a wide range of options available to the licensee to address this degradation and to ensure that safety margins are maintained. The options fall into three principal categories:

1. Improved assessment of the potential for age-related degradation.

2. Actions to prevent age-related degradation.

3. Actions to mitigate the effects of age-related degradation.

Improved maintenance and surveillance activities, such as implementation of a reliability centered maintenance program, may also provide a basis for continued operation. Such activities would improve the assessment of the effects of age-related degradation and lead to actions to prevent or mitigate such degradation.

Life extension options depend largely on the actual assessment of components with due consideration of operational duty and historic performance. Through a combination of better assessment, prevention, and mitigation for those components with the potential for significant age-related degradation, continued safe, reliable, and economic performance can be assured.

The screening methodology will in this way identify the sub-tier attributes within the scope of relevant license renewal inquiry that are expected to be qualified for extended life based on the efficacy of existing licensee operating and maintenance programs. Certain of these items are more easily defineable. For example, the class of equipment now subject to planned maintenance, including refurbishment or replacement, within the scope of the environmental qualification rule (10 CFR 50.49). While subsidiary research in this area might be enlightening, it is not necessary or, therefore, relevant to the license renewal process -- at least as related to the affected class of equipment because effective licensee programs controlled within the existing licensing basis already exist.

The resulting application of this screening methodology to a specific plant will produce a list of systems, structures and components which are safety significant and are susceptible to aging degradation which will affect safety function.

Another major component, PWR steam generators, while potentially relevant to a license renewal inquiry, need not divert our research or regulatory attention beyond those initiatives now underway. Indeed, it serves as an example of the benefits of a focused program on our ability to strategically plan for component aging. Let me be more specific. Steam generators in the U.S. and worldwide have over the past 12 years undergone extensive industry-sponsored study valued at over $100 million.(9) One of the results of this effort has been an improved ability to develop and evaluate strategies to mitigate steam generator reliability and aging. This can be accomplished even with the inherent uncertainty associated with the complexity of many degradation modes in a complex structure with large numbers of potential failure locations. Continuing industry research on aging mechanisms such as wear, intergranular attack (IGA) and stress corrosion cracking (SCC) will further reinforce a strategic initiative that has produced vast improvements in steam generator performance, through improved water chemistry, repair techniques, inspection engineering and understanding of the applicable thermal hydraulic affects.

The point of this discussion is that a focused, comprehensive and well-documented component preservation strategy has already been developed by the U.S. nuclear industry for its steam generators. Although complimentary research to develop refined first principles degradation models may be enlightening and of value to utility planners, it is not necessary, or, therefore, relevant to the license renewal process -- beyond developing appropriate assurance that an applicant has effectively adopted the existing preservation strategies which include chemistry control, state-of-the-art inspection, necessary refurbishment (plugging, sleeving, heat treatment, etc.) and replacement methods based upon by plant specific conditions.

Reliable Data Base

The steam generator provides a particularly useful example in our considerations on how to effectively address license renewal. We have recognized its degraded performance can potentially affect safe plant performance. We have identified aging mechanisms that can contribute to such degraded performance. However, we can demonstrate that sufficient intelligence supported by reliable data exists from which reasonable assurance of safety can be deduced, obviating the need for regulatory intervention through mandatory supplemental research or programmatic prescription. In this instance the data base has been developed by the EPRI sponsored Steam Generators Owners Group[10] and is maintained by EPRI for those owners. Comprehensive controls on the developing data through established water chemistry[11] and examination[12] guidelines ensure this data is reliable and relevant to the evaluation of steam generator service life. As an example, while we may never know the true mechanism for grain boundary attack in steam generator tubing, and as a result, be unable to generate a verifiable first principles IGA/SCC tubing degradation model, we have already developed enough operating empirical data to reliably predict tubing degradation and to respond effectively as a result. This is evidenced by work done over the past four years by Commonwealth Edison in developing a Steam Generator Repair/Replacement Strategy for the PWR units at Zion, Byron and Braidwood. This model is shown schematically in Figure 2.[13] This CECo program has resulted in aggressive actions to improve eddy current technology in use at each of these plants as well as to expand inspections where appropriate. It has also led to the development and implementation of a planned tube sleeving strategy and anti-vibration bar replacement to extend generator life on our older Zion units, as well as tube roll transition shot-peening and U-bend stress relief on our new units to prevent degradation of the type experienced at like units in the U.S. and Europe.

The industry technical initiative includes plans for the development beginning in 1989 of a comprehensive data base similar in concept to that now existing for steam generators, to support defensible owner/utility decisionmaking on systems, structures, and components that are relevant to the license renewal inquiry. Participants in this effort include NUMARC/NUPLEX, EPRI, DOE and Sandia National Laboratory.

The Department of Energy is sponsoring work with the Sandia National Laboratories to develop and maintain this data base. A complimentary program is being developed in conjunction with the Institute of Nuclear Power Operations (INPO) to give better definition to the type of data necessarily included in this inquiry and to evaluate how, if at all, existing industry performance indicators can be modified to support decisionmaking on fitness for service. This latter activity has been focused initially on water chemistry data and controls. It is expected that this activity will, when completed, lead to the development of an industry topical report that can be referenced for license renewal review.

Repeatable Criteria

As should already be apparent, the industry technical program is structured in such a way that generally applicable methods and criteria are generated wherever possible to promote stability and efficiency in the license renewal process. The industry screening methodology has already been discussed. In addition, work is nearing completion on Industry Topical Reports (ITRs) addressing BWR and PWR reactor vessel and internals fitness for service issues and large dry concrete and free-standing steel containments. These reports will be developed to address the potential impacts of age-related degradation on system, structure or component performance as it relates to plant safety function. In addition, these reports will define necessary data preservation/acquisition needs, periodic inspection requirements, and perhaps, most important, necessary follow-up research and code development requirements. It is expected that such topical reports will both guide the effective development of license renewal applications and, through reference in and review of the industry "lead plant" renewal initiative, establish understandable repeatable criteria for use by all applicants in the license renewal process.

Similar topical reports are now being initiated to cover commodity items such as cable and pressure boundary piping and supports. It is hoped that such commodity reports when complete can also be referenced in the system reviews necessary to support the license renewal process. Because of the extensive research, either complete or ongoing in these areas, the centralized documentation of available intelligence should be extremely useful to the U.S. nuclear industry and regulatory authorities.

Although more comprehensive discussion of these and related research activities are documented elsewhere,[14][15] it is important to note that this program supports the stated regulatory objective in the NRC NPAR Program: i.e., the development of a better understanding of the aging process and improved methods for detecting and mitigating aging degradation.[16] A complete understanding with perfect methods is perhaps not achievable. Therefore, resources and research must be properly focused on priority issues.

The development of life estimating methodology for ASME components is a focus of present industry research. In addition, the validation of life estimating methods considering fatigue, neutron and thermal embrittlement, as well as, corrosion induced aging degradation, including irradiation assisted corrosion effects in non-oxygenated water are all receiving heightened industry research attention.

This research is certainly supportive of the defined objectives of the NRC aging research program. There may, in fact, be opportunities for integration of the industry and government research planning as has been done in the past for BWR containment and seismology research. Such integration and focus would better allow for consensus definition of priority issues, with <u>closure of issues</u> a mutual objective.

Aging research, whether it be on humans or nuclear power plants, can take on a certain fanticism, i.e., a redoubling of one's effort when one's aim has been forgotten. Our research aims must be reasonable and directed at preserving life to the extent possible without compromising reasonable assurance in the protection of the public health and safety. Appropriate screening of plant systems, structures or components to delineate in a repeatable, defensible manner, those attributes that require license renewal consideration must, therefore, be accomplished first.

Like man's quest for the "Fountain of Youth," neither a full understanding nor the conquest of aging is likely to be achievable. Our research objectives must provide a practicable alternative that focuses on reliable data that will trigger remedial actions at appropriate times. The evidence of the merit of this approach can be found in the industry steam generator program. While we may not know all we would like to know, we should and will develop sufficient knowledge to control the aging of vital plant components. Rather than diverting our energy to the development of a perfect first principles understanding of the aging of steam generators, we should, and in fact, are similarly focusing our attention on other vital plant attributes, beginning with reactor pressure vessels and their internals, as well as primary containments, and generic commodities such as pressure boundary piping, supports and vital cable.

In fact, the industry is actively developing a strategic plan for mitigating the effects of aging on reactor vessels. This plan, coordinated by EPRI, will be both focused and comprehensive. The plan will address both BWR and PWR reactor vessels and internals. The plan will be comprehensive in that it includes methodologies for evaluating aging, such as: embrittlement, stress corrosion cracking, fatigue, general corrosion, and other degradation mechanisms.

Research in these areas, if properly developed and implemented and if reflecting an effective integration of industry and regulatory needs and expertise, will reduce our present uncertainties related to license renewal.

Conclusion

It has been said that it is better to be profound in clear terms than in obscure terms. With that philosophy in mind, the central objective of this paper and the industry license renewal initiative generally is to ensure that effective actions are taken which will prevent the premature loss of existing nuclear electric generating capacity. The need to accomplish this objective in a manner that supports the existing regulatory findings of reasonable assurance that the public health and safety will be protected is an acknowledged prerequisite to an effective program. This objective will be accomplished most efficiently if a <u>Reasonable scope</u> of technical inquiry is defined and agreed upon, if <u>Reliable data</u> is maintained, or if necessary, generated to support license renewal decisionmaking, and if a <u>Repeatable process</u> of license renewal is developed employing defensible technical methods for assessing fitness for service. The industry technical initiatives follow these three "R's" of license renewal.

The increasing reliance of the American public on electric power is unquestioned. Nuclear generated electric power is a vital existing element of that power source. Our efforts must, therefore, be directed at preserving that resource.

FIGURE 1: A Methodology to Identify and Evaluate Plant Equipment for License Renewal Review

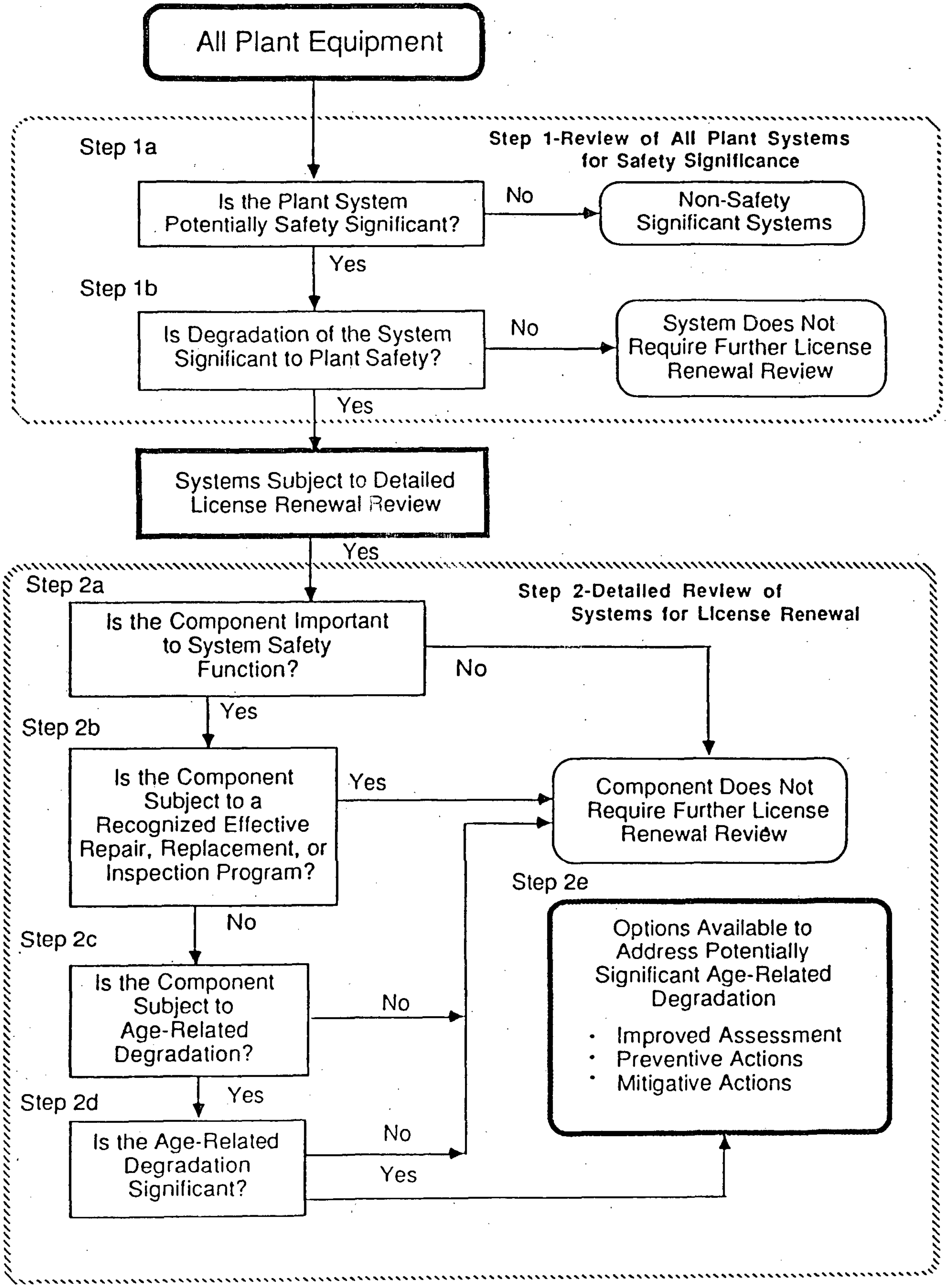
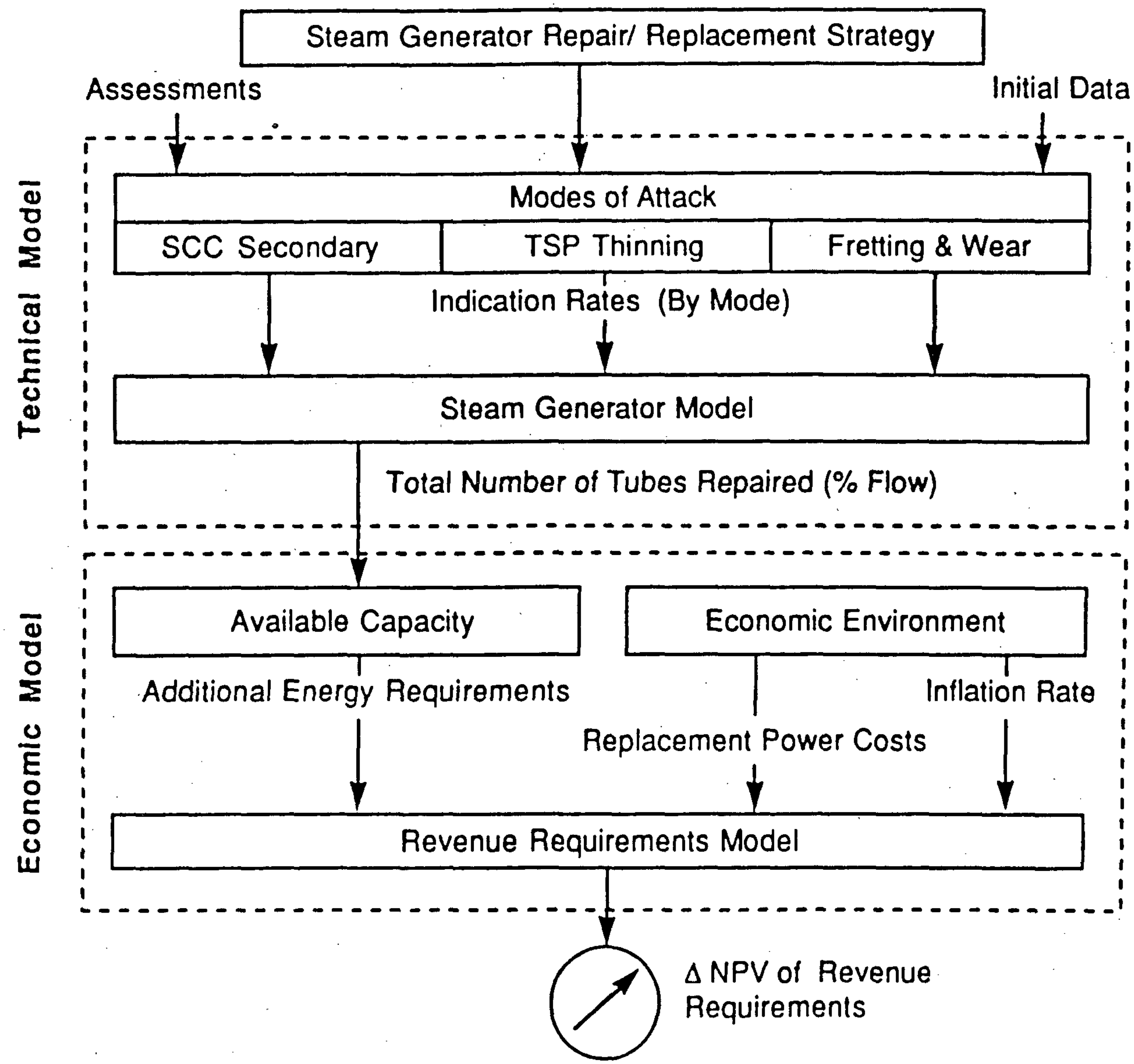


FIGURE 2



Steam Generator Repair/ Replacement Strategy
Assessments
Initial Data
Technical Model
Modes of Attack
SCC Secondary
TSP Thinning
Fretting & Wear
Indication Rates (By Mode)
Steam Generator Model
Total Number of Tubes Repaired (% Flow)
Economic Model
Available Capacity
Economic Environment
Additional Energy Requirements
Inflation Rate
Replacement Power Costs
Revenue Requirements Model
Δ NPV of Revenue Requirements

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Overview of the NRC's Human Factors Regulatory Research Program

Franklin D. Coffman, Jr., Chief
Human Factors Branch

At the NRC, human performance research is directed toward the technology needed to ensure the safe and effective use of commercial nucleonics by people. Human performance research has many factors and is a broadly based technology. The research is grouped into six topical areas to facilitate the research: (1) Human Performance and Human Reliability Assessment, (2) Man-Machine Interfaces, (3) Procedures, (4) Qualifications and Training, (5) Organization and Management, and (6) Performance Indicators. However, even these groups are closely interrelated and not totally separable because the individual elements of human factors research are not readily separable.

At the NRC, human performance research is directed toward the technology needed to ensure the safe and effective use of commercial nucleonics by people. The research proceeds by understanding, measuring, and monitoring the influences that affect human performance. The NRC program researches the many factors shaping human performance and behavior such as cognitive processes, training, qualifications, organizing, supervising, preparing and using procedures, interfacing between the person and the machine, and assessing total reliability. There is both timely research to support regulatory decisions and research to anticipate human performance issues that are potentially safety significant before they develop (foundational research).

The objectives of the research are (1) to broaden our understanding of human performance and to detect the causes of human errors for the purpose of ensuring safe operations in the commercial nuclear industry; (2) to accurately measure human performance for the purpose of enhancing safer operations, precluding critical errors, and contravening adverse human-induced effects on safe nuclear operations; and (3) to provide the technical basis for nuclear regulatory requirements, recommendations, and guidance.

One of our major tasks starting in April 1987 was to revitalize human factors regulatory research. To revitalize this research, we added more than just resources. We added coordination and coherence. We requested and reviewed formal needs from the user offices. To define the projects, we considered ongoing research, past accomplishments, and all pertinent recommendations of the National Research Council in its 1986 report on Nuclear Safety Research. Our considerations involved refining the underlying human performance issues and planning the best approach toward their resolution. In the process, some similar needs were condensed into single projects. The resulting Human Factors Regulatory Research Program addresses all research needs formally requested by the NRC regulatory users. Additionally, the Program currently addresses over eighty percent of the broader human-factors research recommendations to the entire commercial nuclear industry that were identified in a later National Research Council report ("Human Factors Research and Nuclear Safety," February 29, 1988). The NRC's current Human Factors Regulatory Research Program Plan is documented in SECY-88-141, March 23, 1988. The Plan is being revised to reflect developments to date and to increase the coordination among the topical areas.

Program Elements

The research program elements have been placed into topical areas that somewhat follow the factors shaping human performance. However, our topical-area grouping was mainly for convenience, i.e., the grouping is neither disjunctive nor technically unique. The research elements may be regrouped at times when it facilitates the conduct and the administration of the research.

Currently, the six topical areas are (1) Human Performance and Human Reliability Assessment, (2) Man-Machine Interfaces, (3) Procedures, (4) Qualifications and Training, (5) Organization and Management, and (6) Performance Indicators.

Human Performance research is intended to develop an understanding of how people function in the commercial nuclear industry and to systematically model the factors shaping that performance. Human Reliability Assessment research is intended to model, measure, and assess human error rates using both credible and applicable information.

Two of the projects in this topical area are discussed in this session of the meeting. Dr. Lou Buffardi discusses "Selection of Anchor Values for Human Error Probability Estimation," and Dr. David Gertman discusses "Human Performance Data Acquisition and Management for Reliability Evaluations." Other ongoing research projects in this area include the following:

(1) Modeling of human intent formation during emergencies to enable us to focus on causal factors.

(2) Integrating human reliability assessment into probabilistic risk assessments for improving techniques to more accurately account for the risk impact of human performance.

(3) Simulating the effects of changes in maintenance activities on performance for more accurate assessments of maintenance reliability.

In addition to the ongoing research, the planned research in this topical area includes:

(1) Composing an approach for obtaining broader and more credible data on the causes and frequency of human errors directly from operations.

(2) Determining the specific functions essential to success during rare and stressful events and the likelihood of success considering the total situation during an actual event.

(3) Investigating the causes of overexposures in industrial radiography and misadministrations of nuclear medicine.

Man-Machine Interface research is intended to ensure that the man-machine interface permits the full, compatible communication needed for safe operations.

An example from the ongoing research is given by Dr. Jim Higgins in the concurrent session on Generic Safety Issues [in Lecture Room B] entitled "A Probabilistic Risk Assessment of Potential Upgrades of Control Room Annunciators." Other ongoing research projects in this area include the following:

(1) Assessing the potential safety benefits from improvements in local control stations.

(2) Updating the guidance on human factors reviews (currently in NUREG-0700) for possible applications in new designs.

(3) Identifying the need for criteria for or changes in control board annunciators.

(4) Developing measures and criteria for the acceptance of advanced instrumentation and control systems.

(5) Establishing the verification and validation process for advanced software, e.g., expert systems and artificial intelligence.

The planned research in this topical area includes:

(1) Determining the impact of mixing previously installed analog instrumentation and controls with replacement and upgraded digital instrumentation and controls.

(2) Investigating the present classifications for computers used in the commercial nuclear industry.

(3) Ascertaining the current cause of industrial radiography overexposures.

(4) Studying the potential for operators to rely excessively on the Safety Parameter Display System rather than on the Class 1-E control board instruments that comply with Regulatory Guide 1.97.

A substantial portion of the research on man-machine interfaces is being performed through our cooperation with the Halden Project. We expect the Halden Project to provide knowledge on the use of expert systems as operator aids, review criteria for advanced instrumentation, information on computer-based procedures, and simulator data on human performance.

Procedures research is intended to ensure the reliability of rule-based diagnoses and actions, i.e., to minimize safety-significant procedural errors.

The ongoing research projects in this topical area are assessing the potential benefits from improving procedures other than the Emergency Operating Procedures and ascertaining the frequency and safety consequences of procedure violations in U.S. plants. This latter project is a vestige of the Chernobyl accident.

Planned research for this topical area includes:

(1) Exploring the impact that the introduction of advanced instrumentation and controls will have on the presentation formats and contents of emergency operating procedures.

(2) Exploring the many questions associated with accident management research, including:

(a) What is the appropriate level of detail for procedures to manage accident scenarios with large uncertainties in the analyses and experiments?

(b) What portion of operator training should be devoted to unlikely accidents?

(c) What are the best organizational forms and functions for dealing with severe accidents with the more intense interfaces between the central control room, the local control stations, the Technical Support Center, and outside agencies?

Qualifications research is intended to ensure the matching of innate human capabilities with the systems-task requirements. An example in this topical area will be presented by Dr. Paul Lewis on "Impacts of Overtime and Shift Scheduling on Operator Performance."

Training research is to intended to ensure the matching of the required skill levels with the training levels received and maintained for both individuals and teams.

Organization-and-Management research is intended to lay the objective foundations to model, measure, and monitor the influences of organizational and supervisory practices on safe operations. In the next item of this session, Dr. Sonja Haber will present an element from this research area entitled "Methods of Understanding the Influence of Management Factors on Performance Reliability."

Performance Indicators research is intended to provide senior NRC management with additional measures of operational performance to recognize areas of declining safety performance. In this session you will hear from both Dr. John Wreathall on "Programmatic Performance Indicators: Methods and Data for Maintenance and Training" and Mr. Carl Johnson on "Risk-Based Safety System Functioning and Performance Trends."

Although Maintenance research is not identified as a separate topical area, research is being performed on the maintenance that touches all of the other topical areas.

These research projects involve both hardware and humans, as well as the interactions of each project with the others. Since accidents seldom have single causes, the development of solutions needs to be comprehensive in scope. Consequently, the research is being structured in a "total-systems" approach. A "total-systems" approach means that the researchers on each project accommodate the human-performance reality that all else seldom remains equal when a change is made. Some of the research is by its very nature comprehensive. Maintenance addresses all these groups via the work on the Maintenance Personnel Performance Simulation project, Performance Indicators, Reliability Engineering, Human Reliability Assessment, and Event Reporting.

Schedules

The following milestones are examples from each of the topical areas. One of the next milestones in the Human Performance and Human Reliability area is the development and trial application of a cognitive reliability analysis technique in the summer of 1989. And, by the late summer of 1989, we plan to complete interim guidelines for conducting validation and verification of Man-Machine decision aids that utilize expert-systems technology. In Procedures research, we expect to complete by April 1989 the regulatory analysis related to the generic issue of guidelines for upgrading procedures other than the emergency operating procedures. In Qualifications and Training research, we expect to have criteria for evaluating training programs to deal with rare events by 1990. In Organization and Management research, we have developed and plan to have performed a trial application of the method to include the influences of supervisory/management practices in reliability evaluations by December 1988.

Also by December, we expect to complete the retrospective verification of some of the current Performance Indicators. And we will continue to actively contribute to maintenance rulemaking by appropriately including human and operational reliability considerations.

Program Management

Currently there are ten professionals managing the human factors research. The staff is multidisciplinary. The current staff is composed entirely of senior professionals, including internationally recognized experts in man-machine interface designs, human reliability assessment, and simulators. These professionals are qualified to direct research both by formal training and by applied experience.

By formal training, there are four doctorates and ten master's degrees. Every professional has at least one master's degree. Every discipline has at least one master's degree. We are strongest in psychology, with two PhDs, four master's degrees, and four bachelor's degrees.

There is a minimum of eleven years applied experience in any single topical area. The total applied experience is 2 staff-centuries. Over 60 percent of the experience is in human factors and engineering psychology.

Contracts

In contracting projects we look particulary for well-qualified professionals that are experienced in human factors and any of several disciplines related to commerical nucleonics. Currently we have contracts with internationally recognized experts in cognition modeling, maintenance, simulation, advanced man-machine interfaces, circadian physiology, organization and management, psychological experimentation, event investigation, and data management. Most of the research is performed through contractual support. The past fiscal year's contracts were distributed with about 50% at National Laboratories, about 10% at universities (this does not include any work being subcontracted to universities), about 15% at consulting firms, and about 25% at international cooperatives. We expect that the percentages for universities and consulting firms will be increasing.

The Next Steps

The next step for the Human Factors Regulatory Research Program is to reach a level of stable equilibrium. Not stagnation where little is produced. Rather a stable rate of satisfying needs and receiving new needs. The program will need to stay helpfully relevant to the current human factors issues in nuclear regulation. The HFRRP cannot afford either to lag far behind the day-to-day regulatory issues or to advance too far beyond the state of nuclear regulation. Timely research will take some quick thinking and comprehensive understanding of fundamental human performance issues. The foundational research will take some insightful anticipation of developing human performance issues. The HFRRP can ill afford to try using abstract, prolonged, that is, "round" projects to resolve complex, transient,"square" regulatory issues.

Just as sweeteners do their best work when they lose their identity in a drink, so too, human factors specialists appear to do their best work when they lose their identity as specialists in a multidisciplinary team that resolves nuclear safety issues.

When a newcomer to the Office of Nuclear Regulatory Research, I learned about the two challenges faced by researchers. A researcher is challenged both to generate alternative solutions for regulatory problems and to evaluate the alternative solutions. In the first, a researcher is called upon to build on past experience and available information in a flexible, open, unstructured, and unencumbered way to generate creative alternative solutions to complex nuclear safety problems. In the other, at the next stage of the research project, a regulatory researcher is called upon to systematically, cautiously, and vigilantly evaluate and critically test the alternative solutions. To balance both functional responsibilities is a schizophrenic challenge for regulatory researchers.

Conclusions

Human performance research has many factors and is a broadly based technology. Here the discussion grouped the technology into six topical areas to facilitate the research. However, even these groups are closely interrelated and not totally separable because the individual elements of human factors research are not readily separable.

The program will provide a technical basis for supporting nuclear regulatory decisions related to human performance. Also, we expect that the research performed from this program will deepen our overall understanding of the causes of human error for the purpose of reducing its adverse incidence on commercial nuclear operations.

METHODS TO UNDERSTAND THE INFLUENCE OF MANAGEMENT FACTORS ON PERFORMANCE RELIABILITY

Sonja B. Haber and John N. O'Brien
Brookhaven National Laboratory

ABSTRACT

It was the purpose of the project to develop the methods to be used in the assessment of supervisory and management factors on safety performance in a nuclear power plant and to provide a product useful to Nuclear Regulatory Commission personnel, nuclear power plant personnel, and probabilistic risk assessment practitioners. The methods developed in this project provide one means of conceptualizing the human dynamics of a nuclear power plant organization. The model (NOMAM) is descriptive, yet process-oriented and allows for the identification of key supervisory and management influences on plant performance through the evaluation of organizational processes. Examination of standardization processes is critical in understanding the functional dynamics of a nuclear power plant. Behavioral factors unique to supervisory and management influences have been identified, and the actual collection of behavioral data from a nuclear power plant will allow a direct interface with risk and reliability studies.

INTRODUCTION

There has been an increasingly clear need to develop a means for examining and measuring the role of supervisory and management factors at nuclear power plants relative to performance reliability. The assessment of this type of performance would be used in reliability assessments, resolution of Generic Issues, regulatory oversight activities, e.g., Performance Indicator program and probabilistic risk assessments. In order to address this need, an understanding of how nuclear power plant-related supervisory and management functions can be observed, measured, and evaluated was required. Four tasks were identified to achieve the purpose of this project. The first task was to develop an organizational model of a nuclear power plant, specifically those operational units that may exert a direct or indirect impact on plant safety performance. Next, the potentially key supervisory and management functions which may effect plant safety performance were identified. The third task attempted to gain an understanding of the nature and direction of supervisory and management influences on personnel performance. Task 4 identified the methods for analyzing and evaluating supervisory and management factors that may impact plant safety performance. Beyond the scope of these tasks is the utilization of the data to be collected in reliability and risk studies.

Model Development

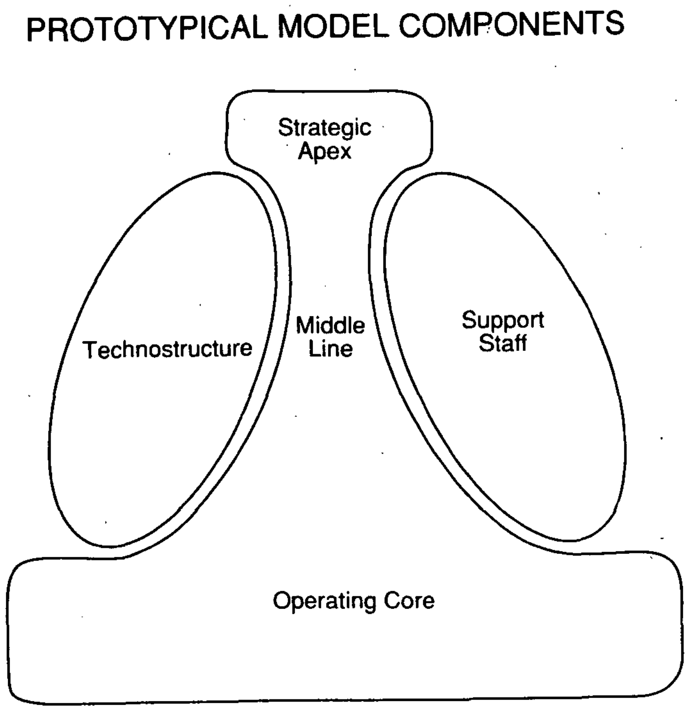
The organizational literature is replete with models and theories of how organizations are structured. An extensive review of the literature uncovered a lucid and robust conceptualization of the material by Henry Mintzberg [1]. After assimilating the literature, both empirical and theoretical, Mintzberg provides a model to define the basic types of organizational structures and the associated variables that are characteristic of each type. Later work by Mintzberg [2,3] elaborates on the types of organizational structures elucidated by him [1], and incorporates new literature into the same basic conceptualizations.

A description of the variables which are addressed in the literature and the generic types of organizational models that have been identified are presented below. The conceptualizations and definitions are taken from Mintzberg [1] so that later discussion will be consistent in terminology and rationale.

Parts of an Organization

The structure of any organization is defined as the sum total of the ways in which it divides its labor into distinct tasks and then achieves coordination among them. Every organization has input and output. The output can be in the form of products or services. There are five basic parts to an organization which comprise its input. The personnel who perform basic work related to the production of products or services are identified as the operating core of the organization. These individuals could be the assemblers in an automobile factory, or the professors in an university, or the maintenance technicians in an NPP. The individuals charged with ensuring that the organization serves its mission in an effective way, and also serve the needs of those people who control or otherwise have power over the organization comprise the strategic apex. Examples are the president of a company, or the superintendent of a school system, or the plant manager of an NPP. Personnel who are a chain of authority between the strategic apex and the operating core are the middle line of the organization. Senior managers to first-line supervisors would fit into this category. Personnel who are responsible for and effect standardization within an organization are described as the technostructure. Depending upon the type of organization, accountants, trainers, or engineers would fit this description. Finally, the individuals who provide support to the organization outside the operating work flow are the support staff. Included in this group are the cafeteria employees, custodial staff, security, and payroll departments.

Mintzberg's conceptualization of the prototypical structure of an organization is presented in Figure 1. As the characteristics and parts of each organization become more or less prominent, the shape of the basic structure changes and defines another type. In order to understand the way these types take shape, a discussion of some of the variables associated with an organization follows.



from Mintzberg (1979)p.20.
Figure 1. Prototypical model components

Characteristics of an Organization

Coordinating Mechanisms

Coordinating mechanisms are the fundamental ways in which an organization coordinates its work. There are five basic coordinating mechanisms delineated by Mintzberg [1] and described below. When one individual is in charge of and responsible for the work of others, the mechanism is known as direct supervision. In the standardization of work, the contents of work for an individual are highly specified. Instructions provided to the consumer by a manufacturer to assemble a product are a good example of this mechanism. Similarly, standard operating procedures used in an NPP also represent standardization of work. The standardization of outputs mechanism standardizes the results of an individual's work in the dimensions of a product or the individual's performance in the case of services. A taxi driver has to arrive at a certain destination, but is not necessarily told which route to take to get there. When the type of training required to perform a certain type of work is specified, the mechanism is defined as the standardization of skills. Hospitals hire doctors from reputable medical schools to insure that they are properly trained to perform their job. The last mechanism identified for the coordination of work is mutual adjustment, which is the simple process of informal communication. This process is used in the simplest and also the most complex organizations.

Design Parameters

There are a number of parameters that can be viewed as defining certain characteristics of an organization. Job specialization defines work in terms of breadth and scope (horizontal specialization) and/or depth of job (vertical specialization). Horizontal job specialization refers to the concept of division of labor, while vertical job specialization relates to how much control an individual has over their job. Training is the process by which job-related skills and knowledge are taught, usually outside of the organization. Indoctrination refers to the process by which organizational norms are acquired, or the socialization of the individual for the organization's benefit. Behavior within the organization is usually formalized in one of three ways: by a job description, by the work flow, or by a set of rules or policies within the organization. Grouping coordinates the work within the organization and can be done on the basis of (1) knowledge and skill, (2) work process and function, (3) time (e.g. shiftwork), (4) output, (5) type of client, or (6) geographical location. The two most typical means of grouping are by function and by output (market). The size of each unit within the organization often determines the type of coordinating mechanism used. Generally, the greater the standardization, the larger the size of the unit. Performance control systems and action planning systems allow the organization to plan its future and evaluate its present. The former regulates the results of a unit by setting objectives, budgets, operating plans, etc., while the latter sets specific actions and decisions for specified points in time. Organizations encourage communication outside formalized channels through liaison devices such as task forces and standing committees. Finally, if all the power in the organization is ultimately in the hands of one individual, the organization is said to be centralized. Centralization can occur both horizontally and vertically.

Contingency Factors

There are certain situations or states that are associated with the use of certain design parameters; these are called contingency factors. The age and size of an organization are two such factors integral to the development of structure. The degree of flexibility of the technical system can dictate a great deal about the structure of an organization. The environment outside of the organization as it relates to the work within the organization is a contingency factor. Power, including the presence of outside control on the organization and/or the personal needs of various members of the organization, is a critical factor in establishing structure.

Structural Configurations

Five "pure" types of structures are identified by Mintzberg[1] when various combinations of the characteristics and variables just described are considered. The simple structure has little or no technostructure and support staff, a loose division of labor, minimal differentiation among its units, and a small managerial hierarchy. A middle-sized retail store would fit this structure. The key part of the machine bureaucracy is the technostructure.

There are large-sized units at the operating level, a functional basis for grouping, centralized power for decision-making, and a sharp distinction between line and staff. A national post office, steel company, or airline are organized in this configuration. A professional bureaucracy relies on the standardization of skills and training and indoctrination for work coordination. It has professionals for its operating core, and gives them considerable control over their work. This structure is common in universities, hospitals, and school systems. The divisionalized form differs from the others in that it is an overall structure superimposed on smaller structures. Each division in this configuration has its own structure held together by a central administrative group. Some of the largest corporations are organized in this configuration. Last, but not least, is the adhocracy which fuses experts drawn from different disciplines into smoothly functioning ad hoc project teams. Little formalization of behavior, high horizontal job specialization and heavy reliance on mutual adjustment characterize this type. The complexity and sophistication of a space agency requires this configuration.

Mintzberg [1] identifies the five "pure" structural types and the list of variables characteristic of each type. This represents a good summary of the information just discussed and a reference for the next sections.

Functional Organization of a Nuclear Power Plant [4]

The nuclear power division of a utility is a somewhat autonomous division within the corporation's structure and is generally headed by a Vice President for Nuclear Operations. It is extensively supported by its own technical and administrative groups, with some interaction with other parts of the utility. For the purposes of this paper, the focus will be on an electric generation and distribution utility with a single nuclear power plant unit at one site. In addition, the plant is under operational control by the utility. It should be noted that multiple units at one site and multiple site arrangements do exist within the nuclear industry.

In general, the entire nuclear power division of a utility is physically located at the plant site. Some utilities do maintain a few groups, such as nuclear engineering, at corporate headquarters. At the site, the Plant Manager (under the V.P. for Nuclear Operations) is directly responsible for all site activities. In general, the two main goals of the nuclear division are the safe operation of the plant and the economical generation of electricity. In addition, the United States Nuclear Regulatory Commission (NRC) oversees the entire operation of each commercial nuclear power plant in the country by maintaining an on-site presence, and ensuring enforcement of many rules and regulations for safe operation.

The functional organization on site at a nuclear power plant generally contains the following units: Operations, Maintenance, Instrumentation + Control (I+C), Quality Assurance (QA), Test & Performance, Health Physics, Chemistry, Independent Safety Engineering Group (ISEG), Administration, Nuclear Licensing, Outage Planning, Reactor Engineering, Design Engineering, Shift Technical Advisors (STAs), Records Management, Spare Parts, Security, and Training. There are also two important standing committees: the Offsite

Review Committee and the Plant Operations Review Committee. A typical site employs 300 to 600 people with increased numbers during major outages. The size of each specific unit varies considerably across plants. Each identified unit has been described in detail elsewhere [5].

An Organizational Model of a Nuclear Power Plant

Using the model provided by Mintzberg, the nuclear power plant (NPP) can be identified as a particular organizational structure. A fit into a "pure" type for the entire plant is not evident, but a basic structure does take shape, and the inconsistencies within the model are very manageable within Mintzberg's theory.

Utility Structure

At the utility level, the divisionalized form best represents the corporate structure. The Division of Nuclear Operations is identified as one division situated in the operating core of the utility. The Vice President for Nuclear Operations represents the middle line of the corporate structure, but will become the strategic apex of the plant structure. The key coordinating mechanism for the utility is the standardization of outputs and the various design parameters and contingency factors associated with this structure conform closely to those described by Mintzberg for the divisionalized form.

Of greater concern in this paper, is the organizational structure of the NPP itself. Its relationship to the corporate structure will be considered when the flow of decision-making and authority within the utility directly impact on the plant. This channel of communication will occur through the Vice President of Nuclear Operations, or another individual in the strategic apex of the plant structure.

Plant Structure

The model of a NPP under consideration in this paper is initially best depicted by the machine bureaucracy structure and is presented in Figure 2. The key part of the machine bureaucracy is the technostructure, and many of the nuclear power plant units fit into that technostructure. Units such as licensing, training, quality assurance, health physics, engineering, shift technical advisors, planning and scheduling, testing and performance, and the independent safety and engineering group, comprise the technostructure of the plant's organization. These are highly developed groups which formalize and standardize the work primarily of the operating core. Much of the work and behavior in a NPP is highly formalized and procedure-based, resulting in the use of standardization of work as a key coordinating mechanism within the organization.

Support staff in a machine bureaucracy are also organized into well developed units to reduce the ambiguity of their function and position within the organizational structure. Records management, payroll, administration, security, cafeteria and housekeeping personnel are examples of the support staff of a NPP.

CONCEPTUAL MODEL OF A NUCLEAR POWER PLANT

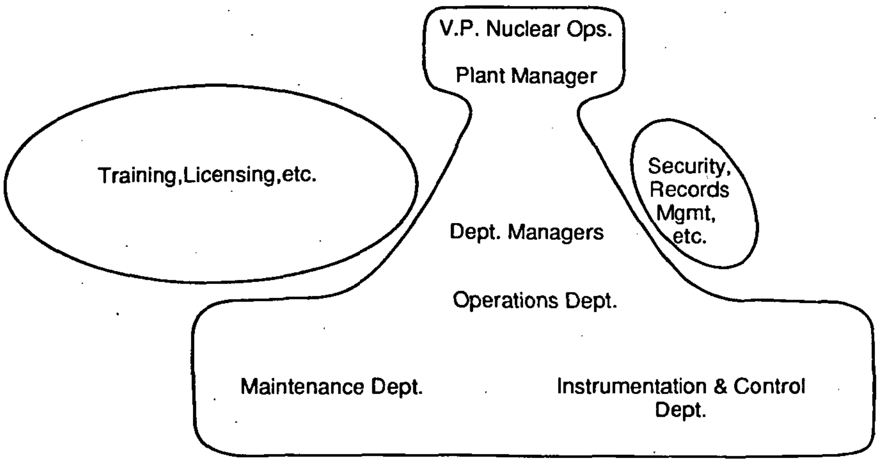


Figure 2. Conceptual model of a nuclear power plant

The strategic apex of the NPP usually consists of the Vice President for Nuclear Operations, from the utility structure, the Plant Manager and Assistant Plant Manager. These individuals are responsible for the fine tuning of the plant, the coordination of functions, and the resolution of any conflicts occurring among the various units in the plant.

At the heart of the plant is the operating core. The operating core is comprised of three different types of units: operations, maintenance, and instrumentation and control (I+C). In a "pure" machine bureaucracy, the entire operating core would be horizontally centralized. In an NPP, the operations unit has some vertical centralization, both functionally and structurally, over the maintenance and I+C units. Therefore, this unit conceptually resides in a different place in the proposed organizational model from the other two units. We propose that part of the operations unit comprises the middle line structure in the NPP. Specifically, the Operations Manager and Shift Supervisor are part of the middle line, while the auxiliary (non-licensed) and reactor operators remain part of the operating core.(Also, included in the middle line are the managers of other units in the NPP.) The position of the senior reactor operators is dependent upon their functional role, which is often dependent upon the operating conditions within the plant. The senior reactor operators could, under certain circumstances, reside in the middle line structure of the plant. Under normal operating conditions, they are part of the operating core.

The maintenance unit within the operating core of the NPP is organized and run like a "pure" machine bureaucracy. The majority of work is routine and standardized, and authority is vertically centralized. The instrumentation and control unit, however, somewhat resembles the structural type of a professional bureaucracy. The personnel in this unit are skilled professionals with much individual autonomy over their work. Centralization in the I+C unit is both vertical and horizontal. One exception to the professional structural type within I+C is that the work being performed is often standardized, eliminating the creativity and discretion of the "true professional."

Key Supervisory/Management Functions and Processes

The nuclear organization and management analysis model (NOMAM) was developed as part of the work scope under Task 1 in this project. The model's basic utility lies in this description of the human organization of a nuclear power plant. NOMAM is a dynamic, interactive, and behavior-oriented characterization of the plant and emphasizes functional relationships between the units in the plant. Consideration is also given to the internal and external forces on the nuclear power plant and how they affect the performance of the organization.

The human organization of a nuclear power plant depends primarily on the standardization of its operating work processes for coordination in meeting its goal of safe operation. For a specific plant, these processes will be empirically validated. The validation will involve identification of the specific managers and supervisors who have key influences on the quality of each process. The nature of these influences will be described as a set of behaviors which each key manager and supervisor exhibits. In addition, management functions and processes will be assess at the organizational behavior level. The attitudes, policies, and behaviors projected by upper management influence the nuclear power plant's organizational climate. This influence is then permeated to the middle- and first-line management functions. All these influences will be assessed during the data collection phase of this project.

If we assume that the nuclear power plant operates in a machine-like manner, then the key process of the organization is the standardization of work as described earlier. Four subprocesses can be considered under the standardization of work; the design of standards (including procedures for hardware and software components of the plant), the application of standards (the conveyance to personnel performing the work which is involved), the feedback on standards (communication and education of refinements to modify the standards) and the override of standards (modification in the event of abnormal conditions).

The key supervisory and management functions can then be identified in terms of the subprocesses of standardization. The development of the standards occurs largely within the technostructure of the organization. Department heads of the units in this component are critical in developing the policies and procedures included in standardization. The middle line managers are responsible for interpreting the standards that are designed. In turn, the supervisors of the operating core ensure implementation of the standards through their employees. Feedback for modifications to the standards should occur across the components of the organization through these supervisory/management functions as well.

Methods for Measuring Organizational and Supervisory/Manager Influences

In order to assess and evaluate nuclear power plants with regard to organizational and management factors, three specific types of data collection methods are proposed for this project. The first is described as a functional analysis of the plant (organization). A description of the organizational structure and functioning is obtained through documentation, interviews, walk-throughs, talk-throughs, and some observation. Networking of communication and work flow are identified as well as key personnel within the components of the organization. The functional analysis will provide a good qualitative description of the organizational factors relevant to the plant and allow for resource allocation of the other methods to be used.

An assessment of the organizational culture/environment of the plant will be obtained through the use of a standardized inventory. The Organization Culture Inventory [6] has been used across many different types of organizations to assess organizational culture. Included in its application has been another high reliability organization, aircraft carriers [7]. Twelve scales describing the environment of the organization have been developed and an additional scale specific to the issue of safety is presently being developed [8]. This inventory is a paper and pencil survey which can be administered in large groups to all levels within the organization.

The final data collection method to be used is an observational technique. Developed by Judith Komaki, [9] the Operant Supervisory Taxonomy Index uses a standardized taxonomy of behaviors as the basis for categorizing and describing behaviors exhibited by managers and supervisors. Based upon the functional analysis of the plant described earlier, selected supervisors and managers will be observed during their workday. Observers shadow the individuals being observed and samples consist of 30-minute observation periods at different times of the day. The number of sample observation periods and the number of individuals to be observed will be decided when a specific organization is identified for the demonstration study. This technique has also been used across different types of organizations. Observers are trained for at least 40 hours prior to the collection of any data. The technique and examples of its application are described elsewhere [10].

SUMMARY

The NPP is probably best described as a machine-like organization with some differences in structure within the operating core. These differences, however, do not significantly effect the overall organization of the plant and how it functions. An important condition that drives this organization to a machine bureaucracy is its special need for safety [11]. Procedures are formalized extensively to ensure that they are carried out and result in safe operation. The airline industry is another example of what Mintzberg identifies as a "safety bureaucracy." Key supervisory and managerial functions are best depicted within the machine bureaucracy, and most authority within the plant is vertically centralized.

An additional point to address when dealing with any aspect of the organization of a nuclear power plant is that it is an organization whose need to structure centers around crisis. In Mintzberg's terms [12], there is a professional overlay to this day-to-day functioning machine bureaucracy. The professionalism is exemplified in the high level of training and skill required for dealing with crisis situations. Several aspects of this idea seem clear: The plant is designed for "no" failure; at the hardware level, through engineering and design, and at the software level though procedures and the technostructure in its standardization of work. The plant is operated for no failure through the training and requirements of its operating core; the specifications for maintaining its hardware and the regulatory control over its operations. Finally, the plant should be able to deal with failures through its structural organization and the level of preparedness at which it operates.

The proposed organizational structure of a NPP that is described in this paper provides one means of conceptualizing the dynamics of a NPP organization. The model as described by Mintzberg is process-oriented and allows for the identification of the key supervisory and managerial influences on plant performance through the evaluation of organizational processes. Examination of the design of standards, both in hardware and software (technostructure), the application of standards through the operating core and the feedback on these standards from the operating core back to the technostructure, are critical in understanding the functional dynamics of an NPP [13]. Evaluation of the design parameters, functional characteristics, and contingency factors associated with the structural type identified will also help to uncover the pathways by which the organization functions. Behavioral factors unique to supervisory and managerial influences can then be identified for further examination.

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SAFETY SYSTEM FUNCTION TRENDS

Carl Johnson

U.S. Nuclear Regulatory Commission

Abstract

This paper describes research to develop risk-based indicators of plant safety performance. One measure of the safety-performance of operating nuclear power plants is the unavailability of important safety systems. Brookhaven National Laboratory and Science Applications International Corporation are evaluating ways to aggregate train-level or component-level data to provide such an indicator. This type of indicator would respond to changes in plant safety margins faster than the currently used indicator of safety system unavailability (i.e., Safety System Failures reported in LERs). Trends in the proposed indicator would be one indication of trends in plant safety performance and maintenance effectiveness. This paper summarizes the basis for such an indicator, identifies technical issues to be resolved, and illustrates the potential usefullness of such indicators by means of computer simulations and case studies.

Introduction

NRC's performance indicator program provides an additional view of plant operational performance and enhances our ability to recognize areas of poor or declining safety performance of operating nuclear power plants. This performance indicator program is led by NRC's Office of Analysis and Evaluation of Operational Data.

One part of NRC's performance indicator program is research to develop and validate risk-based performance indicators. This research has developed and tested methods to analyze data on components or trains of selected safety systems taken out of service to provide indicators of the unavailability and unreliability of important safety systems. The research team includes: J. L. Boccio, M.A. Azarm, J.F. Carbonaro, N. L.Oden, and J.A. Penoyer, BNL; and W. E. Vesely, SAIC.*

Interim results of this research are summarized below.

Basis for Indicator

The likelihood of an accident resulting in core damage is the product of the frequency of initiating events times the probability that safety systems will not respond and the operator will not recover the situation. The probability that plant safety systems will not respond is expressed as their unavailability. Therefore, one measure of the safety-performance of operating nuclear power plants is the unavailability of important safety systems.

* Brookhaven National Laboratory/Science Applications International Corporation.

Also, the Commission defines maintenance as the aggregate of those functions required to preserve or restore safety, reliability, and availability of plant structure, systems, and components (1). Therefore, a degrading trend in the availability or reliability of important safety systems is one indication of ineffective maintenance.

Measuring unavailability of standby safety systems in real time is impossible, because we cannot know whether a standby system will startup and respond to the next demand. However, we can estimate the unavailability based on recent performance data.

NRC's currently used indicator of unavailability of safety systems is the number of safety system failures in Licensee Event Reports. These infrequent instances of loss of system function provide data when the system unavailability equals 1.

Improved indicators of unavailability of safety systems are intended to track changes in safety margins (i.e, loss of redundancy) before loss of system function. Such indicators can be based on loss of component or train function. This approach uses more frequent data, and results in a probabilistic estimate of unavailability; not just 0 or 1. Therefore, this type of improved indicator should more accurately reflect the magnitude and trend of unavailability or unavailability of safety systems before the loss of system function.

Testing of Indicators with Computer Simulations

Several characteristics of indicators of unavailability can be evaluated by computer simulations (2,3). The BNL/SAIC research team used computer simulations to evaluate the characteristics of several formulations of indicators of unavailability and unreliability of safety systems. For example, computer simulations were used to analyze how long the indicator would take to respond to degradation of plant equipment (possibly due to effects of aging.) A typical example is illustrated in Figure 1. In this example, at time 0, the component failure rate in the computer is doubled.

The system failure rate is then tracked by the two indicators: (1) the existing indicator, i.e., safety system failure; and (2) an improved indicator based on train-level data. In this typical case, the first system failure occurred after 11 quarters, and an even longer time would be needed to recognize a trend. On the other hand, the indicator based on train-level data reflects a clear trend within a few quarters.

Testing of Indicators with Plant Data

In addition to computer simulations, available plant data were used to test whether an indicator of unavailability of selected safety systems based on train-level data gives meaningful results. For example, one plant provided maintenance records from which BNL and SAIC calculated indicators of unavailability of selected safety systems. Results are illustrated in Figures 2 and 3.

Figure 2 shows the indicator for unavailability of the average train of emergency diesel generators. The apparent degradation before the shutdown in 1986 and the improving trend after that shutdown are statistically significant trends. During that shutdown, the plant implemented elements of a reliability program.

Similarly, Figure 3 shows the indicator for unavailability of the average train of auxiliary feedwater at the same plant. Here again, the degrading trend before the 1986 shutdown and the improving trend afterwards are statistically significant. Thus, the indicator appears useful to help flag potential problems, and to recognize improved performance.

Issues

Although indicators of unavailability of safety systems appear useful, some issues remain to be resolved.

One issue concerns human intervention into safety systems during power operation. Implementing an indicator of unavailability of safety systems might have the unintended effect of encouraging plants to perform excessive preventive maintenance to ensure that safety systems successfully pass surveillance tests. Such preventive maintenance would be partially picked in the next surveillance test and included in the indicator. However, initiating events caused by errors during maintenance or testing during power operation would not be picked up by this indicator. We plan to explore ways to avoid this unintended potential side effect.

Another concern is to improve the indicator to better evaluate the potential for dependent failures (i.e., potential multiple train failures).

A third concern is that NRC does not currently receive reliable, timely data for this kind of indicator of the unavailability of important safety systems. The potential value of such an indicator and potential sources of data are being evaluated.

Summary

Risk-based indicators can be useful to help monitor the unavailability and unreliability of selected safety systems. Such indicators are being validated with plant data. Improvements are planned to better reflect dependent failures, human intervention, and available data sources.

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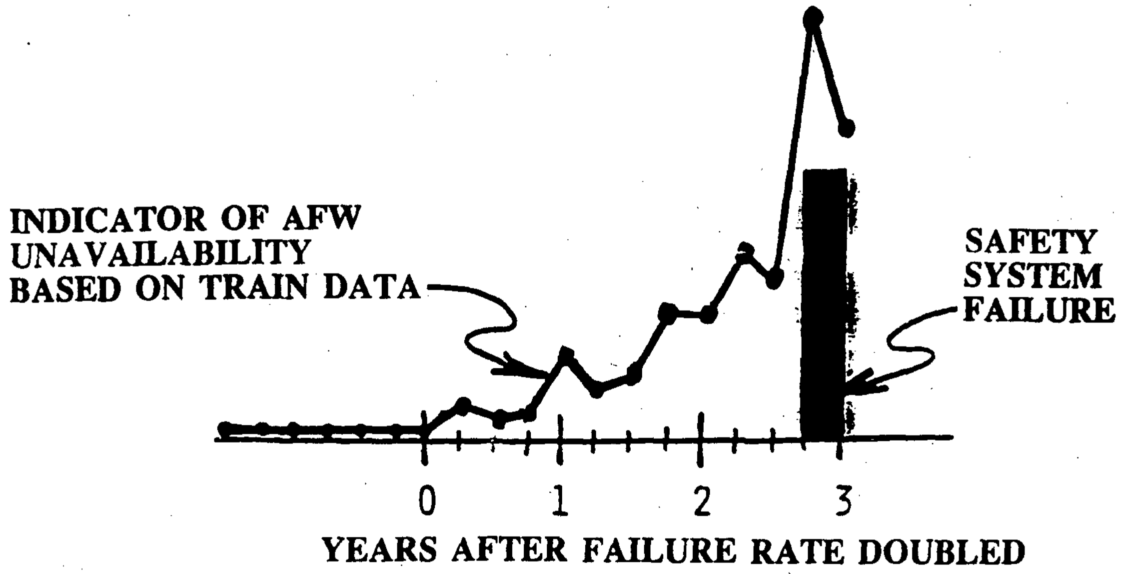

Fig. 1. Example of computer simulation comparing response time of indicator based on train-level data vs. the existing indicator based on loss of system function.

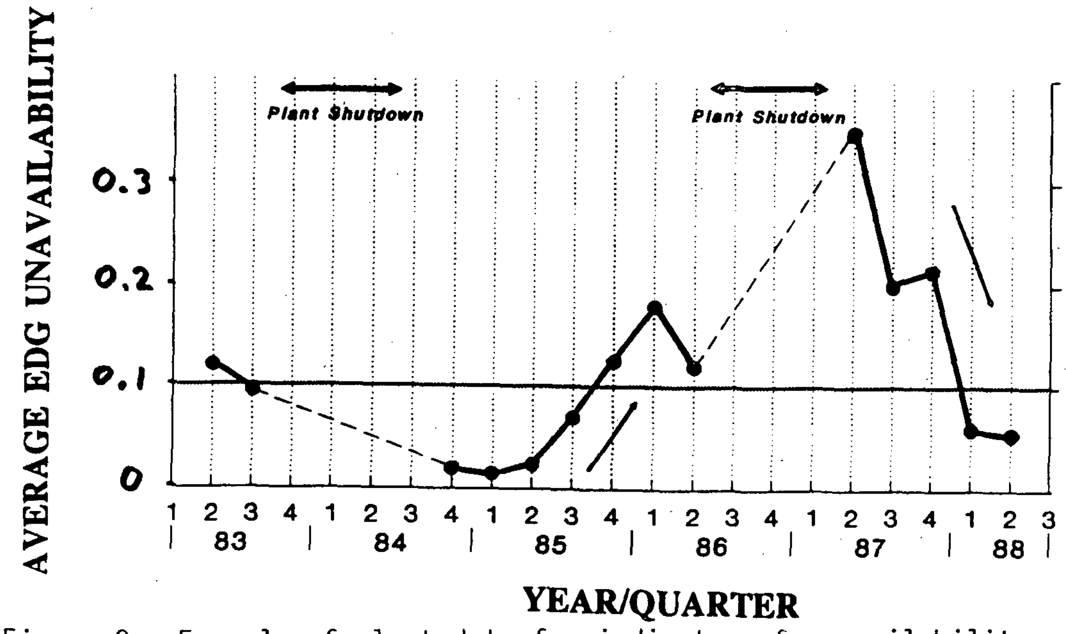

Figure 2. Example of plant data for indicator of unavailability of average EDG.

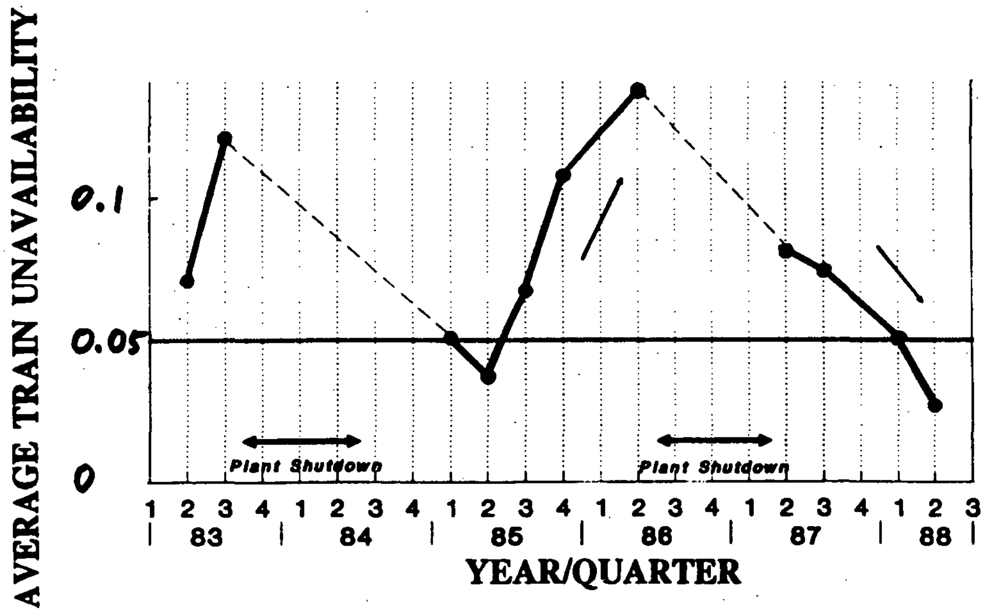

Figure 3. Example of plant data for unavailability of average train of AFW.

SELECTION OF ANCHOR VALUES FOR HUMAN ERROR PROBABILITY ESTIMATION

Louis C. Buffardi **Edwin A. Fleishman** **John A. Allen**

Center for Behavioral and Cognitive Studies
George Mason University

There is a need for more dependable information to assist in the prediction of human errors in nuclear power environments. The major objective of the current project is to establish guidelines for using error probabilities from other task settings to estimate errors in the nuclear environment. This involves: 1) identifying critical nuclear tasks, 2) discovering similar tasks in non-nuclear environments, 3) finding error data for non-nuclear tasks, and 4) establishing error-rate values for the nuclear tasks based on the non-nuclear data. A key feature is the application of a classification system to nuclear and non-nuclear tasks to evaluate their similarities and differences in order to provide a basis for generalizing human error estimates across tasks. During the first eight months of the project, several classification systems have been applied to a sample of nuclear tasks. They are discussed in terms of their potential for establishing task equivalence and transferability of human error rates across situations.

Background of Problem

It is well established that human error plays a major role in the malfunctioning of complex, technological systems and in accidents associated with their operation. Estimates of the rate of human error in the nuclear power industry range from 20-65% of all system failures. In response to this, the Nuclear Regulatory Commission has developed a variety of techniques for estimating human error probabilities for nuclear power plant personnel. Most of these techniques result in the specification of the range of human error probabilities for various tasks. Unfortunately, very little performance data on error probabilities exist for tasks within nuclear power plant environments. It is this shortage of these data that is the most critical factor impeding human reliability index development (Dhillon, 1986). Thus, when human reliability estimates are required (for example, in computer simulation modeling of system reliability), only subjective estimates, usually based on experts' best guesses, are available.

Objective and Strategy

The major objective of the current three-year research project is to establish guidelines for applying error probabilities from *other* task settings to nuclear power plant environments. This involves: 1) identifying critical tasks in nuclear power plant settings, 2) discovering similar tasks in non-nuclear power plant environments, 3) finding human error data bases for non-nuclear tasks, and 4) establishing error-rate values for the nuclear tasks based on the non-nuclear data.

A key feature of the current research is the application of comprehensive classification schemes to nuclear and non-nuclear tasks to evaluate their similarities and differences, thus providing a basis for generalizing human error estimates across tasks. Initial project objectives are to: 1) identify alternative taxonomic schemes that can be applied to tasks, 2) describe nuclear tasks in terms of these schemes, and 3) develop a network of contacts within non-nuclear power plant settings that might have relevant task statements and human error data. The purpose of the current paper is to report the progress made to date during this first year of the project.

Taxonomies of Human Performance

Although tasks are pervasive in everyday life, until recently there have not been many attempts to conceptualize the variables associated with the kinds of tasks that people perform. Much of the method, technique, and research to be discussed was developed by Fleishman (1967, 1975a, 1975b, 1982) and his associates under a contract originally funded by the Advanced Research Projects Agency of the Department of Defense. This project, which came to be called the Taxonomy Project, was monitored and sponsored by the Air Force Office of Scientific Research and later by the U.S. Army Research Institute. Over the years, many prominent investigators and experts in a variety of fields were associated in one way or another with the Project. From the start, one of the major aims was to develop means of conceptualizing tasks and their characteristics in order to resolve important problems concerning human performance. As a result, many systems and schemes designed to conceptualize and classify variables associated with the variety of tasks that people perform have been developed and evaluated (Fleishman & Quaintance, 1984).

Such taxonomies of human performance have a number of practical and scientific implications. Two implications most related to our current purpose are:

1. Generalizing research to new tasks. A human performance taxonomy should assist in extrapolating from previously attained research results to new tasks. A useful taxonomy would tell us if these tasks are in the same or different categories, thus providing a basis for generalizing human error results obtained on one task to the other similar tasks.

2. Establishing better bases for conducting and reporting research studies in order to facilitate their comparison. A comprehensive classificatory system should aid in disclosing the reasons why studies (or data bases) can or cannot be compared. A taxonomic system should provide some guidelines for improving the conduct of research.

Description of Specific Taxonomic Schemes

One accomplishment of this first year of this project is the identification of taxonomic systems which could be applied to nuclear and non-nuclear tasks. Thus far, a comprehensive review of previous taxonomic efforts has identified three such schemes (Fleishman and Quaintance, 1984):

1. **Ability Requirements Approach** (Fleishman, 1975a; 1982). Abilities are relatively enduring attributes of an individual performing the task. The assumption is that specific tasks require certain abilities if performance is to be maximized. Tasks requiring common abilities would be placed within the same category. The Ability Requirements Approach

includes definitions of 52 abilities (e.g., Oral Expression, Problem Sensitivity, Inductive Reasoning, etc.). Each ability has a 7-point rating scale which includes task anchors that provide raters with examples of everyday tasks that reflect high, moderate, and low levels of each ability. (See Table 1 for sample anchors for the Verbal Comprehension ability.)

Table 1. Ability Requirements Approach

Verbal Comprehension

This is the ability to understand English words and sentences.

How Verbal Comprehension is Different from Other Abilities

Understand spoken or written English words and sentences.	vs.	*Verbal Expression:* Speak or write English words or sentences so others will understand.

	Scale	Task anchors
Requires understanding of complex, detailed information which contains unusual words and phrases and involves fine distinctions in meaning among words.		
	7	
	6	
		--Understand in entirety a mortgage contract for a new home
	5	
	4	
	3	--Understand a newspaper article in the society section reporting on a recent party
	2	
Requires a basic knowledge of language necessary to understand simple communications.		--Understand a comic book
	1	

There is an extensive empirical data base associated with this approach. Interrater reliability with current versions of these scales, used to describe jobs and tasks in a wide variety of industrial, governmental, and military settings, tend to be in the .80s and .90s (Hogan et al., 1978; Myers et al., 1979; Cooper et al., 1982). In addition, the scales show evidence of construct and predictive validity. Of particular relevance to the current study, this system has been useful in integrating research data in a variety of areas and in clustering jobs with common attributes.

2. **Generalized Information-Processing System** (Miller, 1973). Miller believed that a useful task vocabulary is needed to provide an exhaustive list of information-processing functions representing all classes of system transactions. As a result, he created descriptive and analytical terminology to represent 24 task functions (e.g., Detect, Transmit, Store, Interpret, etc.). For some of these functions, concrete examples of real-world behavior involving the functions were provided. Thus far, there has been no quantitative evaluation of this system. (See Table 2 for a sample of Miller's categories with definitions and examples.)

Table 2. Generalized Information Processing System

Sample Classifications and Definitions

Classification	Definition
Control	Changing the direction, rate, or magnitude of a physical force that may be acting on objects, processes, or symbols. The stimulus may be embedded in a fixed serial order, or it may consist of feedback test signals.
Detect	Procedures and mechanisms for sensing the presence or absence of a cue or condition requiring that some form of action should be taken by the system.
Message	A pattern of input symbols that is *meaningful* and purposeful in that it activates (or can activate) some processing capability of the system in generating a useful response.
Decide	Rules for selecting a response alternative to given states of affairs. Conceptually, the simplest decision mechanism is a two-way switch in which the input may be in one of two relevant states, each of which selects a response alternative.
Display	Arranging messages into a prescribed format and symbology for human perception and interpretation.
Input select	Rules for admitting a message or message channel into the internal system.
Reset	Purging an old context of status and readiness in order to respond by substituting a new context of status and readiness.
Transmit	Rules and conditions for transmitting a message from one location to another.
Search	Rules for selecting a set of entities for inquiry, for sequencing an inquiry among members of the set to be searched, and rules for applying criteria of *same* or *different* between the objective (search image) for searching and the objects in the search set being examined.

. **Task Characteristics Approach** (Farina and Wheaton, 1973). In contrast to the previous taxonomic schemes, the Task Characteristics Approach conceptualizes tasks *per se*, independent of the human operator's abilities or functions. For example, tasks can be characterized in terms of kind of controls, displays, or various other types of hardware with which an individual may interact during the operation of a system. This approach includes definitions of 21 task characteristics (e.g., dependency on procedural steps, variability of stimulus location, and feedback lag relationship, etc.), with each characteristic having a 7-point rating scale that includes several task anchors along the continuum. (See Table 3 for definitions and anchors for the Precision of Responses task characteristic scale.) Reliability studies with these scales have been encouraging, but not conclusive. With respect to validity, some significant relationships between the task characteristics measures and actual performance have been demonstrated. In general, although this scheme is not yet a definitive taxonomy of task characteristics, it shows considerable promise.

Table 3. Task Characteristics Approach

Scale for Precision of Responses

Tasks may differ in terms of how precise or exact the operator's responses must be. Judge the degree of precision involved in the present task by considering the *most* precise response made in producing an output unit.

Definitions		Examples
High degree of precision--because of small targets, fine scales, sensitive controls, etc., the subject must make responses which are extremely precise.	7	o Using a chemical balance (scales), determine the weight of the following objects to the nearest microgram.
	6	o Replace the mainspring in this wristwatch.
	5	
Moderate precision--relative to the definitions above or below, a moderate degree of precision must accompany subject's responses.	4	o Using your pencil, trace this maze.
Low degree of precision--because of large targets, gross scales, insensitive controls, etc., the subject can make responses which are gross or imprecise.	3	
		o Do 20 push-ups.
	2	o Sort the oranges and lemons into two piles.
	1	

Identification of Nuclear Control Room Tasks

One of the first steps in the process is to identify important tasks in nuclear settings so that the aforementioned taxonomies might be applied. Project staff have been reviewing sources of such data. An example of one source identified is a recent task analysis of nuclear power plant control room crews (Burgy, et al, 1984). The task analysis in this previous study provides data for tasks involved in normal, off-normal, and emergency operating procedures, as well as operator qualification and training requirements. Data collection was conducted at eight nuclear power plant sites that were sampled according to vendor, vintage, simulator availability, architect-engineer, and control room configuration. Twenty-four operating sequences, covering a variety of functions (e.g., generate power, restore plant to a safe condition, mitigate consequences of an accident, maintain plant systems and equipment, coordinate plant support activities) were selected on the basis of frequency and criticality of the sequence in plant operations. Subsequent analyses yielded 470 task statements across the 24 operating sequences. Thus, this document provides a comprehensive list of representative critical tasks in nuclear power plant control rooms for use in the present study. The tasks in this study, along with tasks from other studies examined, are being utilized to develop a core list of nuclear power plant control room tasks.

Applying Taxonomies to Nuclear Control Room Tasks

As a preliminary step, the Ability Requirements Approach and the Generalized Information-Processing System were applied by two members of the project research staff to each of the 470 task statements. The intention is to eventually use the same procedure with the Task Characteristics Approach. Task statements were placed into the category that the rater believed was most appropriate for that taxonomic scheme. The two raters worked independently, accessing the manuals which provided detailed descriptions of the various categories in each of the three taxonomic schemes.

Tables 4 and 5 provide summaries of the average percentage (across judges) of frequency counts and the percentage of scorer agreement for each of the task categories for the Ability Requirements Approach and the Generalized Information-Processing Approach, respectively.

Table 4. Applying Ability Requirements Approach to Nuclear Control Room Tasks

Average Percent of Task Statements	Ability Category	% Scorer Agreement
41.0	Control Precision	95.0
12.9	Perceptual Speed	86.4
10.2	Time Sharing	89.6
10.0	Oral Expression	96.0
5.3	Oral Fact Finding Ability	92.1
5.2	Information Ordering	64.4
5.0	Multilimb Coodination	53.4
2.9	Written Expression	96.4
2.6	Flexibility of Closure	96.0
2.2	Manual Dexterity	87.5

Table 4. Applying Ability Requirements Approach to Nuclear Control Room Tasks (Continued)

Average Percent of Task Statements	Ability Category	% Scorer Agreement
2.3	Written Comprehension	100.0
1.9	Number Facility	75.0
1.6	Problem Sensitivity	75.0
1.3	Deductive Reasoning	68.5
1.3	Selective Attention	75.0
0.5	Rate Control	83.0
0.4	Choice Reaction Time	100.0
0.1	Oral Comprehension	50.0

Table 5. Applying Generalized Information-Processing System to Nuclear Control Room Tasks

Average Percent of Task Statements	Miller Taxonomy	% Scorer Agreement
46.6	Control	98.2
19.7	Detect	96.8
16.0	Message	94.6
3.0	Display	100.0
3.0	Test	100.0
3.0	Transmit	100.0
2.4	Interpret	92.0
1.7	Reset	93.7
1.6	Compute	72.5
1.0	Decide	50.0
0.3	Que to Channel	50.0
0.3	Code	0.0
0.2	Identify	100.0
0.1	Input Select	50.0

With the Ability Requirements Approach, over 89% of the nuclear control room tasks are described by one of seven abilities: control precision, perceptual speed, time sharing, oral expression, oral fact finding, information ordering, and multilimb coordination. The raters agreed on 87% of their judgements, with most of the disagreements concerning whether a given ability was of primary or secondary importance in task performance.

Similarly, using the Generalized Information-Processing System, 94% of the tasks are described by one of six functions: control, detect, message, display, test, and transmit. The two raters agreed on 93% of their judgments.

At this stage, some preliminary evaluation of the taxonomies is possible, particularly with respect to the *internal* validity. Two criteria related to internal validity evident in this initial study are scorer reliability (Do judges agree with the category chosen for a given task?) and discriminability between tasks (Does the classification scheme provide enough different categories to provide useful distinctions among tasks?). Based on these initial results, it would appear that the Ability Requirements Approach and the Generalized Information-Processing Systems are viable taxonomic schemes that provide reasonable discrimination between tasks and demonstrate respectable scorer reliability. In the near future, the Task Characteristics Approach also will be similarly evaluated. The Ability Requirements Approach has the added advantage of an anchored rating scale that could provide even finer distinctions between levels within an ability category. Although these scales were not used in the present pilot study, they will be utilized in the next phase and may be particularly useful in interpolating error rates from data bases on non-nuclear tasks.

Future Directions of the Research

Although two classification systems show promise for reliably describing nuclear control room tasks, additional criterion information is needed to evaluate the adequacy of these schemes. One obvious need is to validate the judgments made by project staff with nuclear power plant personnel serving as subject matter experts, applying the selected taxonomic schemes to the same operating sequence tasks. Such information will provide further evidence on the validity of the taxonomic schemes as well as provide additional reliability data. Furthermore, the points of disagreements between judges could be examined with the possibility of refining the current taxonomies to make them more applicable to nuclear tasks. We will also continue to examine the feasibility of the task characteristic approach to classifying tasks.

Future studies will extend these analyses beyond control room tasks. It is quite possible that human error rate data may be more plentiful for other types of tasks. Hence, similar work has begun in applying the taxonomies to maintenance jobs.

Once the reliability and validity of describing nuclear tasks using standardized taxonomies is fully demonstrated, then the process must be repeated for comparable non-nuclear tasks. An accomplishment of this first year of the project is the establishment of an extensive network of contacts within non-nuclear power plant organizations that may have relevant task and human error information. Currently, work has proceeded on this front with task list information being requested on military, electric power industry, and air-traffic control positions that would appear comparable.

One critical issue to be resolved is the development of criteria for establishing task comparability. Two tasks classified within the same category of a single taxonomic scheme may still differ in other ways. For example, verbal comprehension may well be required for understanding a mortgage contract for a new home and for understanding the directions in a technical manual. However, if one task is done in a quiet room and the other with many alarms sounding simultaneously, the two tasks are likely to have widely different error rates. One way to address this issue would be to apply several different types of taxonomies to the tasks before comparability is established. Alternatively, each category within the currently available taxonomies may have to be refined to provide the greater precision needed to reach that judgment.

Once the comparability of tasks across settings is established, human error rate data on the non-nuclear tasks can be sought and applied to the nuclear tasks. This line of research leads directly to the goal of generalizing error rate data from other environments to nuclear settings, provides greater understanding of system reliability, and should contribute significantly to nuclear power plant safety.

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Human Performance Data Acquisition and Management for Reliability Evaluations [1]

David I. Gertman, Human Factors Research Unit
Idaho National Engineering Laboratory, Idaho Falls, Id 83401 USA

ABSTRACT

The Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) is an automated data base management system used to process, store and retrieve human error probability (HEP) and hardware component failure rate data (HCFD) in a ready to use format. The data base has now been implemented and efforts continue to identify and process qualified sources of data. A Clearinghouse function has been established at the Idaho National Engineering Laboratory (INEL) to: handle requests for data, issue software, answer user questions, and publish NUREG/CR-4639 Volume IV: User's Guide. In order to reflect the continual addition of data to NUCLARR, the Clearinghouse issues update pages to Volume V: Data Manual. This presentation summarizes existing HEP data in terms of the type of plant systems, components and displays, controls and indicators used by plant personnel. Most NUCLARR data comes from either NUREG, laboratory or consensus expert judgement sources. Suggestions are made regarding obtaining and processing additional data from root cause data bases, sister industries, and the international community at large.

1. Introduction to NUCLARR

Due to the perceived need for collecting, processing, storing and managing human error probability data, work to produce a Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) was initiated in 1982. Since HRA was an emerging discipline, the thought at the time was not to limit the data contained in the data bank to only one collection method (source) or calculation technique. There was no logical basis, for example, for accepting laboratory studies or plant operations experience and excluding consensus expert judgement sources. It was determined that the risk analyst would be the final arbiter of data quality and would determine which data should be aggregated. Upon this foundation NUCLARR has been built and is now fully functional, handling both human error probability (HEP) and hardware component failure (HCFD) data.

[1] Requests for reprints should be sent to Dr. David I. Gertman, Human Factors Research Branch, EG&G, Idaho, Idaho Falls, Idaho 83415.

Human error probability (HEP) data, contained within the NUCLARR data bank, are organized in a series of 16 matrices. The analyst searches for data items by combining equipment characteristics and human actions on three levels. Equipment characteristics are arranged by rows, and human actions are arranged by columns. In a typical scenario, the analyst first identifies the job position and type of human error committed. Error types include errors of commission and omission. Next, the analyst must decide whether recovery factors need be modeled. If so, the "recovery considered" option is specified during the on-line search process. Finally, the applicable equipment (system, component, or individual display, control or indicator) is selected. The search may be specified further by specifying acceptable values for performance shaping factors such as stress, training and procedures.

A number of parameters can be used to describe and define error probability and rate-based data. Those presently coded and available for the analyst to review include:

<u>Reference document</u>- All references serving as sources of data entered in NUCLARR are documented in an on-line addressable reference data file . Each data record in the system is also tagged according to reference source. Both the data file and the individual data point reference are available for the user's review.

<u>NSSS vendor</u>- The Nuclear steam system supply vendor is indicated, so that users wishing to search only those data referring to Westinghouse, General Electric, Combustion Engineering, B&W or General Atomic plants may do so.

<u>Taxonomy level</u>- (System, component, individual display or indicator)- Data are entered in NUCLARR as function of the level of equipment specified in the original reference document. The three taxonomy levels correspond to Systems (Level 1), Components (Level 2), and Displays, controls and indicators (Level 3). For example, error rate estimations for operator or crew attempts to initiate high pressure injection is indexed within NUCLARR taxonomy Level 1- Systems. Operator attempts to address a particular pump or valve is indexed under components; and error rates for reading or calibrating devices such as a chart recorder or a single annunciator are categorized as Level 3- individual displays and indicators.

<u>Error type (omission, commission)</u>- Classification of process plant operator response is action oriented and errors are either classified as being the type where responses are absent (omission) or performed improperly (commission). NUCLARR assumes no cognitive model per se.

<u>Recovery factors</u>- During data processing, all HEP data are reviewed to determine if recovery factors have been included in the HEP calculation. During data entry, each data point is labeled as "recovery considered" or "recovery not considered".

<u>Action verb</u>- HEP estimates are based upon human actions which have led to errors. These actions are coded in NUCLARR are keyed to the three taxonomy Levels.

<u>Personnel type(job position)</u>- The following job positions are represented in NUCLARR; control room operator, auxiliary or equipment operator, and maintenance technician.

<u>Task statement</u>- Task statements take a predetermined format which includes: the job position, action verb, and plant systems or equipment addressed.

Plant conditions- Plant conditions are placed in the "comments" field of all NUCLARR data records. Much data recorded to date reflects HEP data for operator actions during steam generator tube rupture, station blackout, anticipated transient without scram, or loss of coolant accident scenarios.

HEP Value and Related Information- Quantitative information including error factor, confidence bounds, number of errors and number of opportunities for error, time available, and time needed by the crew in order to respond are all represented in NUCLARR.

Location- Operator actions are entered as either local or remote in regards to the control room.

Performance Shaping Factors (PSFs)- PSFs are used as additional parameters to aid the analyst in searching the data base. These parameters include training, stress, experiences, procedures, man-machine interface, tagging, and quality of supervision.

Range of Documentation to be searched (by period) - Users have the option to review only those data collected from specific historical periods. For example, it is possible to review only those data collected prior to 1980.

Data Origin- Data are classified as one of the following; field, training simulator, laboratory, consensus expert judgement, simulation, or analytic.

Time to Perform- Operator or crew performance times for a task are entered along with the system time allowable for that task.

2.1 Criteria for HEP Inclusion

HEP data entered into NUCLARR must meet three specific criteria:

They must specify a human action,

They must specify a system or piece of equipment; and

They must be quantitative in nature.

The most preferable data are in the form of an HEP statement with upper and lower confidence bounds. Data presented as median values with errors, or simply as error observed over the number of opportunities for error, are also acceptable.

A human and hardware reliability analysis group (HHRAG) meets on a periodic basis to process data and to provide a quality assurance function for data resident in the data bank. Members of the HHRAG include representatives from experts available in government, industry, and academia.

The NUCLARR Clearinghouse has responds to user requests for data, software, or User's Guides, and has the additional responsibility of issuing periodic updates to Volume V: Data Manual.

2.2 HEP Data Treatment

NUCLARR automatically makes a number of calculations for each HEP data point entered. Depending upon the degree of detail present when data are first entered, the NUCLARR system software will compute upper confidence bounds (UCBs), lower confidence bounds (LCBs), error factors, medians, means, errors and opportunities for error. The system keeps track of which values have been entered by NUCLARR data technicians and which are system calculated values.

Separate aggregation algorithms are applied to compute task statement HEPs, cell HEPs and functional group HEPs. Aggregations are computed for each of the three levels of the NUCLARR systems taxonomy. Thus, each of the aggregations is nested in each of the equipment taxonomy levels.

A description of the types of data combinations performed are listed below:

<u>Task Statement HEP</u>- an estimate of the HEP for similar data all of which share a like task statement. Data are not deemed like if, for example, they are not of the same error type or differ in whether or not recovery has been factored in the error estimate.

<u>Cell HEP</u>- an estimate of the HEP for similar tasks in which equipment are identical but situations may vary. As in the case of task statement HEP, data on commission errors are not combined with data on omission errors and data where recovery has been factored in the HEP estimate are not combined with data where recovery has not been included in the calculation.

<u>Functional Group Summary HEP</u>- an estimate of the HEP for similar tasks in which equipment <u>and</u> situations may vary. As in the case of task statement and cell HEP estimates, only like error type and recovery factors data are combined.

2.3 Computational Aspects

When task statement HEP aggregations are computed, raw source data are compared for consistency using a homogeneity test based upon the binomial distribution. Statistically consistent HEPs are pooled; the task HEP is the total number of errors divided by the total number of opportunities. Based upon binomial distribution characteristics, the UCB and LCB limits are computed.

For computing cell HEP aggregations, HEPs from functionally related tasks are gathered together and are assumed to be lognormally distributed. Therefore, the sum of the logs of the HEPs for a given cell is divided by the number of HEPs, and the antilogorithm is calculated to determine the cell HEP. Calculation of the error factor for the cell HEP is based on taking the root mean square of the log ratios of task statement UCBs to LCBs.

The highest level of aggregation in NUCLARR is the functional group summary level. The aggregation employed combines task HEPs that are functionally grouped across a set of cells. The distribution of HEPs is assumed to be lognormally distributed. The sum of the logarithms of the HEPs for these tasks comprising the functional group is divided by the number of HEPs, and the antilogarithm is calculated to determine the functional group HEP. Calculation of the error factor for the functional group HEP is based on taking the root mean square of the log ratios of task statement UCBs to LCBs.

Data estimates may be reviewed individually or in aggregated form. Originally, it was anticipated that the risk analyst would want to obtain a single aggregated value for all instances where a human action and equipment type were common. This capability is provided on-line to the analyst. who can request either to review raw data contained in the buffer or a value which is the aggregate of these raw data. In both instances, the user may generate a report which captures these value and supporting information pertaining to the task characteristics as noted in the original reference . HEPs are compared for statistical consistency using a homogeneity of variance test based on the binomial distribution. A chi-square statistic is used to compare the variance of the reported HEPs to the variance that the HEPs would have had if they were calculated from independent samples taken from the same binomial distribution. If the null hypothesis can not be rejected at 1 confidence level, the data are pooled and the HEP is computed as the total number of errors over the total number of opportunities for all source data.

3.0 Results to Date

A total of 600 HEP data points are resident in the NUCLARR system. Data collection, processing, and entry is expected to continue through 1989. A sample of the HEP values for components is presented below in Table 1. Each estimate represents data on either the cell or functional group summary level. The estimates are expected to fluctuate slightly as new data are entered into NUCLARR. The HEP values presented in Table 1 represent instances where operators committed errors of omission and recovery actions were factored into the HEP estimate.

TABLE 1. SAMPLE AGGREGATE HEP VALUES FOR COMPONENTS [1]

Component	HEP Value	UCB	LCB
Centrif. pump	.0029	.0029	.0002
Pumps(all)	.0008	.0029	.0002
Valve Operators	.0038	.0083	.0017
Valves(test)	.0058	.0092	.0036
Circuit closures	.0044	.0077	.0026
Flow contrl(diag)	.0831	.1520	.0380
ElectEquip(calib)	.1530	.3060	.0765
ElectEquip(diag)	.1771	.2891	.1085
Switch	.0148	.0405	.0054

Table 2 presents systems for which there currently is data in NUCLARR. Data in the systems category have been collapsed across NSSS vendor and job position. Because of the nesting of such factors as job position within NSSS vendor or plant, it is difficult to draw conclusions. There are still a large number of plant systems for which NUCLARR, as of this presentation, has no data. This is due, in part, to the tendency in some sources to list HEP estimates in terms of plant components or indivdual displays.

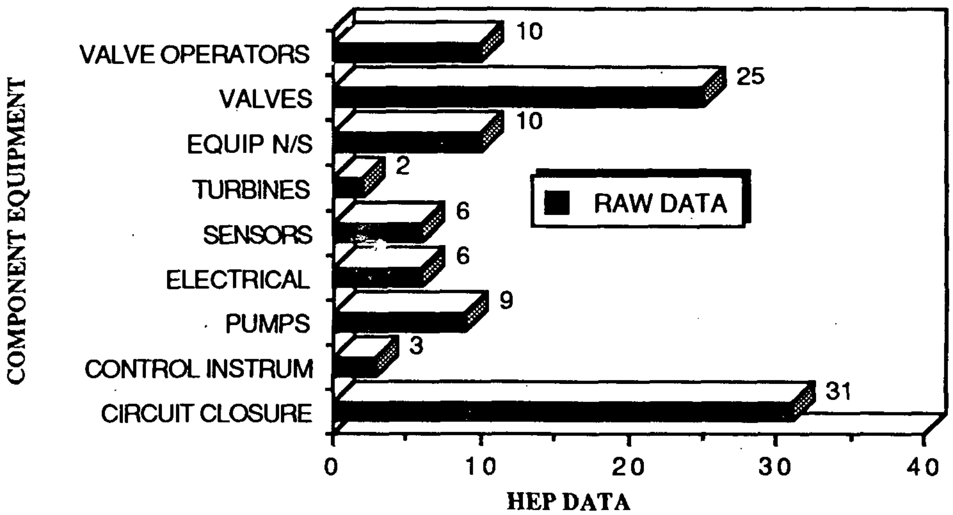
[1] Values presented for UCB and LCB represent the 95th and 5th %tile respectively.

Table 2. NUCLARR HEP Data for Systems

Air	HVAC	ECCS
Condensate	Instrumentation and Control	Feedwater
containment	Reactor Coolant	Generator
Control Rod Drive	Steam	
Electrical Distribution	Main Steam	

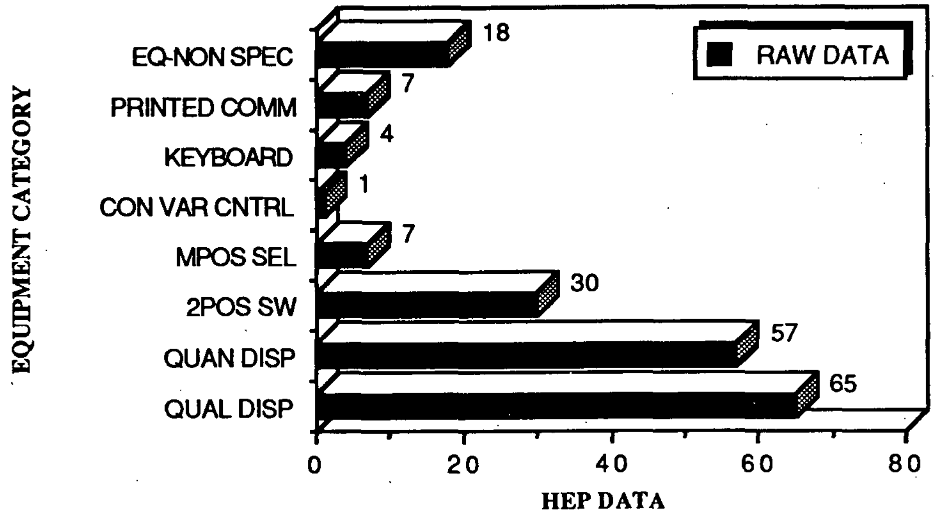
Figure 1 lists NUCLARR HEP data for plant components. The largest number of data entered are related to errors committed by personnel interfacing with circuit closures. A large number of the cases also refer to personnel attempting to realign or to operate valves. The reader is cautioned that these are the number of HEP estimates in the data management system, and those with the highest frequency, are probably studied the most, but do not necessarily contain the highest error probability.

FIGURE 1. NUCLARR HEP DATA FOR COMPONENTS



HEP data for displays, controls and indicators are presented in Figure 2. Most data entered to date refect error rates for personnel reading or monitoring qualitative and quantitative displays. A slighlty lessor number of HEPs deal with use of two-position switches. As in the case of HEP component data, the frequency of estimates for a particular component listed in the data bank is is not an indicator of the actual human error probability estimate for that component.

FIGURE 2. NUCLARR HEP DATA FOR D/C/I



4.0 Introduction to HCFD

The other half of the NUCLARR data management system contains hardware component failure data and is described briefly in the sections which follow. The design of the hardware side of NUCLARR has taken less than two years. The concept is fully operational, has been implemented in NUCLARR and is addressable from the main systems menu. HCFD are hierarchically configured within NUCLARR. All events refer to component failures; the data do not describe train or system-level events.

The data are configured around: category of equipment, type of equipment (mechanical or electrical), design of component, failure mode, normal state, and application. For example, failure modes include: fails to operate, spurious operation, leakage and blockage. Demands, hours, data origin, tolerance bounds, plant code, distribution type, whether the source rate came from a Bayesian update, severity, and a code for aggregation type are just a few of the features provided.

Figures 3 and 4 present the number of failure rates collected for the mechanical and electrical components represented in NUCLARR. In the former, most component failure rate data entered are from studies of either pumps or valves. In the latter, most failure rates are from channels, generators and power electric components.

FIGURE 3. NUCLARR HCFD-MECHANICAL

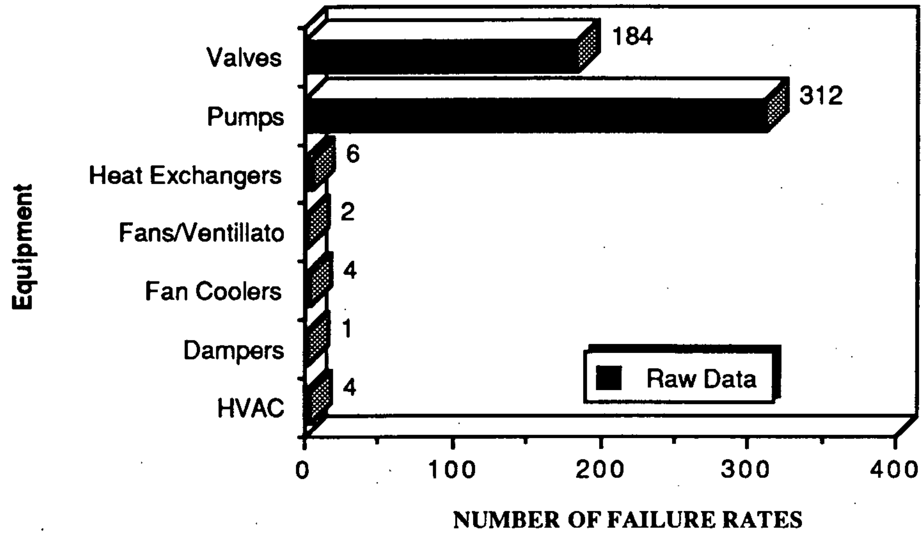
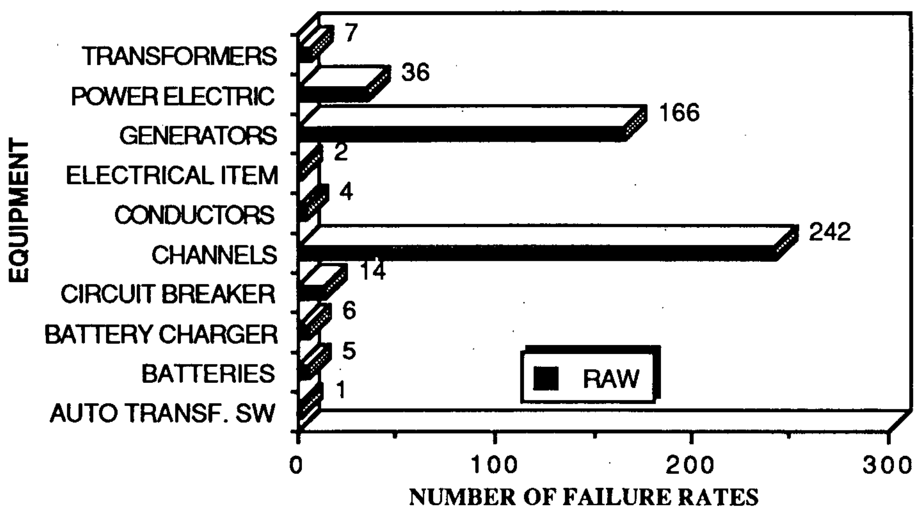


FIGURE 4. NUCLARR HCFD-ELECTRICAL



5.0 Discussion and Summary

NUCLARR has been designed to assist in risk quantification by offering a library of human error probability and hardware component failure rate estimates associated with running process control facilities. The NUCLARR system provides the risk and reliability analysis community with a menu driven, computer-based tool for performing a variety of nuclear power plant risk assessment activities.

The following computer-aided features of the system are fully operational: applications software for HEP data entry, storing, and retrieval; capability for on-line calculation of HEPs and failure rates; report generation and aggregation capability, entry level and expert search protocols; and computerized generation of the NUREG/CR-4639: The Data Manual.

Data that are resident in the NUCLARR data management system come from a number of sources and represent human performance for a variety of conditions which may exist, from time to time, at nuclear power plants. Because much of the industry's research in human performance has been safety-related, the data tend to reflect what experts feel will be likely error rates for personnel during off-normal events at these facilities.

Additional data sources must be pursued if the data bank is to be a complete repository for human error probability data. More data from simulator trials and simulator experimentation are needed. Once a taxonomic equivalence is achieved it may be possible to include human reliability data from other industries. Still another source would be extrapolation from root cause data bases. Root cause data could be transformed if the appropriate denominator estimation exercises were conducted. The knowledge gained from the addition of these data in NUCLARR, would, in the author's opinion well outweigh the costs involved.

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STAFFING, OVERTIME, AND SHIFT SCHEDULING PROJECT

Paul M. Lewis, Ph.D.

Pacific Northwest Laboratory[a]
Richland, Washington

ABSTRACT

Recent events at the Peach Bottom nuclear power plant have demonstrated the need to establish a quantifiable basis for assessing the safety significance of long work hours on nuclear power plant operators. The incidents at TMI-2, Chernobyl, and Bhopal, which all occurred during the late evening/night shift, further highlight the importance of the relationship between shift scheduling and performance.

The objective of this project is to estimate, using statistical analysis on data from the nuclear industry, the effects on safety of staffing levels, overtime, and shift scheduling for operators and maintenance personnel.

Staffing Levels - The Nuclear Regulatory Commission (NRC) currently has no explicit regulation concerning the minimum acceptable levels of staffing in a plant that has an operating license. The NRC has no systematic method for collecting data on the number of licensed operators on the operating crews. Anecdotal evidence indicates that some plants have been understaffed; understaffing leads to routine overtime.

Overtime - In 1982 the NRC recommended that plants write into their technical specifications a model policy on overtime. Currently, 77 nuclear power plant units have the model policy or a modification of it written into their technical specifications; 33 units have no policy on overtime. The model policy sets "limits" on overtime for safety related personnel, although these "limits" can be exceeded with plant manager approval. The NRC does not collect systematic data on overtime. However, evidence exists that some operators have worked considerable overtime. For example, one operator worked 97 hours in 7 days.

Shift Schedules - The U.S. nuclear power industry has three types of shift schedules: 1) forward-rotating 8-hour/day shift schedules, 2) backward-rotating 8-hour/day schedules, and 3) 12-hour/day schedules. Experts agree that forward-rotating shift schedules are generally less fatiguing than backward-rotating schedules. Twelve-hour shift schedules are becoming popular; 20 nuclear plants had 12-hour shifts for operators in 1986.

(a) Operated for the U.S. Department of Energy by Battelle Memorial Institute.

OBJECTIVES AND APPROACH

The objectives of this project are as follows:

1. to gather data on staffing levels, overtime, and shift schedules for operators and maintenance personnel in the nuclear industry

2. to estimate, using statistical analysis, the effects of these factors on safety

3. to recommend to the Performance Indicator Program definitions and methods for collecting data on staffing levels and overtime

4. to supplement, using data from the nuclear industry, the analysis that led to the recommendation for an NRC policy on staffing levels, overtime, and shift schedules (NUREG/CR-4248, Lewis 1985).

In this project, the effects on safety will be estimated using statistical analysis on cross-sectional data on staffing levels, overtime, shift scheduling, operator performance, maintenance performance, plant performance, and suitable control variables. The relationships among these variables are shown in Figure 1. The arrows indicate the direction of presumed causation.

CURRENT STATUS AND CAVEATS

This project began in April, 1988. A considerable amount of data has been collected. However, adequate data on the most important issues, operator staffing levels and operator overtime, are yet to be collected.

Some analyses of the data collected so far have been conducted. However, because this is a progress report of an on-going project, these analyses are only in the earliest stages of exploratory analysis. None of the analyses are complete or final, for the following reasons:

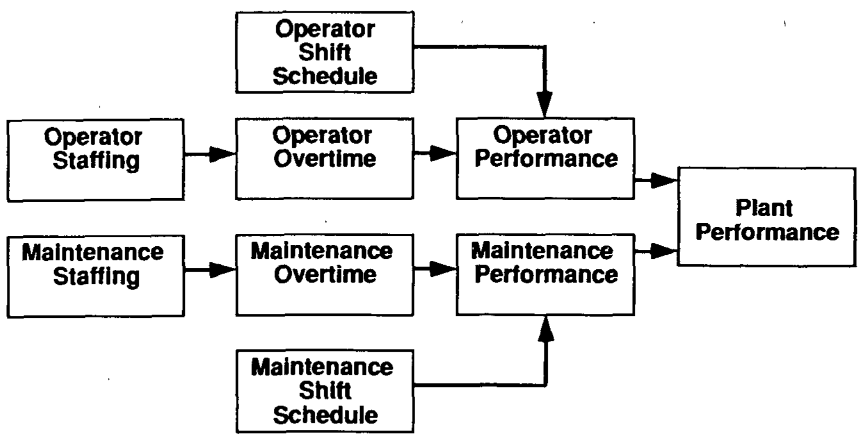

FIGURE 1. Objectives and Approach

1. Only exploratory bivariate statistical analyses have been conducted. Although bivariate analysis is useful as an exploratory device, inferences drawn from bivariate analyses of these issues can be incorrect. The proper technique for analyzing these data is not bivariate but multivariate analysis, which will be used for the final report of this project.

2. The effects of the operator variables (staffing, overtime, shift schedules) should be estimated on operator performance, not plant performance, because many factors other than the operator variables (e.g., maintenance, plant age) affect plant performance. However, the only measure of operator performance now available is the operations Systematic Assessment of Licensee Performance (SALP) score, which has many limitations. (In the future a more appropriate measure of operator performance, derived from the texts of Licensee Event Reports, will be constructed.) A similar caveat applies to the maintenance variables.

3. The figures are not necessarily good representations of cause and effect. For example, in some of the figures the presumed cause is actually data that were collected several years after the presumed effect.

The next six sections will discuss the six issues affecting safety (as shown in Figure 1): operator staffing levels, operator overtime, operator shift schedules, maintenance staffing levels, maintenance overtime, and maintenance shift schedules.

OPERATOR STAFFING LEVELS

The NRC requires that a utility demonstrate adequate staffing levels in order to obtain an operating license. The NRC also requires that a plant have certain minimum numbers of operators at any one time. However, the NRC has no requirement for minimum staffing levels (i.e., employment levels) for plants that have already obtained their operating license. Nor does NRC have a requirement that the utility inform the NRC of the number of licensed operators that are on the operating crews. Anecdotal information exists that some plants have been understaffed; understaffing leads to routine overtime.

Data on Staffing Levels for Licensed Operators

The project staff have obtained information on operator staffing levels from the following four sources.

Office for Analysis and Evaluation of Operational Data (AEOD)

Table 1 shows the number of nuclear power plants with 4, 5 and 6 crews, according to data from the AEOD (Stello 1987).

TABLE 1. Number of Plants With 4, 5, and 6 Crews

Number of Crews	Number of Plants
4	1
5	54
6	50

The number of crews, however, is not always an accurate indication of the number of operators at a plant, as is demonstrated in the following SALP report.

Systematic Assessment of Licensee Performance

According to one SALP report, one plant had fully 5 crews, which is an adequate number of crews, but only 10 licensed operators. The report stated that "a chronic shortage of operators (existed). . . . Only nine operators and one senior operator . . . were staffing five operating crews" (NRC 1985, p. 9). The U.S. Code of Federal Regulations requires that at least 4 licensed operators be on site at operating plants. If a plant were to operate with only 10 operators, each operator would have to work an average of 67 hours per week. Any hours for training would be in addition to the 67 hours required to operate the plant.

NRC Operator Licensing Branch List of Operators

The NRC Operator Licensing Branch keeps an up-to-date computerized list of all individuals who have a Reactor Operator (RO) licence or Senior Reactor Operator (SRO) license for each plant. This list includes all current license holders, whether they work full-time on operations crews or work straight days (i.e., regular daytime hours) at the utility headquarters. The information of primary interest to this project is the number of licensed operators that work full-time on operations crews. Although this source of data is not a direct measure of the number of licensed operators that work full-time on operating crews, it does provide an upper bound for that number.

A copy of this list was obtained in June, 1988. Based on that list, Table 2 shows the number of people with RO and SRO licenses for one- and two-unit sites.

Data From Region 1, Division of Reactor Safety (1988)

To overcome the limitations of the data from the Operator Licensing Branch, Bill Johnson in the Region 1 Division of Reactor Safety conducted a survey of his staff for this project to determine the number of license holders that worked full-time on operating crews. The results of his survey are shown in Table 3.

The project staff is preparing to collect survey data on operator staffing levels for all five regions.

TABLE 2. Number of People With RO and SRO Licenses for One- and Two-Unit Sites

	One-Unit Sites			Two-Unit Sites		
	ROs	SROs	ROs Plus SROs	ROs	SROs	ROs Plus SROs
Minimum	9	16	28	10	17	38
Median	18	33	52	28	42	75
Maximum	30	50	74	52	64	99

TABLE 3. Number of Licensed Operators Working Full-Time on Operating Crews at One- and Two-Unit Sites in Region 1

	One-Unit Sites			Two-Unit Sites		
	ROs	SROs	ROs Plus SROs	ROs	SROs	ROs Plus SROs
Minimum	9	17	28	18	23	47
Median	16	30	49	24	30	48
Maximum	29	44	64	34	45	79

Bivariate Analyses of Operator Staffing Data

Figure 2 shows the operations SALP scores (1986) on the vertical axis, and the number of people with RO and SRO licenses in 1988 (data from the NRC Operator Licensing Branch) on the horizontal axis.

The curved line in Figure 2 is a "smooth." A smooth is an exploratory device to aid in discovering possible non-linear relationships in data. It is not a rigorous statistical test. A smooth is a kind of moving average that is calculated by iteration so that outliers are given little or no weight in the determination of the final curve (Tukey 1970).

The smooth in Figure 2 curves down to the lower right corner of the scatterplot, which seems to suggest that a large number of licensed operators is associated with good SALP scores. However, the smooth is pulled down by a small number of plants.

Figure 3 shows the number of safety system failures on the vertical axis and the number full-time licensed operators in Region 1 on the horizontal axis. The fact that the smooth curves down seems to suggest that a large number of full-time operators is associated with a small number of safety system failures. As was stated earlier, however, a multivariate analysis must be conducted before inferences can be made concerning causal relationships.

OPERATOR OVERTIME

In 1982 the NRC recommended that plants write into their technical specifications a model policy on overtime for safety related personnel. Currently, 77 units have the model policy or a modification of it written into their

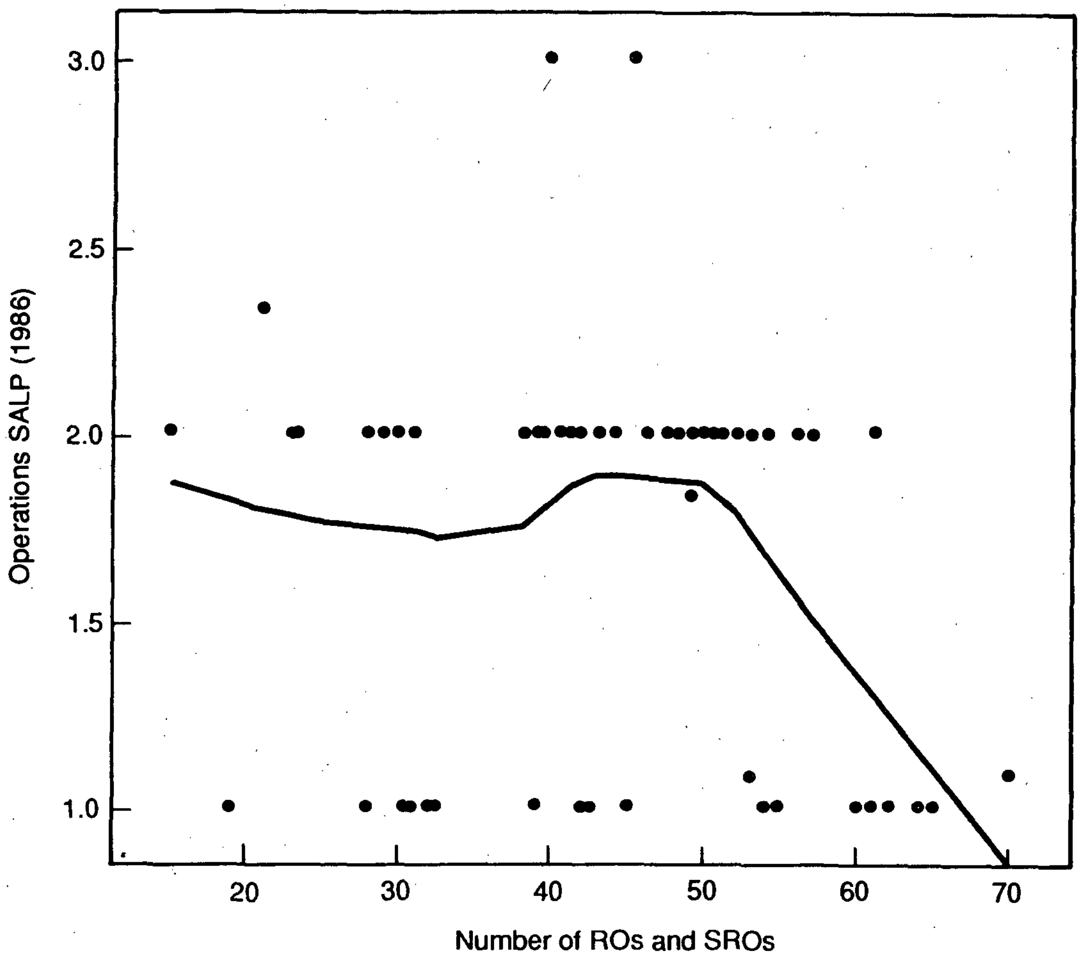

FIGURE 2. Operations Performance Versus Operator Number

technical specifications. Another 33 units have no policy on overtime in their technical specifications, which means that the NRC has no explicit means of overseeing overtime in these units.

The model policy sets "limits" on overtime for safety related personnel for periods of one day, two days, and one week. The plant manager must approve exceeding these "limits". As long as it is approved by the plant manager, the NRC sets no explicit restrictions on the amount of overtime worked above these "limits."

Data on Operator Overtime

Although NRC currently does not systematically collect data on overtime, evidence exists that some operators have worked considerable amounts of overtime. According to a survey by Bauman et al. (1985), 38% of operators

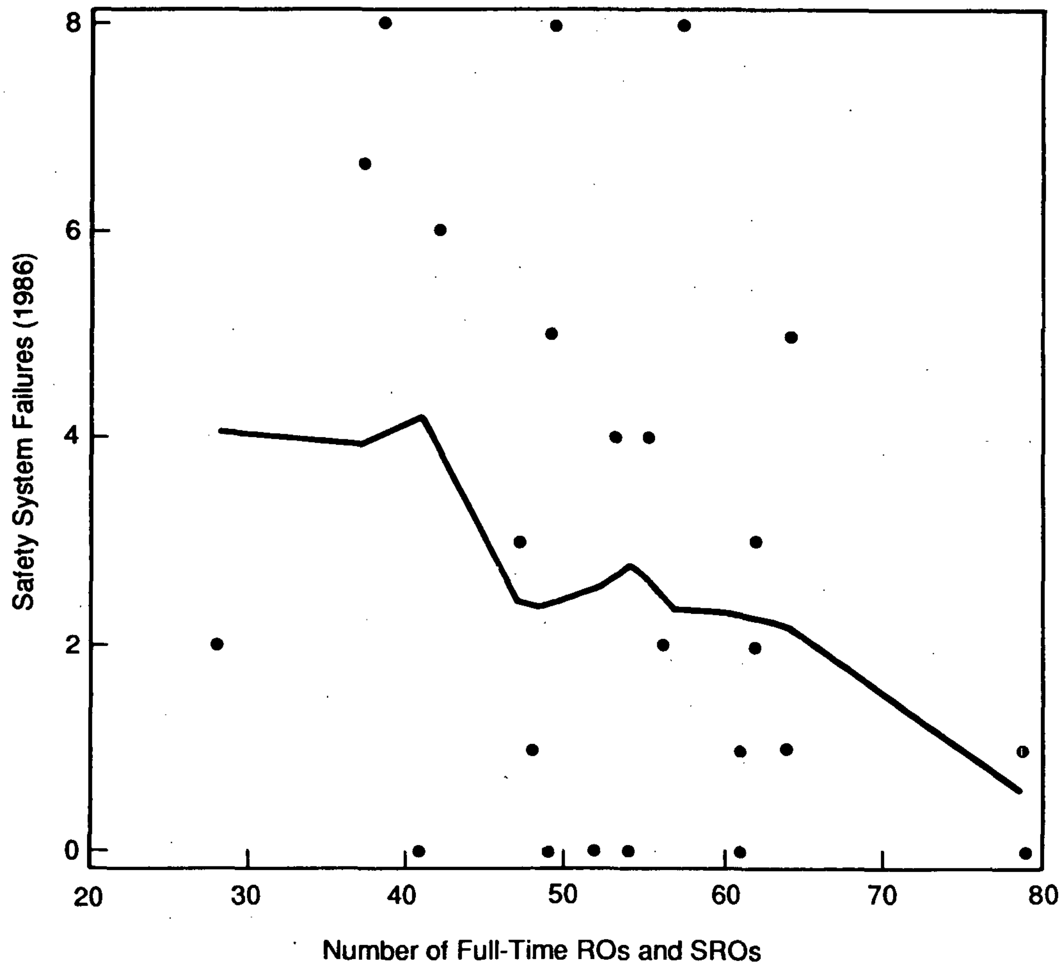

FIGURE 3. Safety System Failures Versus Operator Number

worked 400 or more hours of overtime in one year, and 6% worked 800 or more hours of overtime in one year. Lewis (1985a, p. vii) reports that one crew worked 3,900 hours in one year, which averages nearly 80 hours of work per week. In another plant, one operator worked 97 hours in 7 days (NRC 1985, p. 9). The project staff is preparing to collect survey data on operator overtime.

OPERATOR SHIFT SCHEDULES

The U.S. nuclear power industry uses three types of shift schedule: 1) forward-rotating 8-hour/day shift schedules, 2) backward-rotating 8-hour/day schedules, and 3) 12-hour/day schedules. "Forward rotation" means that the schedule rotates clockwise: from day shift, to afternoon shift, to night shift. "Backward rotation" means that the schedule rotates counter-clockwise: from day shift, to night shift, to afternoon shift. Experts agree that forward rotating shift schedules are generally less fatiguing than backward rotating schedules. Twelve-hour shift schedules are becoming popular; 20 nuclear power plants had 12-hour shifts in 1986.[a]

Table 4 shows the number of plants with each of the three main types of shift schedule for operators (Stello 1987).

Figure 4 shows the SALP scores for operations on the vertical axis, and the three major types of shift schedule for operators on the horizontal axis. A SALP score of "1" indicates the best performance. The distribution of operations SALP scores is almost identical for forward and backward rotating shift schedules. The distribution for 12-hour schedules indicates a higher percentage of both high and low scores. These data seem to suggest that no significant relationship exists between the type of shift schedule and operations SALP scores.

Figure 5 shows the number of scrams while critical in 1986 on the vertical axis, and the three types of shift schedule on the vertical axis.

The box plot in Figure 5 is an exploratory device, a visual means of conveying several attributes of a univariate distribution. The dotted line indicates the median. The bottom and top lines of the box indicate the 25th and 75th percentiles, respectively. The vertical lines extending from the boxes define boundaries that contain most of the data. Data points outside the vertical lines are shown as asterisks; they can be considered outliers. The formula that determines the length of the vertical lines is such that in a normal (gaussian) distribution, only 0.35% of the data points will be shown outside the vertical lines.

Figure 5 shows that the median number of scrams while critical in 1986 for plants with forward-rotating shift schedules was 4. The median for plants with backward rotating schedules was 3. The median for plants with 12-hour shifts was 6.

TABLE 4. Types of Shift Schedules for Operators

Type of Shift Schedule	Number of Plants
Forward-Rotating 8-Hour Shift	63
Backward-Rotating 8-Hour Shift	22
12-Hour Shift	20
Total	105

(a) A brief history of the introduction of the 12-hour shift into the U.S. nuclear industry is recounted in Lewis (1985b, pp. B.1 - B.3). A comprehensive and highly quantified comparison of alertness and performance on 8- and 12-hour shifts at one nuclear facility is reported in Lewis et al. (1986).

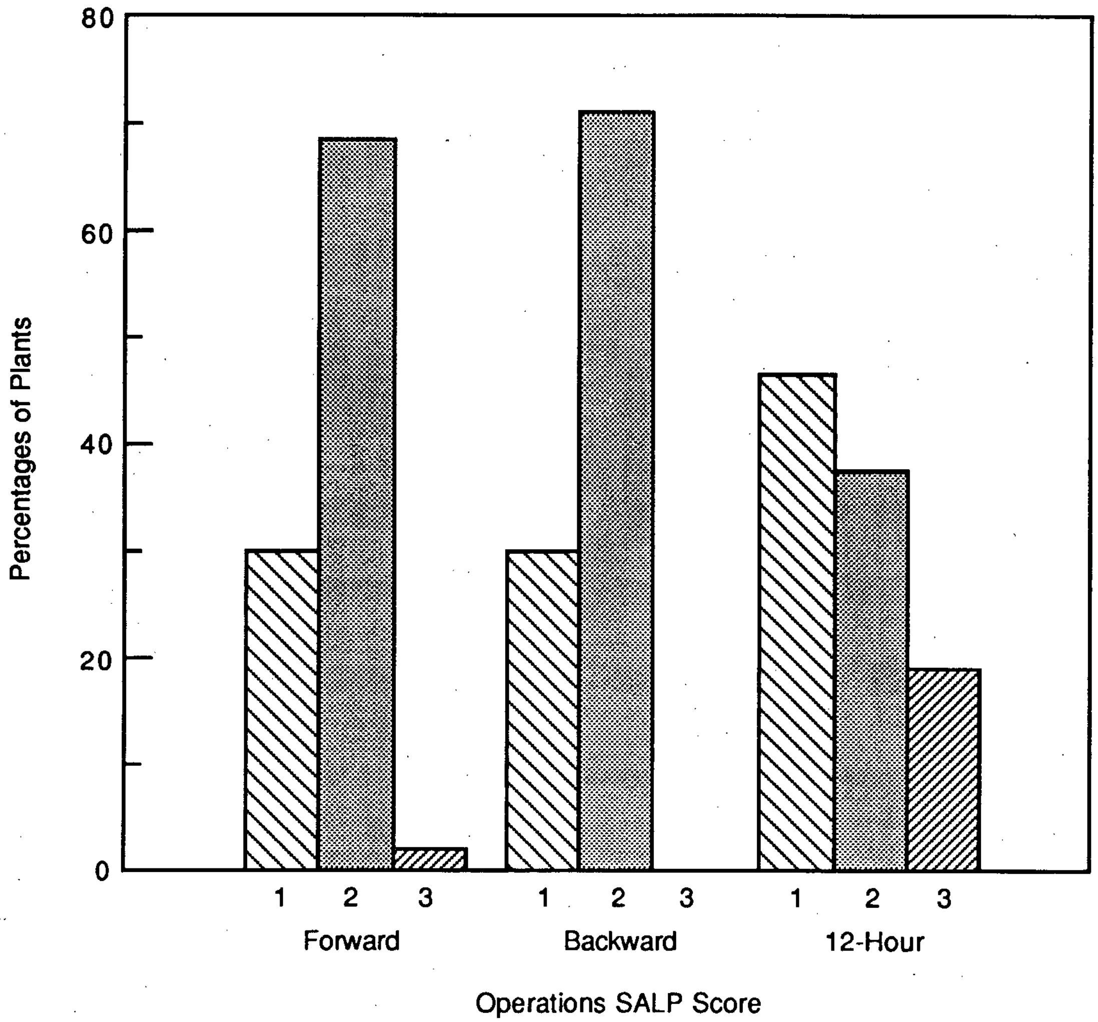

FIGURE 4. Operations Performance Versus Operator Schedule

MAINTENANCE STAFFING LEVELS

The NRC has no regulations on maintenance staffing levels. Analysis of maintenance staffing levels is complicated by the fact the degree to which plants rely on temporary maintenance personnel varies greatly from plant to plant.

Rankin et al. (1986) conducted an extensive survey of maintenance in nuclear power plants, which is the source of the data presented in Tables 5 and 6.

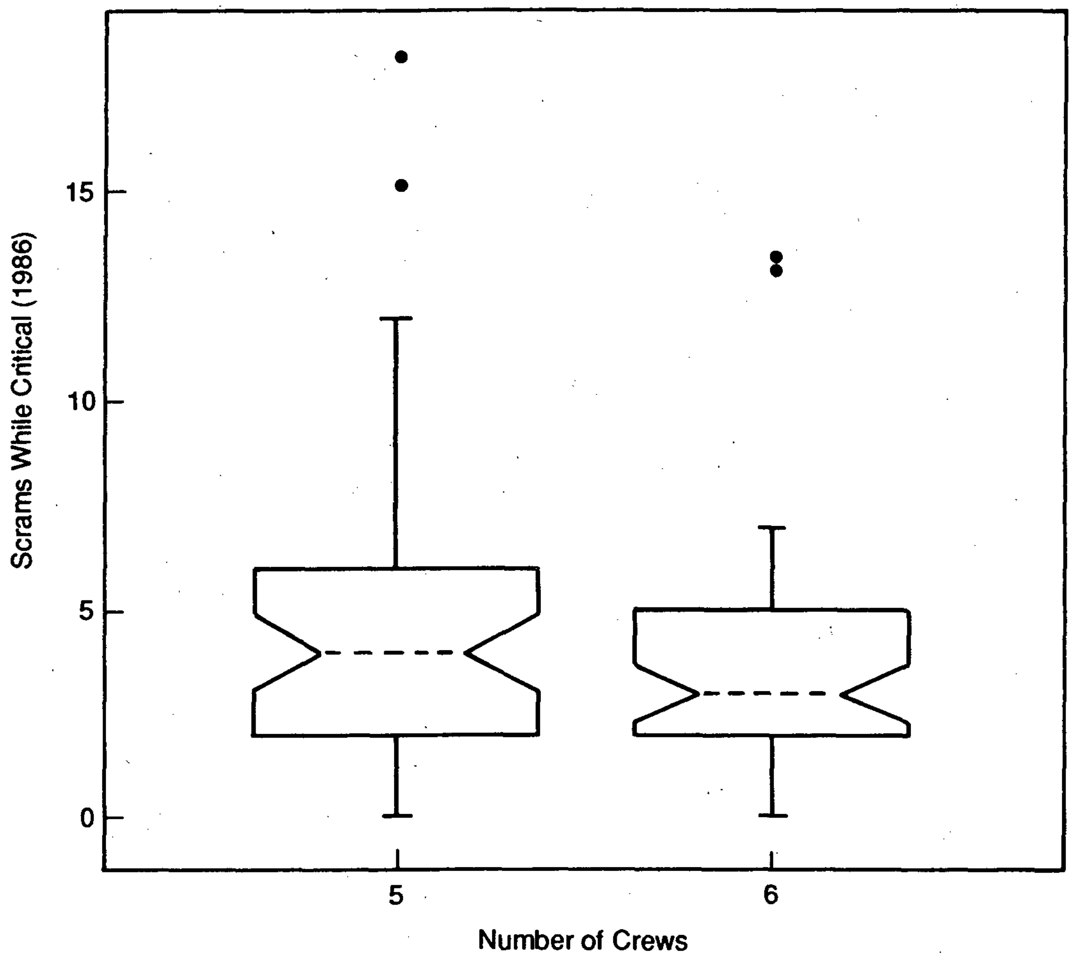

FIGURE 5. Scrams While Critical Versus Operator Number

TABLE 5. Average Number of Maintenance Supervisors and Technicians, at One- and Two-Unit Sites, for Normal Operations and for Outages

	Normal Operations	Outages
1-unit sites	107	164
2-unit sites	141	260

MAINTENANCE OVERTIME

Rankin et al. (1986, p. A-16) collected data on overtime for maintenance craft personnel and maintenance professionals. These data are shown in Table 7 for normal operations and in Table 8 for outages.

TABLE 6. Percentage of Maintenance Workforce on Site That Are Contractor Personnel, for Normal Operations and Outages (Rankin et al. 1986, p. A-15b).

Percentage of Workforce	Percentage of Plants: Normal Operations	Percentage of Plants: Outages
0 - 10%	66%	22%
11 - 20%	9	11
21 - 30%	7	11
31 - 50%	16	26
51 - 75%	2	22
76 - 100%	0	7
Total	100%	100%

TABLE 7. Average Amount of Overtime Hours Worked Per Week by Maintenance Craft Personnel and Professionals, During Normal Operations

Hours of Overtime Per Week	Crafts Personnel	Professionals
0	22%	15%
1 - 4	24	31
5 - 8	22	19
9 - 16	25	15
17 - 20	7	20
Total	100%	100%

TABLE 8. Average Amount of Overtime Hours Worked Per Week by Maintenance Craft Personnel and Professionals, During Outages

Hours of Overtime Per Week	Crafts Personnel	Professionals
0	8%	4%
1 - 8	12	13
9 - 16	31	40
17 - 30	42	34
31 - 40	7	9
Total	100%	100%

MAINTENANCE SHIFT SCHEDULES

Many plants have no maintenance craft personnel on swing shift or night shift during normal operations. These plants, however, are more likely to have maintenance craft personnel on back shifts during outages. Table 9 shows

TABLE 9. Percentage of One-Unit Plants With No Maintenance Craft Personnel of Various Types on Back Shifts During Normal Operations (Rankin et al. 1986, p. A-14b)

	None on Swing Shift	None on Night Shift
Mechanical Maintenance Personnel	48%	57%
Electrical Maintenance	50%	60%
I&C Technicians	43%	47%
Maintenance Warehouse Staff	58%	61%

the percentage of one-unit plants that have no maintenance personnel of specific types on swing and night shifts.

The data collected to date provide evidence that some plants have been understaffed and that some operators have worked considerable amounts of overtime. However, these data are incomplete or cannot be ascribed to specific plants. The project staff are preparing to collect survey data on staffing levels and overtime that can be used to estimate the effect of these issues on performance and safety.

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USI A-45, SHUTDOWN DECAY HEAT REMOVAL

Roy Woods, Senior Task Manager
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Summary

This paper describes the resolution of USI A-45 through planned plant-specific analyses under the Individual Plant Evaluation (IPE) program. The technical basis for this resolution includes important insights gained from decay heat removal risk assessments for six operating reactors. These studies, together with the operating history of decay heat removal (DHR) failures, have led to the conclusions that:

1) risk due to loss of DHR could be unduly high for some plants;

2) DHR failure vulnerabilities, and the optimum corrective actions for those vulnerabilities, are strongly plant specific;

3) detailed plant specific analyses under the IPE program, including anticipated future extension of the IPE program which will require consideration of externally initiated events, will be needed to resolve this issue.

Discussion:

The Commission designated "Shutdown Decay Heat Removal Requirements" as an Unresolved Safety Issue (USI A-45) in March 1981. The USI A-45 program was initiated to evaluate the safety adequacy of the DHR function in the currently operating light water reactor power plants and to assess the value and impact (i.e., the benefit and cost) of alternative measures to improve the overall reliability of the DHR function.

The USI A-45 program employed probabilistic risk assessments and deterministic evaluations of those DHR systems and support systems required to achieve hot-shutdown and cold-shutdown conditions in both pressurized and boiling water reactors. Systems analysis techniques were used to assess the vulnerability of DHR systems to various internal and external events. The analyses were limited to transients, small-break loss-of-coolant accidents, and special emergency challenges such as fires, floods, earthquakes, and sabotage. Cost-benefit analysis techniques were used to assess the net safety benefit and cost of alternative measures to improve the overall reliability of the DHR function.

Six plants were analyzed after an initial selection process which considered vendor, product line, other issues in which each particular plant might be involved, operational status, and utility willingness to participate.

The internal events analyses for these plants proceeded along the well-documented lines used for other PRAs. Additional emphasis was put on supporting systems including service water, component cooling water, and electrical systems. The performance of containment systems was examined for each of the dominant core melt sequences, and the probability of containment failure for each containment failure mode was estimated.

The analyses for fire, earthquake, wind, and external and internal flood proceeded by identifying the significant hazards and their frequency of occurrence. An estimate of the response of the plant to such hazards was then made, utilizing the results of an onsite inspection of the plant and equipment. The appropriate event and fault trees were adjusted to account for common-cause failures, the effects of fires and floods were quantified based on an estimate of the probability of mitigative action, and estimates of the potential contribution of these events to core melt probability were derived.

The above results were combined to calculate a total core damage frequency caused by decay heat removal failure for the six plants studied (see References 1 and 2 for further details). This frequency was found to be quite plant-specific in nature (i.e., there was considerable variation in the results among the six plants studied).

In terms of expected core damage frequency caused by decay heat removal failure per reactor year, the range for the six plants was 7E-05 to 4E-04 with an average value of 2E-04 if credit is allowed for feed and bleed operation on the PWRs and containment venting on the BWRs. If one arbitrarily makes the unrealistic (but limiting) assumption that no such credit should be allowed, then the range becomes instead 1E-04 to 1E-03 (average 4E-04). Neither the above ranges nor the averages were found to be significantly changed when several other existing, reliable PRA results were also included.

On the other hand, the result of one recent industry-sponsored re-analysis (for the Point Beach plant, one of the six plants the staff analysed) was outside of those ranges. The core damage frequency caused by decay heat removal failure re-calculated in this study was 1E-05 per reactor year, a factor of seven below the bottom of the range quoted above, and a factor of thirty lower than the 3E-04 per reactor year that the staff obtained for Point Beach. Reasons for the difference include different assumptions regarding the frequency of certain initiating events and the probability of the operator's taking appropriate mitigative actions (such as initiating the feed and bleed cooling option). The differences are detailed in Appendix D of Reference 1, where it is concluded that the "true" best estimate for Point Beach probably lies above 1E-05 and within the range quoted above, but below 3E-04.

Utilizing any of the above results, the six plants meet the health effects quantitative objectives in the Commission's Safety Goal (i.e., 0.1% of the expected accident or cancer fatality risks from causes not related to nuclear plants). Guidance for an acceptable core damage frequency has not been explicitly provided. However, in order to provide assurance that: (1) core damage due to a decay heat removal failure related event will not occur in the lifetime of the present population of plants; (2) consistency is maintained with the 1E-05 per reactor year contribution to core damage frequency from station blackout expected after resolution of the station blackout USI (A-44, NUREG-1109); and (3) the frequency of a large release will be less than the

Commission's safety goal guidance of 1E-06 per reactor year, the staff selected a goal that core damage due to failure of decay heat removal function should be less than 1E-05 per reactor year. This staff-selected goal is intended only for current application to the resolution of generic issue USI A-45.

The results quoted above indicate that the decay heat removal related frequency of core damage at certain plants may be considerably above this goal. To address the question of whether corrective actions could be cost effective, six possible alternatives addressing potential decay heat removal vulnerabilities were identified and then evaluated. The approximate costs and value/impact ratios of the alternatives were estimated, and are summarized in Tables 1 and 2 with full details in References 1 and 2.

Alternative 1 is to take no action for the resolution of USI A-45, i.e., the status quo described above would be maintained. This alternative was not selected because it appears likely that certain plants have a core damage frequency above the staff selected goal.

Alternative 2 is to have each licensee perform a risk assessment for its plants. This assessment would be done in conjunction with the Individual Plant Examination program. Available options for acceptable risk assessments include performing a Level-1 PRA (enhanced) or performing an analysis using the IDCOR IPEM. This is the proposed alternative (as discussed throughout the remainder of this paper), since the plant risks, and the effects of individual corrective actions, are highly plant specific.

Alternative 3 is to perform a certain specified group of equipment and procedure modifications for each plant (as described for Alternative 3 in References 1 and 2). These modifications generally correspond to several of the current generic unresolved safety issues. This alternative would require the same group of corrective actions for each plant, and is not recommended since many of the vulnerabilities are not the same for each plant.

Alternative 4 is to take whatever actions are necessary at each individual plant to provide and/or enhance the "feed and bleed" heat removal method for PWRs, and the "containment venting" method for BWRs. These diverse heat removal methods are also being considered by other NRC programs. The severe accident integration program (SECY-88-147) is considering a recommendation to enhance BWR Mark 1 containment performance using containment venting, and Generic Issue 84 (CE plants without PORVs) is considering the need to backfit PORVs to provide a PWR feed and bleed heat removal method which would be independent of the steam generators. (A proposal to subsume GI-84 within the IPE program is currently under review).

Thus, Alternative 4 is not recommended as the generic resolution of USI A-45 because: (1) the calculated risk reduction varies from plant to plant depending on the design, for example PWRs with highly reliable auxiliary feedwater systems accrue little benefit; and (2) the issue of operator reliability for initiation of the systems when needed has not been resolved, and it greatly affects the quantitative safety improvement realized by adopting these methods.

Alternatives 5 and 6 are to install new, separate and dedicated decay heat removal systems capable of cooling the plants to a hot (Alternative 5) or cold (Alternative 6) shutdown condition. These alternatives show considerable potential safety benefit, particularly when the advantages to other safety issues, such as the possibility of insider sabotage, are considered. They may have a favorable value/impact ratio only if conventional value-impact methods are modified, for example, to take into account the value of avoiding a nuclear moratorium following a severe accident ("method #3" described below). However, these alternatives cannot be recommended at present due to their high cost (on the order of $100,000,000 per plant) and unfavorable value/impact ratios. Value/impact ratios must be taken into account in cases such as this where the alternatives being considered may be considered as providing additional protection over that necessary for adequate protection.

It was found that the value (safety benefit) and the impact (cost) of Alternatives 2 through 6 varied significantly from plant to plant. To facilitate making a single recommendation that would be applicable to all plants, a set of generic results were derived to represent the overall family of operating U. S. plants. In addition, the value/impact analyses were performed using three separate methods:

1. The value term was limited to the reduction in dose to the population within 50 miles. The impact term was defined as the total cost of implementation with no reduction for the anticipated economic advantages in the form of averted costs.

2. The value term was defined as in method #1 but with reduction in dose to plant personnel taken into account; the net impact used was reduced by the averted onsite costs.

3. The value term was based on the reduction in population dose as in method #2, but was supplemented by the monetary value of other averted cost savings that would affect the public interest, such as consideration of a nuclear moratorium, insider sabotage, other outstanding generic issues, environmental qualification, unquantifiable internal initiating events, and residual risk from special emergency events (assuming $1,000 per averted person-rem). The impact term was defined as in method #2.

Using the more conventional approach of method #1, the value/impact ratios for certain alternatives for the "generic" (i.e., the "average") plant did achieve a value/impact ratio of about $1,000 per person-rem averted. Specifically, Alternatives 3 and 4 achieved this ratio for the BWR, and were close to this ratio for the PWR. Alternative 2 was close to achieving this ratio for the BWR. However, none of those alternatives (i.e., Alternatives 2, 3, or 4) achieved a core damage frequency near or below the staff's goal of 1E-05 per reactor year DHR related core damage frequency. Therefore, the results show that, using method #1, none of the alternatives will simultaneously achieve a value/impact ratio of $1000 per person rem and also reach the staff's selected core damage frequency goal.

Using method #2, value-impact ratios near $1000 per person rem are achieved for most alternatives, except for Alternatives 5 and 6, which are the only two alternatives that reach the staff targeted goal for core damage frequency. (These generic value/impact results are summarized in Tables 1 and 2, and are given in detail in References 1 and 2.)

The results of method #3 show the core damage frequency goal can be achieved by Alternatives 5 or 6 at a value/impact ratio near $1000 per person rem. However, use of method #3 goes beyond value/impact analysis methods previously used for Unresolved Safety Issues. The very high cost of Alternatives 5 or 6 cannot be justified on the basis of conventional value/impact methods (i.e., methods #1 or #2), and they are therefore not recommended.

Summary of Technical Findings:

1) The core damage frequency caused by decay heat removal failure for the six plants studied spanned a broad range with the average value either 2E-04 or 4E-04 per reactor year, depending on whether or not credit is allowed for certain backup core cooling methods. This result includes consideration of all known significant decay heat removal failure related sequences, including those related to station blackout.

Although a safety goal in terms of core damage frequency has not been formalized, the Commission's safety goal policy states that large releases should remain below 1E-06 per reactor year. Assuming one severe release per hundred core damage events, the resulting core damage frequency goal including all causes of core damage is 1E-04 core damage events per reactor year.* Realistic application of this goal can be achieved by requiring that the contribution from any identified broad class of events be less than ten percent of 1E-04 (i.e., less than 1E-05). This was done for USI A-44, "Station Blackout," where a core damage frequency goal of 1E-05 for events involving station blackout was implied (NUREG-1109, January, 1986).

Decay heat removal failure related events constitute another broad class of core damage events, and so a goal of 1E-05 was adopted by the staff as also being appropriate for that class of events. It is likely that the decay heat removal related core damage frequency (which averaged 2E-04 to 4E-04 for the six case studies) may be considerably above this goal at certain plants, which results in the conclusions and recommendations made below.

*More recently, a core damage frequency goal of 5E-05 per reactor year has been proposed under the safety goal implementation program. This is a factor of two (2) lower than the 1E-04 used herein, but is within the uncertainty inherent in calculations and assumptions made when assessing compliance with either goal, and its adoption in lieu of a 1E-04 goal would not affect the conclusions stated in this paper.

[Note that application of one 1E-05 goal for the blackout events and a separate 1E-05 goal for the decay heat removal failure events will result in a combined risk from both types of events above 1E-05 but less than 2E-05 since there is some overlap (i.e., some of the decay heat removal failures are caused by station blackout). This small amount of "double counting" contributes to the margin that will be available for later inclusion of other quantitatively determined core damage frequencies (such as ATWS) without exceeding the overall 1E-04 core damage frequency goal.]

2) The value/impact ratio of the studied alternatives varied significantly from plant to plant. Even using only offsite benefits in the value/impact methodology, some corrective actions do achieve a value/impact ratio that would justify their implementation for certain plants although none of the six alternatives analysed will simultaneously achieve such a value/impact ratio and also reach the staff's core damage frequency goal (i.e, no cost effective corrective actions were identified which would make all plants reach the targeted goal for core damage frequency).

3) Since all of the significant USI A-45 results have been found to be highly plant specific, it is not appropriate to propose a single generic action to be applied uniformly to all plants.

4) For any specific plant, to determine the core damage frequency caused by decay heat removal failure and the value and impact of proposed corrective design changes, detailed analyses of that specific plant are necessary.

Summary of Resolution

The Commission is currently deliberating on proposed staff actions that would implement the Severe Accident Policy (50 FR 32138), and is expected to approve issuance of a generic letter to require all plants currently operating or under construction to undergo a systematic examination termed the Individual Plant Examination (IPE) to identify any plant-specific vulnerabilities to severe accidents. The IPE analysis, which is similar to that needed for item 4 above, is intended to examine and understand the plant emergency procedures, design, operations, maintenance, and surveillance to identify vulnerabilities. The analysis will examine both the decay heat removal systems and those systems used for other functions. It is anticipated that a future extension of the IPE program will require examination of externally initiated events, some of which significantly contribute to DHR failure related core damage frequency. Therefore, the staff decided to subsume A-45 into the IPE program and its anticipated extension as the most effective way of achieving resolution of A-45.

Thus, Alternative #2 was adopted and USI A-45 was subsumed as an integral part of the IPE program. That is, USI A-45 is considered resolved generically. Plant-specific implementation (including the effectiveness of any corrective actions proposed by the licensee and/or required by the Commission) will also be subsumed within the planned IPE activities.

Table 1

Overview of Alternatives on Generic Basis - PWR

(Cost per P-REM - Average Site)

	Extent of Improvement				Cost of Improvement		Cost per Person-REM	
	P(CM)		Popn. Dose				Method 1, Offsite W/Gross Impact ($ per P-REM)	Method 2, Off + Onsite W/Net Impact ($ per P-REM)
Alternative	Initial Value	% Reduction	Initial Value	% Reduction	Gross Impact $	Net Impact $		
2	2.2E-4	75%	3.7E3	62%	9.4E6	5E6	4100	2180
3	2.2E-4	10%	3.7E3	11%	0.56E6	-0.52E6	1370	NO COST
4	4.8E-4	61%	8.3E3	61%	7E6	-6.2E6	1390	NO COST
5	4.8E-4	94%	8.3E3	94%	66E6	46E6	8400	5830
6	5.7E-4	95%	9.9E3	94%	94E6	70E6	10,140	7520

Note:
Alt. 2 & 3 Assume F&B
Alt. 4, 5 & 6 Assume No F&B
Alt. 6 Assumes + 20% for Cold Shutdown

TABLE 2

OVERVIEW OF ALTERNATIVES ON GENERIC BASIS - BWR

(COST PER P-REM - AVERAGE SITE)

	EXTENT OF IMPROVEMENT				COST OF IMPROVEMENT		COST PER PERSON-REM	
	P(CM)		POPN. DOSE				METHOD 1, OFFSITE W/GROSS IMPACT ($ PER P-REM)	METHOD 2, OFF + ONSITE W/NET IMPACT ($ PER P-REM)
ALTERNATIVE	INITIAL VALUE	% REDUCTION	INITIAL VALUE	% REDUCTION	GROSS IMPACT $	NET IMPACT $		
2	2.2E-4	54%	2.3E4	45%	13E6	9E6	1260	870
3	2.2E-4	4%	2.3E4	4%	0.28E6	-0.12E6	300	NO COST
4	2.67E-4	30%	2.7E4	31%	1.1E6	2.7E6	120	NO COST
5	2.67E-4	84%	2.7E4	84%	80E6	69E6	3460	3020
6	3.56E-4	84%	3.6E4	84%	84E6	73E6	2690	2260

NOTE:
ALT. 2 & 3 ASSUME CONT. VENT
ALT. 4, 5 & 6 ASSUME NO CONT. VENT
ALT. 6 ASSUMES + 20% FOR COLD SHUTDOWN

References

1. "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements," NUREG-1289, April, 1988 (DRAFT).

2. "Shutdown Decay Heat Removal Analysis Plant Case Studies and Special Issues Summary Report," NUREG-1292, October, 1987 (DRAFT). Note: this document will be issued in final form as NUREG/CR-5230. The title and date of manuscript completion will remain unchanged.

GENERIC IMPLICATIONS OF THE CHERNOBYL ACCIDENT

George Sege
U.S. Nuclear Regulatory Commission

ABSTRACT

The U.S. Nuclear Regulatory Commission staff's assessment of the generic implications of the Chernobyl accident led to the conclusion that no immediate changes in the NRC's regulations regarding design or operation of U.S. commercial reactors are needed. However, further consideration of certain issues was recommended. This paper discusses those issues and the studies being addressed to them. Although 24 tasks relating to LRW issues are identified in the Chernobyl follow-up research program, only four are new initiatives originating from Chernobyl implications. The remainder are limited modifications of ongoing programs designed to ensure that those programs duly reflect any lessons that may be drawn from the Chernobyl experience. The four new study tasks discussed include a study of reactivity transients, to reconfirm or bring into question the adequacy of potential reactivity accident sequences hitherto selected as a basis for design approvals; analysis of risk at low power and shutdown; a study of procedure violations; and a review of current NRC testing requirements for balance of benefits and risks. Also discussed, briefly, are adjustments to ongoing studies in the areas of operational controls, design, containment, emergency planning, and severe accident phenomena.

INTRODUCTION

The NRC staff's assessment, "Implications of the Chernobyl Accident for the Safety Regulation of U.S. Commercial Nuclear Power Plants" (NUREG-1251), led to the conclusion that no immediate changes in the NRC's regulations regarding design or operation of U.S. commercial reactors are needed. However, further consideration of certain issues was recommended. Most of these issues were found to be largely already under consideration as a part of ongoing NRC work.

This paper briefly describes and discusses the work recommended in NUREG-1251 and is limited to that work. As noted in NUREG-1251, the Chernobyl experience will continue to be taken into account in various areas of reactor safety. But the Chernobyl follow-up program is limited to work where the nexus of issues to Chernobyl is direct, clear, and substantial, but with reasonable extrapolation to account for the large differences in specific design and operational features, i.e., to items recommended in NUREG-1251. Other work that may relate generally to severe accidents will be pursued by the NRC (or considered for pursuit) in accordance with established procedures, outside the Chernobyl follow-up program.

The issues on which further work was judged to be in order were identified in the August 1987 comment issue of NUREG-1251. The public comments received led to only limited modifications in the assessment. The program described here has been updated with respect to those modifications. The work is in progress.

OVERVIEW

Research on issues stemming from the study of Chernobyl consists predominantly of limited influences on ongoing work and includes only four tasks that are new work items initiated or planned as a result of the Chernobyl implications assessment. Twenty-two of the tasks consist of modification of existing relevant programs. These tasks may involve consultation to reflect Chernobyl lessons, added emphasis or modified scope for some aspect of the program, or addition of specific work items to an ongoing program. The new tasks are Task 1.1B, Procedure Violations; Task 1.2B, NRC Testing Requirements; Task 1.4C, Low Power and Shutdown; and Task 2.1A, Reactivity Transients. All the new tasks and all but two of the others pertain to light water reactor issues; two tasks relate to high-temperature graphite reactors. A statistical task breakdown is presented in Table 1; Table 2 lists the specific tasks. (The numerical part of the task number keys the task to the section of NUREG-1251 in which the work is recommended.)

The work is in progress. All tasks are expected to be completed by mid-1989 and an overall report on the work is planned to be issued shortly thereafter. However, some of the tasks involve issues on which work will continue beyond the mid-1989 close-out of the Chernobyl follow-up program (e.g., Task 1.4C, Low Power and Shutdown, and Task 4.4A, Decontamination). Such further work, even when its content is clearly influenced by Chernobyl lessons, will be pursued in the normal course of NRC business. It is not intended that the Chernobyl follow-up program would be extended as a discrete program to encompass such subsequent activities.

TABLE 1: TASK BREAKDOWN

	Number of Tasks		
Issue Area	New	Mod. of Existing	Total
1. Operational Controls	3	7	10
2. Design	1	5	6
3. Containment		2	2
4. Emergency Planning		3	3
5. Severe Accident Phenomena		3	3
6. Graphite Reactors		2	2
TOTAL	4	22	26

TABLE 2: TASK LIST

Task No.	Task	Remarks
Operational Controls		
1.1A	Symptom-Based Emergency Operating Procedures	
1.1B	Procedure Violations	New. Extent, nature.
1.2A	Test and Change Reviews	Criteria, guidelines.
1.2B	NRC Testing Requirements	New. Benefits vs. risks.
1.3A	Regulatory Guide 1.47	Revision. Safety systems bypass.
1.4A	Engineered Safety Feature Availability	Adequacy of requirements for all operating conditions.
1.4B	Tech. Spec. Bases	Consistency with safety analyses.
1.4C	Low Power & Shutdown	New. Risk analysis.
1.6A	Assessment of NRC Requirements on Management	
1.7A	Accident Management	
Design		
2.1A	Reactivity Transients	New. Reconfirmation of safety.
2.3A	Control Room Habitability	At multi-unit sites.
2.3B	Contamination Outside Control Room	At multi-unit sites.
2.3C	Smoke Control	At multi-unit sites.
2.3D	Shared Shutdown Systems	At multi-unit sites. For future plants.
2.4A	Firefighting with Radiation Present	Risk assessment.

TABLE 2: TASK LIST (cont'd)

Task No.	Task	Remarks
Containment		
3.1A	Containment Performance	During severe accidents.
3.2A	Filtered Venting	International information exchanges.
Emergency Planning		
4.3A	Ingestion-Pathway Measures	With Federal Emergency Management Agency.
4.4A	Decontamination	Post-accident measures and their effectiveness. With FEMA.
4.4B	Relocation	Post-accident measures. With FEMA.
Severe Accident Phenomena		
5.1A	Mechanical Dispersal in Fission-Product Release	
5.1B	Stripping in Fission-Product Release	Chemical or thermal stripping of UO_2.
5.2A	Steam Explosion	Fuel-coolant interactions in reactivity-initiated accidents. This task will not be undertaken unless need for it is indicated by Task 2.1A, Reactivity Transients.
Graphite Reactors		
6A	Fort St. Vrain PRA	Licensee PRA under IPE program is under discussion.
6B	Structural Graphite Experiments	Priority under consideration.

NEW STUDIES

Procedure Violations (Task 1.1B)

At Chernobyl, serious procedure violations were a key factor in the causation of the accident. In the United States, extensive measures are taken by the NRC and the industry to keep violations to a minimum. These measures include Technical Specifications containing operability requirements for safety equipment, kept prominent in operators' and management's minds. Violations of procedures nevertheless occur. However, systematic information concerning the frequency and safety impact of the violations of normal, abnormal, and emergency operating procedures, or test and maintenance procedures is lacking. Accordingly, a study has been initiated to determine the frequency, nature, causes, and impact of procedure violations in nuclear power plants. The information to be developed could provide a basis for consideration of measures that might further increase assurance that violations of procedures that could be instrumental in causing an accident or emergency situation or compromising safety margins will not occur.

The NRC has enlisted the services of the Battelle Pacific Northwest Laboratories, with collaboration of Battelle's Human Affairs Research Centers, in the performance of this research.

NRC Testing Requirements (Task 1.2B)

There is a potential for human error when conducting tests to assess equipment capabilities. This potential represents a risk to plant safety which can vary in severity depending both on the nature of the test and the circumstances associated with the test. Tradeoffs between the risks of not testing or of testing at a lesser frequency and the risks associated with such testing have not always been assessed. Accordingly, current NRC testing requirements are being reviewed for balance of benefits and risks, with particular attention to any tests whose conduct may present a sufficient impact on plant safety risk to suggest modification of the test, a reduced test frequency, or elimination of the test. The Chernobyl accident occurred when the plant was used for a test.

Though initially conceived as a new program, this task is now being folded into a broader review of NRC testing requirements under the Technical Specification Improvement Program.

Risks at Low Power and Shutdown (Task 1.4C)

Regulations for commercial nuclear power plants in the United States require that potential accidents that could occur during all conditions of operation (full, low, and zero power) be considered and provided for in the plant design. Such provisions are considered in safety analyses required in support of licensing. Often, analyses assuming full-power operation are found to be limiting cases -- bounding accident risks at low-power operation or when the reactor is shut down. The Chernobyl accident suggests that accident sequences beginning at low power and under shutdown conditions should be reviewed, particularly for situations in which not all engineered safety features are considered necessary to be available.

In this task, the probabilistic risk of a plant in a shutdown or low-power mode will be investigated. An initial, scoping study will be performed on the Surry nuclear power plant. The study is being initiated at Sandia National Laboratories, as part of the Accident Sequence Evaluation Program.

Reactivity Transients (Task 2.1A)

Positive void reactivity coefficients, which are a characteristic of the RBMK graphite-moderated water-cooled reactors, played a central role in determining the severity of the Chernobyl accident. Commercial reactors in the United States are designed very differently from the RBMK reactor at Chernobyl, and have generally a negative void reactivity coefficient. This provides assurance that the kind of superprompt critical excursion that took place at Chernobyl will not occur. However, the NRC has initiated a study to reconfirm that vulnerabilities and risks from possible accident sequences have been adequately factored into safety analysis reports on which design approvals are based, and to identify any potential reactivity transients for further regulatory attention if probability and consequences warrant.

This work is being done at the Brookhaven National Laboratory. It includes probabilistic analyses to estimate the frequency of selected multiple-failure transients as well as deterministic analyses to assess potential consequences. The selection of transients for study, including establishment of criteria for that selection, is part of this task.

Particular attention is being focused on sequences that might involve a positive void coefficient or moderator temperature coefficient; that arise in connection with deliberate bypassing or disabling of any safety feature; and whose causes include human error (commission, omission, or misjudgment).

Initially identified events of interest are as follows:

For BWRs:
- Multiple rod drop
- Control rod ejection
- Overpressurization with limited relief
- Boron dilution during anticipated transient without scram (ATWS)
- ATWS without recirculation pump trip
- Multiple rod bank withdrawal
- Reactivity events with more than one rod stuck out

For PWRs:
- Multiple rod bank withdrawal ATWS
- Multiple rod ejection (low power)
- Injection of cold, unborated emergency cooling water
- Injection of cold, unborated water due to steam generator tube rupture.
- Unlimited boron dilution
- Rod withdrawal, heatup, or depressurization from low temperature with positive moderator temperature coefficient

ATWS with less negative moderator temperature coefficient
Reactivity events with more than one rod stuck out

MODIFICATION OF ONGOING PROGRAMS

Operational Controls

In general, regulatory provisions at nuclear plants in the United States, if properly implemented, are adequate with respect to administrative controls to ensure that reactor operations are conducted within a safe range of operating conditions. These controls address procedural adequacy and compliance, approval of tests and other unusual operations, bypassing of safety systems, availability of engineered safety features, operating staff attitudes toward safety, management systems, and accident management. However, the benefits of certain additional provisions are being examined. The work in this area includes the studies of procedure violations, NRC testing requirements, and risks at low power and shutdown already described. In addition, ongoing programs are being modified to recognize the lessons of Chernobyl in the following areas:

° Symptom-Based Emergency Operating Procedures (EOPs) (Task 1.1A).

  The NRC staff has undertaken an accelerated inspection program of the EOPs, which is aimed at evaluating their technical correctness, ability to be physically carried out, and ability to be correctly carried out. Possible regulatory action to upgrade programs or possible further study of any inconclusive results will be considered in the light of the results of this inspection program.

° Test and Change Reviews (Task 1.2A).

  Planned tests and experiments not described in licensees' safety analysis reports and changes to the facility and procedures described in those reports are required to be evaluated beforehand by licensees in accordance with NRC regulation 10 CFR 50.59, to assure their safety and that NRC be afforded the opportunity for their review where appropriate. Work is underway to improve guidance and criteria for performing reviews of tests, changes, and experiments, to strengthen these reviews.

  The improved guidance is being developed by a joint NUMARC/NSAC Working Group. The NRC is expected to endorse the guidance document, supplementing it with any additional measures needed. The industry and NRC will use the guidance in their reviews of tests, experiments, and changes required by 10 CFR 50.59.

° Regulatory Guide 1.47 (Task 1.3A).

  Safety system bypass was a key part of the cause of the Chernobyl accident. The lessons of Chernobyl will be reflected in ongoing work to revise and improve Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." The work includes evaluation of the implications of bypassing safety systems and recommendation of improved procedures and methods to prevent inadvertent bypassing of safety functions during test or maintenance. Completion of this task will also resolve Generic Issue I.D.3, "Safety System Status Monitoring."

° Engineered Safety Feature Availability (Task 1.4A) and Technical Specifications Bases (Task 1.4B).

  Current Technical Specifications Bases (included in the Technical Specification statements) do not always provide a clear and comprehensive discussion linking specific requirements to the safety analysis assumptions from which they are derived. This can result in operators not being as aware as possible of the safety significance of certain types of Technical Specifications violations. It can also result in changes being proposed to Technical Specifications without adequate consideration of all the relevant safety issues. A related issue is that, in some older Technical Specifications, mode requirements for operability of Engineered Safety Features may not be specified for other than the power operating mode. The Technical Specifications Improvement Program addresses these issues, seeking potential worthwhile improvements.

° Assessment of NRC Requirements on Management (Task 1.6A).

  Management oversight at all levels must be effective to ensure that tests, maintenance, and operations are conducted safely and that NRC requirements are enforced. The NRC is developing improved methods of monitoring licensee management performance to give early warning of management problems and to initiate enforcement mechanisms. Current NRC research efforts seek enhanced assurance of particular management attention to matters important to safety and avoidance of excessive burdens that could divert that attention.

° Accident Management (Task 1.7A).

  Programs for accident management, including training and the development of procedures for coping with severe core damage and for the effective management of the containment, are part of the implementation of the Commission's Severe Accident Policy.

Design

Among the studies of design issues raised by Chernobyl, the reactivity transients work already described occupies a central place. But the Chernobyl lessons are also being recognized in ongoing efforts that relate to multi-unit sites and firefighting, as discussed below.

° Multi-Unit Sites (Tasks 2.3A, B, C, and D).

  The radioactive gas and smoke released during the accident at Chernobyl Unit 4 spread to the other three operating units at the site. The airborne radioactive material was transported to the other units through a shared ventilation system as well as by way of general atmospheric dispersion paths. This raises the question of how accidents at one unit of a multiple-unit site affect the remaining units, and additional questions of how these effects may be compounded when structures, systems, and components are shared between units.

The adequacy of protection of control rooms in the event of an accident at one of the units (Task 2.3A) is included in studies being conducted in furtherance of resolution of Generic Issue 83, "Control Room Habitability." The study utilizes recent research on radionuclide release. Work is underway at Argonne National Laboratory. A related task (2.3B) addresses contamination outside the control room. Its purpose is to identify all plant areas to which human access would be necessary to manage an accident at an affected unit or to maintain other units at a multiple-unit site, to assess the dose consequences to personnel performing needed tasks within those areas, and to identify any worthwhile potential measures for further reducing consequences that may represent a significant risk. The assessment of the risk significance of smoke propagation from one unit to another unit on the site, in case of fire, is the subject of yet another task in this group (2.3C). A final task (2.3D) is aimed at determining restrictions related to sharing of systems required for safe shutdown among units at a multi-unit site. This last task is applicable to future plants.

° Firefighting with Radiation Present (Task 2.4A).

This task addresses the questions of whether there is a significant risk that radiation released during a fire or from the initiating event could limit firefighting capability and what additional measures, if any, such risk may necessitate. The work is being completed at Sandia National Laboratory, as part of an ongoing fire risk scoping study.

Containment

The Chernobyl accident, with its absence of effective containment, has focused attention on the strengths and performance limits of the substantial containments for U.S. light-water reactors. It has led to added recognition of the significance of ongoing work on the issue of whether U.S. containments that were built using criteria based on design-basis accidents have adequate margins available to prevent the release of large quantities of fission products during severe accidents. Challenges include phenomena such as increased pressures from an uncontrolled hydrogen combustion or release of large quantities of noncondensible gases from core-concrete interactions. Venting the containment in case of certain severe accidents could be an effective way to preserve the long-term containment functional integrity and reduce the uncontrolled release of radioactive material. These considerations are being taken into account in the activities already in place in the areas of containment integrity and containment venting. In the area of containment performance under severe accident conditions (Task 3.1A), these ongoing efforts include the Individual Plant Examination, Accident Management, and Reactor Risk Reference Document (NUREG-1150) efforts. In connection with filtered venting of containments (Task 3.2A), the work includes international technical information exchanges concerning venting provisions being proposed and implemented in Europe. The NRC-sponsored venting evaluations are being done at the Idaho Nuclear Engineering Laboratory.

Emergency Planning

Three tasks in the emergency planning area, involving study of ingestion pathway protective measures, decontamination, and relocation experiences after the Chernobyl accident, relate to existing staff efforts to re-assess aspects of emergency planning in light of estimated potential U.S. severe accident releases. The gathering and study of information concerning post-Chernobyl experiences is an interagency Federal and international effort, in which the NRC is participating. The Federal Emergency Management Agency is expected to be the coordinating organization for the U.S. activities. This is clearly a multi-year effort: in the next few months the efforts will focus on plans and arrangements. The specific issues involved are indicated below.

° Ingestion Pathway Protective Measures (Task 4.3A).

  After the Chernobyl accident, human and animal food chains in the Soviet Union and other European countries were contaminated in varying degrees. The Soviet and other affected governmental authorities took measures -- both short-term and longer term -- to protect the public from receiving unacceptably high levels of radiation through consumption of contaminated food. The contamination level findings and the experience with the Soviet and other European control measures could provide important extensions of the data base for planning of protective measures in the U.S. The NRC will participate, with FEMA and other Federal and appropriate international agencies, in planning and eventual execution of efforts to obtain available information on the Soviet and other European post-Chernobyl ingestion pathway contamination and control-measures experience and analyze that information in relation to U.S. understanding of the issue.

° Decontamination (Task 4.4A).

  The practicality and effectiveness of measures to decontaminate structures, land, etc. after a major accident can be a significant factor in evaluation of accident consequences as well as in formulation of plans and approaches for post-accident decontamination. The experience with post-Chernobyl decontamination in the Soviet Union could provide important extensions of the data base. The NRC will participate, with FEMA, other Federal agencies, and appropriate international agencies, in planning and eventual execution of efforts to obtain available information on the Soviet post-Chernobyl decontamination experience and analyze that information in relation to U.S. understanding of available techniques and their effectiveness.

° Relocation (Task 4.4B).

  Notwithstanding cultural and socioeconomic differences, the Soviet experience in connection with the post-accident evacuation and relocation of the population of contaminated towns and villages near the Chernobyl reactor may well offer valuable lessons for U.S. emergency planning. The NRC will participate, with FEMA and other appropriate Federal and international agencies, in developing plans and arrangements for learning about and from the Soviet post-Chernobyl relocation experience.

Severe Accident Phenomena

In view of the Chernobyl occurrences, NRC-sponsored source-term research efforts are being adjusted to recognize possible mechanical dispersal of fuel and chemical or thermal stripping of fuel particle surfaces as potential contributory mechanisms in some circumstances, as discussed below.

° Mechanical Dispersal in Fission-Product Release (Task 5.1A).

The initial release of fission products that occurred at Chernobyl was the result of mechanical dispersion. Although such potential energetic-event mechanisms in LWRs are being studied with regard to their likelihood of occurrence and their consequences, associated mechanical releases of fission products have not been quantified in current source term models, and the study of such releases has only just begun to receive attention. Because some of these phenomena appear to have played a dominant role in the releases at Chernobyl, it is important to understand these phenomena more completely. The Chernobyl lessons are being introduced into ongoing work to improve understanding of mechanical dispersal phenomena and to improve the modeling in NRC source term assessment codes.

° Stripping in Fission-Product Release (Task 5.1B).

The late enhanced release of fission products during the Chernobyl accident may be attributable to the chemical and/or thermal stripping of UO_2 fuel. Such mechanisms have been observed in in-pile and out-of-pile experiments when UO_2 fuel rods were exposed to steam or high temperatures (and other severe degraded core conditions). During the process of thermal stripping, for example, fission products were released in proportion to the amount of UO_2 vaporized. The rate of fission product release is thus controlled by UO_2 vaporization.

Fission product release by chemical and thermal stripping mechanisms is not modeled in current severe accident source terms codes. The Chernobyl accident has demonstrated that such mechanisms can be important in fission product release under some conditions. Accordingly, the Chernobyl lessons are being introduced into the continuing research on chemical and thermal stripping, needed to obtain sufficient data for model development and assessment.

° Steam Explosion (Contingent Task 5.2A).

No specific research is currently underway or planned on reactivity-insertion-accident prompt-burst steam explosions with fuel-vapor-driven fragmentation and mixing of the molten fuel and water that are relevant to the Chernobyl accident. Such work is currently not believed to be necessary, subject to confirmation in the light of results of the Chernobyl follow-up reactivity transient study. Accordingly, this task (5.2A) will not be undertaken unless a need for it is indicated by the results of Task 2.1A, Reactivity Transients.

CONCLUDING NOTE

The relatively limited scope of the work on generic implications of the Chernobyl accident for U.S. reactors contrasts with the extensive generic-issues activity that followed the accident at Three Mile Island. The scope and limits of the Chernobyl follow-up work reflect the assessments in the light of Chernobyl, which have indicated that the causes of the accident have been largely anticipated and accommodated for commercial U.S. reactor designs, partly as a result of improvements since TMI. But the scope also reflects the dependence of nuclear power plant safety on continued vigilance, including alertness to newly apparent issues and reexamination of past regulatory judgments in the light of new experiences and their implications.

GENERIC ISSUE 125.II.7

AUTOMATIC ISOLATION OF AUXILIARY FEEDWATER [a]

by

D. Basdekas, Nuclear Regulatory Commission
S. J. Bruske, EG&G Idaho, Inc
H. J. Welland, EG&G Idaho, Inc

ABSTRACT

Generic Issue 125.II.7 was evaluated to determine if the disadvantages of the automatic Auxiliary Feedwater System (AFW) isolation system, such as reduced AFW system reliability, outweighed the benefits provided by automatic AFW isolation to a steam generator during a steam or feed line break accident. Reactor plants with automatic AFW isolation systems from each Pressurized Water Reactor (PWR) vendor were evaluated by determining the automatic isolation system's contribution to the Core Damage Frequency (CDF) and determining any CDF increase or decrease that would occur if the system were removed. The CDF changes were used to estimate the change to the consequences or public risk due to removing the isolation system. Recent Probabilistic Risk Assessments (PRAs) for a reactor plant from each PWR vendor were used to quantify the changes to CDFs and risks. The findings of this study indicate that the automatic AFW isolation system does not contribute significantly to the CDF or public risk for any of the plants evaluated. Removing the AFW isolation system resulted in a slight decrease in CDF and risk in some plants and caused a slight increase in CDF and risk in others.

1. INTRODUCTION

The purpose of Generic Issue 125.II.7 was to reevaluate the provisions of automatic AFW isolation for a PWR steam generator during a steam or feed line break. This issue was identified from the findings of the Davis-Besse Incident Investigation Team as reported in NUREG-1154, Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985. The investigation identified the concern that the automatic AFW isolation system may add more risk, due to a higher AFW system unavailability, than the risk reduction that would be achieved by the automatic AFW isolation system performing its intended function. There are twenty-seven PWRs that have an automatic AFW isolation system.

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The automatic isolation of AFW from a steam generator is provided to mitigate the consequences of a steam or feedwater line break. The typical AFW isolation system logic closes all main steam isolation valves and also isolates AFW from the broken depressurizing steam generator while continuing to feed the unaffected steam generator(s).

With an automatic AFW isolation system, the blowdown inventory from a steam line or feed line break will be minimized. This will minimize the containment pressure increase and the uncontrolled cooldown of the primary system caused by the break. In some plants, automatic AFW isolation is required to divert AFW from the affected steam generator for an orderly plant shutdown and to meet the single failure criterion in supplying feedwater to the intact steam generator.

The disadvantages of automatic AFW isolation are related to concerns that the automatic isolation system may reduce the reliability of the AFW system. Also, with operator error, the long term success of AFW for main feedwater transients, steam generator tube ruptures, and small-break loss-of-coolant accidents (LOCA) could be compromised. Failures that cause inadvertent actuation of the AFW isolation system could cause loss of all AFW system flow during accidents or transients. Additionally, during a controlled cooldown, the thresholds for automatic AFW isolation may be crossed, which would require that the operator lock out the isolation logic as the steam generator parameters approach the isolation setpoint. During an accident scenario, the accompanying distractions could result in a failure to lock out the automatic isolation, thus AFW would not be available as predicted for the applicable accident analyses.

2. EVALUATION METHODOLOGY

2.1 Evaluation Technique

A plant from each of the PWR vendors was selected for evaluation, and available PRAs for the selected plants were employed for this analysis. AFW systems were studied to determine how they functioned and how the AFW automatic isolation system functioned. Accident sequences that included AFW operation were evaluated to determine the contribution to the CDF of inadvertent or spurious AFW isolation system actuation.

These CDFs were then used to calculate the change in consequences (public risk) by using containment failure probabilities from the respective PRAs and containment release rates from NUREG-0933. A cost estimate for modifications that would disable the automatic AFW isolation system were developed and a cost benefit analysis was performed.

2.2 Postulated Accident Sequences Affected by the AFW Isolation System

Inadvertent or spurious actuation of the AFW Isolation System could in some cases be the cause of a transient. For example, a spurious signal could cause the main steam isolation valves and AFW isolation valves to close. Closure of the main steam isolation valves would effectively trip the main feed pumps in many plants because the pumps are turbine-driven and rely on a steam supply for operation. Thus, the plant would be in a total loss-of-feedwater transient. If the operators cannot recover feedwater flow or initiate and maintain feed-and-bleed in the limited time available, usually about 30 minutes, the transient will lead to core damage due to a loss of heat removal capability.

Spurious or inadvertent actuation of the AFW isolation system could be a significant contributor to the unavailability of the AFW system. Recovery actions may not be simple operations; at the June 1985 Davis-Besse event, the operators had to manually initiate the opening of the isolation valves because the valve motor torque limit had been improperly set, and the pumps had to be manually restarted.

During a long term cooldown, secondary system conditions that cause actuation of AFW isolation system will eventually be reached, i.e. low steam generator pressure. If the operator has not locked out or bypassed the isolation system, the AFW system will be lost and some type of recovery action will be required. The added stress (caused by this additional event) may cause the operators to make other errors, complicating recovery from an accident sequence, and eventually leading to core damage.

Isolation of a depressurizing steam generator, caused by a feedwater line break, is required to prevent either the diversion of flow from unaffected steam generators or the failure of all the AFW system due to pump runout or cavitation caused by higher than normal flow rates. If the automatic AFW isolation system is removed or disabled, isolation of the affected steam generator would have to be performed manually. During a postulated accident sequence consisting of a feedwater line break, followed by the failure to isolate the affected steam generator, and the failure of feed-and-bleed (where this technique can be performed), the core damage contribution may increase because timely operator action under stressful accident conditions is required to isolate the affected steam generator.

If the AFW flow to a steam generator with a broken main steam line is not stopped, steam will continue to be released to the containment, which could lead to containment overpressure. Disabling or removing the automatic AFW system would increase the consequences associated with this scenario since operator action would be required to isolate the affected steam generator.

3. PLANT ANALYSIS

3.1 Plant A

3.1.1 System Description

Plant A is a Combustion Engineering (CE) designed reactor system that has two U-tube steam generators. The AFW system (Figure 1) has one turbine-driven pump and one motor-driven pump. Each pump supplies both steam generators through a separate header, i.e., there are flow control and isolation valves for the turbine-driven pump on a header separate from the flow control and isolation valves for the motor-driven pump to steam generator 1. There is another turbine-driven pump, but it must be manually lined up and started by the operator. Also, the motor-driven pump from Unit 2 can be cross-connected to Unit 1. The extra turbine-driven pump and cross-connecting of the motor-driven pump from Unit 2 were considered recovery actions for the applicable accident sequences by the PRA used for this plant.

Each steam generator has its own automatic AFW isolation system, actuated by two independent channels (A and B). The AFW system isolation is accomplished by two valves in series in each header supplying each steam generator for a total of eight isolation valves. The valves have no other purpose in the system and are normally open. Isolation initiation circuit A closes one valve in each header on the affected steam generator and circuit B closes the other valve. Only one valve in each affected header (motor-driven and turbine-driven pump headers) must be closed to isolate AFW flow to the desired steam generator.

During a steam or feedwater line break, the main steam isolation valves on both steam generators will close when the pressure of either steam generator drops below 500 psig. The AFW isolation valves on the affected steam generator will close on coincident low steam generator water level (less than 50 in.) and high steam generator differential pressure (greater than 100 psid). If the isolation signal is generated while the AFW system is in operation, half of the isolation signal will already be present because the AFW is initiated by the low steam generator water level signal. The actuation signals for each circuit are based on two of four coincidence from four independent transducers on each steam generator.

Each header has a throttle valve set to limit flow to 200 gpm; thus, failure to isolate AFW from a ruptured steam generator will not cause the loss of AFW to the other steam generator due to flow being diverted to the steam generator with the lower pressure or cause the pumps to fail because of cavitation or runout due to higher than normal flow. The PRA used to evaluate this plant indicates that the AFW system is assumed to fail if less than 400 gpm is delivered to one or both steam generators.

The AFW system is also used to maintain steam generator levels during startup and shutdown when the reactor power level is low (less than 5%).

3.1.2 Plant A (CE) Sequence Analysis

At Plant A (CE), two automatic AFW isolation system failure sequences were evaluated. The first sequence was failure of the isolation system that results in inadvertent isolation of both steam generators, and the second sequence was inadvertent isolation of one steam generator when one of the AFW pump trains is inoperable. If both steam generators are isolated, all AFW flow will be lost. If one steam generator is isolated

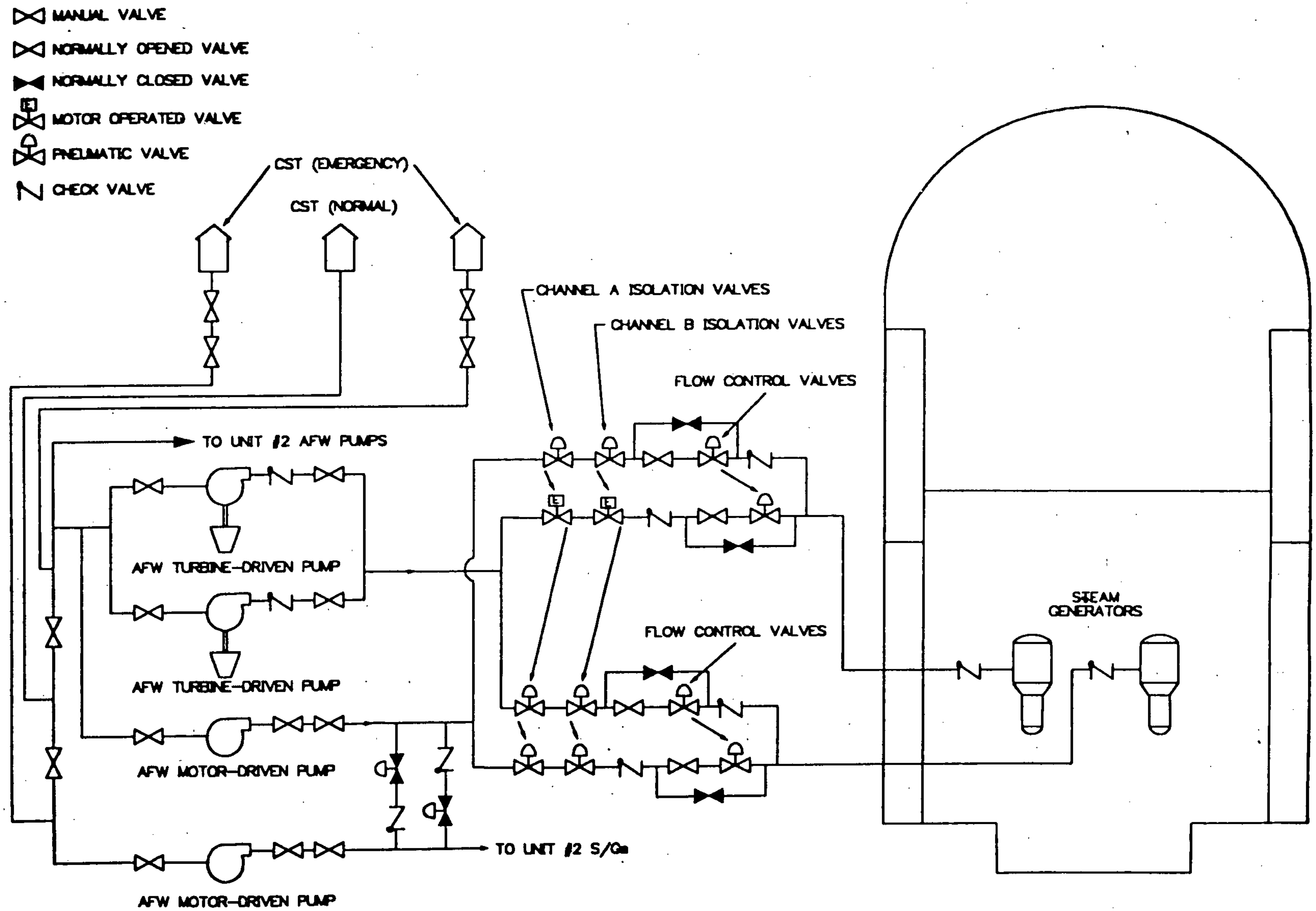

Figure 1. Plant A (CE) Auxiliary Feedwater System Flow Schematic. MGH01701

while one of the pump trains is inoperable, the AFW flow will drop to 200 gpm. The operator must then take manual control of the appropriate throttle valve to increase AFW flow to greater than 400 gpm, or he must open one of the isolation valves to maintain adequate heat removal. Feed-and-bleed was not considered effective at Plant A because of the relatively low discharge head (1275 psi) of the high pressure injection pumps and the uncertainty as to whether or not the pressure could be reduced below the pumps' discharge head.

AFW system failure caused by spurious actuation of the AFW isolation system would consist of; (a) a spurious signal to isolate one steam generator, (b) a common mode failure of the logic module to isolate the other steam generator, and (c) the operator failing to recover flow to greater than 400 gpm. The Plant A (CE) PRA indicates a spurious isolation has a probability of 7.2E-05. If a common mode failure probability (0.05) similar to that used in the NUREG-0933 evaluation of this event and a failure of recovery probability (0.04) similar to that used in the PRA are assumed, the resulting cutset will have a probability of 7.2E-05 * 0.05 * 0.04 = 1.44E-07.

Because there are two actuation channels, two cutsets contributed to the change in CDF due to the aforementioned factors; thus, the AFW system failure probability due to the automatic AFW isolation system was calculated to be 2.88E-07.

Failure events involving isolation of both steam generators caused by independent spurious signals to both steam generators have a probability of occurrence of about 5.0E-09 and were judged to be insignificant; therefore, they were not included in this evaluation.

Failure events that could lead to core damage caused by spurious isolation of one steam generator when the turbine-driven or motor-driven pump trains are not operating for some reason (such as during maintenance or valve failure) consist of (a) a spurious signal to isolate one steam generator and (b) the operator fails to recover AFW to greater than 400 gpm. The probability of a spurious signal actuating the AFW isolation system to isolate one steam generator and the operator failing to recover was calculated as 7.2E-05*0.04 = 2.88E-06.

Again, two cutsets contribute to this accident sequence because there are two isolation actuation channels; thus, the AFW system failure probability due to spurious isolation of one steam generator was calculated as 5.76E-06. These cutsets are only applicable when one of the pumping systems is inoperable for reasons not related to the spurious isolation.

In the preceding discussions, only the AFW is isolated by the spurious signals. The main steam isolation valves are on a different logic module and will require a different signal to cause them to close.

3.1.2.1 <u>Spurious AFW Isolation Caused Transients</u>. A spurious actuation of the AFW isolation system at Plant A (CE) will not cause a total loss of feedwater transient because the signal to close the main steam isolation valves (which will cause the main feed pumps to trip) comes from a different logic module than the signal that closes the AFW system isolation valves. Thus, two spurious actuation signals are required to initiate a transient and fail AFW. This accident sequence is covered in the plant PRA by the loss of the Power Conversion System (PCS) followed by loss of the AFW system accident sequence. The PCS comprises the main feedwater and condensate system, the steam generators, and the main steam system which includes the turbines, the turbine bypass, the atmospheric dump valves, and the safety relief valves.

3.1.2.2 Spurious AFW Isolation During Transients Requiring AFW. Six dominant accident sequences contain AFW system failures; the automatic AFW isolation system contributes 5.5E-07/Rx-yr to the core damage frequency in these accident sequences.

3.1.2.3 Operator Failure To Lock Out Isolation System During Cooldown. No operator action is required to lockout the automatic AFW isolation system at Plant A (CE) because the AFW isolation signal is generated from a high steam generator differential pressure. During a long term cooldown the steam generators will remain at approximately the same pressure; thus, the isolation signal will not be actuated.

3.1.2.4 Feedwater Line Break Initiated Transient. This accident sequence considers the impact of the AFW isolation system following a feedwater line break. Generally, the affected steam generator is isolated to prevent pumping water out of the break. Failure to isolate the affected steam generator could lead to the failure of the remaining trains of AFW due to the diversion of a sufficient amount of flow from the break, which would fail the AFW function or the pumps could fail due to cavitation or runout problems due to the higher than normal flow. Because Plant A (CE) has flow limiting valves in the system headers, this sequence is not affected by the removal of the AFW isolation system. No operator action is required to prevent the diversion of AFW flow out of the break.

3.1.2.5 Main Steam Line Break Initiated Transient. This accident sequence evaluates the impact of removing the AFW isolation system on a steam line break accident. This accident involves a transient initiated by a steam line break. Steam line break accident sequences that lead to core damage have such a low CDF that they were not included in this analysis. The primary concern due to this postulated accident sequence is containment failure due to overpressurization. The frequency of occurrence was calculated on a generic basis in NUREG-0933 as 1.0E-06/Rx-yr. The NUREG-0933 analysis was used because the pipe rupture frequency and operator error and containment failure rates are consistent with similar events found in the PRAs used in this evaluation and other documents that contain generic failure rates. The NUREG-0933 evaluation is:

1.0E-03 * 0.1 * 0.01 = 1.0E-06/Rx-yr

where the frequency of steam and feedwater line breaks is estimated as 1.0E-03/Rx-yr, with 10% (0.1) of these assumed to be steam line breaks. Failure of the operator to manually isolate the affected steam generator has been estimated as 0.01. NUREG-0933 also assumes that, given the occurrence of this sequence of events, the probability of containment failure is 0.03. It should be noted that NUREG-0933 considers this a "highly conservative assumption." Using this value, the estimated frequency of containment failure due to a steam line break is 3.0E-08/Rx-yr. This value is used later in determining the impact this issue will have on consequences (total man-rem).

3.1.3 Total CDF Contribution for Plant A (CE)

Removing the automatic AFW isolation system was calculated to decrease the core damage frequency by 5.5E-07/Rx-yr.

Removal of the AFW isolation system is expected to have a negative impact on consequences through the steam line break accident sequence, because of the increased probability of containment failure. No increase in the core damage frequency due to feedwater line breaks is expected because the plant has flow limiting valves in the AFW supply headers which will prevent flow being diverted to the depressurized steam generator or pump failure due to cavitation or pump run out. Therefore, the net impact of this issue (i.e., removal of the AFW isolation system) is a CDF reduction of 5.5E-07/Rx-yr.

3.2 Other Plants

Plant B, a Babcock & Wilcox (B&W) plant with once-through steam generators and an upgraded emergency feedwater initiation and control system (EFIC), Plant BB the same plant with the original emergency control system, and Plant C, a Westinghouse (W) designed reactor with four U-tube steam generators were evaluated using the same methods as those described for Plant A. The change to the CDF caused by removing the automatic AFW isolation system is shown below in Table 1.

TABLE 1. CHANGE TO CDF CAUSED BY REMOVING THE AUTOMATIC AFW ISOLATION SYSTEM

Plant	Decrease In CDF Caused By Deactivating AFW Isolation System (per Rx-yr)	Increase In Main Feed Line Break CDF Caused By Deactivating AFW Isolation System (per Rx-yr)	Total Change To CDF (per Rx-yr)
A (CE)	5.50E-07	0	-5.5E-07
B (B&W)	4.40E-08	1.4E-07	+9.6E-08
BB (B&W)	1.04E-06	1.4E-07	-9.0E-07
C (W)	4.00E-08	4.4E-07	+4.0E-07

The increase in the frequency of containment failure due to a steam line break was 3.0E-08/Rx-yr for plants A, B, and BB. For plant C the increase in containment failure frequency was 1.0E-06/Rx-yr. Plant C is different because the facility has an ice condenser containment and it was felt the containment failure probability used for the other plants was not appropriate. A conservative containment failure probability of 1.0 was used for this plant.

4. COST BENEFIT ANALYSIS

4.1 Cost Benefit Analysis Methodology

The cost benefit analysis was performed by estimating the costs for disabling the automatic AFW isolation system at each of the four plants and the costs for providing flow limiting devices at the two plants that require flow limiting, and comparing the total cost to the risk reduction.

The containment failure probabilities and the release categories for a specific accident sequence were extracted from the PRAs used in the analysis. The release categories are those defined in WASH-1400.

Estimated public dose in terms of man-rem were assigned to the WASH-1400 release categories in accordance with the data presented in NUREG-0933. The data presented in NUREG-0933 was calculated based on a typical mid-west site adjusted to reflect the mean of the population density within a 50-mile radius of U. S. nuclear power plants.

4.2 Risk Evaluation

The risk evaluation was performed by utilizing the reduction in CDF calculations presented in Section 3 of this report and the methodology identified in Section 4.1 to determine the containment failure rate and the offsite dose releases. Total risk change was estimated using the following relationship:

Change in CDF (events/yr)	X	Containment Failure Probability	X	Offsite Radiation Dose (man-rem)	=	Risk Change (man-rem/year)

To calculate the total change to the potential population exposure or risk per plant life due to this issue, the above relationship was extended over the plant life, taking into account plant down-time. The total change in population exposure over the remaining plant lifetime is calculated as follows:

Change in Risk (man-rem/ year)	X	Remaining Plant Life (years)	X	Plant Utilization Factor	=	Total Change in Population Risk (man-rem)

It was estimated that each plant had a remaining lifetime of 23 years, with an associated utilization factor of 75%. These values were taken from the NUREG-0933 analysis.

To provide additional information with regard to the potential impact from implementing the proposed modification, simple sensitivity analyses were performed, which consisted of utilizing the NUREG-0933 containment failure probabilities to calculate a lower bound for the change in risk.

Table 2 shows the estimated change in risk (man-rem) for the plants evaluated (23 years at 75%).

TABLE 2. RISK CHANGE DUE TO PROPOSED AFW SYSTEM MODIFICATION

Plant	Plant PRA Data (man-rem)	NUREG-0933 Data (man-rem)
A (CE)	36.2 Decrease	1.5 Decrease
B (B&W)	4.5 Increase	0.25 Increase
BB (B&W)	44.4 Decrease	3.0 Decrease
C (W)	13.3 Increase	

Table 2 shows that the change in risk values calculated using the NUREG-0933 data are much lower than those determined by using the plant specific containment failure probabilities. The NUREG-0933 values are based on additional information gained over the past several years due to the significant amount of research performed on the response of the containment under accident conditions and are considered more realistic than the Plant PRA release data.

4.3 Proposed Modifications Cost Analysis

Cost estimates for disabling the automatic AFW isolation system and adding flow limiting devices (when required) were developed using NUREG/CR-4568, A Handbook for Quick Cost Estimates, as a guide and a cost estimate was prepared by the NRC staff. The estimated costs are $351,000 to disable the automatic AFW isolation system only and $768,000 to disable the automatic AFW isolation system and install flow limiting devices. These estimates assumed the work could be accomplished during a scheduled outage. If a special outage has to be scheduled, approximately $6,000,000 for replacement power must be added to the costs for plants that require flow limiter modifications.

4.4 Cost Benefit Analysis

The cost effectiveness of the proposed modification for each plant was determined by use of the following equation:

Estimated cost of Modification ($)	/	Change in Risk (man-rem)	=	Cost/Benefit ($/man-rem)

The cost/benefit ratio for a plant that does not require flow limiter modifications is:

$$\$351{,}000/44.4 = \$7905/\text{man-rem}$$

and for a plant that requires flow limiter modifications, the cost/benefit ratio is:

$$\$768{,}000/44.4 = \$17{,}290/\text{man-rem}.$$

The largest risk change was used in the above calculations for conservatism, even though it did not apply to the plant that does not require flow limiter modifications.

5. UNCERTAINTIES

5.1 Consequence Uncertainties

The uncertainty in the risk calculations was estimated by assuming an error factor of 10 on the risk changes calculated by this evaluation. No uncertainty was assumed for the containment failure probabilities, or the release rates. The assumption of an error factor of 10 was confirmed to be acceptable based on a Monte-Carlo analysis of Plant A.

The Monte-Carlo analysis calculated the risk change (man-rem) that resulted from removing the automatic AFW isolation system from Plant A. The accident sequences used in the analysis contributed 96% of the CDF for sequences associated with failure of the AFW caused by the automatic AFW isolation system. Uncertainties were assigned to all elements of the accident sequences except the remaining plant life and the plant utilization factor. The results showed a calculated mean of 35.0 compared to a mean of 36.2 calculated by the point estimate method of this report. Ninety percent of the Monte-Carlo solutions were between 0.665 and 136.0 compared to 3.62 and 362 based on the error factor of 10. The results of the Monte-Carlo analysis show that assuming an error factor of 10 was conservative on both ends of the error bounds.

5.2 Sensitivity of Cost Benefit Summary

Table 3 presents the base information utilized in performing the sensitivity analysis. This table is a compilation of data previously presented. Table 4 presents the results of the sensitivity analysis. Cost/benefit ratios were not calculated for those plants (Plants B and C) for which implementing the proposed modifications caused a net increase in the CDF.

Table 4 shows that all of the estimated upper cost/benefit ratios are above $1000/man-rem with the exception of Plant A using the PRA containment response information.

TABLE 3. BASE DATA EMPLOYED IN THE SENSITIVITY ANALYSIS

Plant	Total Change In CDF (per Rx-year)	Offsite Consequences (Total man-rem) PRA	NUREG-0933	Cost ($1000)	Cost/Benefit Ratio ($1000/man-rem)
A (CE)	5.5E-07 (decrease)	36.2	1.5	351	9.7
B (B&W)	9.6E-08 (increase)	4.5	0.25	768	*
BB (B&W)	9.0E-07 (decrease)	44.4	3.0	768	17
C (W)	4.0E-07 (increase)	13.3	**	351	*

* Cost/benefit ratios were not calculated for plants where the implementation of this issue would result in an increase in the estimated risk.

** Consequences using the NUREG-0933 information were not estimated for this plant since the resulting value would not be comparable to the plant specific value. The values are not comparable due to the different assumptions and techniques employed in the two analyses to determine offsite consequences.

TABLE 4. SENSITIVITY ANALYSIS RESULTS

		Cost/Benefit Ratio ($/man-rem)	
Plant		PRA Containment Failure Information	NUREG-0933 Containment Failure Information
A (CE)			
	Upper Bound	970	23,400
	Best Estimate	9700	234,000
	Lower Bound	97,000	2,340,000
BB (B&W)			
	Upper Bound	1700	25,600
	Best Estimate	17,000	256,000
	Lower Bound	172,000	2,560,000

6. CONCLUSIONS

Four PWRs, one each CE and W designs and two B&W designs, were evaluated to determine the AFW isolation system's contribution to CDF. Three of the plants selected did not have flow limiters to limit flow to a ruptured steam generator, one of them could not be cooled successfully by feed-and-bleed, and one had a very diverse AFW isolation system.

The evaluation indicates that the effects of the AFW isolation system are strongly dependent on the particular plant's design. The estimated contribution to CDF due to AFW isolation system were reasonably low, but the difference between the highest and the lowest value was an order of magnitude. Disabling the automatic AFW isolation system would result in a slight decrease in risk at two of the plants and a slight increase at the other two plants. The cost/benefit ratios for making the necessary modifications to disable the automatic AFW isolation system range from $8,000 to $17,000 per man-rem depending on whether flow limiting devices are required or not.

Based on the results of the cost/benefit analysis, and the insights gained during assessment of the pros and cons of removing the automatic AFW isolation system, it was concluded that no backfit requirement to remove or disable the system was warranted.

Additional information about the evaluation of this issue can be found in NUREG/CR-5178, "Evaluation of Generic Issue 125.II.7, Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break" and in NUREG-1332, "Regulatory Analysis for the Resolution of Generic Issue 125.II.7, Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break," dated September 1988.

EMERGENCY DIESEL GENERATOR RELIABILITY PROGRAM

Aleck W. Serkiz
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Reactor and Plant Safety Issues Branch

ABSTRACT

The need for an emergency diesel generator (EDG) reliability program has been established by 10 CFR Part 50, Section 50.63, "Loss of All Alternating Current Power," which requires that utilities assess their station blackout duration and recovery capability. EDGs are the principal emergency ac power sources for coping with a station blackout. Regulatory Guide 1.155, "Station Blackout," identifies a need for (1) an EDG reliability equal to or greater than 0.95, and (2) an EDG reliability program to monitor and maintain the required levels.

The resolution of Generic Safety Issue (GSI) B-56 embodies the identification of a suitable EDG reliability program structure, revision of pertinent regulatory guides and Tech Specs, and development of an Inspection Module. Resolution of B-56 is coupled to the resolution of Unresolved Safety Issue (USI) A-44, "Station Blackout," which resulted in the station blackout rule, 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout."

This paper discusses the principal elements of an EDG reliability program developed for resolving GSI B-56 and related matters.

ELEMENTS OF AN EDG RELIABILITY PROGRAM

A reliability program process is shown in Figure 1. The concept is a series of activities designed to monitor performance, to assess operational characteristics, to identify problem causes and corrective actions, and to compare performance against target reliability levels. Although the concept is straightforward, the definition and implementation of an effective reliability program for a specific application is not simple. The intermittent fast-start operational requirements placed on EDGs at nuclear power plants further complicate implementation of an effective reliability program.

A diesel generator reliability program should be based on the philosophy that specified reliability targets can be achieved by understanding the factors that drive a diesel generator's reliability and operational characteristics. The application of sound reliability and engineering considerations can then be used to develop a reliability program concept such as discussed below.

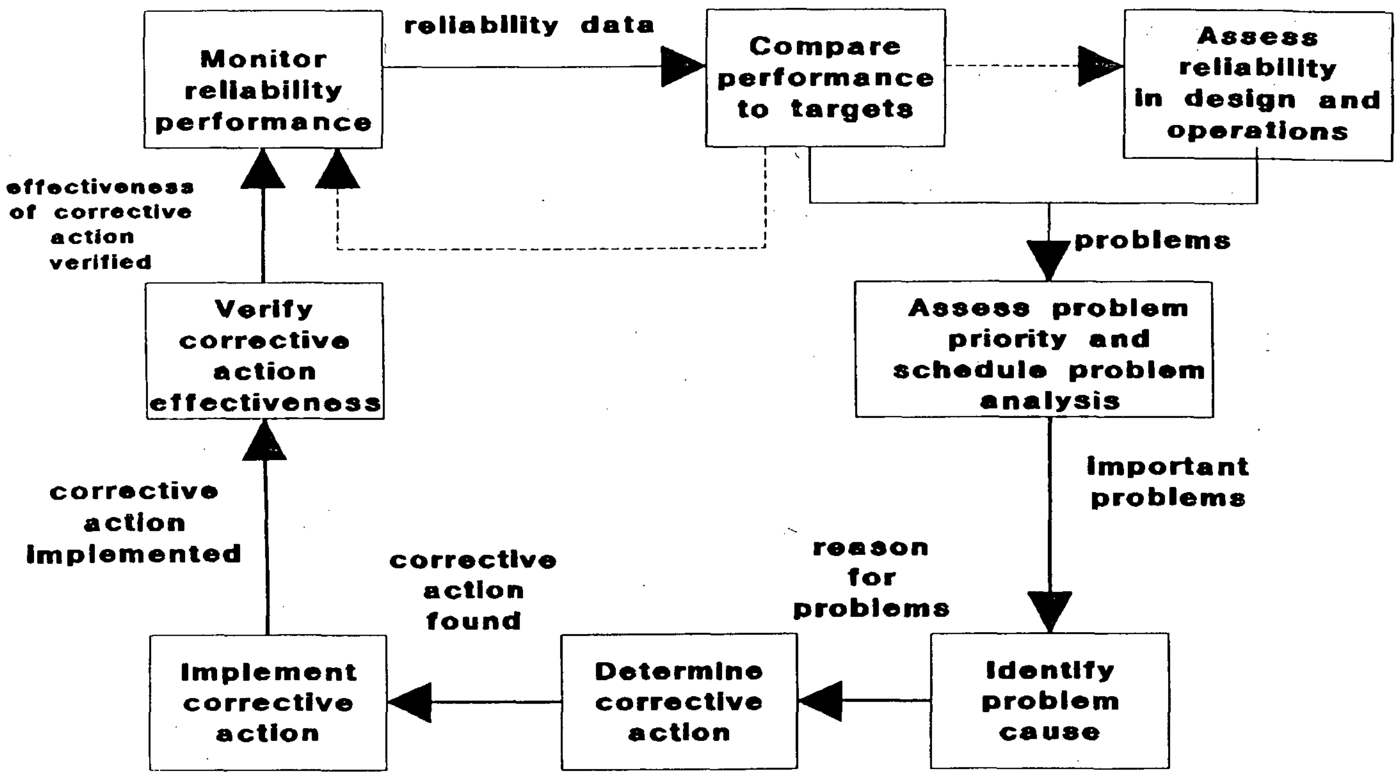

FIGURE 1 Reliability program process

A balanced EDG reliability program should be comprised of the following activities or elements:

1. A RELIABILITY TARGET level of => 0.95

2. SURVEILLANCE REQUIREMENTS that identify EDG equipment boundaries, subsystems, and surveillance criteria for monitoring and judging acceptable performance. Maintenance and aging considerations should be incorporated into this element.

3. PERFORMANCE MONITORING requirements and criteria that are designed to spot degradations in advance of failures.

4. A MAINTENANCE PROGRAM that has a reliability focused approach that includes preventive maintenance, vendor recommendations, spare parts considerations, and proven operational experience.

5. FAILURE ANALYSIS and ROOT CAUSE INVESTIGATION PROCEDURES that will systematically reduce identified problems to correctable causes.

6. PROBLEM CLOSEOUT procedures that have established criteria for problem closeout when a reliability problem is detected and that provide for followup monitoring to ensure that the correction has been successful.

7. A "real time" DATA SYSTEM that provides for data gathering, storage, and retrieval with sufficient capability to service reliability monitoring and maintenance activities and all parts of the EDG reliability program.

8. RESPONSIBILITIES and MANAGEMENT CONTROLS that ensure that responsibilities have been clearly defined, qualified personnel committed, authority lines established, and management oversight and control procedures developed to ensure effective functioning of the reliability program.

Although these elements can be viewed individually, elements 1 and 8 are most important. The importance of management commitment to such a program and the assignment of qualified staff cannot be overemphasized.

Figure 2 illustrates how these elements would likely interact. The target reliability is a goal that must be maintained. Plant management must ensure that targets are being met and that the program is functioning effectively. The data system should service all elements of the program and is therefore shown centrally located. The right-hand side of Figure 2 shows the maintenance and operations oriented elements; the monitoring activities are shown on the left. All elements need to interact effectively.

EDG HISTORICAL PERFORMANCE

Industrywide EDG reliability levels have generally exceeded 0.98, with only a few plants being in a lower range. The Electric Power Research Institute's (EPRI) estimated industrywide reliability levels (as derived from NSAC-108) are shown in Figure 3. For the most part, high levels of reliability have been achieved except for a limited number of plants. The tracking system of the Institute of Nuclear Power Operations (INPO) exhibits a similar pattern.

Nonetheless, EDG failures do continue to occur in a random and unpredictable manner. Reported EDG failures and affected subsystems can be used to develop insights for development of monitoring and maintenance activities. Figure 4 illustrates failure distribution by subsystem derived from the data base of the Nuclear Plant Reliability Data System (NPRDS). The high number of failures occurring in the air start subsystem and the EDG itself warrant consideration in developing a reliability program. The importance of a reliable air start system cannot be underestimated. Unfortunately, the NPRDS data base does not contain enough information to calculate equipment on-line time, number of demands, etc., and thus failure rates cannot be estimated. A plant-specific reliability program would benefit from an onsite data base for the onsite EDGs used and should incorporate performance experience (particularly failures) from other sites using the same vendor-type EDG.

NUREG/CR-4590 provides a failure data base that can be used to obtain an insight by various subsystems for a particular diesel manufacturer. There are nine different suppliers of diesels, these being: ALCO, Allis-Chalmers, Caterpillar, Cooper-Bessemer, Electro-Motive Division of General Motors, Fairbanks-Morse, Nordberg, Transamerica Delaval and Worthington. Figure 5 illustrates reported failures by subsystem from this data base. Instrumentation and control (I&C) reported failures are the dominant class, followed by fuel, starting, switchgear, cooling, and lubrication subsystems. Except for I&C failures, the distribution shown in Figure 5 is fairly flat and illustrates the difficulty of singling out a particular subsystem for corrective action.

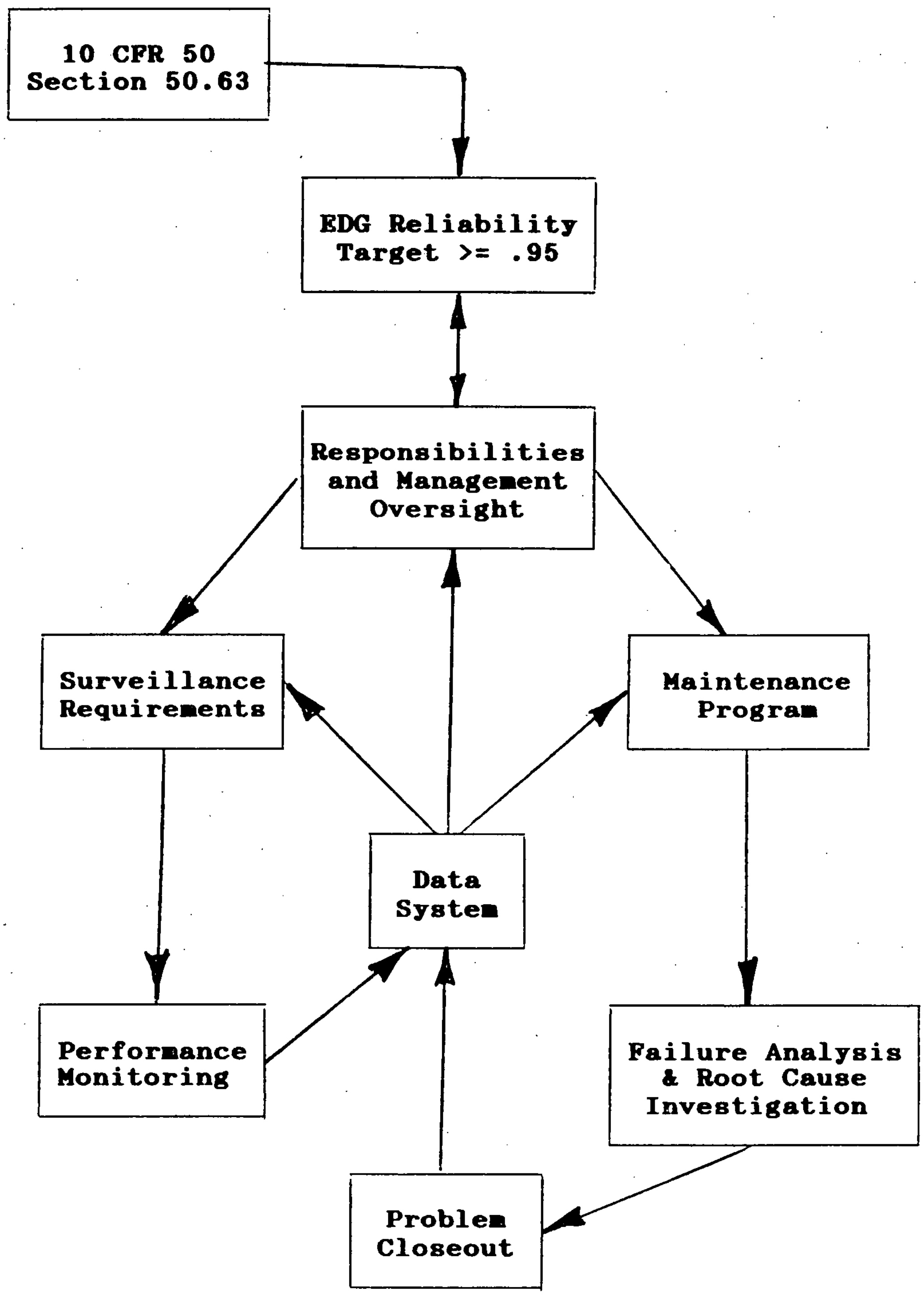

FIGURE 2 Interaction of EDG reliability program elements

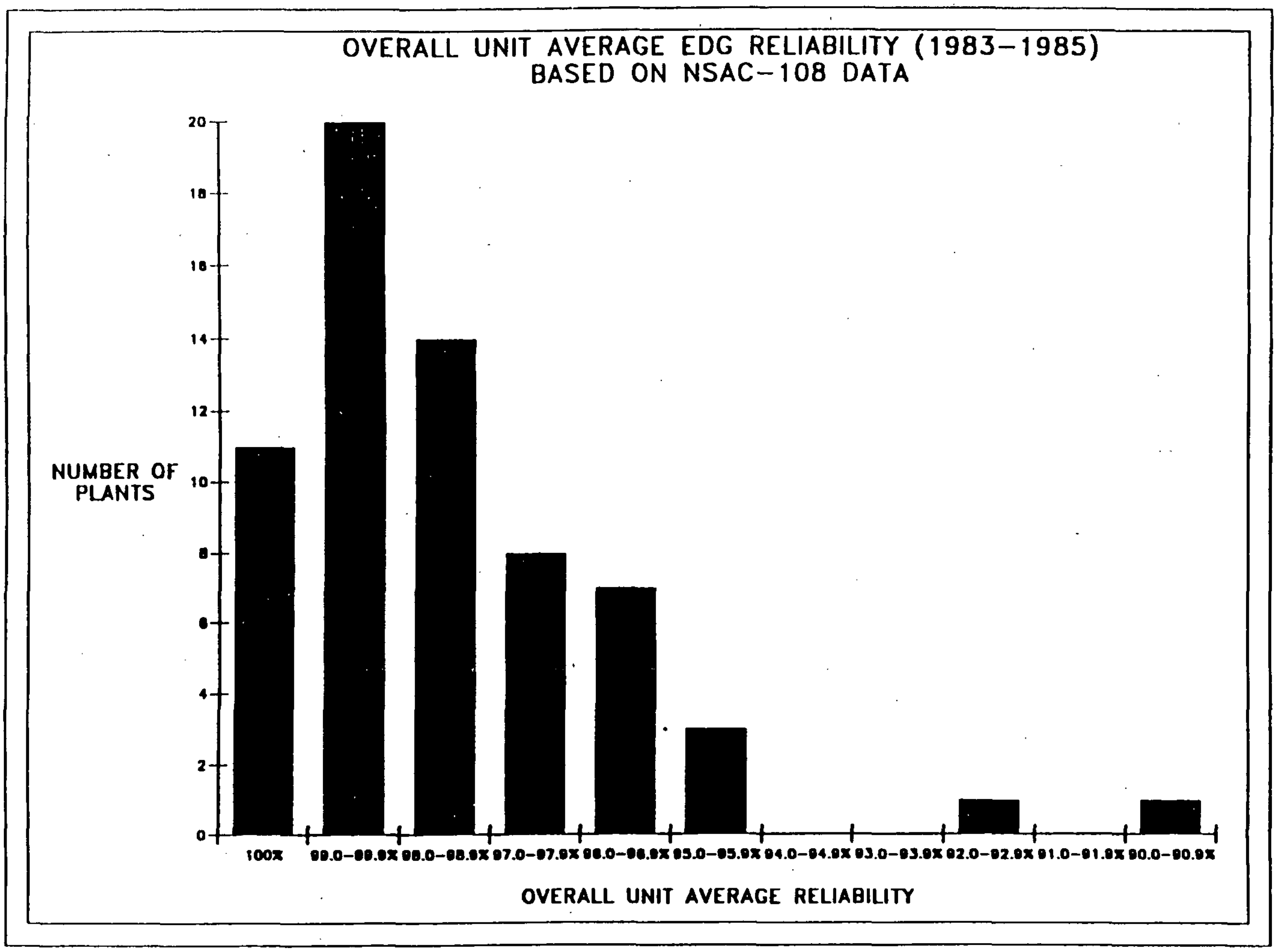

FIGURE 3 Industry-wide EDG reliabilities

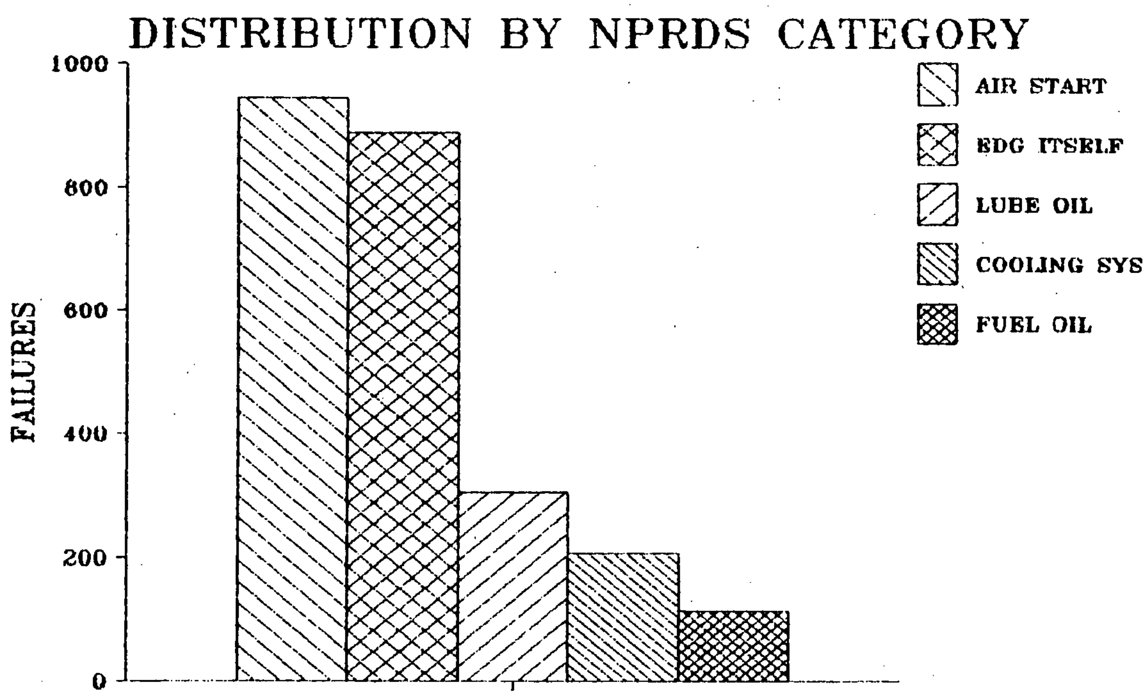

FIGURE 4 EDG failure data from NPRDS

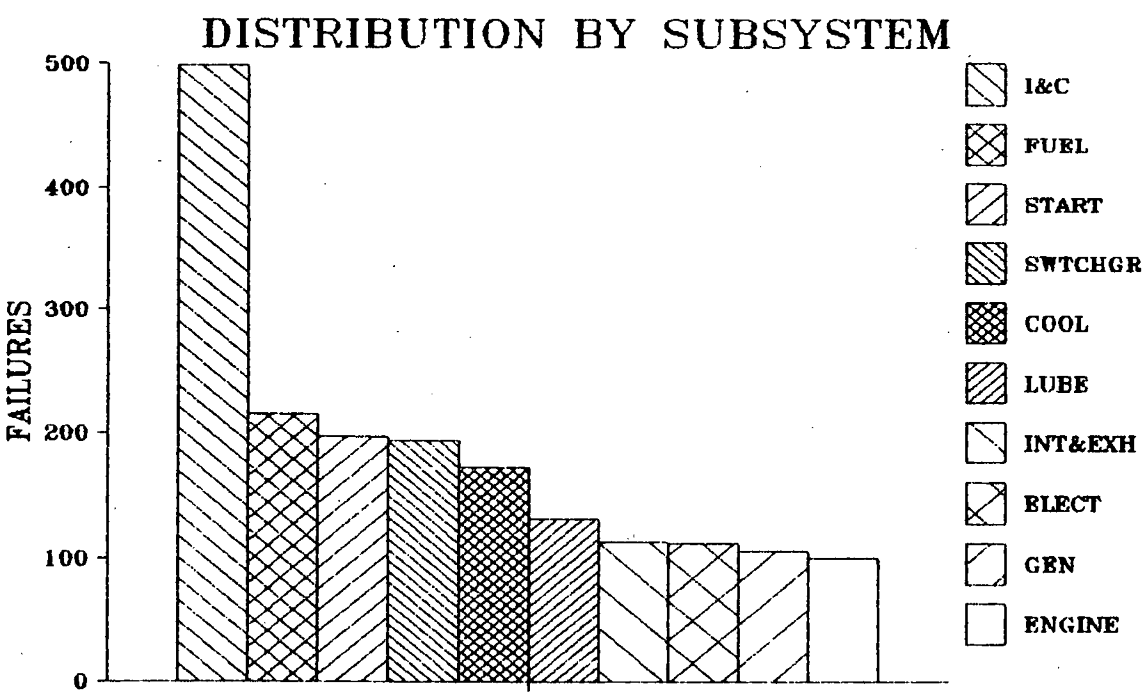

FIGURE 5 EDG failure data from NUREG/CR-4590

NUREG/CR-5078, Vol. 1, Appendix I provides more details on these data bases as well as the failure distribution by EDG vendor. In addition, maintenance and operator errors (see Fig. 6) contribute significantly to failures. Although Figure 6 represents only a 3-month time frame, licensee event report (LER) submittals over the past year exhibit a similar pattern, with "people" induced underlying causes being a high percentage.

This apparently high industrywide EDG reliability versus unpredictable failures and underlying causes supports the need for a suitably designed EDG reliability program that makes use of proven reliability program techniques to anticipate failures and to take corrective actions in advance.

REGULATORY TESTING REQUIREMENTS

The USNRC requires that EDGs be load-tested on a monthly basis (e.g. Regulatory Guide 1.108 and Generic Letter 84-15) and, if a start failure occurs in the last 2 of 20 start attempts, that the test interval be reduced to weekly tests to verify operability and to estimate EDG reliability. Such accelerated testing imposes high stress conditions, particularly if cold fast starts are run. Generic Letter 84-15 allowed for pre-conditioning of the EDG prior to demand load testing and also relaxed other aspects of Regulatory Guide 1.108. The currently proposed Revision 3 to Regulatory Guide 1.9 attempts to combine the better features of Regulatory Guide 1.108 and Generic Letter 84-15, and outlines an EDG reliability program such as discussed above.

The regulatory criteria for estimating EDG reliabilities are based on a failure-on-demand approach as deduced from the last 20, 50, and 100 tests. This approach is susceptible to statistical uncertainties (particularly for a 20-test sample) and prior maintenance effects. EPRI has recommended that only the 50 and 100 demand test sample be used for estimating reliability levels to avoid limited sample size effects.

FAILURE EVALUATION CRITERIA (or ALERT LEVELS)

If failure tracking is going to be the principal regulatory tool for deriving reliability levels, a failure evaluation criterion needs to be developed. Failure tracking can be used to establish an ALERT system that allows time for taking corrective action(s) and that minimizes accelerated testing. Furthermore, a properly defined failure tracking methodology, with pre-defined action criteria, would enhance EDG reliability and verify operability of the implemented reliability program.

EMERGENCY DIESEL GENERATOR FAILURES
UNDERLYING CAUSES

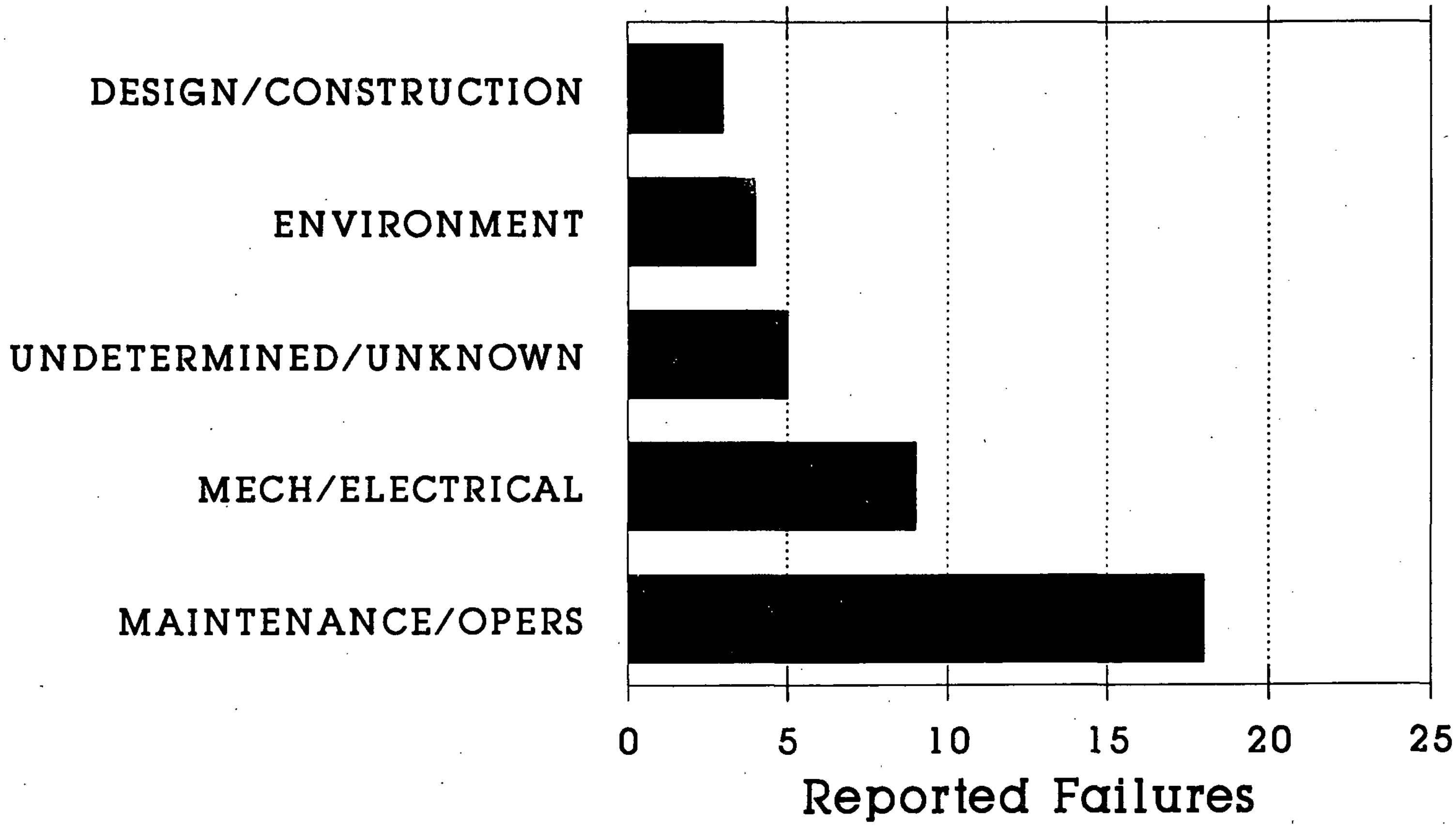

FIGURE 6 EDG failure underlying causes

Such an approach is discussed in NUREG/CR-5078, Vol. 1, and is based on interpreting failure progression for determining what action should be taken. For example:

Failure Progression #1(IMMEDIATE ACTION REQUIRED)

For EDGs with Reliability Target of 95%	For EDGs with Reliability Target of 97.5%
=> 2 failures in 20 demands	=> 1 failure in 20 demands
=> 5 failures in 50 demands	=> 3 failures in 50 demands
=> 10 failures in 100 demands	=> 6 failures in 100 demands

This condition is unacceptable and requires the licensee to take immediate action, declaring the EDG inoperable and entering the required Limiting Conditions of Operation (LCO). Such a continued state of failure progression strongly suggests that the EDG reliability program has been deficient for a long period of time and that the program needs correcting before entering the EDG into service.

Failure Progression #2 (STRONG ALERT)

For EDGs with Reliability Target of 95%	For EDGs with Reliability Target of 97.5%
=> 2 failures in 20 demands	=> 1 failure in 20 demands
=> 5 failures in 50 demands	=> 3 failures in 50 demands
< 10 failures in 100 demands	< 6 failures in 100 demands

This is a strong alert condition, with strong evidence that the EDG reliability has been deteriorating over time or has been deficient for some time. Action is warranted by both the utility and the NRC to determine if the diesel generator(s) should be placed into the inoperable category upon encountering the next start failure.

Failure Progression # 3 (MILD ALERT)

For EDGs with Reliability Target of 95%	For EDGs with Reliability Target of 97.5%
=> 2 failures in 20 demands	= > 1 failure in 20 demands
< 5 failures in 50 demands	< 3 failures in 50 demands
< 10 failures in 100 demands	< 6 failures in 100 demands

This is a condition that provides a mild alert to the NRC and licensees that an EDG may be experiencing performance problems. EDGs with an acceptable reliability of 0.95 can be expected to experience 2 failures in 20 demands about 26% of the time. EDGs having an acceptable reliability of 0.975 can be expected to experience 2 failures in 40 demands about 26% of the time. Also, more than 2 failures in the last 20 (or last 40) demands should be taken as cause for heightened concern as they may indicate progressive failure of the reliability program.

Failure progression #3 is also illustrative of the informational content of the last 20 tests, but without undertaking accelerated testing. This example is also likely to be the most commonly found industrywide. Statistical uncertainties associated with the 20 test samples have been discussed previously.

A sample set of potential failure progressions and recommended actions for a target reliability of 0.95 is shown in Table 1.

CONCLUDING ACTIONS

The resolution of GSI B-56 will be based on implementation of a reliability program similar to that discussed above. NUMARC-8700 (the industry's guidelines to evaluating station blackout capabilities) notes the need for a reliability program, and Regulatory Guide 1.155 identifies very similar elements. The staff plans to revise Regulatory Guide 1.9, to identify such a program and to incorporate certain recommended testing actions from Generic Letter 84-15, and to issue a proposed Revision 3 for public comment. NUMARC has formed a B-56 working group to interact with the staff in concluding B-56 and to ensure that successful aspects of existing EDG reliability programs are used.

EDG FAILURE EVALUATION CRITERIA
FOR EDGs WITH RELIABILITY TARGET OF 95%

Evaluation Criteria (# Failures/# Demands)	Combinations of Failure Evaluation Criteria	Time Period (1 Demand/2 Wks)	False Alarm Rate
≥ 2/20	Y Y Y N N N N Y	~ 10 Months	26%
≥ 5/50	Y Y N N N Y Y N	~ 2 Years	11%
≥ 10/100	Y N N N Y Y N Y	~ 4 Years	3%
Failure Progression	1 2 3 4 5 6 7 8		

Legend: Y = Yes
N = No

Interpretations of the Failure Progressions

Failure Progression	Interpretation
1. ≥ 2 failures in 20 demands ≥ 5 failures in 50 demands ≥ 10 failures in 100 demands	This is an unacceptable condition requiring immediate action to declare the EDG inoperable. There is strong evidence that the long-term EDG unreliability is larger than the target value and no evidence that it is improving. The EDG reliability program should be improved or enhanced before the EDG can be declared operable again.
2. ≥ 2 failures in 20 demands ≥ 5 failures in 50 demands < 10 failures in 100 demands	This is an alert condition where action is recommended to declare the EDG inoperable. There is evidence that the EDG is deteriorating over time and that the current reliability is unacceptable. The action taken may depend on other circumstances and information from the plant.
3. ≥ 2 failures in 20 demands < 5 failures in 50 demands < 10 failures in 100 demands	This is a mild alert condition where no action by the NRC is recommended unless there are other recent indications of EDG deterioration. EDGs with acceptable unreliabilities will display this condition about 26 percent of the time. Although some concern is justified, a single failure, with no evidence of degraded performance, should not lead to excessive concern.
4. < 2 failures in 20 demands < 5 failures in 50 demands < 10 failures in 100 demands	This is an acceptable condition. No concrete evidence of unacceptable performance.
5. < 2 failures in 20 demands < 5 failures in 50 demands ≥ 10 failures in 100 demands	This is an acceptable condition. There is an indication of a past problem that has probably been corrected. Low-level vigilance is prudent to ensure continued acceptable operation.
6. < 2 failures in 20 demands ≥ 5 failures in 50 demands ≥ 10 failures in 100 demands	This is an acceptable condition but one that needs continued vigilance. There is indication that a continuing past problem is being corrected, but the evidence is not convincing enough to warrant a decrease in vigilance.
7. < 2 failures in 20 demands ≥ 5 failures in 50 demands < 10 failures in 100 demands	This is an acceptable condition but one that needs continued vigilance. The interpretation of this condition is similar to the interpretation of condition 6 above, except that the history of unacceptable performance is less extensive.
8. ≥ 2 failures in 20 demands < 5 failures in 50 demands ≥ 10 failures in 100 demands	The interpretation of this condition is somewhat similar to the interpretation of condition 3, except that there is a history of a performance problem that may have been corrected, or partially alleviated. This situation is an ambiguous one, requiring a more detailed evaluation. The assessment would be different if there were 2 failures in the last 50 demands and 2 failures in the last 20 demands than if there were 5 failures in the last 50 and 2 in the last 20. An alert condition is indicated by this condition.

Table 1 EDG failure evaluation criteria

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Nuclear Management and Resources Council, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC-8700, Nov. 20, 1987.

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"REACTOR COOLANT PUMP SEAL FAILURES"

J. E. JACKSON

U.S. NUCLEAR REGULATORY COMMISSION

Failures of reactor coolant pump (RCP) seals that could result in a loss-of-coolant accident (LOCA) are of current concern to the NRC. This could occur either where leakage through the seals exceeded the capacity of the normal makeup systems, as has occurred in operating plants, or under station blackout conditions where loss of seal cooling represents a common mode failure for all RCPs.

Reactor coolant pump seals limit the leakage of reactor coolant along the pump shaft, directing the majority of this flow back to the chemical and volume control system with the remainder being directed to the reactor coolant drain tanks. In limiting the reactor coolant leakage to containment, the RCPs use a series of primary and secondary seals. Therefore, these seals become part of the reactor coolant system pressure boundary. The primary seals (metallic oxides, carbides and graphite) limit the leakage of reactor coolant across the interface between rotating and stationary RCP elements. The secondary seals (elastomer O-rings, U-cups and teflon channel seals) prevent leakage between stationary mechanical elements of the RCP seal or those elements which have only a slight relative motion. Both the primary and secondary seals require continuous cooling during pump operation and at hot shutdown conditions with RCPs stationary.

Excessive leakage resulting from RCP seal failures can occur as a result of loss of seal cooling or mechanical failures. In addition to the mechanical failures caused by the lack of adequate seal cooling, mechanical failures may result from other causes such as excessive pump vibration, defective parts, introduction of contaminants, high frictional torque, secondary seal failure, pressure, temperature or flow transients, improper maintenance, faulty assembly, and installation or adjustment. The RCP seal failures which have occurred to date have not resulted in a direct threat to health and safety of the public. However, the potential does exist for seal failures which could have significant safety consequences. Seal failures have occurred in which the loss of primary coolant to the containment was greater than the normal makeup capacity of the plant. The potential, therefore, exists for seal failures which can result in a small LOCA.

In all of the seal failures that have occurred to date, emergency makeup capability was available to replenish reactor coolant lost through seal leakage. However, none of these incidents involved complete loss of the component cooling water (CCW) system which provides cooling water to the seal cooling heat exchangers. On some plants, the high pressure coolant injection pumps are also cooled by the CCW system and cannot operate with the CCW system inoperable. Therefore, on complete loss of CCW, the equivalent of a small-break LOCA could occur, due to seal degradation, with no high pressure coolant injection pumps available for reactor coolant system makeup. This sequence of events could lead to core melt. Station blackout can also lead to core melt since station blackout results in a common mode failure of the RCP seal cooling systems through the loss of all A/C power.

In the early 1980's the staff made a preliminary review of Licensee Event Reports which indicated that a relatively large number of RCP seal and seal auxiliary system failures of varying degrees of severity had occurred at the operating plants. Some of the more significant seal failure events identified at that time in terms of quantity of coolant leaked to containment were as follows:

	Date of Event	Total Leakage/ Estimated Leakage Rate
Oconee 2	January 1974	50,000 gal/90 gpm
H. B. Robinson 2	May 1975	200,000 gal/300-500 gpm
Indian Point 2	July 2, 1977	90,000 gal/75 gpm
Haddam Neck 1	August 1, 1977	4020 gal
Salem 1	October 21, 1978	15,000 gal
ANO-1	May 10, 1980	60,000 gal/200-300 gpm
Brunswick 2	August 4, 1975	1,500 gal
Brunswick 2	September 5, 1975	2,600 gal

The Reactor Safety Study, WASH-1400, published in October 1975 indicated that breaks in the reactor coolant pressure boundary in the range of 0.5 to 2 inches in diameter occurred with a frequency of 10^{-3} per reactor-year and contributed the largest probability to PWR core melt. A 1980 staff study based on RCP seal failures experienced at operating plants showed that RCP seal failures, with leak rates equivalent to those of small-break LOCAs, had actually occurred at a frequency of about 10^{-2} per reactor year, an order of magnitude greater than the pipe break frequency used in WASH-1400. The conclusion reached was that the overall probability of core melt due to small breaks could be dominated by RCP seal failures.

As a result of these concerns, the staff assigned a high priority to the investigation of RCP seal failures (NUREG-0933, "A Prioritization of Generic Safety Issues" dated November 10, 1982). The NRR Operating Plan for FY 83 included the review of RCP seal failures as Generic Issue 23, "Reactor Coolant Pump Seal Failures" and authorized work in October, 1983. Another related task, Generic Issue 65, "Probability of Core Melt Due to Component Cooling Water System Failures" was also assigned a high priority. Because of the close relationship between GI-65 and GI-23, Generic Issue 65 was incorporated into the task action plan for Generic Issue 23. In addition, because of the dependence of RCP seal cooling on AC power supplies, RCP seal failures are linked to the reliability of onsite and offsite electrical supplies. These concerns were heightened by the Indian Point and Zion probabilistic safety studies station blackout results, staff comparisons between Westinghouse SNUPPS design and the British Sizewell-B plants, and the fact that both the French and British systems use steam driven power sources to maintain seal cooling under station blackout conditions. Consequently, it was decided that Generic Issue 23 would consider the effects of station blackout on RCP seal performance to the extent they were not addressed by the unresolved safety issue A-44, "Station Blackout."

The purpose of Generic Issue 23 is to evaluate the adequacy of current licensing requirements relating to RCP seal integrity and to determine if further NRC action is necessary to assure that RCP seal failures and seal auxiliary system failures do not pose an unacceptable risk. This generic issue has two main objectives, 1) determine the need to improve the reliability of RCP seals during normal operations and 2) preventing small-break LOCAs resulting from RCP seal failure during station blackout.

The technical work on Generic Issue 23 is essentially complete. The remaining documents to be completed are the regulatory analysis of the proposed resolutions including a cost/benefit analysis of these resolutions, and the technical findings document which summarizes all the technical findings to date in one document.

The major sources of information for resolving Generic Issue 23 were as follows:

1. Idaho National Engineering Laboratory/Atomic Energy of Canada Limited.

   a. "Reactor Coolant Pump Shaft Seal Behavior During Station Blackout", NUREG/CR-4077, April 1985.

   b. "Reactor Coolant Pump Shaft Seal Stability During Station Blackout", NUREG/CR-4821, May 1987.

2. Atomic Energy of Canada Limited.

   a. "Report on the EDF-Montereau Full Scale Test of RCP Seals Under Station Blackout Conditions", NUREG/CR-4907P, July 1985.

   b. "Review of the Westinghouse Owners Group Report WCAP-10541, Revision 2, "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power", NUREG/CR-4906P, January 1988.

3. Energy Technology Engineering Center.

   "Leak Rate Analysis of the Westinghouse Reactor Coolant Pump", NUREG/CR-4294, July 1985.

4. Brookhaven National Laboratory.

   a. "The Impact of Mechanical- and Maintenance-Induced Failures of Main Reactor Coolant Pump Seals on Plant Safety", NUREG/CR-4400, December 1985.

   b. "Evaluation of Core Damage Sequences Initiated by Loss of Reactor Coolant Pump Seal Cooling", NUREG/CR-4643, August 1986.

   c. "Reactor Coolant Pump Seal Related Instrumentation and Operator Response", NUREG/CR-4544, December 1986.

d. "Indian Point 2 Reactor Coolant Pump Seal Evaluations", NUREG/CR-4985, August 1987.

e. "Technical Findings Related to Generic Issue 23: Reactor Coolant Pump Seal Failure", NUREG/CR-4948, not published at this time.

5. Scientech, Inc.

"Cost/Benefit Analysis for Generic Issue 23 "Reactor Coolant Pump Seal Failures", NUREG/CR-5167, not published at this time.

6. Westinghouse.

a. "Westinghouse Owners Group Report; Reactor Coolant Pump Seal Performance", WCAP-10541, Revision 2, November 1986 (plus Supplements 1 and 2).

b. Numerous meetings of the NRC staff with Westinghouse and the Westinghouse Owners Group.

7. Meetings of the NRC staff with Babcock & Wilcox, Combustion Engineering, Byron Jackson Pump Division and Bingham International.

The various research programs have identified a need for improving quality control over seal materials and fabrication, installation and maintenance, as well as seal operations. There is also a need for improving instrumentation and monitoring capabilities to identify degraded seal performance early enough to take corrective action to mitigate seal failure. In the area of station blackout research, certain secondary seal materials were found to be inadequate to survive the conditions of station blackout in a functional condition. Also, "popping open" has been identified as the most serious seal failure concern under station blackout conditions. Seal "popping open" can occur due to seal face flashing, increased axial seal friction or partial extrusion and jamming of the axial seal.

Early in the resolution of USI A-44, the station blackout issue, it was necessary to make certain assumptions in order to proceed with the resolution. One of these assumptions defined the interface with Generic Issue 23, namely that the RCP seals would leak no more than 25 gpm per pump during station blackout conditions. It was assumed that GI-23 would study the station blackout RCP seal failure issue and determine that the expected leakage would be 25 gpm or less or, as an alternative GI-23 would take the necessary regulatory actions that would limit the leakage to this value.

Actual testing of the RCP seals under station blackout conditions has been very limited to date. The complete reactor coolant pump including the seal package has not been tested under controlled station blackout conditions for an adequate duration. The major test events which are applicable to station blackout have been; 1) test of Byron-Jackson St. Lucie seal cartridge, 2) test of the French 7-inch Westinghouse type RCP seal package, and 3) some isolated loss of cooling events of short duration, mainly during plant startup testing.

Therefore the majority of information regarding the RCP seal performance under station blackout conditions has come from analysis and scale model testing of seal components. In addition, the majority of this research has been aimed at the Westinghouse hydrostatic type RCP seal which has been considered the most likely to have large leakage under station blackout conditions. The Westinghouse seal is a high leak rate seal under normal cooled conditions (3 gpm), the majority of the pressure drop is taken across the first stage (2000 psi), the present O-ring material used has shown a high probability for failure (NUREG/CR-4077) and the seal package must be properly managed in order to operate in an optimum manner for a station blackout event.

Studies by Westinghouse (WCAP-10541, Rev. 2), however, claim the probability of RCP seal failure during station blackout is negligibly low, if a better high temperature material is used in conjunction with a proper Q/A program. However, a more recent study (NUREG/CR-4821) has shown a potential for failure in the hydrodynamic type RCP seals, such as, Byron-Jackson, Bingham and CE-KSB. Therefore the regulatory decisions to preclude RCP seal failure during station blackout have been made based on analysis, probabilistic risk assessment, risk estimates and only limited test results.

Three alternative resolutions are being evaluated as candidates for resolution of Generic Issue 23.

Alternative 1

Under the provisions of Alternative 1, the RCP seals will be defined to be part of the primary reactor coolant pressure boundary, with the appropriate QA/QC requirements. The adoption of Alternative 1 will mean a tighter system of quality control over materials and fabrication methods used for the manufacture of RCP seals, to ensure that all RCP seals are capable of meeting specified performance requirements. Also increased control over the installation and maintenance of RCP seals will ensure that seal integrity is not compromised by such actions. This control will be realized through the use of detailed procedures for RCP seal installation and maintenance. Also, increased procedural control over the operation of the RCP pumps will ensure compliance with manufacturer specifications, particularly during startup and shutdown when the seals are most susceptible to damage.

Alternative 2

Alternative 2 will require additional instrumentation where necessary in order to fully monitor the RCP seal performance with alarm capability and on-line analysis of certain functions as required. This requirement will increase efforts to detect incipient RCP seal leaks in time to take corrective action and insure the seal is operated only within the manufacturer's design envelope of intended operation.

Alternative 3

Provide an independently powered method of cooling RCP seals during station blackout conditions. This system will be capable of maintaining seal temperatures under specified limits for the duration of the blackout. This will ensure the survival of the RCP seals during station blackout or on complete loss of component cooling water.

In accordance with the Commission requirements on backfitting (10 CFR 50.109) the alternatives will be evaluated on a cost/benefit basis. Upon completion of the evaluation of these alternatives, a regulatory decision will be made giving the preferred approach and the requirements for implementation.

INTEGRATION OF GENERIC ISSUES

by Dale Thatcher
U. S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Engineering Issues Branch

Abstract

The NRC has recognized the need to integrate generic issues (GIs). The GI process includes a number of phases, all of which should recognize the potential for overlap and conflict among related issues. In addition to the issues themselves, other related NRC and industry programs and activities need to be factored into the GI process.

Integration has taken place, or is taking place, for a number of GIs. Each case of integration involves a specific set of circumstances and, as a result, the way in which integration proceeds can vary.

This paper discusses the integration of issues in the generic issue process and provides a number of examples.

Introduction

The resolution of a generic issue (GI) has the potential to lead to a significant investment of resources by both the NRC and the utilities. With the large number of GIs, and the possible addition of more in the future, the NRC has recognized the need to ensure that the issues and their resolutions do not overlap or conflict. For example, two or more related issues could lead to similar resolutions and, if addressed separately, significant overlapping work could result. A worse case would be where previous work to resolve an issue would prove later to have been completely unnecessary or that a major portion of the effort involved would have to be repeated.

A generic issue is an issue that is applicable to all, several, or a class of reactors or reactor-related facilities. Generic issues can arise from various concerns and, accordingly, are classified into one of the following four categories. A generic safety issue (GSI) is a generic issue that involves a safety concern that may affect the design, construction, operation, or decommissioning of all, several, or a class of reactors or facilities and may have a potential to require licensees to make safety improvements and/or require the issuance of new or revised requirements or guidance. A regulatory impact issue is a generic issue not related to improving safety, but to modifying current NRC requirements or guidance, with the primary purpose of reducing the regulatory impact, usually cost, of requirements on licensees or applicants. An environmental issue is a generic issue involving impacts on those items protected by the National Environmental Policy Act (NEPA). A licensing issue is a generic issue related to actions the NRC staff could take to increase knowledge, certainty, and/or understanding in order to increase confidence in assessing levels of safety; improve or maintain the NRC capability to make independent assessment of safety; establish, revise, and carry out programs to identify and resolve safety issues; document, clarify, or correct current requirements and guidance; or improve the effectiveness or efficiency of the review of applications.

This discussion is mainly concerned with the generic safety issues. Some GSIs have been designated unresolved safety issues (USIs) based on their receiving a HIGH priority ranking and a recommendation that the issue be a USI based on criteria documented in NUREG-0705 (Ref. 1). This discussion of the "Integration of Generic Issues" uses the general term, generic issue, but focuses on generic safety issues and includes USIs under that broader category.

The Generic Issue Process

The generic issue process consists of six phases: identification, prioritization, resolution, imposition, implementation, and verification. The integration of generic issues as discussed here mainly deals with the first three phases, i.e., identification, prioritization, and resolution. However, the integration of the actions resulting from the resolution of GSIs can continue into the imposition, implementation, and verification phases.

Potential generic issues may be suggested by organizations or individuals within the NRC, the Advisory Committee on Reactor Safeguards (ACRS), the nuclear power industry, or the public. Generic issues may also be suggested as an outcome of reactor research programs. These identification and the subsequent prioritization procedures (Ref. 2) were developed to provide a consistent method to document new safety concerns with existing and future reactors and to have the staff formally evaluate these concerns for safety significance and appropriate action.

The identification and prioritization phases of GIs were formally implemented through the publication of NUREG-0933 (Ref. 3). The original preparation of 0933 involved evaluation of a backlog of issues that had been identified up to that time. With the issuance of 0933, the staff created a single document to catalogue GIs. As stated in the introduction section to 0933, issues are often complex and usually interrelated with other issues, and therefore careful definition of an issue's scope and bounds is essential. This will often highlight the possible interfaces and relationships with other existing GIs. In the simplest case, a detailed description and consideration of related issues may show that the proposed issue is identical to or part of an existing issue. In that case, there would be no need to repeat the issue.

During the original preparation of NUREG-0933, some issues were identified as being identical to other issues. This type of duplication was largely due to the fact that at that time the generic issues often existed in different documents. Since this original cataloguing is over, this type of duplication is now minimal. Specifically, the identification and prioritization procedures (Ref. 2) state that during the identification process the staff will screen proposed GIs for duplication or overlap.

Although the identification and prioritization phases involve the type of integration mentioned above, the most significant aspects of integration occur during the resolution phase. Even though issues may be designated as separate issues, almost all are related to other issues. These interrelationships need to be recognized during the resolution process.

The procedures for the resolution phase of generic issues (Ref. 4) emphasizes the importance of consolidation and integration of issues and their resolution to achieve the most safety benefit. All issues need to be integrated, however the initial part of the resolution process includes a quick review to determine if an issue may be best handled by combination with other generic issues.

Based on the results of this quick review, the process of preparing a plan to resolve the issue begins. The plan (often referred to as a Task Action Plan) includes a description of the concern, including background and history. Also to be included is a discussion of the relationship to other generic issues and programs. In fact, the procedures and guidance for resolution (Ref. 4) repeatedly emphasize that the integration and coordination of the resolution of GIs with other GIs, other NRC activities, such as generic licensing actions and research programs, and outside activities are essential.

The prime responsibility for the integration and coordination during the resolution phase clearly lies with the Task Manager of the issue. To quote the guidance (Ref. 4):

> "The Task Manager must take the initiative to seek out all related issues and programs, assure coordination and integration, resolve differences and elevate inconsistencies when necessary."

As a GI progresses through the resolution process described in the Task Action Plan or other plan document, the Task Manager must periodically review the status of related generic issues, other NRC activities, and outside activities to continually ensure coordination. As a regulatory analysis is prepared, these relationships with other issues and activities must be included.

The focus of this discussion is the NRC process of integration as it relates to the generic issues among other generic issues; however, as mentioned, the overall integration of GIs takes into consideration other NRC activities and industry activities. For example, the Office of Nuclear Reactor Regulation (NRR) has the responsibility for issuing generic communications to the licensees. To keep the Institute of Nuclear Power Operations (INPO) abreast of generic communications being considered or under development by the staff, the Generic Communication Branch provides a biweekly listing of future bulletins, information notices and generic letters. The listing to INPO receives wide distribution inside NRC. This provides a good reference document for Task Managers to track other NRC activities that might be related to the resolution of their generic issues.

A recent example of the effectiveness of this tracking system involves Generic Issue 51, "Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems," and Generic Issue 130, "Essential Service Water Pump

Failures at Multiplant Sites." Specifically, the Office of Nuclear Regulatory Research, NRR, and the Office of the Analysis and Evaluation of Operational Data (AEOD) are presently coordinating the resolution of these two GIs with the concern raised in an AEOD case study on service water systems. Potentially one generic letter will be issued by NRR.

USI A-44, "Station Blackout", which is discussed later, is a case where industry initiatives were integrated into the resolution of a GI.

Examples of Issue Integration

The identification, prioritization, and resolution of GIs has involved a number of cases of integration. Integration involves special sets of circumstances and, for purposes of discussion, the examples have been divided into four categories. It should be noted that although issues may be integrated with one another, each issue retains its identity in NUREG-0933 for future reference.

1. Subsuming of one issue into another issue

   When a generic issue is defined to involve a fairly focused concern, there is the potential that another broader issue can cover the issues by including the more focused concern into the broader concern. This has been described as "subsuming" one issue into another.

   The decision to subsume one issue into another issue can be straight forward, but it can also present problems. The resolution process often involves a number of draft proposals that are subject to revision. The Task Manager must ensure that the final resolution does continue to cover the subsumed issues.

   USI A-17, "Systems Interactions in Nuclear Power Plants," is by its title a potentially very broad issue. All nuclear power plants include many systems that interact by design. As a result of extensive evaluation of this issue, the staff focused the major concerns to a limited number of areas. One of these areas involved the concern for disabling vital equipment by water intrusion and internal flooding. A specific aspect of the flooding and water intrusion concern is the possibility that some subtle pathways may not have been adequately considered in previous analyses performed by licensees.

   As this concern was evaluated further, the potential direct relationship with existing Generic Issue 77, "Flooding of Safety Equipment Compartments by Back Flow Through Floor Drains," was identified. GI 77 was concerned with the more focused area of the plant drains as a pathway for communication of water and moisture and had been given a separate priority ranking of HIGH. The drains were one of the pathways identified as a concern in USI A-17.

   Because of the broad nature of A-17, the decision to subsume GI 77 was delayed until a clearer resolution for A-17 was defined. After the draft proposed resolution for A-17 was prepared, which included the specifics of GI 77, the decision to subsume GI 77 into A-17 was made.

In this case, it was necessary to wait until this rather late stage because of the very broad nature of USI A-17 and the possibility that the proposed resolution would not include enough specific treatment to be able to subsume GI 77.

USI A-45, "Shutdown Decay Heat Removal Requirements," was initiated to evaluate the safety adequacy of the decay heat removal (DHR) function in currently operating light-water reactors and to assess the value and impact of alternative measures to improve the overall reliability of the DHR function. Extensive analysis was performed on six plants and various alternatives were evaluated. One of the alternatives considered was to have plants perform risk assessments of the plant's DHR function.

The Commission is currently planning to implement the severe accident policy and will issue a generic letter to require all plants currently operating or under construction to undergo a systematic examination termed the Individual Plant Examination (IPE) to identify any plant-specific vulnerabilities to severe accidents. The IPE analysis is intended to examine and understand the plant emergency procedures, design, operations, maintenance, and surveillance to identify vulnerabilities. The analysis will examine both the systems used for the DHR function as well as systems used for other functions. It is anticipated that a future extension of the IPE program will require examination of externally initiated events, some of which significantly contribute to DHR failure-related core damage frequency. Therefore, it was decided to subsume A-45 into the IPE program and its anticipated extension as the most effective way of achieving resolution of A-45.

As an additional note, the IPE generic letter also includes the option that a utility may chose to address any other generic safety issue (including USIs) as part of their IPE. This provides an opportunity to integrate a large number of generic issues into one program.

2. <u>Coordinating of related issues to avoid overlap in their separate treatment</u>

A Task Manager's review of issues related to his or her issue may conclude that the resolution may best proceed with separate treatment of the related issues and a clear delineation of the bounds and interfaces. This coordination approach to integration can best be used in cases where the issues are clearly defined and overlap can be clearly avoided.

For this approach, and depending on the schedule for resolution of the related issues, it may be necessary to make some assumptions about the resolution of the related issues. Then care must be taken to ensure that the other issues do indeed meet the assumptions. A contingency must be defined for the case where the interface assumption is not met.

USI A-44, "Station Blackout," involves the concern for the possibility that a nuclear power plant would lose all offsite ac power sources and the onsite diesel generators would fail to provide ac power. The evaluation of this issue required a number of assumptions about the performance of certain plant systems and components. In a number of instances, the plant systems and components are the subject of other related generic issues.

The three related generic issues are:

a. GI B-56, "Diesel Reliability,"
b. GI 23, "Reactor Coolant Pump Seal Failures," and
c. GI A-30, "Adequacy of Safety-Related DC Power Supplies."

The resolution of A-44 assumes that a plant has reliable diesel generators, but does not have any specific requirements for ensuring diesel reliability. GI B-56 is addressing that aspect separately.

The analysis assumes that under blackout conditions, the reactor coolant pump seals will not create a loss of coolant that could result in core uncovery during the plant specific coping period. This assumption is to be factored into the resolution of GI 23 involving reactor coolant pump seal failures. One of the more significant potential seal failure scenarios involves the station blackout sequence.

The analysis also assumes that safety-related dc power will be available to supply power during the blackout. Because the blackout conditions can place greater demand on the dc power supplies than may have been assumed in their original sizing, the coping analysis for A-44 must demonstrate that sufficient capacity is available. However, GI A-30 is exploring potential improvements to ensure the availability of battery power under all plant conditions, not just station blackout. The improvements involve aspects of maintenance, surveillance, and monitoring. In this regard, the A-30 resolution can support the assumption of the A-44 resolution.

In addition, industry actions were also factored into the resolution. The resolution of A-44 requires that a plant perform a coping analysis for a station blackout condition. Guidelines for the coping analysis on this issue have been prepared by the industry group, and an NRC regulatory guide has been prepared to endorse that guidance.

3. <u>Combining the resolution of two or more issues into one set of requirements to be implemented</u>

In some instances, separate generic issues may be proceeding to resolution independently; however, each may involve the same subject area such as the same plant structure, system, or component. If both are proceeding within the same time frame, it may be possible to combine their proposed resolutions into one set of requirements and guidance. However, if one issue turns out to involve significant delays, it can potentially hold up the resolution of the other issue(s). At that time, it may be necessary to make a decision to separate the resolutions.

Generic Issue 70, "PORV and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection," were being handled separately, but both involved the potential for new guidance on the power-operated relief valves. As the proposed resolutions were being drafted, it was considered desirable to issue a single generic letter and simplify the implementation for both NRC and the licensees. A single draft generic letter now includes separate enclosures, one for the staff position for GI 70 and one for the staff position for GI 94.

4. <u>Combining two or more issues into a single issue or program to address all the individual issues</u>

In a few instances, the staff has determined that a number of issues, originally prioritized for separate resolution, could best be addressed by a single program. In these instances, the individual issues involve the same subject area (e.g., same system or component) and all the issues have already received a priority ranking of at least MEDIUM.

The objective of creating one program is to provide a more cohesive and logical approach to the resolution of all the combined issues. As the program proceeds, it may be possible to include all the resolutions in one set of actions such as in one generic letter.

GI 128, "Electrical Power Reliability," is an example of a case where a group of issues were integrated or combined into one issue for the purpose of addressing them in one program. The separate issues, which all involve the electrical power system, are:

GI 48, "LCOs for Class 1E Vital Instrument Buses,"
GI 49, "Interlocks and LCOs for Class 1E Tie Breakers," and
GI A-30, "Adequacy of Safety-Related DC Power Supplies."

GI 48 involves the low-voltage ac instrumentation power supplies and the possibility that they could be operated in modes that could place the plant in situations not considered in the analysis of the plant. The potential solution was believed to be an implementation of technical specifications for limiting conditions of operation (LCOs).

GI 49 also involves the electric power systems and a concern that connections between redundant and independent electrical buses may be left closed. This issue also had a potential solution involving the plant technical specifications. GI 49 had been given a MEDIUM priority.

GI A-30 involves the safety-related dc power supplies and includes concerns that these important power supplies appeared to have some potential for common cause failures that could disable redundant and independent dc power supplies. A number of possible improvements in maintenance, surveillance, and monitoring had been identified. In addition, tie breakers were identified as a possible compromise for independence of the dc sources. This aspect clearly overlapped with GI 49.

With the common subject area, potential solutions with common aspects, and the potential overlap, it was decided to integrate these issues into one program, GI 128. At present, GIs 48 and 49 are to be addressed in a generic letter. GI A-30 will probably be addressed by a separate generic letter.

Summary

The generic issue process involves the integration of a variety of issues. The individual circumstances surrounding the issues can dictate the ways in which integration proceeds. The resolutions of the GIs also integrate aspects of other NRC activities and industry activities where possible. The staff recognizes the potential downside of integration. The process balances the benefits of combining issues with the need to proceed with generic issues in a timely manner, and the potential for added delays.

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A Value-Impact Assessment of Potential Upgrades to Control Room Annunciators

J. Higgins, D. Crouch, and W.J. Luckas, Jr.
Brookhaven National Laboratory

ABSTRACT

A human factors analysis was performed to assess the importance of identified upgrades or improvements to nuclear power plant control room annunciators. The effect of these upgrades on human performance during accident scenarios was also analyzed, and a value-impact assessment was performed.

1. INTRODUCTION

1.1 General Discussion

Typical control room annunciator systems in nuclear power plants (NPPs) serve an important service to alert and inform operators in a timely fashion of certain circumstances which warrant special attention and possible further consideration. Since the Three Mile Island accident almost a decade ago, a number of potential upgrades to present-day, hard-wired NPP control room annunciator systems have been identified. An assessment of the safety significant benefit of specific annunciator upgrades needs to be explored in terms of their potential for human reliability improvement. When the cost associated with implementing the upgrades is determined, a value-impact assessment can be made to determine the feasibility of implementing these upgrades at all commercial nuclear power plants.

1.2 Background

The accident at the Three Mile Island (TMI) nuclear power plant confirmed for the nuclear industry that there were a number of significant problems associated with their control rooms. Analysis of this accident indicated that the annunciators were of limited use to the operators during the first 2 1/2 hours of the accident (NUREG/CR-3217).

Annunciator system problems identified as a result of the TMI accident included: an overwhelming amount of information was available to the operator during the accident, much of which was not useful; the annunciator tiles and legends did not have standard formats or designs; many annunciators were positioned in less than optimal areas which limited their usefulness; there were too many alarmed conditions which occurred at the same time; and perhaps most significantly, the annunciator lights continued to flash and the auditory alarms continued to sound which distracted operators and reduced their ability to diagnose the event.

As a result of the TMI accident, the Nuclear Regulatory Commission (NRC) developed the TMI Task Action Plan (NUREG-0660). The purpose of this plan was to provide both a systematic and detailed approach to improve the overall safety level of commercial NPPs. One portion of the Action Plan established a requirement for control room design reviews at NPPs. A follow-up publication, "Guidelines for Control Room Design Reviews," (NUREG-0700), provided the necessary guidance to carry out these Detailed Control Room Design Reviews (DCRDRs). The DCRDRs were conducted at all licensed NPPs as well as at those awaiting licensing.

These DCRDRs identified a number of human factors related design deficiencies in control rooms. Improvements identified have not yet been completed at all NPPs. Many of the identified deficiencies were subsequently classified as generic issues, indicating that more research and evaluation was needed before improvements were mandated at all NPPs.

One such generic issue was human factors generic issue 5.2, which currently addresses annunciator improvements or upgrades to control rooms.

1.3 Upgrades for Annunciator Systems

Much of the work done to date with annunciator systems is described in a report entitled, "Near-Term Improvements for Nuclear Power Plant Control Room Annunciator Systems," (NUREG/CR-3217). This report provided guidance to utilities for near-term improvements in their conventional, hard-wired control room annunciator systems consistent with the guidelines presented in NUREG-0700. It did so by providing a philosophy for the annunciator system, as well as the functional criteria and design principles necessary to implement the philosophy. The report also identified a number of potential upgrades which could help to improve the NPP annunciator systems. NUREG/CR-3217 describes a prioritization of the upgrades and an evaluation on the ease of implementation of the upgrades, determined by cost estimates and hardware considerations. As requested by the NRC, this study was used as a beginning point for the current project, in that the specific annunciator upgrades listed in Section 3 and considered as part of this value-impact assessment are those identified in NUREG/CR-3217.

1.4 Organization of Report

Section 1 of this paper defines the problem and provides a background into the issues surrounding this problem. Section 2 states the objective of the current work. Section 3 defines the specific annunciator upgrades under consideration. Section 4 details the methodology for the project. Sections 5, 6, and 7 provide the details of the implementation of the project in accordance with the methodology. Section 8 provides conclusion and Section 9 lists the references.

2. OBJECTIVE

The objective of the research in this project was to evaluate the human factors safety issues associated with upgrades to control room annunciator systems at commercial nuclear power plants and to perform a value-impact assessment for these upgrades. The results of the value-impact assessment are also evaluated. This effort is important to provide a scoping determination of the impact on public health and safety of these upgrades and to also obtain an appreciation of how cost-beneficial the upgrades appear to be. The project also provides a human factors related prioritization of the upgrades and identifies further actions necessary to resolve these issues.

3. DEFINITION OF ANNUNCIATOR UPGRADES

This section presents the upgrades from NUREG/CR-3217, which were considered for detailed analysis in this project. Many of the upgrades, as presented in NUREG/CR-3217, can be implemented in a number of different ways. The manner in which the upgrade is implemented can depend on many things, including the perceived importance of the upgrade, the cost of adding the upgrade, and the difficulty associated with learning to use the upgrade.

An alphabetical list of the upgrades and a description of how it would be implemented for this study follows.

Auditory Signal Intensity: An annunciator upgrade which allows the operator to control the auditory signal loudness in order to ensure that the auditory signal is always 10dBs above ambient noise level.

Blackboard for Normal Operations: An annunciator upgrade which requires tiles to be lit at normal full power operations only when there is a real problem (i.e., correct unnecessarily lit tiles).

Controls (Separate Silence and Acknowledge): This allows the operator the capability of silencing the auditory signal while still allowing the annunciator tile to flash by providing a separate control for both silence and acknowledge.

Elimination of Grouping: Grouping is a design feature where more than one alarm point is grouped within a single alarm tile. This upgrade calls for separate alarm tiles for each alarm point.

Fail-Safe: An annunciator upgrade that ensures that all annunciator tile lights are working as they should. This is accomplished by actuating the test switches once every four hours. (An alternate method is double light bulbs.)

First-Out Panel: This upgrade requires a separate panel for reactor and turbine trips on which only a tile lights, indicating the initial trip parameter.

Inhibit: A design feature that keeps various alarm points from operating while other alarm points are tripped. This feature includes the elimination of nuisance alarms (i.e., tank level low inhibited when tank level low/low annunciated). It also includes the use of a silence control for auditory alarms during major transients. (More sophisticated inhibit schemes appear potentially useful, but were not considered here.)

Keying Procedures to Tiles: An upgrade that ensures that annunciator tiles are keyed to the alarm response procedures in a manner that is straight forward and does not increase the operator's workload (e.g., not needing to use an index).

Prioritization: An annunciator upgrade that allows the operator to focus attention on the most critical annunciated conditions and leave the less important alarms until the situation has stabilized through the use of a color coding scheme.

Reflash: An annunciator upgrade that can be implemented any time one annunciator tile receives input from more than one function (grouping). With one function having alarmed the tile, reflash allows the alarm tile to reinitiate when another function associated with the same tile deviates from it's unalarmed state. The tile cannot return to normal until all related functions return to normal. (NOTE: You cannot have reflash unless you have grouping).

Relocation of Tiles: Involves moving annunciator tiles so they are located in such a way that they are easily visible from the relevant control/display panel. This is necessary only where an immediate response to an alarmed condition is needed.

Ringback: An annunciator upgrade that informs the operator that an annunciated condition has cleared by providing a distinct indication that the process or system condition that has tripped an alarm point has returned to normal. The operator must take action to reset the alarm after it ringbacks.

Tile Legibility/Intelligibility: An annunciator upgrade which ensures that all annunciator tiles are legible and intelligible from the operator's work position (consistent abbreviations and inscriptions, permanent labels, large lettering, etc.).

Table 3.1 below, extracted from NUREG/CR-3217, summarizes these upgrades and also gives the functional prioritization and ease of implementation determined in this NUREG/CR-3217.

Table 3.1. Upgrades to be Evaluated and the Priority and Ease of Implementation Established for Each by NUREG/CR-3217

Upgrades	**Priority**	**Ease of Implementation**
First-out Panel	1,3	1-2
Reflash	1	1
Inhibit	1	1,3
Prioritization	1	1,3
Fail-Safe	1	1,3
Auditory Signal Loudness	1	1
Tile Legibility/Intelligibility	1	1
Keying Procedures to Tiles	1	1
Relocation of Tiles	1	2
Controls (Separate Silence and Acknowledge)	2	1-3
Ringback	2	1,3
Black Board for Normal Operations	2	2-3
Grouping	3	3
Flashrate	3	1,3

Table 3.1. Continued

Priority:	Ease of Implementation:
1 = needed to meet functional criteria	1 = relatively easy/inexpensive to implement
2 = helps to meet functional criteria but is not essential	2 = moderately hard/expensive to implement
3 = helps minimally in meeting functional criteria	3 = hard/expensive to implement

* This table was taken directly from NUREG/CR-3217, Appendix B, Table B1, "Priority of the Upgrades and Ease of Implementing the Upgrades."

4. METHODOLOGY

In order to meet the objective of this project, a methodology was established, whereby the effects of the various annunciator upgrades could be quantified. The methodology involved a number of various stages which are depicted in Figure 4.1. Several of these stages were able to proceed in parallel until they needed input from another stage of the project. The overall methodology will be briefly described here using Figure 4.1, with more detailed descriptions provided in the following sections.

Path #1

Due to the somewhat limited scope of the project, a determination was made to select one nuclear power plant (NPP) with an acceptable probabilistic risk assessment (PRA) to use as a sample demonstration of the value of the upgrades. The Sequoyah NUREG-1150 PRA was selected. The human errors (HEs)and their associated human error probabilities (HEPs) were extracted from the PRA for later use. At the same time, contacts were made with both NRC and Tennessee Valley Authority (TVA) personnel familiar with the Sequoyah annunciator system. TVA sent detailed drawings to BNL. This allowed BNL to determine which annunciator upgrades Sequoyah already had implemented (either in new design or as a backfit) and also which HEs from the PRA would potentially be affected by use of the annunciator system. This Path is discussed in detail in Section 5 below.

Path #2

The process of quantitatively determining the effect of hardware modifications, such as annunciator upgrades, on human performance is a difficult and necessarily subjective one. This path thus forms the major portion of the methodology. The first step in the process was the determination of human factors criteria by which the individual upgrades could be quantified. Using these criteria, an expert panel was then established to actually quantify the upgrades. This was done using the PC-based computer program SLIM-MAUD (Success Likelihood Index Methodology - Multi-Attribute Utility Decomposition). As a result of this quantification, the individual upgrades were placed into a prioritized list and the most important upgrades from a human performance standpoint were determined.

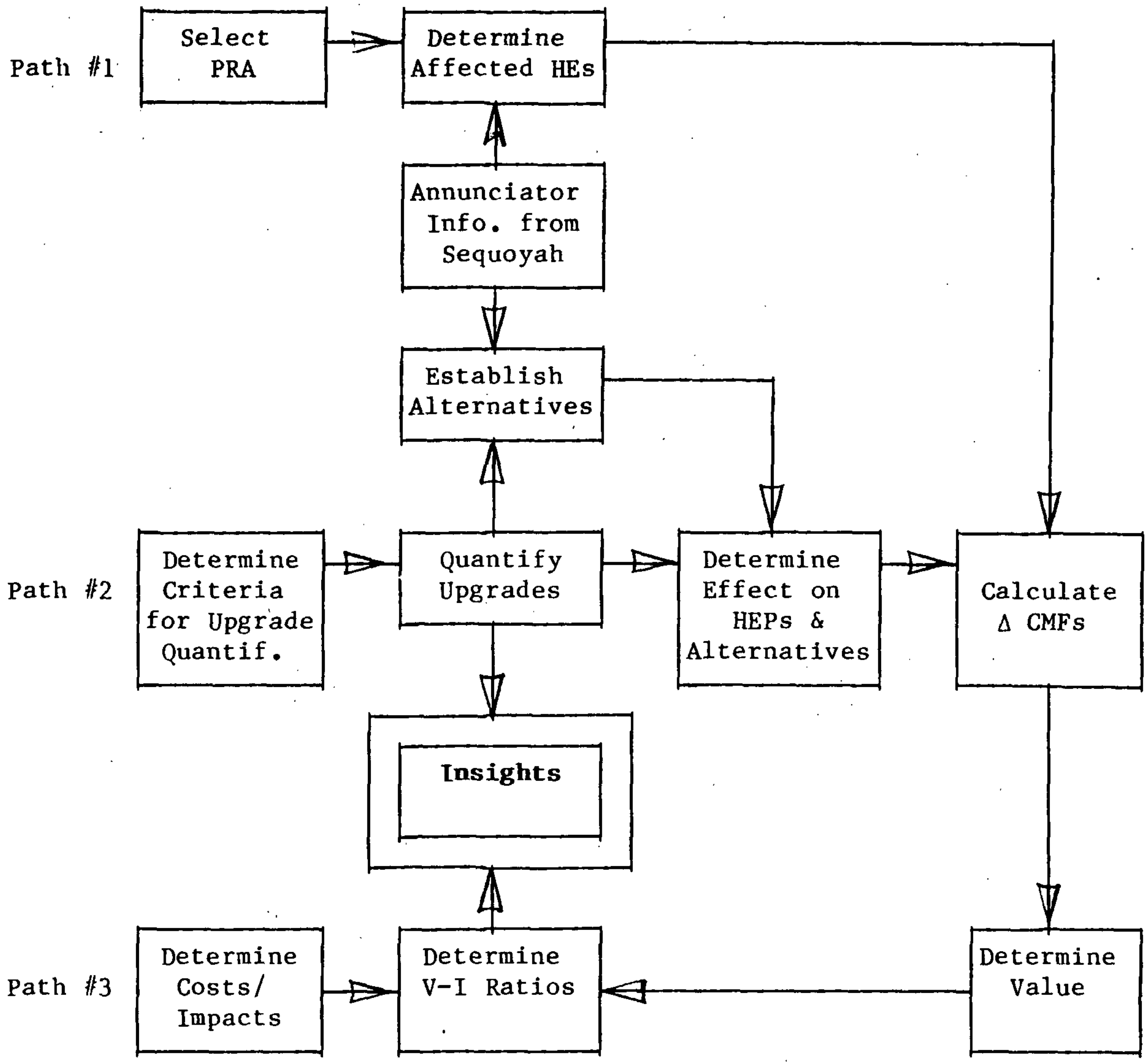

Figure 4.1. Methodology outline

Part of the value-impact analysis process is to establish alternative proposals for implementing the desired changes and then evaluating quantitatively the various alternatives. In this case, groups of the 14 upgrades were combined into alternatives, taking into consideration the upgrades that Sequoyah already had and also the prioritization schemes both in NUREG/CR-3217 and from this project. Three alternatives were developed.

Next, using the quantified list of upgrades, scaling factors were established to define the limit of the effect that the upgrades could have on the human error probabilities (HEPs) from the PRA. Then, based on the quantified list and the scaling factors, the actual effect of each of the three alternatives on the HEPs of the PRA were determined. Once new HEPs were available, revised core melt frequencies (CMFs) could be calculated using the PC-based PRA model available for Sequoyah. Change in CMF (ΔCMF) from the base case was also computed for each of the three alternatives cases. This path is discussed in detail in Section 6.

Path #3

A generic conversion factor for Pressurized Water Reactors (PWRs) was used to convert the ΔCMFs from Path #2 into offsite public dose as measured in person-rem. In all three cases, the CMF decreased, resulting in a less offsite dose to the public. These values of averted public dose were the value portion of the analysis.

A scoping type impact or cost analysis was also performed. In it, the costs for each upgrade were separately estimated. These were then combined to obtain the cost or impact of each of the three alternatives. Finally, these results were combined with the results of the Value analysis to determine the Value-Impact (V-I) ratios. These ratios were compared with the standard baseline of $1,000/person-rem. It was found that they all met this test, but some alternatives appeared more favorable than others. This Path is discussed in Section 7.

5. PLANT AND PRA SELECTION

To assess the safety significance of upgrades to the control room annunciator systems, a probabilistic risk assessment (PRA) of a nuclear power plant (NPP) was used as a sample application. The selection of the PRA and sample NPP are described in this section, which generally corresponds to Path #1 of the Methodology described in Figure 4.1.

5.1 Selection of PRA Application Type

To facilitate the analysis of annunciator upgrade impacts on plant safety, the System Analysis and Risk Assessment (SARA) program (NUREG/CR-5022) was selected. SARA, a microcomputer-based system, contains PRA data for the dominant accident sequences of the five NUREG-1150 NPPs and descriptive information about the NPPs including event trees and system model diagrams. Using SARA, the failure rates of basic events and initiating events in the plant systems can be easily changed including human error (HE) events. The effects of these changes can then be evaluated in terms of the resultant changes in core melt frequency (CMF), a measure of safety significance. The interactive capability to requantify human error probabilities (HEPs) and to recalculate resulting CMFs quickly and easily makes the use of SARA an appropriate PRA related tool for the sample application to assess the safety significance of annunciator upgrade alternatives.

5.2 Selection of the NPP

The SARA program was developed for each of the five NUREG-1150 NPPs. Each of the PRAs, as modeled on SARA, were evaluated for ease of application, modeling completeness, and the character of the human errors. Also considered was the availability of actual control room annunciator design information for the later stages of this project. Each of the PRAs had a moderate number of human errors modeled, some of which could be associated with the annunciator system, although usually not directly. The Sequoyah NPP was selected because it appeared reasonably representative, had sufficient human errors modeled, and offered the possibility of obtaining the necessary detailed design information. For information, each unit at the Sequoyah site consists of a 3400 Mwt, four-loop, Westinghouse Pressurized Water Reactor (PWR). One should note that the NUREG-1150 PRAs as modeled on SARA contain fewer human error events than a full-scale commercial PRA, such as Oconee, NSAC-60.

5.3 Sequoyah Annunciators Associated with SARA Human Error Events

The overall methodology for this project is to vary the PRA human error probabilities as upgrades are added to the annunciator systems. Thus, it was important to verify that the human errors, as modeled in the SARA PRA, could in fact be affected by the annunciator system.

In order the make this determination, the Tennessee Valley Authority (TVA), the owner and operator of Sequoyah, was contacted. BNL obtained from TVA: a copy of all control annunciator panel drawings (containing the actual tile engravings); pertinent emergency operating procedures and instructions; and information regarding the status of the annunciator system design upgrades. A review was then performed to compare the SARA human errors with the annunciator tiles and emergency procedures. It is acknowledged that the new symptom-oriented emergency procedures do not make extensive use of annunciators in the course of significant transients. However, the review performed, identified annunciator tiles that would typically provide useful information at key portions of the transients events, and which could be closely matched with the SARA human error events.

5.4 Sequoyah Annunciators - Existing Upgrades

As mentioned above, the status of the annunciator upgrades at Sequoyah was obtained from TVA. Additionally, NRC personnel knowledgeable with the Sequoyah control room were also contacted. After this, the status of the annunciator system was documented by BNL and sent to TVA for their information (Higgins, June 27, 1988).

Based on discussions with personnel knowledgeable with NPP control rooms and also based on a review of NUREG/CR-3217 and EPRI NP-5795, it was determined that the set of upgrades Sequoyah has is reasonably typical. Those upgrades in place at Sequoyah are not uncommon, and the total number is fairly typical. The actual upgrades at any given plant varies considerably.

6. EFFECTS OF UPGRADES ON HUMAN ERRORS AND CMF

This section generally corresponds to Path #2 of the methodology described in Section 4 and details how the effects of the annunciators upgrades on Human Error were quantified.

Table 5.1. Recommended Annunciator Upgrades to Sequoyah 1

Annunciator Upgrades	Perceived Status of Annunciator at Sequoyah	Status
First out panel	Two existing panels - turbine trip and reactor trip.	Yes
Reflash	Some (multi-point) panels have reflash, others do not.	Partial
Inhibitors	No inhibitors at Sequoyah.	No
Prioritization	Sequoyah prioritizes by red and white colors.	Yes
Fail-safe	All panels have either two light bulbs per tile, or one light per tile and associated test button - tested once per shift.	Yes
Auditory (signal intensity)	Loudness is adjustable, normally not varied.	Yes
Tile (legibility and intelligibility)	Several HEDs (e.g., 0208 and 0207) discuss this. BNL assumes status at time of NUREG-1150 PRA same as findings described in HEDs.	Partial
Keying Procedures to Tiles	Sequoyah has separate procedures for each panel. Panel numbers are keyed to a procedure.	Yes
Relocation of Tiles	Many HEDs refer to tile relocation, such as 0321 and 8083.	Partial
Controls (separate silence and acknow.)	Sequoyah has one button for silence and acknowledge. Also has 30-second timer to stop horn.	No
Ring Back	No ringback now at Sequoyah. Tile lights always remain solid until manually reset.	No
Black Board (normal oper.)	Almost 100% blackboard concept now at Sequoyah (three tiles lit).	Partial
"Grouping"	Sequoyah uses "grouping" for backpanel alarms (such as Radwaste and Second Plant Water). Also some grouping for 2 or 3 inputs (such as hitemp/lopress) from single component or inputs from 2 or 3 components.	No
Flash Rate	Flash rate is 3 - 5 cps.	Yes

6.1 Determination of Criteria

Before the annunciator upgrades could be quantified, suitable criteria first had to be determined by which they would be evaluated. It was desired to establish human factors criteria by which one could judge how much a particular upgrade would affect human performance in the control room. As a result, four criteria were developed to represent and judge these effects. The four criteria chosen are not likely to encompass 100 percent of the factors which could influence the annunciator upgrade effectiveness, but are believed to encompass a large portion of the variables which could influence their effectiveness. The criteria considered in evaluating each annunciator upgrade's effectiveness on human performance were:

1) Does the annunciator upgrade minimize sensory stimuli and maximize information? This factor first considers whether or not a given upgrade is providing more or less sensory stimuli (through sight, sound, smell or touch) while simultaneously increasing or decreasing the amount of information that is available for the operator's use.

   This factor deals with some of the issues involved in a concept known as alarm filtering. Alarm filtering calls for the elimination or inhibition of less important information and alarms, thereby maximizing important alarm information. This relates back to some of the critical issues uncovered during the accident at TMI.

   It was the opinion that the better upgrades would minimize sensory stimuli while maximizing information, and that the worst case for an upgrade would be if it maximized sensory stimuli while minimizing information.

2) Does the annunciator upgrade provide unique information to the operator? This factor requires consideration of other information that may be available to an operator in a control room and a determination of whether a given annunciator upgrade is providing new and unique information to the operator that is not available anywhere else in the control room. Or alternatively, is a given annunciator upgrade redundant because the same information is available from another source that is utilized by the operator?

3. Is a given annunciator upgrade useful in an accident scenario? The objective of this study was to evaluate the annunciator upgrades for implementation into a PRA; it was therefore desirable to determine how useful an upgrade would be in the types of situations typically modeled in PRA. Therefore, those annunciator upgrades which were judged to be useful in an accident scenario were considered more effective than those annunciator upgrades which were judged not useful or potentially even detrimental in the accident scenario.

4. What is the amount of training necessary for an operator to properly utilize an annunciator upgrade? Optimally, an annunciator upgrade, by this factor, would require little training for effective utilization. Conversely, a poor rating for an annunciator upgrade on this

factor would be if an upgrade required a large amount of training before it could be effectively utilized. An important variable which has a tremendous influence on this factor is a concept termed negative transfer of learning. Negative transfer of learning is a phenomenon that can occur when someone must unlearn a task in order to learn a new task. The old task, though seemingly unrelated to learning the new task, may actually inhibit learning the new task. It takes much more time to train when there is a potential for negative transfer of learning.

6.2 Quantification of Individual Upgrades by SLIM-MAUD

The Success Likelihood Index Method/Multi-Attribute Utility Decomposition (SLIM/MAUD) (NUREG/CR-3518) was utilized to obtain the relative ranking based on expert opinion of each annunciator upgrade considered in this study.

Four expert judges were involved in the SLIM-MAUD process. The aggregate experience of the judges included a former Senior Reactor Operator for two NPPs, a former NPP startup manager, a former NPP instrumentation and calibration manager, a NRC Senior Resident Inspector, reactor operators with experience in navy nuclear power, human factors specialists, an experimental psychologist, and individuals with Probabilistic Risk Assessment (PRA) Human Reliability Analysis (HRA) experience.

SLIM-MAUD was originally conceived as a method to develop Human Error Probabilities (HEPs) utilizing group expert judgement. Although this portion of the research was concerned with evaluating and quantifying annunciator upgrades and not quantifying HEPs, SLIM-MAUD was still well-suited for the purpose.

SLIM-MAUD was utilized by the expert panel on an IBM-PC (personal computer) to rank annunciator upgrades on the four human factors criteria discussed above. In SLIM-MAUD, each of the 14 annunciator upgrades was separately considered as it applied to the four criteria described above. Every upgrade was evaluated on a rating scale of 1 to 9 for each criteria or factor, with 1 always representing the best possible case for that factor and 9 representing the worst possible case.

The four factors were then weighted by the judges on relative importance to one another, as follows. Two hypothetical annunciator upgrades which had different values on two of the criteria scales were compared. The judges were asked which one of the two hypothetical upgrades was more effective and then successively degraded on of the criteria on the more effective upgrade and improved one of the criteria on the less effective upgrade, until the judges decided to reverse their opinion on which upgrade was the most effective. The process was repeated until all criteria had been evaluated and then, based on the experts' judgement, SLIM-MAUD determined the relative weights for each human factors criteria on a scale from 0.00 to 1.00. The lower value, 0.00, would indicate that the human factors criteria was evaluated as insignificant, and a value of 1.00 would indicate that the criteria was very important, in relation to the other criteria.

What follows are the weights for the five criteria, as perceived by the judges. Whether an annunciator upgrade provided unique versus redundant information was given the highest relative importance of 0.32 by the judges. Whether an upgrade minimized sensory stimuli while it maximized information and whether an annunciator was considered to be useful in an accident were both given relative importance weights of 0.26. Finally, the amount of training necessary to effectively utilize any given upgrade was given a relative importance of 0.16, the lowest of the four factors.

Once each upgrade had been evaluated on each of the four dimensions and the dimensions had been assigned relative importance weights, SLIM-MAUD calculated final index values for each annunciator upgrade. The range of potential final index values is from 0.0 (indicating that the upgrade scored the worst on every evaluated factor) to 1.0 (indicating that the upgrade scored the best on every factor). Table 6.1 lists each potential annunciator upgrade, in order of rating, and its final index value, as provided by SLIM-MAUD.

Table 6.1 Potential Annunciator Upgrades and Final Index Values as Computed by SLIM-MAUD

Potential Annunciator Upgrades	SLIM-MAUD Final Index Value
Prioritization	0.84
Separate Silence and Acknowledge	0.84
Inhibit	0.72
First-Out Panel	0.66
Reflash	0.52
Elimination of Grouping	0.48
Tile Legibility/Intelligibility	0.48
Keying Procedures to Tiles	0.47
Blackboard for Normal Operations	0.45
Relocation of Tiles	0.45
Flashrate	0.28
Fail-Safe	0.25
Auditory Signal Intensity	0.20
Ringback	0.18

Although SLIM-MAUD can be used to compute HEPs, it was not possible to do that in this study. To compute probability, it is necessary to evaluate two calibration items with known values which SLIM-MAUD then uses as anchor points to calculate all other items. However, because there were no known values for anything that approximated an annunciator upgrade, it was necessary to take final index values provided by SLIM-MAUD for the upgrades and devise a separate method to compute the values for each.

It is instructive at this point to note a few of these differences between this prioritization, based on human factors considerations, and that of NUREG/CR-3217, based on functionality. The main difference is that this scheme provides a full prioritization and not just a grouping. Thus, the earlier scheme has nine upgrades equally ranked as Priority 1. Regarding specific upgrades, two upgrades (Auditory Loudness and Fail-Safe) were Priority 1 per the NUREG, but came out quite low on the new scheme. One upgrade (Separate Silence and Acknowledge) moved from Priority 2 to high on the new scheme. One upgrade (Ringback) moved from Priority 2 to low on the new scheme. And finally, one upgrade (Eliminate Grouping) moved from Priority 3 to the middle of the new list.

6.3 Effect of Individual Upgrades on HEPs

Based on expert judgement, it was decided that no individual annunciator upgrade being evaluated in this study could affect an HEP by more than a factor of 2. The value is believed to be reasonably conservative as an upper bound on the effect on one upgrade. As will be seen later, a composite limit was also placed on multiple upgrades, so that as more upgrades are added, there is a decreasingly positive effect. Therefore, a scale from 1.0 to 2.0 was devised and based on the final index values provided by SLIM-MAUD, each annunciator upgrade was placed on this scale. The lower value of 1.0 was chosen because it was believed that none of the proposed annunciator upgrades were detrimental and therefore none could fall below the value of 1.0, which represents an upgrade having a neutral effect on the HEP.

A SLIM-MAUD final index value of 0.84 as computed for both prioritization and Separate Silence and Acknowledge, translates into a potential HEP factor change of 1.8 on the devised scale (when 0.84 is rounded to 0.8). On the other end, a SLIM-MAUD final index value of 0.18 for Ringback, represents a HEP factor change of 1.2 when that index value is converted to the scale value (0.18 is rounded to 0.2). Table 6.2 presents the potential annunciator upgrades and their associated potential HEP factor change.

Table 6.2 Potential Annunciator Upgrades and their Associated HEP Factor

Potential Annunciator Upgrades	Potential HEP Factor Change
Prioritization	1.8
Separate Silence and Acknowledge	1.8
Inhibit	1.7
First-Out Panel	1.7
Reflash	1.5
Elimination of Grouping	1.5
Tile Legibility/Intelligibility	1.5
Keying Procedure to Tiles	1.5
Blackboard for Normal Operations	1.5
Relocation of Tiles	1.5

Table 6.2. Continued

Potential Annunciator Upgrades	Potential HEP Factor Change
Flashrate	1.3
Fail-Safe	1.3
Auditory Signal Intensity	1.2
Ringback	1.2

A choice of 2.0 as an upper limit for any single annunciator upgrade was, admittedly, a subjective one; however, it is also believed to be realistic. It is important to note that no single upgrade was placed at the lower end of 1.0 and also that no single upgrade realized the maximum potential effect of 2.0. Future work planned on the annunciator issue will include an experiment to validate this work.

6.4 Selection of Alternative Upgrade Groups

A Value-Impact assessment typically considers more than one alternative situation for implementation. This process allows a more complete examination of the proposal under consideration. For example, in this study, perhaps some of the upgrades may be beneficial and not others. Or perhaps all of the upgrades should be implemented. In order to obtain as broad an evaluation as possible within the scope of the project, three alternatives were settled upon. However, it was only possible to decide upon the alternatives after determining exactly what annunciator upgrades Sequoyah already had and after the prioritization of the annunciator upgrades based on the SLIM-MAUD session was completed. Each of the alternatives are described below.

Alternative 1

The first alternative consisted of the addition of all of the Priority 1 upgrades per NUREG/CR-3217. Since Sequoyah has several of these Priority 1 upgrades already installed, the assessment of this alternative required two steps. The first step consisted of taking out all six upgrades Sequoyah already had in place in their annunciator system. The upgrades involved in this step are: Prioritization, First-Out Panel, Keying Procedures to Tiles, Flashrate, Fail-safe, and Auditory Signal Intensity. The second step of this alternative consisted of adding the nine upgrades which had been established as Priority 1 by NUREG/CR-3217. These upgrades are: First-Out Panel, Reflash, Inhibit Prioritization, Fail-safe, Auditory Signal Intensity, Tile Legibility/Intelligibility, Keying Procedures to Tiles, and Relocation of Tiles. The net result is the addition of all Priority 1 upgrades to a NPP, which had no upgrades at all.

Alternative 2

The second alternative to be used in the Value-Impact analysis involved implementing all upgrades which Sequoyah did not already have in place. This consisted of implementing the following seven upgrades: Separate Silence and Acknowledge Controls, Inhibit, Reflash, Ringback, Tile Legibility/Intelligibility, Blackboard for Normal Operations, and Relocation of Tiles. Although Sequoyah does not have elimination of grouping, it is not possible to eliminate grouping and add reflash. Therefore, Reflash was implemented because it had a slightly higher final index value than Elimination of Grouping had.

Alternative 3

The third alternative considered was based on the human factors prioritization by the BNL expert panel (using SLIM-MAUD) for the various upgrades. In this scenario, all upgrades which had a final index rating of 0.50 or above, and which were not already being utilized in the Sequoyah control room annunciator system, were implemented. This alternative involved three annunciator upgrades, namely: Separate Silence and Acknowledge Controls, Inhibit, and Reflash. These are the upgrades judged most effective by this project.

These three alternatives represent three ways in which the various annunciator upgrades can be combined to determine their safety significance, although there are numerous other variations which can also be implemented and analyzed.

6.5 Quantification of Alternatives

The effect of individual upgrades on the HEPs was determined in 6.3 above. This section describes the method developed for quantifying the effect that multiple upgrades, when combined into an alternative, will have on the HEPs. A method was needed that met the following constraints:

1) It should provide one single factor for an alternative grouping of upgrades to be applied to the HEPs.

2) The factor for an alternative should increase incrementally as more upgrades are included in the alternative.

3) The alternative factor should increase slowly. That is, the effect of two upgrades should be less than the product of the two upgrades implemented separately.

4) The formula should be mathematically consistent.

5) The total impact of all upgrades should not be more than about five times.

Within these constraints, the equation below was devised to combine the various upgrades within an alternative to obtain an overall factor for each to be applied to the Sequoyah HEPs.

$$\sqrt{a^2 + b^2 + c^2 + \dots} = \text{alternative factor to be applied to HEP}$$

where a, b, and c represent the factors to be applied for each individual upgrade in an alternative.

This equation provided factors for the alternatives that were reasonable to the expert judges. The model is such that one upgrade does not detract from another upgrade. Also, as the number of upgrades within an alternative increases, each upgrade's overall percentage effect decreases.

It is important to emphasize that this equation is hypothetical and subjectively derived. It is not known how these upgrades actually combine or if they are all equally important when combined. However, this is an empirical question, and a better understanding of how they combine may be obtained when the future work employing experimental data is conducted.

Using the above equation, the three annunciator upgrade alternatives were quantified and the following results obtained:

Alternative 1

First, a factor had to be developed for the upgrades that Sequoyah has. these upgrades are to be theoretically removed. Hence, a factor is needed by which the Sequoyah HEPs will be degraded, when these upgrades are removed. When the factors for the individual upgrades already in place were combined to obtain an overall factor to be taken out of the Sequoyah HEP values, the resulting value was 3.6. This number was then multiplied for each applicable HEP in SARA, as discussed in Section 7, to obtain HEPs which assume no upgrades in place. Next, a factor was needed for all of the Priority 1 upgrades, which would be added to the base case plant of no upgrades. This factor would be used to improve the HEPs. When all factors for annunciator upgrades classified as Priority 1 by NUREG/CR-3217 were combined using the above equation, the previously modified HEPs were improved by a factor of 4.6.

Alternative 2

This alternative consists of putting in place all the upgrades which Sequoyah did not already have. This resulted in the base case HEPs in SARA for Sequoyah being improved by a factor of 4.1.

Alternative 3

Finally, when all factors for the three highly-ranked upgrades not already in place at Sequoyah were implemented in the equation, the Sequoyah HEP values were improved by a factor of 2.9.

These factors are summarized in Table 6.3 below.

With the above factors calculated for the three upgrade alternatives using the developed equation, the Sequoyah base case core melt frequency (CMF) was modified as described in the following section.

Table 6.3. Overall Alternative Factors for Variation in HEPs

	HEP Degrade Factor	HEP Improve Factor
Alt. 1	3.6	4.6
Alt. 2	---	4.1
Alt. 3	---	2.9

6.6 Core Melt Frequency (CMF) Calculations

6.6.1 Calculations

The base case CMF for Sequoyah is given in SARA (NUREG/CR-5022) as 8.58E-5. This base case represents the Sequoyah plant with its annunciator upgrades as now in place (see Table 5.1). The alternative upgrade groups are described in Section 6.4 of this report. Using the alternative factors given in Section 6.5, each of the applicable human error events were changed in the SARA program and the program was then run to produce a new CMF.

For Alternative 1, when the existing annunciator upgrades were taken out, HEPs increased by 3.6 times and the CMF increased to 2.11E-4. When all priority 1 upgrades from NUREG/CR-3217 were put in, HEPs decreased by 4.6 times and the CMF became 7.80E-5. The difference between these two values or the ΔCMF was therefore 1.33E-4.

Alternative 2 added in all the annunciator upgrades that Sequoyah does not have to the SARA base case, and the new CMF was computed with the SARA program. The resulting CMF dropped from the base case of 8.58E-5 to 5.99E-5. The ΔCMF was thus 2.59E-5.

With only the three most significant annunciator upgrades (per this analysis) added to the base case (Alternative 3), the CMF decreased to 6.31E-5. The ΔCMF from the base case was thus 2.27 E-5.

The change in CMF (ΔCMF) for the three alternatives is summarized in Table 6.4 below.

6.6.2 Discussion of Results

For Alternative #1, the CMF was first increased by a factor of about 2.5 when the six existing upgrades at Sequoyah were removed. Then, when the nine priority 1 upgrades were added, the CMF decreased by a factor of 3.7. The net result is a fairly large change in CMF of 1.33E-4. A few important points should be noted about the results of this alternative. First of all, as discussed in Section 5, the control room annunciator upgrade status (six fully

Table 6.4. Alternative Sequoyah Upgrades

	ΔCMF
Alt. #1	
Put in all 9 priority 1 upgrades to a base case of no upgrades	1.33E-4
Alt. #2	
Put in all 7 upgrades Sequoyah does not have	2.59E-5
Alt. #3	
Put in the 3 upgrades Sequoyah does not have	2.27E-5

implemented) at Sequoyah is fairly typical. There is probably no plant in the country that currently has an annunciator system with zero of the upgrades. Hence, no plant would experience such a dramatic improvement in CMF. However, this is illustrative, since it shows that the fewer upgrades a plant has, the bigger the payback by implementing some upgrades. The main reason for this is the large sensitivity of CMF to degraded human performance. Thus, if there is a poor annunciator system, human performance would likely also be less than optimal, resulting in higher CMFs, which creates the potential for a large benefit by improving the annunciator system.

For Alternative #2, the CMF showed a ΔCMF of 2.59E-5, which is an improvement of about 30% (or a decrease by a factor of 1.43). This change in CMF in absolute terms is not nearly as dramatic as for alternative #1, however, it is still notable. The main reason for the large difference from Alternative #1 is that one is starting from the Sequoyah base case, which is already a reasonably good annunciator system, and thus there is not nearly as much room for improvement in the CMF.

For Alternative #3, the ΔCMF improved to 2.27E-5, an improvement of about 26% (or a decrease by a factor of 1.36). This last alternative with only three annunciator upgrades showed a ΔCMF almost as good as Alternative #2, which had seven upgrades. This could be a significant finding for those considering upgrading annunciator panels. When the costs are considered in the next section, it will be seen that this alternative is more cost-beneficial than alternative #2.

7. VALUE-IMPACT ASSESSMENT

The function of a value-impact assessment is to estimate the relevant values (or benefits) and the associated impacts (either costs or savings) likely to result from a proposed NRC action. Values considered are generally only radiological effects, such as person-rem of exposure incurred or averted as a result of an action. The person-rem to the public is often calculated from the change in core melt frequency obtained from the PRA portion of the V-I analysis. For this study, the only benefit attribute calculated is Public Health dose reduction as a result of the reduction in core melt frequency. The impact attributes considered in this study were industry implementation costs, NRC development costs, and NRC implementation costs.

For the Public Health calculations, NUREG/CR-3568 suggests the possibility of using a single generically applicable conversion factor of 2E7 person-rem per CM event. However, subsequently NRC (Thompson, 1985) has stated that this value is too high and recommends a factor of 1.1E6 person-rem per CM event for PWRs. Thus, for this analysis 1.1E6 is used to compute the person-rem averted. Utilizing this conversion factor the value portion of each of the three alternatives was computed from the changes in CMF derived in the preceding section.

For the impact portion, costs were estimated for each of the control room annunciator upgrades. The estimation process is described in detail in the report for this project. The costs were then combined to obtain an overall cost for each alternative. Overall, NRC costs were also estimated and combined with the individual alternative costs per NUREG/CR-3568.

Once the values and the impacts (or costs) were completed for each alternative, the Value-Impact Ratios were formed. These are shown below in Table 7.1. Also displayed are the reciprocal or cost-benefit ratios and the number of upgrades contained in each alternative.

Table 7.1. Value-Impact and Cost-Benefit Ratio Summary

Alternative	Value/Impact (person-rem/$million)	Cost/Benefit ($/person-rem)	Number of Upgrades Added
1	17,628.	56.7	9
2	3,472.	288.0	7
3	9,948.	100.5	3

The results indicate that the value-impact ratios of the three alternatives considered are about the same order of magnitude even though Alternative 3 includes only three upgrades, while Alternatives 1 and 2 have nine and seven, respectively. As discussed previously in Section 6.6, the value-impact ratio of Alternative 1 may be somewhat high, since it considers a hypothetical plant with no existing upgrades, and the increased value from this assumption outweighs the increased costs. With this in mind, the results of this scoping assessment indicate that each of the alternatives considered indicate a favorable action when compared to the normally used $1000/person-rem equivalence factor.

8. CONCLUSIONS

As stated in Section 2, the objective of this project was to evaluate the human factors safety issues associated with control room annunciator systems and to perform a Value-Impact Assessment for the associated upgrades. A sample nuclear power plant, Sequoyah, was selected for the analysis.

The analysis of the human factors issues surrounding these upgrades was described in Section 6 and indicates that none of the upgrades exert a negative effect on operator performance, and in fact, would have a positive effect. The analysis also prioritized all of the upgrades from a human factors standpoint, giving an indication of which ones should be most beneficial for performance. The top three upgrades were Prioritization, Separate Silence and Acknowledge, and Inhibit. The full prioritization is provided in Table 6.1. As discussed in Section 6.2, the new prioritization scheme provided somewhat different results than that of NUREG/CR-3217. For example, this scheme provides a full prioritization, whereas the NUREG/CR scheme has nine upgrades equal as Priority 1. Also, two upgrades (Auditory Loudness and Flash Rate) were Priority 1 in the NUREG, but were ranked quite low in the new scheme. One upgrade (Separate Silence and Acknowledge) was raised from a Priority 2 to high on the new list.

The analysis also used the Sequoyah nuclear power plant probabilistic risk assessment (PRA) to aid in estimating the effect on plant risk and hence, public health that would result from implementation of these annunciator upgrades. In this process, the effect of the annunciator upgrades on the PRA human error probabilities (HEPs) was first determined and the revised core melt frequencies were calculated based on the improved HEPs. Three different sets of upgrades (or alternatives) were analyzed. The next phase of this project will entail a small-scale simulator experiment, which will obtain the actual data regarding operator performance as annunciator systems are upgraded.

The change in core melt frequency (ΔCMF) was used to determine the public health dose aversion or value portion of the Value-Impact (V-I) analysis. The costs of annunciator upgrades were also estimated and formed the Impact portion of the analysis. The final results of the V-I analysis are summarized in Section 7 and show a favorable V-I ratio for each of the three alternative combinations of upgrades when compared with the NRC guideline of $1,000 per person-rem. In fact, the three alternatives have ratios of about $50, $100, and $300 per person-rem. During the next phase of this project, if the simulator experiment reveals significant differences in performance from those estimated herein, the V-I analysis will be redone.

The three separate alternatives that were evaluated provided some added insights. For example, Alternative #1 entailed the implementation of all the NUREG/CR-3217 Priority 1 upgrades into a plant with no upgrades at all. This provided the largest ΔCMF and the best cost-benefit ratio. This plant is unrealistic, in that all plants have some upgrades. However, it does show that the fewer upgrades that a plant has, the larger the benefit of adding additional ones. Alternative 2 added seven upgrades to the Sequoyah base case, and Alternative 3 added only the three highest ranked upgrades. The ΔCMF for

Alternative 3 was nearly as high as for Alternative 2, and hence, the cost-benefit ratio was noticeably better. This illustrates the merit of the human factors type analysis performed.

The overall results of this project show that implementation of the defined annunciator system upgrades should improve operator performance and would be cost beneficial.

9. REFERENCES

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THE DEVELOPMENT OF A PRA MODELS AND RESULTS DATA BASE *

Donna J. Fink, EG&G Idaho, Inc.
Martin B. Sattison, EG&G Idaho, Inc.
Dale M. Rasmuson, U. S. Nuclear Regulatory Commission

on work performed at
The Idaho National Engineering Laboratory
Idaho Falls, Idaho

ABSTRACT

The Nuclear Regulatory Commission's Office of Nuclear Regulatory Research (NRC-RES) is currently funding the development of the PRA Models and Results Data Base (PRA-DB) at the Idaho National Engineering Laboratory (INEL). The initial prototype is completely personal computer (PC)-based and provides a menu-driven user interface. The PRA-DB's primary functions are to serve as a data repository and data manager for NUREG-1150 data and other permanent probabilistic risk assessment (PRA) data, and to provide conversion capabilities on data utilized by the Integrated Reliability and Risk Analysis System (IRRAS), the System Analysis and Risk Assessment (SARA) system, the Set Equation Transformation System (SETS) and the Top Event Matrices Analysis Code (TEMAC).

The PRA-DB manages the transfer and storage of data between the mainframe and PC codes. Having logic models, failure rate data, minimal cut sets and other PRA-related information available in a readily accessible form allows different analysts to perform studies on the same plants starting with the same baseline PRA information.

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INTRODUCTION

For several years, the U. S. Nuclear Regulatory Commission (NRC) has been using the Integrated Reliability and Risk Analysis System (IRRAS)[1] and the System Analysis and Risk Assessment (SARA)[2] system to evaluate and analyze generic issues and multi-plant actions. These personal computer (PC)-based software tools have proven to be extremely useful, readily accessible, and cost-effective. During the development of these tools, the large Probabilistic Risk Assessments (PRAs) sponsored by the NRC were using mainframe computer tools such as the Set Equation Transformation System (SETS)[3] and the Top Event Matrices Analysis Code (TEMAC)[4]. While these codes are powerful, they are also expensive to use for multiple sensitivity studies and follow-on analyses, and they are not as readily available to the many different users that would like to use PRA information and techniques in their projects.

The PRA Models and Results Data Base (PRA-DB) was developed by the Idaho National Engineering Laboratory (INEL) under contract to the U. S. Nuclear Regulatory Commission (USNRC). The PRA-DB's primary functions are to serve as a data repository and manager for NUREG-1150 data and other permanent data, and to provide conversion capabilities on data utilized by IRRAS, SARA, SETS, and TEMAC software systems and programs. The data base structure is entirely PC-based. The menu-driven program provides the means to easily store the input and output from these codes, convert from one code's data file format to another, and output stored data in any of the desired formats.

Event tree, sequence, fault tree, basic event, and plant damage state data are loaded into the data base. Other descriptive information are manually entered for these data types and also for accident consequences, failure modes, locations, system types, component types, and class attributes.

The regular user's view of the data base is limited to that of retrieval and reporting of the contents of the data base, with the additional capability provided to convert data files between the formats of the different PRA tools. The master user controls the contents of the data base and therefore can load data into the data base, modify the contents of the data base, and archive selected data to another device.

This paper describes the data types stored within the PRA-DB, the files that can be processed for each of the PRA tools, and all of the features available to the master user. Detailed information on the operation of the PRA-DB software and the formats of the data files can be found in the PRA Models and Results Data Base User's Guide[5] and the PRA Models and Results Data Base Master User's Guide[6].

PRA-DB DATA BASE STRUCTURE

The PRA-DB's data base is organized by families. A family is any logical grouping of fault trees, event trees, and sequences with their associated basic events, cut sets, reliability data, and descriptions. For example, a nuclear power plant could be a family. Access to any portion of the data base is obtained by selecting the appropriate family.

Data across families are independent (i.e. class attributes, failure modes, component types, and system types, and locations are identified for each family rather than using a single list that all families must abide by.) The term "family" is used rather than "plant" to allow grouping of data for a single study on a plant. The data stored in the PRA-DB is organized in the following manner:

<u>Data Organization</u>

Family
- description
- 1. Event Trees
  - description
  - logic
  - graphics
  - 1.1 Sequences
    - description
    - logic
    - cut sets
    - PDS number
- 2. Plant Damage States (PDS)
  - description
  - containment release mode probabilities
- 3. Fault Trees
  - description
  - logic
  - graphics
  - cut sets
- 4. Basic Events for all cut sets in the family
  - descriptions
  - attribute information
  - failure rate data
- 5. Accident Consequences
  - description
  - probabilities of early death and latent cancer
- 6. Basic Event Attribute Descriptions
  - locations
  - system types
  - component types
  - failure modes
  - class attributes

A family may contain data for each of the data types. The event tree logic creates sequence logic. The event tree graphics (when available from IRRAS) will depict all of the valid sequences. An initiating event is associated with each of the event trees. The sequence logic defines the success or failure of each system (fault tree) in an event tree. An initiating event begins a sequence and the end state is a plant damage state. However, for event trees, a sequence may end in a transfer to another tree, but this is only a part of the entire accident sequence. The sequence cut sets are the minimal cut sets for a sequence's logic as derived from the fault tree logic. The sequence logic and cut sets are stored for each sequence. The analysis parameters used in determining the minimal cut sets such as the mission time, random number seed, sample size, probability cut off value, and cut set size cut off value may also be entered.

The probability of plant damage for each of twenty containment release modes (with and without direct heating), a description, and a frequency are stored for each damage state.

The accident consequences provide the probability of early deaths and the probability of latent cancers for each of the containment release modes.

A fault tree is composed of gates (AND, OR, NOR, NAND, etc.) and, at the bottom level, basic events. The combination of failure probabilities for the basic events determines the end probability for the fault tree. The graphics, logic, and cut sets for a fault tree may be loaded into the PRA-DB. As with sequences, the analysis parameters may also be stored.

A basic event has a name, description, failure rate, and descriptive attributes. The failure rate may be a probability value or may be calculated from a combination of lambda, tau, and the mission time. The attribute descriptions include the location, system type, component type, and failure mode of the event. Sixteen (16) class attributes may be set as applicable or not to the event.

A relational data base structure is used to store the data within the PRA-DB. The design of the data base allows the user to store multiple families within the data base and to interconnect data. For example, sequences are tied to both event trees and damage states. Also basic events are derived from fault trees and are tied to the textual attribute descriptions. The data for a single family are easily distributed and maintained. Any changes to the type of data stored can easily be made.

DATA FILES

Data files created and read by IRRAS, SARA, SETS, and TEMAC are loaded into the PRA-DB. These same files can be output from the PRA-DB or converted from one format to another by the PRA-DB. Figure 1 contains a matrix of the PRA tools (IRRAS, SARA, SETS, and TEMAC) and data types. A file extension within an intersection indicates a conversion to this data tool's format is available.

IRRAS	SARA	SETS	TEMAC	Convert From : File Description	File Ext.	Convert To : IRRAS	SARA	SETS	TEMAC
I				Cut Sets	.CUT	===	.SQS	.DNF	.TCS
	S			Cut Sets	.SQS	.CUT	===	.DNF	.TCS
		S		Cut Sets	.DNF	.CUT	.SQS	===	.TCS
			T	Cut Sets	.TCS	.CUT	.SQS	.DNF	===
	S			Damage States	.PDS		===		
I				Fault Tree Graphics	.DLS	===			
I				Fault Tree Logic	.TRE	===		.SET	
		S		Fault Tree Logic	.SET	.TRE		===	
I	S	S		Basic Event Descr.	.DES	===	===	===	
I	S			Basic Event Rates	.RAT	===	===	.VBK	
		S		Basic Event Rates	.VBK	.RAT	.RAT	===	
			T	Listing - rates & cut sets	.OUT	.RAT .CUT	.RAT .SQS	.VBK	
		S		Listing - cut sets	.LIS	.CUT	.SQS		

Figure 1. File Definition Table.

The previous section described the data that may be stored in the data base. As indicated in Figure 1, only a portion of the data contained in the data base may be loaded from data files. The remainder of the descriptive data are currently manually entered by the master user. This descriptive data and the event tree graphics and logic files will be loaded once sufficient data are available and the file formats are defined.

The IRRAS files currently loaded and output from the data base are:

- Sequence cut sets
- Fault tree graphics
- Fault tree logic
- Basic event failure rates.

The SARA files currently loaded are output by the PRA-DB are:

- Sequence cut sets
- Damage state names and descriptions
- Basic event descriptions
- Basic event failure rates.

The SETS files currently loaded and output by the PRA-DB are:
- Sequence cut sets
- Fault tree logic
- Fault tree cut sets
- Basic event descriptions
- Basic event failure rates.

Cut sets may also be loaded from the SETS output listing with a variable occurrence table written to a separate file.

The TEMAC cut set files are currently loaded and output by the PRA-DB. In addition the cut sets and basic event and initiating event failure rates may be loaded from the TEMAC output listing.

PRA-DB Features

The PRA-DB has been structured so that the various functions are contained in individual modules or program units. Each module is activated by selecting an option on the main menu. The main modules are UPDATE Defaults, CONVERT Data Files, OUTPUT Data Files, REPORT Data Base contents, SUMMARY Results, LOAD Data Files, MODIFY Data Base, and ARCHIVE Data Base. The last three options are only available to a master user of the data base. Currently the master user designation is assigned to Martin B. Sattison at the INEL.

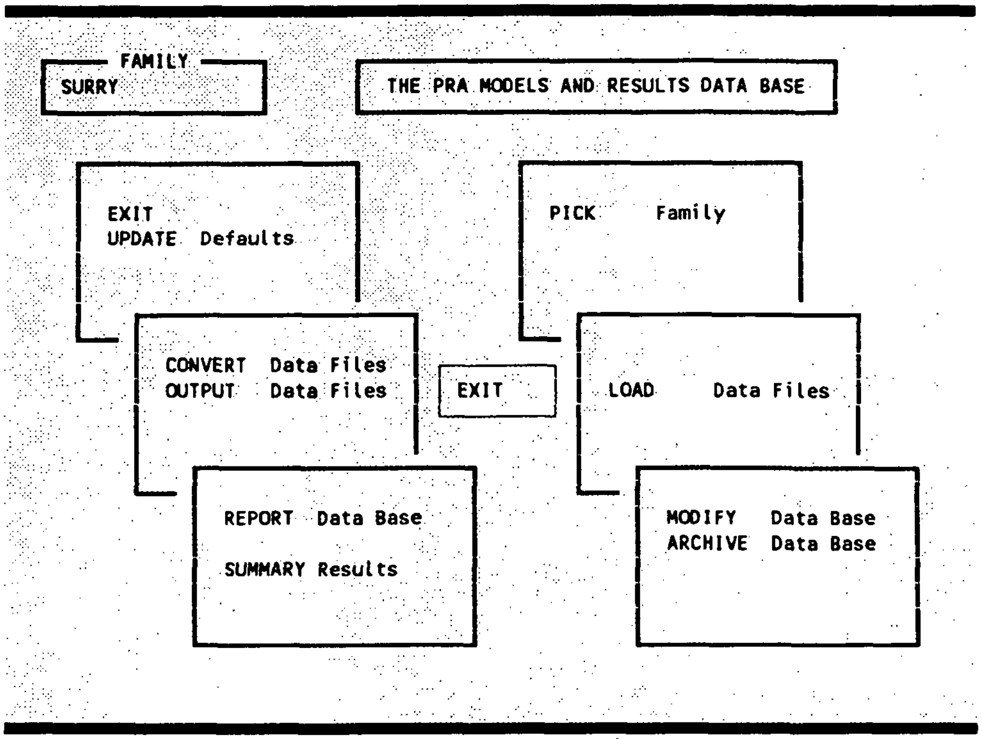

Figure 2. PRA-DB Main Menu.

PICK Family

The PICK Family option is used by the user to select a current family from the data base. When PICK Family is chosen, the user selects a family from a list of all the displayed family names and descriptions. The family stays selected until a new family is PICKed. The current family is saved upon EXIT from the PRA-DB program.

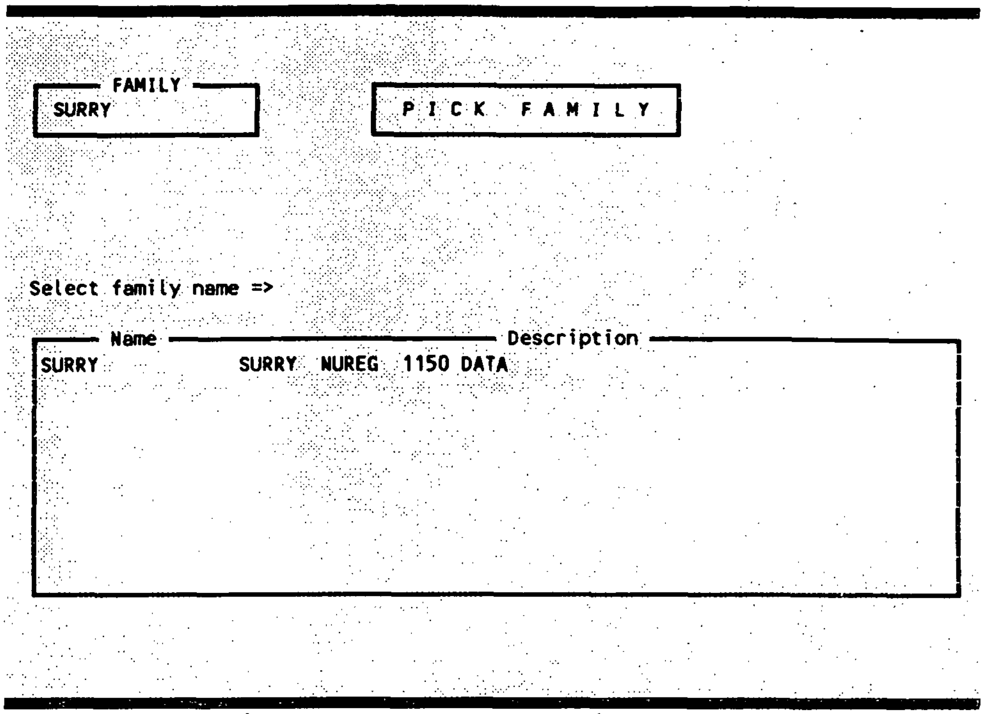

Figure 3. PICK Family Menu.

UPDATE Defaults

The UPDATE Defaults options allows the user to select a report output device (printer, console, or file), specify the default disk drive for the PRA-DB data base, specify an archive disk drive, set a directory path for locating files to convert, and select the type of basic event name they choose to use for searching the data base (primary or alternate name).

```
                  U P D A T E           USER ID INFORMATION

     Report output device :  FILE (PRN=printer  CON=console  FILE)
     Default disk drive   :  C:   (C: D: ...)
     Archive disk drive   :  A:   (A: B: ...)
     Directory path       :  C:\PRADB\DATA\

     Use Primary Basic Event Names ? : Y (N = Use Alternate Names)
```

Figure 4. UPDATE Defaults Menu.

CONVERT Data Files

The CONVERT Data Files option allows all valid conversions as shown in Figure 1. This option serves as a data interface between one code and another (e.g., converting cut sets generated by SETS into the SARA format for loading and further analysis.)

The CONVERT module allows files to be converted between the following PRA tools: IRRAS to SARA, IRRAS to SETS, IRRAS to TEMAC, SARA to IRRAS, SARA to TEMAC, SETS to IRRAS, SETS to SARA, TEMAC to IRRAS, and TEMAC to SARA. The data files are either converted directly from one file to another or are temporarily stored in the data base. This is a powerful tool for the analyst to use for transferring working data between the different tools.

```
C O N V E R T

Select DATA conversion =>

0 - EXIT
1 - IRRAS to SARA
2 - IRRAS to SETS
3 - IRRAS to TEMAC
4 - SARA  to IRRAS
5 - SARA  to TEMAC
6 - SETS  to IRRAS
7 - SETS  to SARA
8 - TEMAC to IRRAS
9 - TEMAC to SARA

0 - Option
```

Figure 5. CONVERT Data Files Menu.

OUTPUT Data Files

Data may be output from the PRA-DB through the OUTPUT Data Files option. The data is written in the requested IRRAS, SARA, SETS, TEMAC, or GENERIC file format. The GENERIC format is the IRRAS file format in all cases where that data type is applicable. In all other cases it is the SARA data file format. The GENERIC format is for the analyst who simply wants to output the data without regard to a specific tool. The output files are ASCII files that may be directly edited, except for the graphics files which are the DLS (display list) format used within IRRAS 2.0.

Only SURRY data Draft NUREG-1150 are loaded into the PRA-DB for demonstration purposes. The received NUREG-1150 data will be loaded as it becomes available. Note that this project is developing the data base structure and not the loading data.

A data tool type is selected on the first menu, followed by a data type, and the specific data (logic, cut sets, or graphics for a fault tree, event tree, sequence, or all basic events or damage states) to be output.

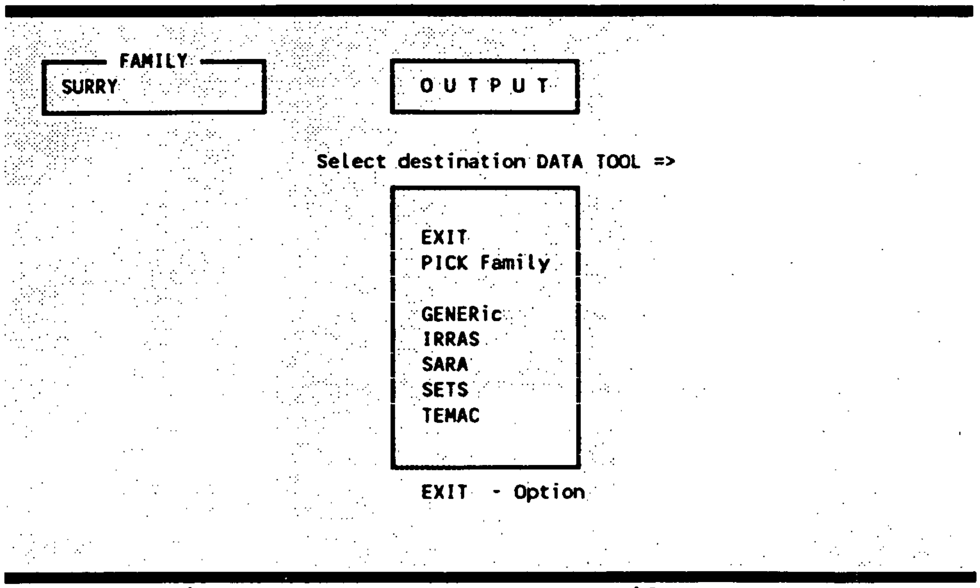

Figure 6. OUTPUT Data Files Menu.

REPORT Data Base

Three types of reports are available: Data Summary Report, Data Location Report, and the Detailed Data Report.

The Data Summary Report provides a map of what data are available for a single family or for all families. It contains the name and description of all data and an indication if associated cut sets, logic, and graphics data are available.

The Location Report lists where the data for each family are located.

The Detailed Data Report lists the details of the data stored in the data base including all descriptive information (e.g. basic event calculation type and values and attribute descriptions.) It may be produced for a single data type or for all of the data for the currently specified family.

The user can select from the UPDATE Defaults module whether they choose to have the report printed, recorded in a file, or shown on the console. It is suggested that an analyst always produce a Data Summary Report for the family of interest. A Detailed Data Report should be produced for a data type that is output in order to obtain analysis parameters such as the mission time used in obtaining the frequencies.

```
                              Data Summary Report

SURRY                 SURRY NUREG 1150 DATA
C:\PRADB\SURRY

EVENT TREES
    1  SURRYEVE  All Sequences will be stored in this event tree
       LOGIC:  N   GRAPHICS:  N
       S2-H1
         LOGIC:  N   CUT SETS:  Y
       S2-H2
         LOGIC:  N   CUT SETS:  Y
       S2-D1
         LOGIC:  N   CUT SETS:  Y
       T-K-R-Z
         LOGIC:  N   CUT SETS:  Y

FAULT TREES
    1  ACC-D5
       LOGIC:  Y   GRAPHICS:  Y   CUT SETS:  N
    2  ACC4
       LOGIC:  Y   GRAPHICS:  Y   CUT SETS:  N
    3  AFW-L
       LOGIC:  Y   GRAPHICS:  Y   CUT SETS:  N
    4  AFW1
       LOGIC:  Y   GRAPHICS:  Y   CUT SETS:  N

DAMAGE STATE
   1  Large LOCA, RWST injd, CHR avail. and CSI and CSR avail.
   2  Large LOCA, RWST injd, no CHR and CSI only available
   3  Large LOCA, RWSR injd, no CHR and CSI and CSR available
   4  Large LOCA, RWST injd and no CHR or CSS available
   5  Large LOCA, no inj. of RWST and no CHR or CSS available

BASIC EVENT
  ACC-PSF-LF-ACCB

  ACC-PSF-LF-ACCC

  AFW-PSF-FC-XCONN  FLOW DIVERSION TO UNIT 2 THROUGH PIPE SEGMENT PS94

  AFW-CCF-LK-STMBD  UNDETECTED LEAKAGE THROUGH CHECK VALVES CV27, CV58, OR CV89

  AFW-PSF-LF-PTRN2  FAULTS IN PS80 (TURBINE DRIVEN PUMP TRAIN 2)

  OEP-BETA-DGENFR   COMMON CAUSE FAILURE OF ALL 3 DIESEL GENS TO RUN FOR 6 HRS
```

Figure 7. Example Data Summary Report.

```
                         Detailed Data Report

                          FAMILY DESCRIPTION

 Family Name    : SURRY
 Description    : SURRY NUREG 1150 DATA
 Mission Time   :  2.400E+001
 Ref. Document  : NUREG/CR-4550 Volume 2.
 Location       : C:\PRADB\SURRY

EVENT TREES
  Event name      :                      Event tree ID    :    1
  Description     : All Sequences will be stored in this event tree
  Initiating event: ----
  Logic     ?     : N
  Graphics  ?     : N

SEQUENCES
  Event tree name : SURRYEVENTTREE1    Event tree ID :     1
  Sequence name   : A-C-F1             Sequence ID   :    16
  Description     :
  Damage state no.:   0                Probability cut off:  1.000000E-015
  Logic     ?     : N                  Size cutoff        :  6
  Cut sets ?      : Y                  Random number seed :    0
  Point estimate  :  1.000000E+000     Sample size        : 1000
                                       Mission time       :  2.400E+001
FAULT TREES
   Fault Tree ID   :     1
   Fault tree name:
   Level           : --
   Description     :
   Logic      ?    : Y                    Probability cut off : ---------E----
   Graphics   ?    : Y                    Size cutoff         : --
   Cut sets   ?    : N                    Random number seed  : ----
   Point estimate  : ---------E----       Sample size         : ----
                                          Mission time        : ------E----

BASIC EVENT

  Basic Event ID   :    1
  Basic Event Name: ACC-PSF-LF-ACCB  Alternate Name            : ACC-PSF-LF-ACCB
  Description      :
  Calculation Type: 1                Uncertainty Correlation : -
  Component ID     :                 Uncertainty Distr. Type : L
  Component System:                  Uncertainty Value.      :  1.000000E+000
  Component Type   :                 Probability             :  3.980000E-004
  Failure Mode     :                 Lambda                  : +0.000000E+000
  Location         :                 Tau                     : +0.000000E+000
  Class Attributes: 1:N  2:N  3:N  4:N  5:N  6:N  7:N  8:N
                    9:N 10:N 11:N 12:N 13:N 14:N 15:N 16:N
```

Figure 8. Example Detailed Data Report.

SUMMARY Results

The SUMMARY module displays several summaries of different arrangements of the data. This is currently a hard-coded prototype of the interface and is included for the user to provide suggestions. A working Summary and Results data base will be implemented during the coming year. It is anticipated that this portion of the data base will interface in some fashion with SINET.

```
S U M M A R Y   R E S U L T S   A N D   I N S I G H T S

E - Exit
P - Plants
S - Systems
C - Components
Q - Sequences
I - Initiators
A - Across Plants

E - Option
```

Figure 9. SUMMARY Results and Insights Menu.

LOAD Data Files

The data files that may be loaded are shown in Figure 1. The TEMAC and SETS output listings are not loaded in entirety, but instead only the cut sets and basic event information are extracted from the listing. This option is available only to the master user.

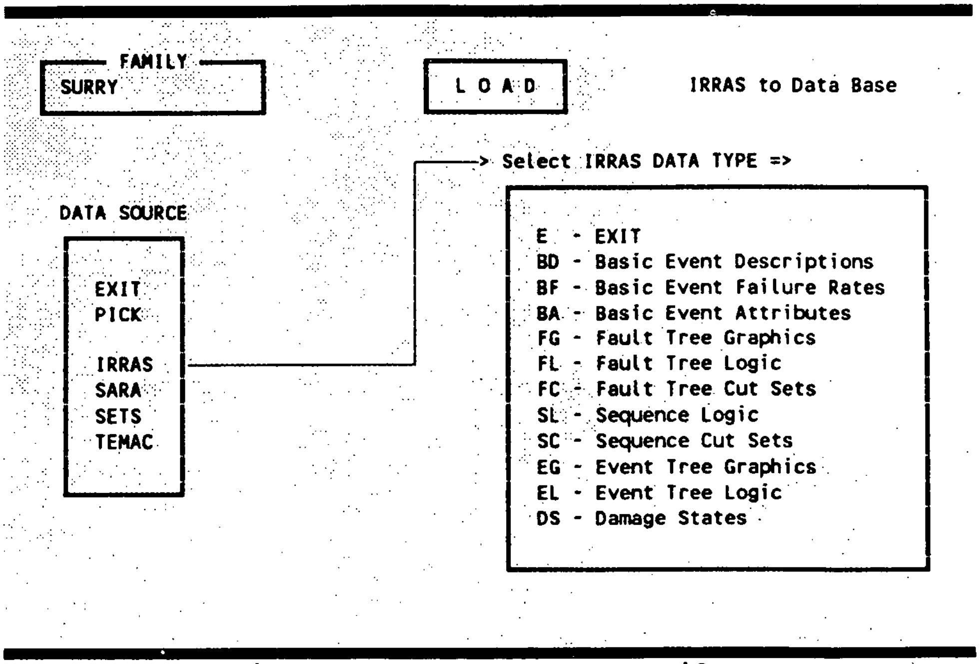

Figure 10. LOAD Data Files Menu.

MODIFY Data Base

The MODIFY Data Base option allows the master user to add descriptive (non-graphics, logic, or cut set) information to the data base. There are some data types which may only be entered through this option.

```
FAMILY
SURRY                    M O D I F Y

              E - Exit
              P - Pick Family

              F - Family
              A - Attribute Description
              B - Basic Events
              T - Fault Trees
              S - Sequences
              V - Event Trees
              D - Damage States
              C - Accident Consequences

              E - Option
```

Figure 11. MODIFY Data Base Menu.

ARCHIVE Data Base

The master user may archive an entire family's data to another media to save disk space on the permanent device used for the PRA-DB. The data may be processed directly from this alternate device, although this method is not recommended due to the access speed. The Data Location Report provides a way to catalog where the data are currently stored.

```
FAMILY
SURRY                         A R C H I V E

              Select a process =>

               E - Exit
               P - Pick Family

               A - Archive a family
               R - Retrieve archived family

               E - Option
```

Figure **12**. ARCHIVE Data Base Menu.

CONCLUSIONS

Anticipated follow-on work will determine the data base needs of other analysis tools such as CAFTA, NUPRA, RISKMAN, FRANTIC, and MELCOR. The types of data stored will be refined and enhancements to the data base schema and user interface will be made based on comments of analysts from their experiences with the prototype PRA-DB system.

A Summary and Results interface is a planned addition to the PRA-DB. A user interested only in the high-level data, such as the percent contributions of sequences, components, or systems to a plant damage state, will obtain reports without the detailed information. Across plant comparisons will also be allowed.

Acknowledgments

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A VALUE IMPACT ANALYSIS UTILIZING PRA TECHNIQUES COMBINED WITH A HYBRID PLANT MODEL[a]

J. L. Edson
D. W. Stillwell[b]
Idaho National Engineering Laboratory, Idaho Falls, Idaho

ABSTRACT

A value impact analysis (VIA) has been performed by the INEL to support a NRC Regulatory Analysis for resolution of Generic Issue (GI) 29, "Bolting Degradation or Failure in Nuclear Power Plants". A VIA for replacing the reactor coolant pressure boundary (RCPB) bolts of BWRs and PWRs was previously prepared by Pacific Northwest Laboratories (PNL) in 1985 under instructions limiting the VIA to the potential for failure of primary pressure boundary bolting. Subsequently the INEL was requested to perform a VIA that included non primary systems and component support bolts to be compatible with the resolution of the broader issue. Because the initial list of systems and bolting applications that could be included in the VIA was very large, including them all in the VIA would likely result in analyzing some that have little if any effect on public risk. This paper discusses how PRA techniques combined with a hybrid plant model were used to determine which bolts have the potential to be significant contributors to public risk if they were to fail, and therefore were included in the VIA.

INTRODUCTION

Generic Issue (GI) 29, "Bolting Degradation or Failure in Nuclear Power Plants", was issued as a result of an increase in the number of bolting-related incidents reported by the licensees of operating reactors and reactors under construction.[1] A large number of the reported bolting incidents have been related to primary pressure boundary applications and major component support structures. A Value-Impact Analysis (VIA) for replacing the reactor coolant pressure boundary (RCPB) bolts of BWRs and PWRs was prepared by Pacific Northwest Laboratory (PNL) in 1985 under instructions limiting the VIA to the potential for failure of primary pressure boundary bolting[2]. In addition, recent research has been sponsored by the Atomic Industrial Forum (AIF) and EPRI for industry resolution of generic issue GI-29 and is reported in a two-volume draft report issued in June 1987[3]. The report presents results of the detailed investigations into the bolting issue and makes recommendations for resolution of the bolting problem. Subsequently the INEL was requested to perform a VIA that includes non primary systems to be compatible with the resolution of the broader issue, as stated in a NRC

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b. Currently employed by Houston Lighting & Power Co.

letter from the Chief of the Engineering Issues Branch to the Chief of the Probabilistic Risk Analysis Branch[4]. By NRC-RES direction, the scope of the VIA was restricted to PWRs, internal bolts in primary system components were not within the scope of this VIA, and bolts covered by the previous VIA performed by PNL were not to be considered.

Because the number of systems and bolting applications that could be affected was very large, it became necessary to perform engineering and risk evaluations to identify a realistic boundary for systems, bolts, and causes of stress (stressors) to bolts that would be analyzed in the VIA. This paper describes the techniques used to perform the bounding studies.

ENGINEERING EVALUATIONS

Engineering evaluations utilizing studies that had already been performed were used to provide an initial list of systems, bolting applications, and stressors. These studies included evaluations of reported bolt failures, research performed by EPRI, and evaluations of water hammer events. The engineering evaluations were designed to provide information concerning 1) how frequent are bolt failures, 2) where are bolt failures expected to cause a loss of system function, and 3) what stressors are important to bolt failures.

A review of bolt failure data reported in NUREG-0943[5], and the research reported by EPRI[3], show that the frequency of bolting failures is low. Therefore it is assumed that simultaneous failure of multiple systems is very unlikely and may be excluded from this analysis. Research sponsored by EPRI, "Degradation and Failure of Bolting in Nuclear Power Plants", Research Project 2520-7, June 1987, considered the frequency of degraded bolting (bolts that had either completely failed or showed evidence of degradation) in RCPB applications for each nuclear plant from criticality to September 30, 1984. There were 796 degraded bolts out of 31569 bolts at risk, in a total of 52 nuclear plants accumulating about 477.5 plant years of operation since initial criticality. This corresponds to a degradation rate of 1.67 per plant year (0.28% of the bolts at risk in a plant per year).

The study of bolt failure events, sponsored by NRC and reported in NUREG-0943, "Threaded-Fastener Experience in Nuclear Power Plants", showed that events usually involved more than one bolt failure, although the exact number was not always stated. This study provides a list of the reported bolt failure events from 1964 to March 1982. A total of 43 events are reported, 39 in PWRs and the remainder (4) in BWRs. Eleven PWR component support events were reported for 47 PWRs accumulating about 350 plant years of operation. This corresponds to 0.031 events per plant year. If each event involved 10 failed or degraded bolts the rate of bolt failures would be 0.31 failures per plant year. (Note that the low number of bolt failures reported in BWRs supports the NRC direction to limit the VIA to PWRs.)

The rate of bolt failures, or degradation, reported by EPRI for RCPB applications (1.67 rejects/py) appears to be conservative for all bolting applications. Because most bolted joints involve several bolts, this low rate of bolt failures supports the conclusion that simultaneous bolted joint failures in multiple systems may be excluded from this analysis.

Research reported by EPRI[3] supports the conclusion that bolted pressure boundary joints will fail with a leak-before-break sequence permitting failures to be detected and fixed prior to a catastrophic failure that initiates a transient within the plant. Preliminary analyses of several bolted RCPB joints were performed to assess a leak-before-break strategy. The preliminary analyses considered the following closures:

Manway cover (16 studs)
Manway cover (20 studs)
Reactor coolant pump flange (16 studs)
6-inch check valve (12 studs)
10-inch check valve (16 studs)

Results of the analyses show that at 1 GPM leakage a computed safety margin of 2.2 to 3.2 exists with 7.8% (pump) to 27.7% (6-inch valve) of the studs in one component failed. Since the number of failed studs is expected to be small, as previously discussed, and the leak rate is sufficient to be detected, it is reasonable to assume that degraded pressure boundary joints will be identified by a noticeable leak and repaired before a catastrophic failure initiates a reactor system transient or the affected system becomes unavailable to perform its function.

Because of the leak-before-break assumption, failures of pressure boundary bolts are not likely to initiate serious or significant accident sequences. Other initiating events such as water hammers and earthquakes may result in sufficient stresses to cause leakage at degraded pressure boundary bolt connections. But because the number of degraded bolts is expected to be small and a significant number of bolts must be degraded at a single bolted joint for the joint to fail, the leaking bolted pressure boundary joints will not cause systems to fail to function. It was concluded that pressure boundary bolts need not be considered further in the VIA.

Water hammers are not produced in service water, component cooling water, residual heat removal, low pressure injection, high pressure injection, and charging systems during normal and accident operations because they are always filled with single phase water. A study of water hammers, NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants", March 1984[6], was performed to resolve Generic Safety Issue A-1, "Water Hammer". The report identified 67 known or suspected water hammer events in PWRs from 1969 to 1981. Feedwater water hammer including those in the feedring of the steam generator accounted for 59.7% of the total while occurrences in the main steam system and reactor primary system accounted for 11.9% and 7.5% respectively. There were no reported water hammer events involving the diesel generators and so they are not considered in this study. Therefore, water hammers will be considered for only the main steam, feedwater, and auxiliary feedwater systems.

NUREG-0943 reported 6 events involving internal bolting of PWR components with all but 2 being in the reactor vessel. One of the 2 involved a main steam isolation valve stud and the other involved a service water pump impeller capscrew. Because the reported failure rate of internal bolts in components outside the primary system is very low and internal bolting of primary system components is not within the scope of this study, internal bolting was not included in the VIA.

Degradation of bolts in component supports is not expected to initiate accident sequences but is important as a result of accidents that are initiated by events that significantly stress the support systems, such as water hammers or seismic events. While Large LOCAs in the primary system also stress component supports, PWRs have multiple primary loops and a LOCA in one loop will not result in large stresses to component supports in the unbroken loops. Even though component supports in the broken loop are stressed and could fail, the loop is already broken and added failures will not increase the severity of the accident. Therefore, large LOCAs as initiating events were not considered in the VIA.

In summary, the failure of pressure boundary bolts is not expected to result in the initiation of accident sequences and is not expected to result in failure of system function during accident sequences that involve water hammer or earthquake initiators. Water hammers are not expected to be a factor in many systems but are considered for the main feedwater, auxiliary feedwater, and main steam systems. Degradation of component supports is not expected to initiate accident sequences but is important for seismic and water hammer events. Table 1 shows the systems and bolting applications that were considered further in the study.

RISK EVALUATIONS

Development of Hybrid PRA

A hybrid probabilistic risk assessment (PRA) model was developed to evaluate the changes in core damage frequency caused by bolting failures in the systems of interest. The need for the hybrid PRA was determined after review of several PRA's which included seismic risk in the total contribution to plant damage. No one model contained accident sequences which included all of the systems of interest in this VIA. Rather than limit the number of systems included, a decision was made to develop a hybrid plant model which would contain representative seismic core damage sequences for at least a majority of the systems of interest.

This model was constructed after review of six PRA's which included seismic risk in the total core damage frequency. The plants reviewed were:

Zion
Indian Point 2
Indian Point 3
Seabrook
Millstone 3
Oconee

The total core melt frequency and the frequency associated with seismic events for each of these plants is presented in Table 2.

The procedure followed in developing this hybrid model is as follows:

1) Identify the dominant accident scenarios from each PRA.

2) Identify and study the sequences initiated by seismic events, focusing on those seismic failures caused by component bolting, either support or pressure boundary.

3) Assemble a specialized plant model which contains representative dominant sequences from the plant reviewed.

4) Assemble sequences from the six PRA's that cover the systems of interest to the resolution of the VIA.

5) Quantify this list of dominant sequences using typical (generic) values for component failures, initiating event frequencies, and seismic fragilities.

6) Establish a "representative" core damage frequency from the hybrid model, e.g. baseline the model.

The sequences developed for the hybrid model are identified in Table 3. The baseline core damage frequency from this model is 3.1E-04 per reactor year. The seismic sequences included in the model contribute approximately 10% (3.0E-05 per reactor year) to the total core damage frequency. These core damage frequencies are higher than those listed for the six PRA's, Table 2, because the hybrid PRA is a compilation of all the systems that are in the individual PRA's. None of the individual PRA's include all the systems that are included in the hybrid PRA. However the seismic contribution of the hybrid PRA (10%) is nearly identical to the average of the seismic contributions of the six PRA's (12%), thus providing some evidence that the hybrid PRA produced valid results.

<u>Bounding studies</u>

Bounding calculations were performed using the hybrid model for sequences that include seismic and water hammer initiating events and for the systems that were listed for consideration. For the bounding calculations, the affected systems were assumed to fail with a probability of one, given a seismic initiating event having a magnitude equal to the design basis earthquake. As can be seen from Table 4, failures of most systems changed total core melt by less than 5%. Failure of the diesel generators or the service water system resulted in 50% change in total core damage. The 50% change in core melt frequency is essentially the same as the frequency of the seismic event. These results indicate that the electric power system and its support systems play a significant role in the prevention of core damage after a seismic initiating event. This conclusion is not surprising when one realizes that components in the offsite power distribution system typically are more fragile than the components at a nuclear power plant. Failure in the Reactor Coolant System (LOCA) resulting from seismic events has the same effect on core damage as failure of the electric power system. This is due primarily to the assumption that more than one loop will be affected during the seismic event and the assumption that any seismic event fails the system. Core damage is guaranteed if more than one loop fails.

The systems which were identified as being susceptible to water hammer failure were quantified under bounding conditions similar to the systems affected by the seismic events. The initiating frequency for total loss of main feedwater was increased by 50%, from 0.33 events per reactor year (the value used in the baseline) to 0.50 events per reactor year, and the initiating event frequency for steam line breaks was increased 50%, from 6.0E-03 to 9.0E-3 events per reactor year to model an increase in bolt failures given a water hammer event. The results shown below indicate that even with this increase, large changes in core damage frequency are not seen and bolting failures as a result of water hammer are not a problem.

WATER HAMMER EVENTS

System	Core Damage Total	Core Damage Change
FEEDWATER	3.1E-04	3.2E-06
STEAM	3.1E-04	5.0E-06

Analysis of Risk

The bounding calculations described previously indicate that this analysis should focus on the likelihood of system bolt failure (component supports, pressure boundary, etc.) during a seismic event. The assumption of system failure due to bolting failure for any earthquake at or above the design basis earthquake acceleration is felt to be extremely conservative and so the bounding calculations are used as the high estimate. For the purpose of evaluating the VIA, a more realistic quantification was necessary. Since there are no data to support a change in component failure frequency due to bolt degradation, engineering judgement is used to determine a best estimate likelihood of the effects of degraded bolting on core damage frequency. For the purpose of calculating the best estimate, it was assumed that if degraded component support or pressure boundary bolting exists, component failure would be guaranteed at twice the design basis earthquake acceleration. The plant model was requantified using this assumption. This assumption is similar to that used in NUREG-0577, Rev. 1[7] which investigated degraded pump and steam generator support failures during seismic events. The results of the quantification using this assumption are shown below.

RESULTS OF BEST ESTIMATE CALCULATIONS
SEISMIC FAILURE

System	Core Damage Total	Core Damage Change
Base Case	3.08E-04	-----
AFW	3.17E-04	8.9E-06
HPI	3.15E-04	7.0E-06
LPI	3.15E-04	6.9E-06
CCW	3.19E-04	1.1E-05
EPS / SWS	3.40E-04	3.2E-05
LOCA	3.40E-04	3.2E-05

Under the assumption described above, the maximum change in core damage frequency due to failure of degraded bolts at twice design basis earthquake accelerations is seen in the electric power or reactor coolant systems and the change was approximately 10% of the total core damage frequency. Because of the plant damage sequences that are included in the hybrid model, the maximum change in core damage frequency is approximately equal to the frequency of exceeding an earthquake acceleration of twice design basis.

Table 5 lists the changes in sequence frequency under the best case assumptions. Table 3 lists the changes in sequence frequency under high estimate assumptions. Table 6 describes the top event coding used in Tables 3 and 5. The use of Table 6 is described in the following example.

The second sequence in Table 3 and Table 5 is: LOSP * GA * GB * ER. The first entry is the initiating event, in this case a loss of offsite power. The remaining entries describe subsequent system/top event failures in the hybrid model. Referring to Table 6,

GA Diesel Generator Train A fails
GB Diesel Generator Train B fails
ER Electric Power System Recovery Models

The sequence listed is a loss of offsite power followed by failure of diesel generator A and diesel generator B with failure to recover offsite or onsite power before core damage occurs. The remainder of the sequences in Tables 3 and 5 are read in a similar manner.

For the remainder of the analyses performed to support the VIA, the change in core damage frequency associated with the LOCA initiating event was used as a measure of the risk associated with degraded bolting. This decision was made for the following reason:

The LOCA case was analyzed under the assumption that all RCS loops would be affected by degraded bolting (e.g. perfect coupling). Under this assumption, failure of the RCS is guaranteed to result in core damage because of the inability to inject water into the vessel. Thus the change in core damage frequency can be directly related to initiating event frequency, thereby simplifying the calculations.

There were three separate calculations of core damage frequency for this VIA. The benchmark calculation was taken to be the base case and assigned to the 'LOW' category for the purposes of evaluating this bolting issue. The first bounding calculation for the LOCA initiating event was assumed to be the high estimate case. The best estimate case was the best estimate described above for the LOCA initiating event. The following table summarizes the results for the low, best, and high estimates of core damage frequency for the bolting VIA.

	Core Damage Frequency	Change From Base Case
Low	3.08E-04	0.00
Best Estimate	3.40E-04	3.2E-05
High Estimate	6.03E-04	2.9E-04

These results were used in the calculation of the various inputs to the VIA including averted man-rem and averted offsite property costs.

CONCLUSIONS

The use of a hybrid plant model coupled with recognized PRA methods has proven to be an effective method for analyzing large, complicated systems when a complete model of an actual system does not exist. The use of a hybrid plant model in the Value Impact Analysis (VIA) of actions to reduce the risk of bolt failures in a PWR provided results that were more complete than would have been obtained had only one incomplete plant model been used.

In addition, the use of a hybrid plant model combined with standard engineering evaluations proved to be an effective method of providing realistic bounds to a very large, unbounded VIA task. This technique should be equally valid for other analysis tasks for which a PRA is needed but for which only partial system models have previously been developed.

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Table 1 SELECTION OF SYSTEMS AND BOLTS FOR BOUNDING CALCULATIONS

System	Bolting Application		
	Pressure Boundary	Component Supports	Component Internals
Primary System	N (1)	SEISMIC	N(2)
Secondary to Isol.	N	WH, SEISMIC	N
Main Feedwater	N	WH, SEISMIC	N
Auxiliary FW	N	WH, SEISMIC	N
HPI	N	SEISMIC	N
LPI	N	SEISMIC	N
Accumulator Inject	N	SEISMIC	N
Charging System	N	SEISMIC	N
Service Water	N	SEISMIC	N
Component Cooling	N	SEISMIC	N
Diesel	N	SEISMIC	N

Notes: 1. Reactor coolant pressure boundary bolts studies by PNL
2. By NRC direction

WH - Water Hammer could be an initiator
Seismic - Seismic events could be an initiator
N - Not considered further in the VIA

Table 2 Core Melt Frequency and Seismic Contribution[a]

Plant	Total Core Melt	Seismic Contribution
Indian Point 2	1.4E-04	7.7E-06
Indian Point 3	1.4E-04	3.6E-06
Zion	5.4E-05	5.6E-06
Seabrook	2.3E-04	2.0E-05
Millstone3	5.0E-05	9.1E-06
Oconee	7.5E-05	2.1E-05

a. The seismic hazard curves from the Seabrook Station Probabilistic Safety Assessment were used to determine the frequency of exceedance values for the seismic initiating events. The results would be similar for a PWR located in the Midwest or east coast.

Plants on the west coast can experience earthquakes of larger magnitude and therefore are designed to a higher earthquake acceleration. Because the components are designed for higher accelerations, the results of this VIA would be similar if a west coast site were considered.

Table 3 RESULTS OF QUANTIFICATION - BOUNDING CASE

	Base	AFW	HPI	LPI	CCW	EPS/SWS	STEAM	WATER
Core Damage Total	3.08E-04	3.23E-04	3.21E-04	3.20E-04	3.19E-04	6.02E-04	3.13E-04	3.08E-04
LOSP * GA * GB * ER	3.97E-05	----	----	----	----	----	----	----
RTRIP * PA * PB	2.11E-05	----	----	----	----	----	----	----
PLMFW * PA * PB	1.71E-05	----	----	----	----	----	----	----
SLOCA * L1 * L2	1.69E-05	----	----	----	----	----	----	----
TTRIP * PA * PB	1.32E-05	----	----	----	----	----	----	----
RTRIP * SA * SB * OR	1.17E-05	----	----	----	----	----	----	----
L1DC * EF * FR * FR	1.14E-05	----	----	----	----	----	----	----
SLBI * ON	9.39E-06	----	----	----	----	----	1.40E-05	----
EXFW * PA * PB	9.34E-06	----	----	----	----	----	----	----
FCRCC	9.00E-06	----	----	----	----	----	----	----
TTRIP * SA * SB * PR	7.29E-06	----	----	----	----	----	----	----
LOMF * SA * SB	5.69E-06	----	----	----	----	----	----	6.01E-06
E0.7T * OG * GA * GB	5.65E-06	----	----	----	----	1.86E-05	----	----
SLOCA * EA * L2	5.55E-06	----	----	----	----	----	----	----
SLOCA * EB * L1	5.55E-06	----	----	----	----	----	----	----
LOSP * PA * PB	4.96E-06	----	----	----	----	----	----	----
FPCC	4.20E-06	----	----	----	----	----	----	----
LOPF * PA * PB	3.79E-06	----	----	----	----	----	----	----
FSRCC	3.60E-06	----	----	----	----	----	----	----
SLOCA * WA * L2	3.26E-06	----	----	----	----	----	----	----
SLOCA * WB * L1	3.26E-06	----	----	----	----	----	----	----
RTRIP * WA * WB * SR	3.14E-06	----	----	----	----	----	----	----
LOSP * GA * GB * EF * ER * FR	3.01E-06	----	----	----	----	----	----	----
LOSP * WA * WB * ER	2.57E-06	----	----	----	----	----	----	----
PLMFW * WA * WB * SR	2.54E-06	----	----	----	----	----	----	2.54E-06
LOSW	2.52E-06	----	----	----	----	----	----	----
SLOCA * WB * EA	2.41E-06	----	----	----	----	----	----	----
SLOCA * WA * EB	2.41E-06	----	----	----	----	----	----	----
FTBLP * GA * GB * ER	2.33E-06	----	----	----	----	----	----	----
LOSP * GA * GB * EF * ER	2.33E-06	----	----	----	----	----	----	----
E0.5T * OG * GA * GB	2.32E-06	----	----	----	----	1.84E-05	----	----
LOSP * GA * WB * ER	2.12E-06	----	----	----	----	----	----	----
LOSP * GB * WA * ER	2.12E-06	----	----	----	----	----	----	----
RTRIP * ON	2.11E-06	----	----	----	----	----	----	----
E0.4T * OG * GA * GB	2.08E-06	----	----	----	----	3.60E-05	----	----
TTRIP * WA * WB * SR	1.95E-06	----	----	----	----	----	----	----

Table 3 RESULTS OF QUANTIFICATION - BOUNDING CASE (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS	STEAM	WATER
V SEQUENCE	1.84E-06	----	----	----	----	----	----	----
PLMFW * ON	1.71E-06	----	----	----	----	----	----	----
FCRSW	1.65E-06	----	----	----	----	----	----	----
E0.7T * OG * SA * SB * PA * PC	1.63E-06	----	----	----	5.85E-06	0.00E+00	----	----
FCRAC	1.62E-06	----	----	----	----	----	----	----
FLSW	1.60E-06	----	----	----	----	----	----	----
E0.7T * OG * GA * GB * EK	1.59E-06	2.85E-06	----	----	----	5.26E-06	----	----
E1.0T * OG * GA * GB	1.25E-06	----	----	----	----	2.16E-06	----	----
LLOCA * LA * LB	1.14E-06	----	----	----	----	----	----	----
E0.2T * OG * GA * GB	1.09E-06	----	----	----	----	1.27E-04	----	----
EXFW * ON	9.34E-07	----	----	----	----	----	----	----
RTRIP * EF * L1 * L2	8.34E-07	----	---	----	----	----	----	----
RTRIP * EF * L1 * L2	8.34E-07	----	----	----	----	----	----	----
RTRIP * EF * O3	8.32E-07	----	----	----	----	----	----	----
E0.7T * OG * SA * SB * EK * RW	7.98E-07	1.43E-06	----	2.34E-06	0.00E+00	0.00E+00	----	----
E0.7T * OG * EK * H2	7.00E-07	1.32E-06	2.50E-06	0.00E+00	----	0.00E+00	----	----
E0.5T * OG * GA * GB * EJ	6.86E-07	1.81E-06	----	----	----	5.43E-06	----	----
LOMF * RT * OH	6.59E-07	----	----	----	----	----	----	6.97E-07
E1.0L * OG * GA * GB	6.52E-07	----	----	----	----	1.12E-06	----	----
LOSP * WA * WB * EF * ER * FR	6.34E-07	----	----	----	----	----	----	----
E0.5T * OG * EJ * RW	5.97E-07	1.75E-06	----	4.26E-06	----	0.00E+00	----	----
E0.7L * OG * SA * SC	5.57E-07	----	----	----	----	0.00E+00	----	----
E0.7L * OG * GA * GB	5.38E-07	----	----	----	----	1.77E-06	----	----
TTRIP * EF * L1 * L2	5.20E-07	----	----	----	----	----	----	----
TTRIP * EF * O3	5.18E-07	----	----	----	----	----	----	----
E0.5T * OG * EJ * H2	5.13E-07	1.51E-06	3.67E-06	0.00E+00	----	0.00E+00	----	----
E0.4T * OG * GA * GB * EI	5.06E-07	1.83E-06	----	----	----	8.73E-06	----	----
FSRAC	4.90E-07	----	----	----	----	----	----	----
E0.4T * OG * EI * H2	4.53E-07	1.96E-06	6.47E-06	0.00E+00	----	0.00E+00	----	----
TTRIP * RT * OH	4.49E-07	----	----	----	----	----	----	----
E0.7T * OG * SA * SB * EK * H2	4.34E-07	7.78E-07	1.55E-06	0.00E+00	0.00E+00	0.00E+00	----	----
E0.5T * OG * SA * SB * PA * PB	4.24E-07	----	----	----	3.53E-06	0.00E+00	----	----
E0.4T * OG * EI * RW	3.83E-07	1.66E-06	----	6.38E-06	----	0.00E+00	----	----
LOPF * ON	3.79E-07	----	----	----	----	----	----	----
LOSP * DA * EF * ER * FR	3.75E-07	----	----	----	----	----	----	----
LOSP * DB * EF * ER * FR	3.75E-07	----	----	----	----	----	----	----

Table 3 RESULTS OF QUANTIFICATION - BOUNDING CASE (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS	STEAM	WATER
EXFW * EF * L1 * L2	3.68E-07	----	----	----	----	----	----	----
EXFW * EF * O3	3.66E-07	----	----	----	----	----	----	----
E1.0L * OG * SA * SC	3.31E-07	----	----	----	----	0.00E+00	----	----
LOSP * WA * WB * ON * ER	3.28E-07	----	----	----	----	----	----	----
E1.0T * OG * SA * SB * EL * RW	2.95E-07	3.95E-07	----	4.75E-07	----	0.00E+00	----	----
LCV * ON	2.83E-07	----	----	----	----	----	----	----
SLBI * L1 * L2	2.73E-07	----	----	----	----	----	4.09E-07	----
SLBI * O3	2.72E-07	----	----	----	----	----	4.08E-07	----
ELOCA	2.66E-07	----	----	----	----	----	----	----
SGTR * EF * OD	2.62E-07	----	----	----	----	----	----	----
MLOCA * L1 * L2	2.54E-07	----	----	----	----	----	----	----
RTRIP * OG * GA * GB * ER	2.44E-07	----	----	----	----	----	----	----
C1MSIV* ON	2.39E-07	----	----	----	----	----	----	----
LOSP * GA * WB * EF * ER * FR	2.31E-07	----	----	----	----	----	----	----
LOSP * GB * WA * EF * ER * FR	2.31E-07	----	----	----	----	----	----	----
E1.0T * OG * GA * GB * RT * EL	2.24E-07	3.01E-07	----	----	----	3.88E-07	----	----
TLMFW * ON	2.24E-07	----	----	----	----	----	----	3.36E-07
E0.5T * OG * SA * SB * EJ * RW	1.65E-07	4.35E-07	----	1.17E-06	0.00E+00	0.00E+00	----	----
LCV * EF * O3	1.52E-07	----	----	----	----	----	----	----
TTRIP * OG * GA * GB * ER	1.52E-07	----	----	----	----	----	----	----
LOSP * WA * WB * EF * ER	1.51E-07	----	----	----	----	----	----	----
LOSP * DA * GB * ER	1.48E-07	----	----	----	----	----	----	----
LOSP * DB * GA * ER	1.48E-07	----	----	----	----	----	----	----
E0.5T * OG * SA * SB * EJ * H2	1.42E-07	3.74E-07	1.01E-06	0.00E+00	0.00E+00	0.00E+00	----	----
E0.5T * OG * SA * SB * PA * PB * EJ	1.38E-07	3.65E-07	----	0.00E+00	1.15E-06	0.00E+00	----	----
LOSP * GB * WA * EF * ER	1.25E-07	----	----	----	----	----	----	----
LOSP * GA * WB * EF * ER	1.25E-07	----	----	----	----	----	----	----
LOSP * GA * WB * ON * ER	1.19E-07	----	----	----	----	----	----	----
LOSP * GB * WA * ON * ER	1.19E-07	----	----	----	----	----	----	----
EXFW * OG * GA * GB * ER	1.07E-07	----	----	----	----	----	----	----
E0.2T * OG * GA * GB * EG	9.56E-08	1.09E-06	----	----	----	1.10E-05	----	----
MLOCA * EB * L2	9.22E-08	----	----	----	----	----	----	----
MLOCA * EA * L1	9.22E-08	----	----	----	----	----	----	----
E0.7T * OG * GA * GB * RT * EK	9.15E-08	1.64E-07	----	0.00E+00	----	3.02E-07	----	----
E1.0T * OG * GA * GB * EL	8.11E-08	1.08E-07	----	0.00E+00	----	1.40E-07	----	----
LOSP * WB * EF * OR * ER * FR	6.94E-08	----	----	----	----	----	----	----
LOSP * WA * EF * OR * ER * FR	6.94E-08	----	----	----	----	----	----	----

Table 3 RESULTS OF QUANTIFICATION - BOUNDING CASE (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS	STEAM	WATER
SLOCA * WA * WB * SR	6.82E-08	----	----	----	----	----	----	----
E0.2T * OG * GB * WA	6.26E-08	----	----	----	----	0.00E+00	----	----
E0.2T * OG * GA * WB	6.26E-08	----	----	----	----	0.00E+00	----	----
LLOCA * LC * LD	6.12E-08	----	----	----	----	----	----	----
E0.2T * OG * WA * WB	5.47E-08	----	----	----	----	0.00E+00	----	----
LOSP * DA * GB * EF * FR * ER	5.07E-08	----	----	----	----	----	----	----
LOSP * DB * GA * EF * FR * ER	5.07E-08	----	----	----	----	----	----	----
LLOCA * EA * LB	3.60E-08	----	----	----	----	----	----	----
LLOCA * EB * LA	3.60E-08	----	----	----	----	----	----	----
SGTR * WA * WB * SR	3.26E-08	----	----	----	----	----	----	----
LOSP * WA * PB * ER	2.87E-08	----	----	----	----	----	----	----
LOSP * WB * PA * ER	2.87E-08	----	----	----	----	----	----	----
LOSP * DA * WB * ER	2.15E-08	----	----	----	----	----	----	----
LOSP * DB * WA * ER	2.15E-08	----	----	----	----	----	----	----
E0.3T * SA * SB * EH * C2	2.03E-08	1.25E-07	----	----	----	----	----	----
MLOCA * WB * L1	1.96E-08	----	----	----	----	----	----	----
MLOCA * WA * L2	1.96E-08	----	----	----	----	----	----	----
E0.4T * SA * SB * EI * C2	1.90E-08	6.90E-08	----	----	----	----	----	----
L1DC * PA * EF * FR * FR	7.35E-09	----	----	----	----	----	----	----
L1DC * PB * EF * FR * FR	7.35E-09	----	----	----	----	----	----	----
SLOCA * L1 * L2 * XA	5.94E-09	----	----	----	----	----	----	----
E0.4T * SA * SB * RW * C2	3.93E-09	----	----	6.56E-08	0.00E+00	----	----	----
E0.5T * SA * SB * RW * C2	3.36E-09	----	----	2.40E-08	0.00E+00	----	----	----
E0.3T * SA * SB * H3 * C2	2.64E-09	----	1.25E-07	----	----	----	----	----
E0.3T * SA * SB * RW * C2	2.47E-09	----	----	1.23E-07	0.00E+00	----	----	----
Other Sequences Not Included in Model	8.00E-06	----	----	----	----	----	----	----
Core Damage Due to Seismic Events	2.92E-05	4.39E-05	4.23E-05	4.18E-05	3.99E-05	3.23E-04		
E0.7T * OG * GA * GB	5.65E-06	----	----	----	----	1.86E-05		
E0.5T * OG * GA * GB	2.32E-06	----	----	----	----	1.84E-05		
E0.4T * OG * GA * GB	2.08E-06	----	----	----	----	3.60E-05		
E0.7T * OG * SA * SB * PA * PC	1.63E-06	----	----	----	5.85E-06	0.00E+00		
E0.7T * OG * GA * GB * EK	1.59E-06	2.85E-06	----	----	----	5.26E-06		
E0.3T * SA * SB	1.54E-06	----	----	----	----	----		
E0.3T * OG * GA * GB	1.34E-06	----	----	----	----	7.34E-05		
E0.7T * OG * EK * RW	1.28E-06	2.43E-06	----	3.79E-06	----	0.00E+00		
E1.0T * OG * GA * GB	1.25E-06	----	----	----	----	2.16E-06		

Table 3 RESULTS OF QUANTIFICATION - BOUNDING CASE (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS
E0.2T * OG * GA * GB	1.09E-06	----	----	----	----	1.27E-04
E0.7T * OG * SA * SB * EK * RW	7.98E-07	1.43E-06	----	2.34E-06	0.00E+00	0.00E+00
E0.7T * OG * EK * H2	7.00E-07	1.32E-06	2.50E-06	0.00E+00	----	0.00E+00
E0.5T * OG * GA * GB * EJ	6.86E-07	1.81E-06	----	----	----	5.43E-06
E1.0L * OG * GA * GB	6.52E-07	----	----	----	6.52E-07	1.12E-06
E0.5T * OG * EJ * RW	5.97E-07	1.75E-06	----	4.26E-06	----	0.00E+00
E0.7L * OG * SA * SC	5.57E-07	----	----	----	----	0.00E+00
E0.7L * OG * GA * GB	5.38E-07	----	----	----	----	1.77E-06
E0.5T * OG * EJ * H2	5.13E-07	1.51E-06	3.67E-06	0.00E+00	----	0.00E+00
E0.4T * OG * GA * GB * EI	5.06E-07	1.83E-06	----	----	----	8.73E-06
E0.4T * OG * EI * H2	4.53E-07	1.96E-06	6.47E-06	0.00E+00	----	0.00E+00
E0.7T * OG * SA * SB * EK * H2	4.34E-07	7.78E-07	1.55E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * PA * PB	4.24E-07	----	----	----	3.53E-06	0.00E+00
E0.4T * OG * EI * RW	3.83E-07	1.66E-06	----	6.38E-06	----	0.00E+00
E1.0L * OG * SA * SC	3.31E-07	----	----	----	----	0.00E+00
E1.0T * OG * SA * SB * EL * RW	2.95E-07	3.95E-07	----	4.75E-07	----	0.00E+00
E1.0T * OG * GA * GB * RT * EL	2.24E-07	3.01E-07	----	----	----	3.88E-07
E0.3T * OG * GA * GB * EH	2.09E-07	1.29E-06	----	----	----	1.14E-05
E0.4T * OG * SA * SB * PA * PB	2.03E-07	----	----	----	4.07E-06	0.00E+00
E0.5T * OG * SA * SB * EJ * RW	1.65E-07	4.35E-07	----	1.17E-06	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * EJ * H2	1.42E-07	3.74E-07	1.01E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * PA * PB * EJ	1.38E-07	3.65E-07	----	0.00E+00	1.15E-06	0.00E+00
E0.2T * OG * GA * GB * EG	9.56E-08	1.09E-06	----	----	----	1.10E-05
E0.7T * OG * GA * GB * RT * EK	9.15E-08	1.64E-07	----	0.00E+00	----	3.02E-07
E1.0T * OG * GA * GB * EL	8.11E-08	1.08E-07	----	0.00E+00	----	1.40E-07
E0.2T * OG * GB * WA	6.26E-08	----	----	----	----	0.00E+00
E0.2T * OG * GA * WB	6.26E-08	----	----	----	----	0.00E+00
E0.2T * OG * WA * WB	5.47E-08	----	----	----	----	0.00E+00
E0.3T * SA * SB * EH * C2	2.03E-08	1.25E-07	----	----	----	----
E0.4T * SA * SB * EI * C2	1.90E-08	6.90E-08	----	----	----	----
E0.4T * SA * SB * RW * C2	3.93E-09	----	----	6.56E-08	0.00E+00	----
E0.5T * SA * SB * RW * C2	3.36E-09	----	----	2.40E-08	0.00E+00	----
E0.3T * SA * SB * H3 * C2	2.64E-09	----	1.25E-07	----	----	----
E0.3T * SA * SB * RW * C2	2.47E-09	----	----	1.23E-07	0.00E+00	----

Table 4 RESULTS OF BOUNDING CALCULATIONS INCLUDING BOLT FAILURES DURING SEISMIC EVENTS

System	Core Damage Frequency Total	Core Damage Frequency Change
Base Case	3.1E-04	---
Failure of:		
- AFW	3.2E-04	1.5E-05
- HPI	3.2E-04	1.3E-05
- LPI	3.2E-04	1.3E-05
- CCW	3.2E-04	1.1E-05
- EPS / SWS	6.0E-04	2.9E-04
- LOCA	6.0E-04	2.9E-04
- ACCUMULATOR	No Change	(1)
- CHARGING	No Change	(2)
- STEAM	No Change	(3)
- FEEDWATER	No Change	(4)

(1) Failure of the accumulators during a seismic event is assumed not to cause an initiating event directly. Because the Large LOCA initiated by seismic failure is assumed to cause failure of multiple RCS loops, success or failure of the accumulators does not affect core damage likelihood.

(2) The charging system is included with failure of the HPI system for the purposes of this analysis. Failure of the normal charging function during seismic events does not significantly affect the likelihood of core damage.

(3) An increase in the likelihood of failure of the steam lines downstream of the MSIV's during a seismic event has no effect on the sequences modeled. The frequency of core damage is dominated by the failures in the other plant systems.

(4) An increase in the likelihood of failure of the feedwater system during a seismic event has no effect on the sequences modeled. The frequency of core damage is dominated by the failures in the other plant systems.

Table 5 Change in Core Melt Frequency - Best Case

	Base	AFW	HPI	LPI	CCW	EPS/SWS
Core Damage Total	3.08E-04	3.17E-04	3.15E-04	3.15E-04	3.19E-04	3.40E-04
LOSP * GA * GB * ER	3.97E-05	----	----	----	----	----
RTRIP * PA * PB	2.11E-05	----	----	----	----	----
PLMFW * PA * PB	1.71E-05	----	----	----	----	----
SLOCA * L1 * L2	1.69E-05	----	----	----	----	----
TTRIP * PA * PB	1.32E-05	----	----	----	----	----
RTRIP * SA * SB * OR	1.17E-05	----	----	----	----	----
L1DC * EF * FR * FR	1.14E-05	----	----	----	----	----
SLBI * ON	9.39E-06	----	----	----	----	----
EXFW * PA * PB	9.34E-06	----	----	----	----	----
FCRCC	9.00E-06	----	----	----	----	----
TTRIP * SA * SB * PR	7.29E-06	----	----	----	----	----
LOMF * SA * SB	5.69E-06	----	----	----	----	----
E0.7T * OG * GA * GB	5.65E-06	----	----	----	----	1.86E-05
SLOCA * EA * L2	5.55E-06	----	----	----	----	----
SLOCA * EB * L1	5.55E-06	----	----	----	----	----
LOSP * PA * PB	4.96E-06	----	----	----	----	----
FPCC	4.20E-06	----	----	----	----	----
LOPF * PA * PB	3.79E-06	----	----	----	----	----
FSRCC	3.60E-06	----	----	----	----	----
SLOCA * WA * L2	3.26E-06	----	----	----	----	----
SLOCA * WB * L1	3.26E-06	----	----	----	----	----
RTRIP * WA * WB * SR	3.14E-06	----	----	----	----	----
LOSP * GA * GB * EF * ER * FR	3.01E-06	----	----	----	----	----
LOSP * WA * WB * ER	2.57E-06	----	----	----	----	----
PLMFW * WA * WB * SR	2.54E-06	----	----	----	----	----
LOSW	2.52E-06	----	----	----	----	----
SLOCA * WB * EA	2.41E-06	----	----	----	----	----
SLOCA * WA * EB	2.41E-06	----	----	----	----	----
FTBLP * GA * GB * ER	2.33E-06	----	----	----	----	----
LOSP * GA * GB * EF * ER	2.33E-06	----	----	----	----	----
E0.5T * OG * GA * GB	2.32E-06	----	----	----	----	1.84E-05
LOSP * GA * WB * ER	2.12E-06	----	----	----	----	----
LOSP * GB * WA * ER	2.12E-06	----	----	----	----	----
RTRIP * ON	2.11E-06	----	----	----	----	----
E0.4T * OG * GA * GB	2.08E-06	----	----	----	----	----

Table 5 Change in Core Melt Frequency - Best Case (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS
TTRIP * WA * WB * SR	1.95E-06	----	----	----	----	----
V SEQUENCE	1.84E-06	----	----	----	----	----
PLMFW * ON	1.71E-06	----	----	----	----	----
FCRSW	1.65E-06	----	----	----	----	----
E0.7T * OG * SA * SB * PA * PC	1.63E-06	----	----	----	5.85E-06	0.00E+00
FCRAC	1.62E-06	----	----	----	----	----
FLSW	1.60E-06	----	----	----	----	----
E0.7T * OG * GA * GB * EK	1.59E-06	2.85E-06	----	----	----	5.26E-06
LOSP * GA * GB * ON * ER	1.56E-06	----	----	----	----	----
E0.3T * SA * SB	1.54E-06	----	----	----	----	----
FLLP * GA * GB * ER	1.43E-06	----	----	----	----	----
EXFW * WA * WB * SR	1.38E-06	----	----	----	----	----
LPCC	1.38E-06	----	----	----	----	----
E0.3T * OG * GA * GB	1.34E-06	----	----	----	----	----
TTRIP * ON	1.32E-06	----	----	----	----	----
E0.7T * OG * EK * RW	1.28E-06	2.43E-06	----	3.79E-06	----	0.00E+00
TCTL * GA * GB * ER	1.28E-06	----	----	----	----	----
E1.0T * OG * GA * GB	1.25E-06	----	----	----	----	2.16E-06
LLOCA * LA * LB	1.14E-06	----	----	----	----	----
E0.2T * OG * GA * GB	1.09E-06	----	----	----	----	----
EXFW * ON	9.34E-07	----	----	----	----	----
RTRIP * EF * L1 * L2	8.34E-07	----	----	----	----	----
RTRIP * EF * L1 * L2	8.34E-07	----	----	----	----	----
RTRIP * EF * O3	8.32E-07	----	----	----	----	----
E0.7T * OG * SA * SB * EK * RW	7.98E-07	1.43E-06	----	2.34E-06	0.00E+00	0.00E+00
E0.7T * OG * EK * H2	7.00E-07	1.32E-06	2.50E-06	0.00E+00	----	0.00E+00
E0.5T * OG * GA * GB * EJ	6.86E-07	1.81E-06	----	----	----	5.43E-06
LOMF * RT * OH	6.59E-07	----	----	----	----	----
E1.0L * OG * GA * GB	6.52E-07	----	----	----	----	1.12E-06
LOSP * WA * WB * EF * ER * FR	6.34E-07	----	----	----	----	----
E0.5T * OG * EJ * RW	5.97E-07	1.75E-06	----	4.26E-06	----	0.00E+00
E0.7L * OG * SA * SC	5.57E-07	----	----	----	----	0.00E+00
E0.7L * OG * GA * GB	5.38E-07	----	----	----	----	1.77E-06
TTRIP * EF * L1 * L2	5.20E-07	----	----	----	----	----
TTRIP * EF * O3	5.18E-07	----	----	----	----	----

Table 5 Change in Core Melt Frequency - Best Case (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS
E0.5T * OG * EJ * H2	5.13E-07	1.51E-06	3.67E-06	0.00E+00	----	0.00E+00
E0.4T * OG * GA * GB * EI	5.06E-07	6.96E-07	----	----	----	----
FSRAC	4.90E-07	----	----	----	----	----
E0.4T * OG * EI * H2	4.53E-07	6.69E-07	----	----	----	----
TTRIP * RT * OH	4.49E-07	----	----	----	----	----
E0.7T * OG * SA * SB * EK * H2	4.34E-07	7.78E-07	1.55E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * PA * PB	4.24E-07	----	----	----	3.53E-06	0.00E+00
E0.4T * OG * EI * RW	3.83E-07	5.66E-07	----	----	----	0.00E+00
LOPF * ON	3.79E-07	----	----	----	----	----
LOSP * DA * EF * ER * FR	3.75E-07	----	----	----	----	----
LOSP * DB * EF * ER * FR	3.75E-07	----	----	----	----	----
EXFW * EF * L1 * L2	3.68E-07	----	----	----	----	----
EXFW * EF * O3	3.66E-07	----	----	----	----	----
E1.0L * OG * SA * SC	3.31E-07	----	----	----	----	0.00E+00
LOSP * WA * WB * ON * ER	3.28E-07	----	----	----	----	----
E1.0T * OG * SA * SB * EL * RW	2.95E-07	3.95E-07	----	4.75E-07	----	0.00E+00
LCV * ON	2.83E-07	----	----	----	----	----
SLBI * L1 * L2	2.73E-07	----	----	----	----	----
SLBI * O3	2.72E-07	----	----	----	----	----
ELOCA	2.66E-07	----	----	----	----	----
SGTR * EF * OD	2.62E-07	----	----	----	----	----
MLOCA * L1 * L2	2.54E-07	----	----	----	----	----
RTRIP * OG * GA * GB * ER	2.44E-07	----	----	----	----	----
C1MSIV* ON	2.39E-07	----	----	----	----	----
LOSP * GA * WB * EF * ER * FR	2.31E-07	----	----	----	----	----
LOSP * GB * WA * EF * ER * FR	2.31E-07	----	----	----	----	----
E1.0T * OG * GA * GB * RT * EL	2.24E-07	3.01E-07	----	----	----	3.88E-07
TLMFW * ON	2.24E-07	----	----	----	----	----
LOSP * GA * EF * OR * ER * FR	2.10E-07	----	----	----	----	----
LOSP * GB * EF * OR * ER * FR	2.10E-07	----	----	----	----	----
RTRIP * EF * OR * FR * FR	2.09E-07	----	----	----	----	----
E0.3T * OG * GA * GB * EH	2.09E-07	----	----	----	----	----
E0.4T * OG * SA * SB * PA * PB	2.03E-07	----	----	----	4.07E-06	----
LOSP * EF * OR * ER * FR	2.02E-07	----	----	----	----	----
LOSP * GA * PB * ER	1.98E-07	----	----	----	----	----
LOSP * GB * PA * ER	1.98E-07	----	----	----	----	----

Table 5 Change in Core Melt Frequency - Best Case (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS
PLMFW * OG * GA * GB * ER	1.97E-07	----	----	----	----	----
E0.5T * OG * SA * SB * EJ * RW	1.65E-07	4.35E-07	----	1.17E-06	0.00E+00	0.00E+00
LCV * EF * 03	1.52E-07	----	----	----	----	----
TTRIP * OG * GA * GB * ER	1.52E-07	----	----	----	----	----
LOSP * WA * WB * EF * ER	1.51E-07	----	----	----	----	----
LOSP * DA * GB * ER	1.48E-07	----	----	----	----	----
LOSP * DB * GA * ER	1.48E-07	----	----	----	----	----
E0.5T * OG * SA * SB * EJ * H2	1.42E-07	3.74E-07	1.01E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * PA * PB * EJ	1.38E-07	3.65E-07	----	0.00E+00	1.15E-06	0.00E+00
LOSP * GB * WA * EF * ER	1.25E-07	----	----	----	----	----
LOSP * GA * WB * EF * ER	1.25E-07	----	----	----	----	----
LOSP * GA * WB * ON * ER	1.19E-07	----	----	----	----	----
LOSP * GB * WA * ON * ER	1.19E-07	----	----	----	----	----
EXFW * OG * GA * GB * ER	1.07E-07	----	----	----	----	----
E0.2T * OG * GA * GB * EG	9.56E-08	----	----	----	----	----
MLOCA * EB * L2	9.22E-08	----	----	----	----	----
MLOCA * EA * L1	9.22E-08	----	----	----	----	----
E0.7T * OG * GA * GB * RT * EK	9.15E-08	1.64E-07	----	0.00E+00	----	3.02E-07
E1.0T * OG * GA * GB * EL	8.11E-08	1.08E-07	----	0.00E+00	----	1.40E-07
LOSP * WB * EF * OR * ER * FR	6.94E-08	----	----	----	----	----
LOSP * WA * EF * OR * ER * FR	6.94E-08	----	----	----	----	----
SLOCA * WA * WB * SR	6.82E-08	----	----	----	----	----
E0.2T * OG * GB * WA	6.26E-08	----	----	----	----	----
E0.2T * OG * GA * WB	6.26E-08	----	----	----	----	----
LLOCA * LC * LD	6.12E-08	----	----	----	----	----
E0.2T * OG * WA * WB	5.47E-08	----	----	----	----	----
LOSP * DA * GB * EF * FR * ER	5.07E-08	----	----	----	----	----
LOSP * DB * GA * EF * FR * ER	5.07E-08	----	----	----	----	----
LLOCA * EA * LB	3.60E-08	----	----	----	----	----
LLOCA * EB * LA	3.60E-08	----	----	----	----	----
SGTR * WA * WB * SR	3.26E-08	----	----	----	----	----
LOSP * WA * PB * ER	2.87E-08	----	----	----	----	----
LOSP * WB * PA * ER	2.87E-08	----	----	----	----	----
LOSP * DA * WB * ER	2.15E-08	----	----	----	----	----
LOSP * DB * WA * ER	2.15E-08	----	----	----	----	----
E0.3T * SA * SB * EH * C2	2.03E-08	----	----	----	----	----

Table 5 Change in Core Melt Frequency - Best Case (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS
MLOCA * WB * L1	1.96E-08	----	----	----	----	----
MLOCA * WA * L2	1.96E-08	----	----	----	----	----
E0.4T * SA * SB * EI * C2	1.90E-08	2.61E-08	----	----	----	----
L1DC * PA * EF * FR * FR	7.35E-09	----	----	----	----	----
L1DC * PB * EF * FR * FR	7.35E-09	----	----	----	----	----
SLOCA * L1 * L2 * XA	5.94E-09	----	----	----	----	----
E0.4T * SA * SB * RW * C2	3.93E-09	----	----	----	0.00E+00	----
E0.5T * SA * SB * RW * C2	3.36E-09	----	----	2.40E-08	0.00E+00	----
E0.3T * SA * SB * H3 * C2	2.64E-09	----	----	----	----	----
E0.3T * SA * SB * RW * C2	2.47E-09	----	----	----	0.00E+00	----
Other Sequences Not Included in Model	8.00E-06	----	----	----	----	----
Core Damage Due to Seismic Events	2.92E-05	3.81E-05	3.62E-05	3.61E-05	3.99E-05	6.14E-05
E0.7T * OG * GA * GB	5.65E-06	----	----	----	----	1.86E-05
E0.5T * OG * GA * GB	2.32E-06	----	----	----	----	1.84E-05
E0.4T * OG * GA * GB	2.08E-06	----	----	----	----	----
E0.7T * OG * SA * SB * PA * PC	1.63E-06	----	----	----	5.85E-06	0.00E+00
E0.7T * OG * GA * GB * EK	1.59E-06	2.85E-06	----	----	----	5.26E-06
E0.3T * SA * SB	1.54E-06	----	----	----	----	----
E0.3T * OG * GA * GB	1.34E-06	----	----	----	----	----
E0.7T * OG * EK * RW	1.28E-06	2.43E-06	----	3.79E-06	----	0.00E+00
E1.0T * OG * GA * GB	1.25E-06	----	----	----	----	2.16E-06
E0.2T * OG * GA * GB	1.09E-06	----	----	----	----	----
E0.7T * OG * SA * SB * EK * RW	7.98E-07	1.43E-06	----	2.34E-06	0.00E+00	0.00E+00
E0.7T * OG * EK * H2	7.00E-07	1.32E-06	2.50E-06	0.00E+00	----	0.00E+00
E0.5T * OG * GA * GB * EJ	6.86E-07	1.81E-06	----	----	----	5.43E-06
E1.0L * OG * GA * GB	6.52E-07	----	----	----	----	1.12E-06
E0.5T * OG * EJ * RW	5.97E-07	1.75E-06	----	4.26E-06	----	0.00E+00
E0.7L * OG * SA * SC	5.57E-07	----	----	----	----	0.00E+00
E0.7L * OG * GA * GB	5.38E-07	----	----	----	----	1.77E-06
E0.5T * OG * EJ * H2	5.13E-07	1.51E-06	3.67E-06	0.00E+00	----	0.00E+00
E0.4T * OG * GA * GB * EI	5.06E-07	6.96E-07	----	----	----	----
E0.4T * OG * EI * H2	4.53E-07	6.69E-07	----	----	----	----
E0.7T * OG * SA * SB * EK * H2	4.34E-07	7.78E-07	1.55E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * PA * PB	4.24E-07	----	----	----	3.53E-06	0.00E+00

Table 5 Change in Core Melt Frequency - Best Case (Cont'd)

	Base	AFW	HPI	LPI	CCW	EPS/SWS
E0.4T * OG * EI * RW	3.83E-07	5.66E-07	----	----	----	0.00E+00
E1.0L * OG * SA * SC	3.31E-07	----	----	----	----	0.00E+00
E1.0T * OG * SA * SB * EL * RW	2.95E-07	3.95E-07	----	4.75E-07	----	0.00E+00
E1.0T * OG * GA * GB * RT * EL	2.24E-07	3.01E-07	----	----	----	3.88E-07
E0.3T * OG * GA * GB * EH	2.09E-07	----	----	----	----	----
E0.4T * OG * SA * SB * PA * PB	2.03E-07	----	----	----	4.07E-06	----
E0.5T * OG * SA * SB * EJ * RW	1.65E-07	4.35E-07	----	1.17E-06	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * EJ * H2	1.42E-07	3.74E-07	1.01E-06	0.00E+00	0.00E+00	0.00E+00
E0.5T * OG * SA * SB * PA * PB * EJ	1.38E-07	3.65E-07	----	0.00E+00	1.15E-06	0.00E+00
E0.2T * OG * GA * GB * EG	9.56E-08	----	----	----	----	----
E0.7T * OG * GA * GB * RT * EK	9.15E-08	1.64E-07	----	0.00E+00	----	3.02E-07
E1.0T * OG * GA * GB * EL	8.11E-08	1.08E-07	----	0.00E+00	----	1.40E-07
E0.2T * OG * GB * WA	6.26E-08	----	----	----	----	----
E0.2T * OG * GA * WB	6.26E-08	----	----	----	----	----
E0.2T * OG * WA * WB	5.47E-08	----	----	----	----	----
E0.3T * SA * SB * EH * C2	2.03E-08	----	----	----	----	----
E0.4T * SA * SB * EI * C2	1.90E-08	2.61E-08	----	----	----	----
E0.4T * SA * SB * RW * C2	3.93E-09	----	----	----	0.00E+00	----
E0.5T * SA * SB * RW * C2	3.36E-09	----	----	2.40E-08	0.00E+00	----
E0.3T * SA * SB * H3 * C2	2.64E-09	----	----	----	----	----
E0.3T * SA * SB * RW * C2	2.47E-09	----	----	----	0.00E+00	----

Table 6 Top Event Coding

INITIATING EVENTS		
C1MSIV	Closure of One MSIV	
E0.2T	Seismic Initiating Event - 0.2g	(3.6E-04)
E0.3T	Seismic Initiating Event - 0.3g	(1.1E-04)
E0.4T	Seismic Initiating Event - 0.4g	(4.3E-05)
E0.5T	Seismic Initiating Event - 0.5g	(2.0E-05)
E0.7L	Seismic Initiating Event - LLOCA, 0.7g	(1.8E-06)
E0.7T	Seismic Initiating Event - General Transient, 0.7g	(1.9E-05)
E1.0L	Seismic Initiating Event - LLOCA, 1.0g	(1.1E-06)
E1.0T	Seismic Initiating Event - General Transient, 1.0g	(2.2E-06)
ELOCA	Excessive LOCA	
EXFW	Excessive Feedwater Flow	
FCRAC	Fire, Control Room - Loss of Power	
FCRCC	Fire, Control Room - Loss of PCC	
FCRSW	Fire, Control Room - Loss of Service Water	
FLLP	Flood - LOSP, Turbine Building	
FLSW	External Flood - Service Water	
FPCC	Fire, PCC Area	
FSRAC	Fire, Spreading Room - Loss of AC Power	
FSRCC	Fire, Cable Spreading Room - Loss of PCC	
FTBLP	Fire, Turbine Building	
L1DC	Loss of One DC Bus	
LCV	Loss of Condenser Vacuum	
LLOCA	Large LOCA	
LOMF	Loss of Main Feedwater (Total and Partial)	
LOPF	Loss of Primary Flow	
LOSP	Loss of Offsite Power	
LOSW	Loss of Service Water	
LPCC	Loss of Primary Component Cooling	
MLOCA	Medium LOCA	
PLMFW	Partial Loss of Main Feedwater	
RTRIP	Reactor Trip	
SGTR	Steam Generator Tube Rupture	
SLBI	Steam Line Break Inside Containment	
SLOCA1	Small LOCA (nonisolable)	
SLOCA2	Small LOCA (isolable)	
TCTL	Truck Crash into Transmission Line	
TLMFW	Total Loss of Main Feedwater	
TTRIP	Turbine Trip	
V	Interfacing Systems LOCA	

Table 6 (cont'd)

SYSTEM EVENT TREE TOP EVENTS

C2	Containment Purge Isolation
DA	DC Power Train A
DB	DC Power Train B
EA	Engineered Safety Features Actuation System (ESFAS) Train A
EB	Engineered Safety Features Actuation System (ESFAS) Train B
EF	Auxiliary Feedwater System
EG	Auxiliary Feedwater System - Earthquake 0.2g
EH	Auxiliary Feedwater System - Earthquake 0.3g
EI	Auxiliary Feedwater System - Earthquake 0.4g
EJ	Auxiliary Feedwater System - Earthquake 0.5g
EK	Auxiliary Feedwater System - Earthquake 0.7g
EL	Auxiliary Feedwater System - Earthquake 1.0g
ER	Electric Power System Recovery Models
FR	Auxiliary Feedwater Recovery
GA	Diesel Generator Train A
GB	Diesel Generator Train B
H2	High Pressure Injection (HPI) for SLOCA, etc.
H3	High Pressure Injection (HPI) for ATWS Events
L1	Low Pressure Injection (LPI) Train A - MLOCA Miniflow
L2	Low Pressure Injection (LPI) Train B - MLOCA Miniflow
LA	Low Pressure Injection (LPI) Train A - LLOCA
LB	Low Pressure Injection (LPI) Train B - LLOCA
LC	Low Pressure Recirculation (LPR) Train A
LD	Low Pressure Recirculation (LPR) Train B
O3	Operator Action - LPR or HPR
OD	Operator Action to Depressurize Steam Generators
OG	Electric Power Systems - Offsite Grid
OH	Operator Action - Manual Reactor Shutdown, ATWS
ON	Operator Action - Plant Stabilization
OR	Operator Action - Feed and Bleed, SGTR Break Flow
PA	Primary Component Cooling Train A
PB	Primary Component Cooling Train B
PR	PORV's in Feed and Bleed
RT	Reactor Protection System
RW	Refueling Water Storage Tank (RWST)
SA	Solid State Protection System (SSPS) - Train A
SB	Solid State Protection System (SSPS) - Train B
SC	Main Steam Functions - Steam Generator Relief
SR	Service Water System Recovery
WA	Service Water System - Train A
WB	Service Water System - Train B
XA	Containment Spray Recirculation Train A

IRRAS 2.0 - More than a Fault Tree Code

K.D. RUSSELL, M.B. SATTISON, D. RASMUSON**

Idaho National Engineering Laboratory
EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

**United States Nuclear Regulatory Commission

Abstract

The Integrated Reliability and Risk Analysis System (IRRAS) is being developed at the Idaho National Engineering Laboratory (INEL) as the U.S. Nuclear Regulatory Commission's (NRC's) state-of-the-art microcomputer-based probabilistic risk assessment (PRA) model development and analysis tool to address key nuclear plant safety issues.

IRRAS is an integrated PRA software tool that gives the user the ability to create and analyze fault trees and accident sequences using an IBM-PC. This program provides functions that range from graphical fault tree construction to cut set generation and quantification. Also provided in the system is an integrated full-screen editor for use when interfacing with remote mainframe computer systems.

The INEL role in the IRRAS program is that of software developer and interface to the user community, including training and technology transfer. Version 1.0 of the IRRAS program was released in February of 1987 to prove the concept of performing this kind of analysis on microcomputers. This version contained many of the basic features needed for fault tree analysis and was received very well by the PRA community. Since the release of Version 1.0, many user comments and enhancements have been incorporated into the program providing a much more powerful and user-friendly system. This version will be designated "IRRAS 2.0" and will be released in October of 1988.

IRRAS has all the capabilities and functions required to create, modify, reduce, and analyze fault tree models used in the analysis of complex systems and processes. IRRAS uses advanced graphic and analytical techniques to achieve the greatest possible realization of the potential of the microcomputer. When the needs of the user exceed this potential IRRAS can call upon the power of the mainframe computer. Version 2.0 of IRRAS provides all of the same capabilities as Version 1.0 and adds a relational data base facility for managing the data, improved functionality, and improved algorithm performance.

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Introduction

The Integrated Reliability and Risk Analysis System (IRRAS) is a software tool which gives the user the ability to create and analyze fault trees using an IBM-PC. This program provides the functionality required for probabilistic risk assesment (PRA) functions ranging from graphical fault tree construction to cut set generation and quantification.

At the center of the PRA analysis is the fault tree model. This model, along with the component reliability data, provides the basis for risk analysis in most PRA methodologies. Prior to the development of IRRAS, an analyst would generate models using an alphanumeric text editor. The information would be input in a format compatible with the analysis software used to process the information. When changes to the model were needed, the analyst would return to the text file and modify the card images to reflect the changes. This process is prone to errors and is difficult for the analyst to visualize. IRRAS eliminates these problems.

IRRAS was developed to automate the model creation, manipulation, modification, and quantification processes. Designed for the IBM-PC, IRRAS is readily accessible and portable. Taking advantage of the state-of-the-art in computer graphics and analysis algorithms, IRRAS is powerful and efficient. However, it is recognized that there are limitations to a PC-based program, so IRRAS has been designed to easily communicate with a mainframe computer for extremely large and complex models. An integrated full-screen editor and various conversion utilities have been provided for communicating with remote computer systems and software. This communication package includes a comprehensive full screen editor with the ability to upload and download files. This integrated editor lets the user interface with remote computers for those tasks which are beyond the ability of todays microcomputers.

IRRAS simplifies the analysis process and automates the construction of input to the analysis software. The analyst can graphically construct and modify fault trees. This program gives the user better visualization of the fault tree and simplifies the construction and maintenance. All of the basic constructs involved in fault tree analysis are supported, including the ability to input tables of events. Once the tree is constructed, the program will automatically generate the input for the analysis software. The graphical output from IRRAS can also be printed on a laser printer for report quality documentation of the work.

After constructing the PRA models, the analyst can store these models in an integrated relational data base. IRRAS then manages this data for the user during the analysis process. Included in the management facilities is the detection of changes to the model information and automatic recalculation of associated data. This powerful feature can greatly reduce the time required to track and propagate model changes throughout a complex system.

IRRAS version 2.0 also includes the ability to link fault trees according to analyst-determined logic to create core melt sequence cut sets. These sequence cut sets can then be analyzed using the same powerful tools provided for fault tree cut sets.

Many of the features of mainframe codes have also been incorporated into IRRAS 2.0. Improved fault tree reduction techniques such as identification of independent subtrees and coalescing gates have demonstrated significant performance improvements over IRRAS 1.0. Many more error checking routines have been provided to aid the user in debugging and checking the completed fault trees.

The addition of a module to automatically generate fault trees on the PC directly from the alphanumeric input used by mainframe codes such as SETS lets the analyst quickly load existing models and data from prior studies into IRRAS for modification and re-analysis.

The IRRAS fault tree analysis tool and the graphical fault tree editor provide the basis for an integrated analysis capability. The improved fault tree reduction methods in the analysis program and the modern graphical techniques in the fault tree editor provide a tool with a level of functionality and automation which is superior to other systems.

IRRAS History

The IRRAS software development project was started as a result of a recognized need for microcomputer based software to aid a PRA analyst. The initial scope of the project was to provide a software package which could demonstrate the feasability of using the microcomputer as a workstation for performing PRA analyses. This package did not necessarily need to perform all of the functions required, however, it did need to provide certain essential functions such as fault tree construction, failure data input, cut set generation, and cut set quantification. The result of this software development project was IRRAS 1.0. This version of the software was released in February of 1987 and contained only the essential concepts mentioned above.

IRRAS 1.0 was an immediate success and clearly demonstrated not only the tremendous need, but also the feasability of performing this work on a microcomputer. As a result of this success, IRRAS 2.0 development was begun. This package was designed to be a comprehensive PRA analysis package and include all the functions necessary for a PRA analyst to perform his work. The areas which were not treated in version 1.0 were addressed and a complete integrated package was developed. Since IRRAS version 2.0 was a complete rewrite from version 1.0, a thorough test plan was necessary. The major features of IRRAS 2.0 along with an Alpha test was completed in early March of this year. Following the Alpha test, approximately 15 sites were selected from among the sites currently using IRRAS 1.0. These sites were sent a Beta test version of IRRAS 2.0. In May of this year we completed the Beta test and began work on fixing any bugs found and including those desired new features which we could reasonably encorporate into version 2.0. IRRAS 2.0 is now ready for distribution.

IRRAS 2.0 Features

The success of IRRAS 1.0 demonstrated the great need for easily accessable and useful PRA tools. Even the basic tools provided in IRRAS 1.0 were received with much enthusiasm. With these concepts in mind, IRRAS 2.0 was designed. This system is a major rewrite from the software contained in Version 1.0. It contains many features which significantly improve the usefulness and flexibility of the system. The Beta test of IRRAS provided much positive feedback on the user interface and capabilities of IRRAS 2.0. As a result, IRRAS 2.0 features combine to provide a fault tree analysis tool that is powerful enough to solve the complex problems associated with fault tree analysis, yet is simple enough to be convenient and easy to use.

Relational Data Base Facility

IRRAS 1.0 used a very simple flat file system for the storage and retrieval of the PRA data. This system lacked the necessary features to allow the user to manage very complex data structures. IRRAS 2.0 includes a relational data base for managing this data. This data base allows IRRAS to automatically maintain the PRA data and track changes to the data. These changes can then be propagated throughout the system during the update phase of the analysis. The structure of the data base is shown in Figure 1.

The design of the data base allows the user to store multiple "families" in each data base. A "family" usually represents the data for one power plant. In each family, a list is maintained of all basic events in the system. This list includes a description, failure rate information, uncertainty data, and various attributes associated with each event. Space is also provided for the storage of an alternate name for associating the PRA name to the plant-specific name. The user may specify that the primary or the alternate name be used anywhere events are displayed. Any changes the user makes to the event data are automatically maintained and propagated throughout the system.

Each "family" also contains many fault tree records. Each fault tree record usually represents a single page of a fault tree. The system allows the user to assign a level to each fault tree. This level can be used in the analysis to determine which trees are "zero" level pages and which are "sub pages" to be included in other fault trees. Only "zero" level fault trees can be analyzed independently. "Zero" level fault trees also have cut sets and quantification information associated with them.

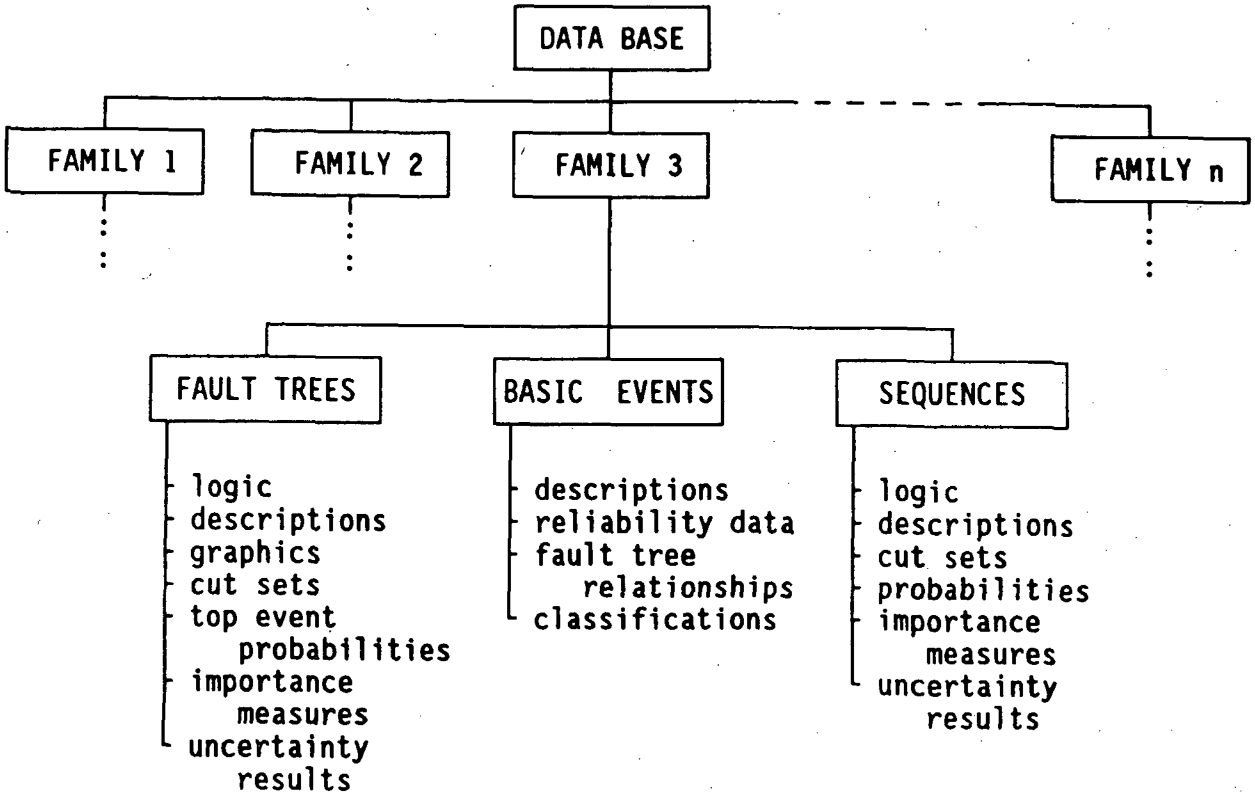

Figure 1. IRRAS 2.0 data base structure.

Each "family" may also contain event tree and accident sequence information. The user defines sequences by representing failure or success of system fault trees in the same "family". IRRAS can then calculate and store cut sets and quantification information for each sequence in the "family".

The relational data base provides an environment for the maintenance of all the information associated with a PRA. The design of the data base allows for easy modification and inclusion of data as needs change. The result is a very flexible data base that meets current needs and can be easily expanded to meet future needs. As needs change, the data base can be easily modified and populated data bases can be easily restructured to include the new data element.

Options and Menus

The most striking change to IRRAS has been the redesign of the menus. This redesign was done to provide a more user-friendly interface and to incorporate many new features. All of the menus use a hierarchical structure with state-of-the-art windows and online integrated help features. The main IRRAS menu is shown in Figure 2. This menu displays the options currently available in Version 2.0 of IRRAS.

```
                    I R R A S - P C
  Integrated Reliability and Risk Analysis System
                      Version 2.0

            EXIT     IRRAS 2.0

            BUILD    Fault Trees
            FAULT    Tree Analysis

            CREATE   Event Trees
            SEQUENCE Analysis

            MODIFY   Data Base
            REPORT   on Data Base

            UTILITY  Options


  Option... EXIT      ...Selected
```

Figure 2. IRRAS 2.0 main menu.

BUILD Fault Trees Option

The "BUILD Fault Trees" option allows the user to perform the graphical fault tree-related functions including extracting from the data base, editing, plotting, and alphanumeric-to-graphics conversions. The IRRAS Build menu is shown in Figure 3.

```
I R R A S - P C
Fault Tree Graphics System

Currently Selected Family ==> DEMO FAMILY

E - Exit

S - Select Family

X - eXtract Fault Trees

B - Build Graphics Trees

P - Plot Tree (plotter)
R - Rasterize Tree
D - Define plotter pens
A - Alpha to Graphics

Option... E
```

Figure 3. IRRAS 2.0 build menu.

Perhaps the most well recieved feature of IRRAS is the graphical fault tree contruction facility. This module is basically the same as Version 1.0; however, many cosmetic changes have been included. These changes resulted primarily from user comments and include the following:

(1) enhanced plotter output capabilities,

(2) the elimination of unused and redundant graphical symbols,

(3) the addition of a table of events, "TBL", symbol,

(4) the inclusion of "NAND" and "NOR" gates,

(5) the elimination of the "Level" feature,

(6) the elimination of the failure rate definition facility, (A more powerful capability is provided in the data base.)

(7) the inclusion of support for all 2 and 3 button mice,and

(8) the inclusion of more friendly menus and pick options.

A major new feature of the grahical editor is the ability to generate the graphical fault tree representation from an alphanumeric input format. This feature allows any fault tree generated for other codes to be easily converted to the IRRAS graphical format. This feature also allows the analyst to make changes to the alphanumeric representation of the fault tree and have these changes easily added to the graphical representation. An example of the fault tree generated from the alphanumeric representation shown in Figure 4 is displayed in Figure 5.

```
TOPGATE AND GATE1 GATE2 GATE3
GATE1 OR EVENT1 GATE4
GATE2 OR EVENT1 EVENT2 GATE5
GATE3 OR EVENT1 EVENT2 EVENT3
GATE4 2/3 EVENT8 EVENT9 EVENT10
GATE5 AND EVENT1 EVENT2
```

Figure 4. Sample alphanumeric fault tree logic.

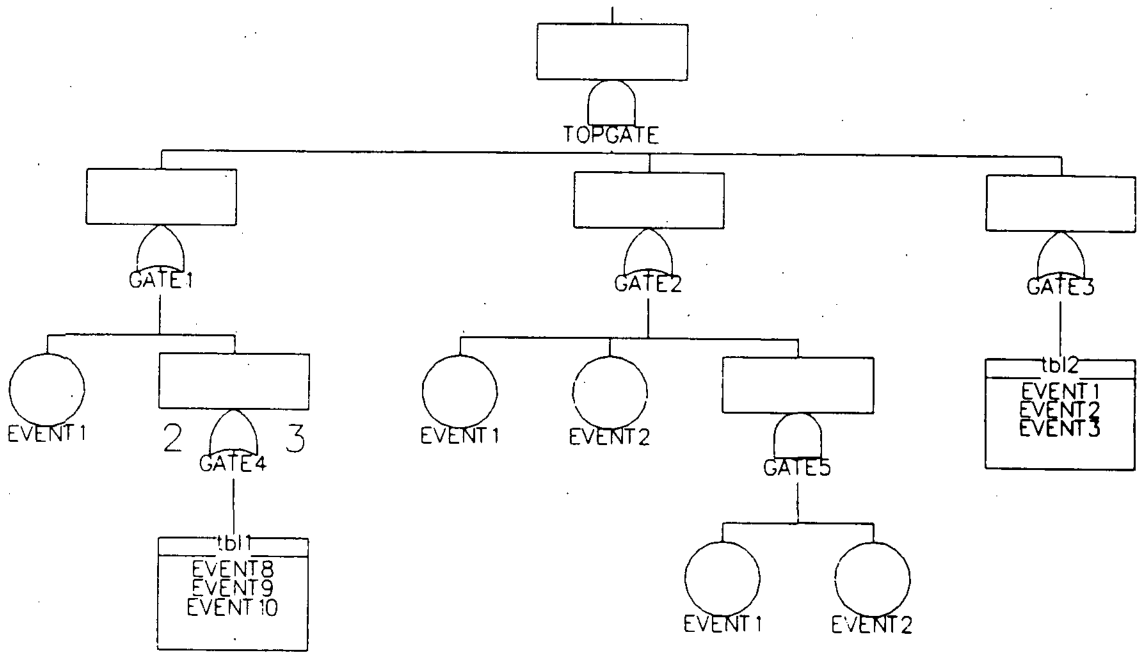

Figure 5. Sample graphical fault tree logic.

These changes significantly improve the usefulness and power of the graphical fault tree editor in IRRAS.

Fault Tree Analysis Option

The "FAULT Tree Analysis" option allows the user to perform the fault tree analysis functions including basic event modifications, cut set generation, quantification, uncertainty calculations, and modification and display of the results. The Fault tree analysis menu is shown in Figure 6.

I R R A S - P C
Fault Tree Analysis System

Currently Selected Family ==> DEMO FAMILY

E - Exit

S - Select Family
M - Modify Event Data
A - Analyze Fault Trees
C - Cut Set Editor
F - Fault Tree Editor
D - Display Fault Trees

Option... E

Figure 6. Fault Tree analysis menu.

Modify Event Data Option

One of the most important features of a good risk analysis system is the ability to define and maintain the event failure and uncertainty data. IRRAS 2.0 provides the user with a very powerful method of performing this function. IRRAS 2.0 maintains a "base case" and a "current" failure and uncertainty data table. These tables allow the user to maintain data which is considered to be the "operating" values as well as a set of data considered to be "temporary". With this arrangement, the user can perform sensitivity analysis on the data while still maintaining the base case or "operating" values associated with plant specific technical specifications.

IRRAS 2.0 also provides the user with the ability to change single event probabilities or to make "bulk" or "class" changes to a group of events. The user input screen that provides this function is shown in Figure 7. In this screen, the user may select any of the fields displayed to limit or define the class to which the uncertainty and failure data changes are to apply. The limiting fields may also be further segmented by using "don't care" and "wildcard" characters in the specific fields. This feature thus allows the user to make changes to classes of events such as, all pumps or all valves in a feedwater system. Multiple class modifications are accumulated until the user "clears" them back to the default "base case".

```
                    ┌──────────────────────────────┐
                    │ E v e n t   C l a s s   C h a n g e s │
                    └──────────────────────────────┘
                  Currently Selected Family ==> DEMO FAMILY

==========================Event=Attributes===============================
   Event Name     Fail Mode  Component Type  System Type  Component ID

                             1  2  3  4  5  6  7  8  9  10 11 12 13 14 15 16
 Location     Class Attributes

 Note:  Enter the attributes of the events to be changed.  Fields which are
        left blank are  not used.  The ? may  be used for positional "don't
        care"  characters.  The * may be  used for non-positional  variable
        length "don't care" strings.
        e.g.  AB?CD*  matches  ABXCD  through  ABXCDXXXXXXXXXX

   =======Uncertainty=Data=============  ===========Failure=Data===========
   Distribution Type ========> L        Calculation Type ========> 1
   (press Esc for list of types)        (press Esc for list of types)
   Value ==========> ------E----        Prob     ========> ------E----
   Correlation Class=> ----             Lamda    ========> ------E----
                                        Tau      ========> ------E----
```

Figure 7. Failure rate maintenance menu.

The user may select from eleven different methods of defining the failure data for IRRAS. The various options are displayed in Figure 8. As shown, all of the standard methods for defining failure data are provided. The user may also specify that the current data be modified by a percent or fixed amount, as well as define any event to be a house event. These options provide the user with a very powerful method of performing sensitivity analysis.

Failure Data Calculation Types	
Type	Calculation Method
1	Probability
2	Lamda * Mission Time
3	1 - Exp(-Lamda * Mission Time)
4	Lamda * Min(Mission Time, Tau)
5	Operating Component with Repair (Full Eq)
6	Lamda * Tau / 2.0
7	1 + (EXP(-Lamda*Tau)-1.0) / (Lamda * Tau)
8	Base Probability + Probability
9	Base Probability * Probability
T	Set to House Event (Failed, Prob=1.0)
F	Set to House Event (Successful, Prob=0.0)
Calculation Type ===>1	

Figure 8. Failure data calculation types.

The user may select from six different uncertainty distributions for each basic event. The various distributions are displayed in Figure 9. The user can also specify the input parameters to the selected distribution and a correlation class to assign a basic event to. This class of events will then be treated as if they were 100% correlated when the uncertainty analysis is performed.

Uncertainty Distribution Types	
Type	Distribution Values
L	Log Normal, Error Factor
N	Normal, Standard Dev.
B	Beta, b of Beta(a,b)
G	Gamma, a of Gamma(a)
E	Exponential, none
U	Uniform, Upper End Pt.
Distribution Type===>L	

Figure 9. Uncertainty distribution types.

Analyze Fault Trees Option

This option allows the analyst to perform the cut set generation, quantification, and uncertainty analysis on Fault trees. The cut set generation algorithm has been completely rewritten for Version 2.0. The rewrite of the algorithm was necessary to include more comprehensive fault tree reduction techniques. Some of the features added to the new algorithm are:

(1) coalescing like gates in the fault tree,

(2) automatic optimization of independent subtrees,

(3) complimented event and gate processing,

(4) error detection for "Loops", multiple top and other common errors,

(5) full implementation of paged fault trees, and

(6) the ability to handle larger fault trees.

These features provide a much more powerful fault tree reduction algorithm with significant performance improvements. Cut set generation using a sample fault tree from an existing plant showed a performance improvement from 2-1/2 hours to 20 seconds. These performance improvements were achieved because the new algorithm was able to take advantage of a large amount of independence in the tree. Most fault trees reductions will not improve this much; however, all should detect some improvement.

Fault Tree and Cut Set Editors

Two new editors have been provided in IRRAS 2.0. These editors allow the user to edit the alphanumeric logic of the Fault Tree and the generated cutset lists. Examples of the screens for the Fault tree and Cut set editors are shown in Figure 10 and 11 respectively. These two tools provide the analyst with an integrated method of creating or modifying the logic associated with a fault tree without needing to use the graphical editor and allow the analyst to edit cut sets to apply recovery. Both of these editors use a very simple to use full screen editing concept with single keystroke functions and are a powerful addition to IRRAS 2.0.

F A U L T   T R E E   E D I T O R

Exit / Add / Modify / Delete / Locate / Next / Previous
Insert / Replace / Search / Options / Undo Delete / Esc for more help

Gate Name	Type	Inputs		
C-MOV-1-FAILS	OR	C-MOV-1	DG-B	
CCS-SUPPLY	OR	C-MOV-1-FAILS	TANK	
CCS-TOP	OR	CCS-SUPPLY	CCS-TRAINS	
CCS-TRAIN-A	OR	C-CV-A	C-MOV-A	DG-A
		C-PUMP-A		
CCS-TRAIN-B	OR	C-CV-B	C-MOV-B	C-PUMP-B
		DG-B		
CCS-TRAINS	AND	CCS-TRAIN-A	CCS-TRAIN-B	

Figure 10. Fault Tree Editor menu and window.

C U T   S E T   E D I T O R

Exit / Add / Modify / Delete / Locate / Next / Previous
Insert / Replace / Search / Options / Undo Delete / Esc for more help

Set #	Event Names			
1	DG-B			
2	TANK			
3	C-MOV-1			
4	C-MOV-A	C-MOV-B		
5	C-MOV-A	C-CV-B		
6	C-MOV-A	C-PUMP-B		
7	DG-A	C-MOV-B		
8	DG-A	C-CV-B		
9	DG-A	C-PUMP-B		
10	C-PUMP-A	C-MOV-B		
11	C-PUMP-A	C-CV-B		
12	C-PUMP-A	C-PUMP-B		
13	C-CV-A	C-MOV-B		
14	C-CV-A	C-CV-B		
15	C-CV-A	C-PUMP-B		

Figure 11. Cut Set Editor menu and window.

Cut Set Display and Partition Options

The user is provided with a very versatile screen display of the cut sets for a fault tree or sequence in IRRAS 2.0. This module allows the user to page through the cut sets and display various attributes of the basic events included in the cut sets. The user may also choose to display basic event importance measures or cut set uncertainty information.

This facility takes full advantage of hierarchical menus and windows to display the cut set data, thus providing an integrated method for the user to display the results of a fault tree analysis without leaving the IRRAS system. This feature significantly reduces the time required to perform sensitivity analysis, whereby the user makes changes to fault trees or component reliability information and desire to see the resulting effect of the changes before making further modifications.

Also provided in IRRAS 2.0 is an option to allow the analyst to perform some additional analysis on the cutsets by using the partition option. An example of the "Partition" option is shown in Figure 12. This option allows the analyst to partition the cutsets by selecting only those cutsets which contain a "Qualified" event.

```
                 Partition Cut Sets

      Currently Selected Family ===> DEMO FAMILY
      Total Family Events =======>   18
      Total Qualified Events ====>    6

  Option |I|  Exit / Include / eXclude / Compliment / Reset / View Events

=========================Event=Attributes=========================
   Event Name     Fail Mode  Component Type  System Type  Component ID
  *MOV*
                             1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16
 Location     Class Attributes

 Note:  Enter the attributes of the events to be included. Fields which are
        left blank are  not used.  The ? may  be used for positional "don't
        care"  characters.  The * may be  used for non-positional  variable
        length "don't care" strings.
        e.g.  AB?CD*  matches  ABXCD  through  ABXCDXXXXXXXXXX
```

Figure 12. Partition included events display.

Events can be qualified in groups by "Including" or "eXcluding" them or individually by selecting the "View Events" option. In this example we selected to include all events which contained the string "MOV" somewhere in the name. The Figure 12 also shows the result of this operation.

The analyst can also specify other attributes to further qualify an event or use "eXclude" to remove some of the qualified events from the list. The View Events" options allows us to display the basic events in this family. The screen in Figure 13 gives and example of this option.

```
                    V i e w   E v e n t s

          Option |I| Exit  / Include event  <= choose Event

 Event Name        Event Description
    C-CV-A         CCS Train A pump discharge check valve
    C-CV-B         CCS Train B pump discharge check valve
  + C-MOV-1        CCS suction isolation valve
  + C-MOV-A        CCS Train A pump discharge isolation valve
  + C-MOV-B        CCS Train B pump discharge isolation valve
    C-PUMP-A       CCS Train A motor-driven pump
    C-PUMP-B       CCS Train B motor-driven pump
    DG-A           Emergency diesel generator A
    DG-B           Emergency diesel generator B
    E-CV-A         ECS Train A pump discharge check valve
    E-CV-B         ECS Train B pump discharge check valve
  + E-MOV-1        ECS suction isolation valve
```

Note: Use <PgUp> or <PgDn> to display more Events. Included events are indicated by "+".

Figure 13. View Events display.

The currently qualified events are displayed with a "+" in front of the name. These events can be individually toggled from qualified to not qualified by highlighting the event and pressing the enter key. Once we exit the "Partition" menu and get back to the Cut Set display menu. The screen in Figure 14 is displayed.

F A U L T   T R E E   C U T   S E T S

Family Name DEMO FAMILY   Fault Tree Name CCS

Min Cut 2.120E-002 Num 15   Part. ==> 1.156E-003   5.45% Num 7

Option |P| Exit / Partition / Report / Basic Events   <= choose a cut set below

Num	%	Frequency	Event Names	
1	4.72	1.000E-003	C-MOV-1	
2	0.47	1.000E-004	DG-A	C-MOV-B
3	0.12	2.500E-005	C-MOV-A	C-MOV-B
4	0.07	1.500E-005	C-PUMP-A	C-MOV-B
5	0.07	1.500E-005	C-MOV-A	C-PUMP-B
6	0.00	5.000E-007	C-MOV-A	C-CV-B
7	0.00	5.000E-007	C-CV-A	C-MOV-B

Use <PgUp> or <PgDn> to display more Cut Sets

Figure 14. Partitioned Cut Set display.

Notice that only those cutsets including an event whose name contains the string "MOV" are displayed. Also note that the top of the display now contains different values for the current partition contributions.

To get a hard copy report of the partitioned cut sets, the analyst can select the "Report" option. This option generated the report shown in Figure 15.

Partition Cut Set Report

Family Name ->DEMO FAMILY   Fault Tree Name ->CCS
Mincut Upper Bound 2.120E-002   This Partition   1.156E-003

Cut No.	% Total	% Cut Set	Freq.	Cut Sets
1	4.7	4.7	1.0E-003	C-MOV-1
2	5.2	.5	1.0E-004	DG-A, C-MOV-B
3	5.3	.1	2.5E-005	C-MOV-A, C-MOV-B
4	5.4	.1	1.5E-005	C-PUMP-A, C-MOV-B
5	5.4	.1	1.5E-005	C-MOV-A, C-PUMP-B
6	5.4	.0	5.0E-007	C-MOV-A, C-CV-B
7	5.5	.0	5.0E-007	C-CV-A, C-MOV-B
8	5.5	.0	9.0E-006	C-PUMP-A, C-PUMP-B
9	5.5	.0	2.0E-006	DG-A, C-CV-B
10	5.5	.0	5.0E-007	C-CV-A, C-MOV-B
11	5.5	.0	5.0E-007	C-MOV-A, C-CV-B
12	5.5	.0	3.0E-007	C-CV-A, C-PUMP-B
13	5.5	.0	3.0E-007	C-PUMP-A, C-CV-B
14	5.5	.0	1.0E-007	TANK
15	5.5	.0	1.0E-008	C-CV-A, C-CV-B

Figure 15. Partitioned Cut Set report.

Thus, the analyst can perform some very powerful partitioning of the cut sets to determine the contribution of systems, component types or other categories needed to gain insights into the data generated by IRRAS. This feature goes far beyond most systems in aiding the analyst to understand the large amounts of information generated in a PRA.

CREATE Event Tree Option

The "CREATE Event Tree" option will be implemented in version 3.0 of IRRAS. This option will provide the user with a tool to graphically construct Event trees and link them similar to the way the Fault trees are currently constructed in IRRAS 2.0.

SEQUENCE Analysis Option

The "SEQUENCE Analysis" option provides the same functions for sequences that is provided for fault trees in the "FAULT Tree Analysis" option. IRRAS 2.0 provides a sequence analysis module which generates the cut sets for an accident sequence and quantifies the sequence. Cut set generation is accomplished by combining the cut sets for the fault trees that make up the sequence. Comparisons are made among the cut set lists for successful and failed systems to eliminate impossible failure combinations. Once the sequence cut sets are generated, the sequence may be quantified, importance measures may be calculated, and uncertainty analysis may be performed. The results can be displayed on the screen, sent to a printer, or written to a file for later use.

MODIFY Data Base Option

The "MODIFY Data Base" option provides access to the data base maintenance facilities in IRRAS. With this option, the user can make changes to the data stored in the relational data base. This feature allows access to all of the data generated by a PRA in an organized fashion. Simple state-of-the-art menus are utilized to make the process even easier.

REPORT on Data Base Option

The "REPORT on Data Base" option provides access to the report generation facility. This facility is a comprehensive report generation facility that allows the user to generate a report of any information contained in the data base. The user is allowed to specify the title for the report and the output device. The user may choose to output the report to the console, the line printer, or to a file for later modification or printing. The following list represents the reports available in IRRAS 2.0.

(1) Data Base Families report,

(2) Basic Event - Summary, Probabilities, and Uncertainty Values,

(3) Fault Tree - Logic, Expanded Logic, Modified Logic, Importance Measures, Cut Sets, Summary, and Uncertainty Values,

(4) Sequence - Logic, Importance Measures, Cut Sets, Summary, and Uncertainty Values reports.

UTILITY Options

The "UTILITY Options" function contains a number of utilities including data conversion, mainframe communications, and data recovery.

The conversion utilities have been added to IRRAS 2.0 to allow the analyst to convert between various formats. These formats include both IRRAS 1.0 and SETS formats. Both of these conversion utilities allow conversion to and from the specified formats and include fault tree logic, failure rate data, and both sequence and fault tree cutsets. These utilities make it easy for the analyst to interface with these programs and still be able to use the power of IRRAS 2.0.

Summary

The technological advances in the fields of probabilistic risk assessment and microcomputers have made possible the development of powerful tools for both PRA practioners and those involved in the risk management of nuclear power plants. Risk-based decision-making in the areas of plant design, operations, and regulation is possible now more than ever before.

The IRRAS 2.0 system is the result of significant modifications and improvements to the IRRAS 1.0 system. This new fault tree analysis system is more comprehensive and easier to use than IRRAS 1.0 and incorporates features which greatly improve the PRA analyst's ability to perform risk assessment.

Continued research in the area of fault tree analysis tools being conducted by the U.S. Nuclear Regulatory Commission and the great strides in microcomputer performance and useability provide a bright future for more and better tools for performing risk analysis.

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IMPACTS OF MULTIPLANT ACTIONS ON PLANT RISK

by
Harry J. Reilly, Richard D. Fowler, Henry J. Welland, Carol Mancuso
EG&G Idaho, Inc.

and
Bharat B. Agrawal
Nuclear Regulatory Commission

ABSTRACT

Idaho National Engineering Laboratories (INEL) has established a program that estimates the effects of Multiplant Actions (MPAs) on plant risk. In this program, NRC data bases are accessed to acquire status information on each MPA that has been imposed but not yet implemented at each plant, and details of the imposition for the plant. An available Probabilistic Risk Assessment (PRA) of the plant is then used to analyze the effect that implementation of the MPA is expected to have on the best-estimate core damage frequency (CDF) for the plant. The analyses are documented and loaded into the data base of a PC program called the Multiplant Action Evaluation Display System (MEDS). The user of the PC can view or print any of the files in this data base.

Experience with development of MPA Evaluations indicates that acquisition of files from the NRC data bases can be accomplished with reasonable ease using a PC with a modem. Some MPAs may be close enough to implementation that evaluations may be inappropriate. Some MPAs do not lend themselves to evaluations by PRA methods, because they are administrative issues or they involve possible occurrences or failure modes that were not within the scope of the available PRA. For these MPAs, only a text screen is prepared for MEDS.

Best-estimate values for the projected changes in CDF can be obtained for some of the MPAs that can be addressed by the available PRA. More often, the evaluation can develop only a maximum range of improvement that could possibly result from implementation of the MPA. For some MPAs, only a qualitative discussion is provided.

The results of the MPA Evaluation Program provide a perspective on the value of generic issue resolutions at operating nuclear power plants. They also provide a perspective on the usefulness of PRAs for such purposes.

INTRODUCTION

The Nuclear Regulatory Commission (NRC) has an active program in which generic issues are resolved within NRC and actions are taken to implement changes at affected facilities in accordance with the resolutions. The status of the actions are tracked by NRC management systems. Actions that affect more than one plant are called Multiplant Actions (MPAs). Idaho National Engineering Laboratory (INEL) is conducting a program that estimates the effects of these MPAs on plant risk. In this program, the NRC Safety Issues Management System (SIMS) is accessed to obtain status information on each MPA that has been imposed but not yet implemented. NRC's Nuclear Document System (NUDOCS) is also accessed to obtain details of the imposition for the plant. An available Probabilistic Risk Assessment (PRA) of the plant is used to analyze the effect that implementation of the MPA is expected to have on the best-estimate core damage frequency (CDF) for the plant. The analyses are documented and loaded into the data base of a PC program called Multiplant Action Evaluation Display System (MEDS). The user of the PC can view or print any of the files in this data base. MEDS also displays the CDF-effects of all the MPA evaluations on a common bar graph. The program helps NRC decision-makers by providing concise descriptions of pending actions, and by allowing them to determine which actions merit the most attention at a plant.

The results of the MPA Evaluation Program provide a perspective on the value of generic issue resolutions at operating nuclear power plants. They also provide a perspective on the usefulness of PRAs for such purposes. This paper describes the program and its experience through the end of FY-1988.

DESCRIPTION OF THE OVERALL PROGRAM

The workflow in the program is the same for all the plants that are evaluated. First, INEL and the NRC agree on which plant to evaluate. Up to this time, only plants for which PRAs are available have been selected. Then the Safety Issues Management System (SIMS) is accessed using a PC with a remote data link of some sort: at INEL the Local Terminal Network interfaces with a Renex modempool maintained by NRC. SIMS provides information on the status of all MPAs for all plants, and information on code-numbers under which documents are filed in the Nuclear Document System (NUDOCS). The MPAs of interest are those that have been imposed but not yet implemented at the plant. NUDOCS is accessed with the same PCs that were used to access SIMS. Several different searches must be conducted in NUDOCS to find all the relevant documents pertaining to each MPA identified by SIMS; no one search seems to ensure completeness of the acquisition. Having identified the relevant documents, the analyst then accesses microfiche files to obtain the documents. These documents give the analyst details about the MPA, insofar as it applies to a specific plant. From the documents, it is possible to determine the current status in detail.

INEL reviews its conclusions based on the data acquisitions with NRC personnel, who are responsible for the regulation of the plants, to check the accuracy and currentness of the information. The MPAs that appear to be still active are then evaluated for the effect they may be expected to have on the core damage frequency at the plant. The evaluation is performed using a PRA that is available for the plant. (Sometimes more than one PRA may be used, if the most recent PRA has a different scope than some older PRA that is applicable.) Each of the evaluations is written up using a standardized format, and the writeup is installed in a PC data base. This PC data base and its data-base manager program are provided to NRC, for its use in assessing the relative importance of MPAs that are still "open" at the plant. The NRC user can view or print any of the files in the data base. He can also view or print a bar graph that compares the relative values of all the MPAs for a plant.

KINDS OF MPA EVALUATIONS

Experience indicates that there are five different kinds of evaluations that can be done (see Table 1).

TABLE 1. KINDS OF EVALUATIONS

1. No evaluation, because MPA is complete or nearly complete.
2. Qualitative evaluation only, because effect is beyond the scope of the available PRA.
3. Only a range of possible effect can be calculated.
4. A best estimate can be calculated.
5. A combination of range and best estimate.

Some MPAs do not lend themselves to evaluations by PRA methods, because they are administrative issues or they involve possible occurrences or failure modes that were not within the scope of the available PRA. Also, there is insufficient information to provide a quantitative analysis for some MPAs that may affect the PRA. Usually, such MPAs could be analyzed if there were time to acquire more information by plant visits, searches of data bases, or additional supporting analysis, but such efforts are beyond the scope of the task. For these kinds of MPAs, only a qualitative discussion is provided.

For the MPAs that can be addressed by the available PRA, best-estimate values for the projected changes in CDF can be obtained for a few of the MPAs. More often, the evaluation can develop only a maximum range of improvement that could possibly result from implementation of the MPA. In such cases, basic events in the PRA fault trees that will be affected by the MPA can be identified, but it is conjectural how much the basic event unavailability values will be affected by the implementation of the MPA. For this kind of MPA, the unavailability values are usually adjusted to their lower bounds, or to zero, to represent the maximum effect the MPA could possibly have. If the analyst has an opinion regarding the real effect within the range, that opinion may be entered in the text file, but the opinion is not reflected in the bar graph unless there is a demonstrable basis for the opinion.

MULTIPLANT ACTION EVALUATION DISPLAY SYSTEM

The MPA evaluations that are performed for a given plant are added to a data base in a program called the Multiplant Action Evaluation Display System (MEDS). Basically four kinds of information are available in the data base: text descriptions of the evaluation of each MPA open at a plant (Figure 1), a text summary of the statuses and evaluations of all the MPAs still open for the plant (Figure 2), a summary screen for each MPA at each plant (Figure 3), and a bar graph that is generated by the computer on demand. The bar graph (Figure 4) displays the estimated effects for all the MPAs open for the plant. The bar graph also displays the best-estimate CDF before the MPA is implemented (baseline value), as well as the best-estimate CDF, if any, after it is implemented, or the range value if that is all that is available. Note that a logarithmic scale is used to compare the values.

MEDS uses a menu-driven selection process to allow the user to find the data he wants. The user can print as well as view the data. The program operates on an IBM PC/XT.

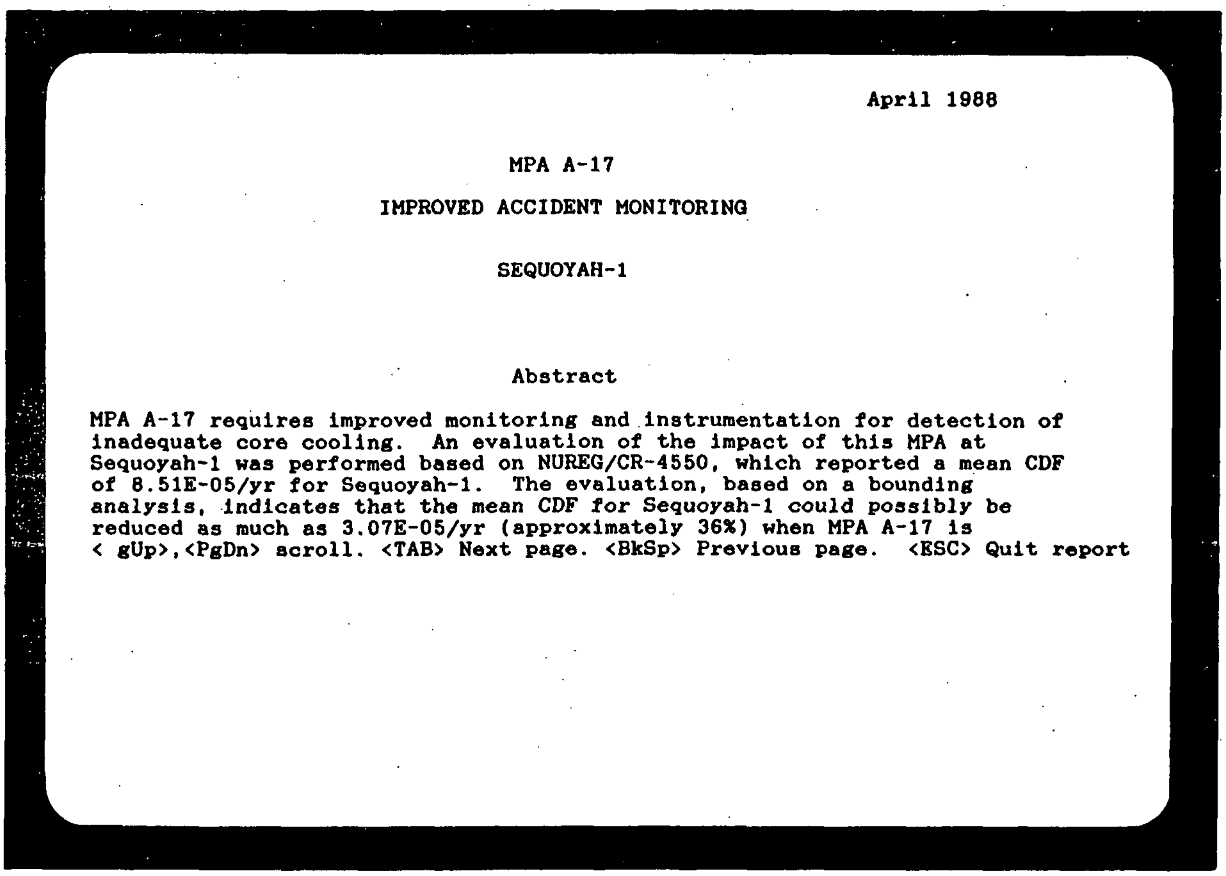
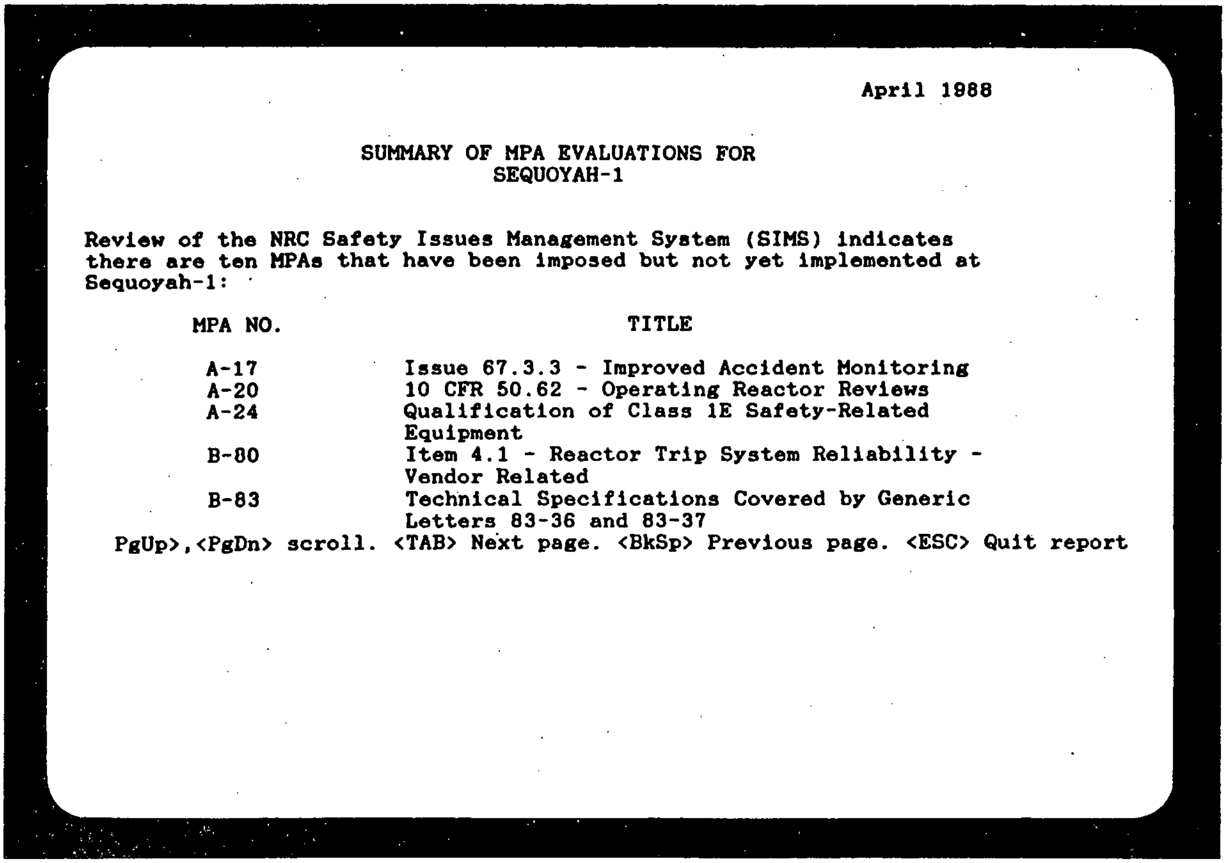


Figure 1. Screen from MPA text file.



April 1988

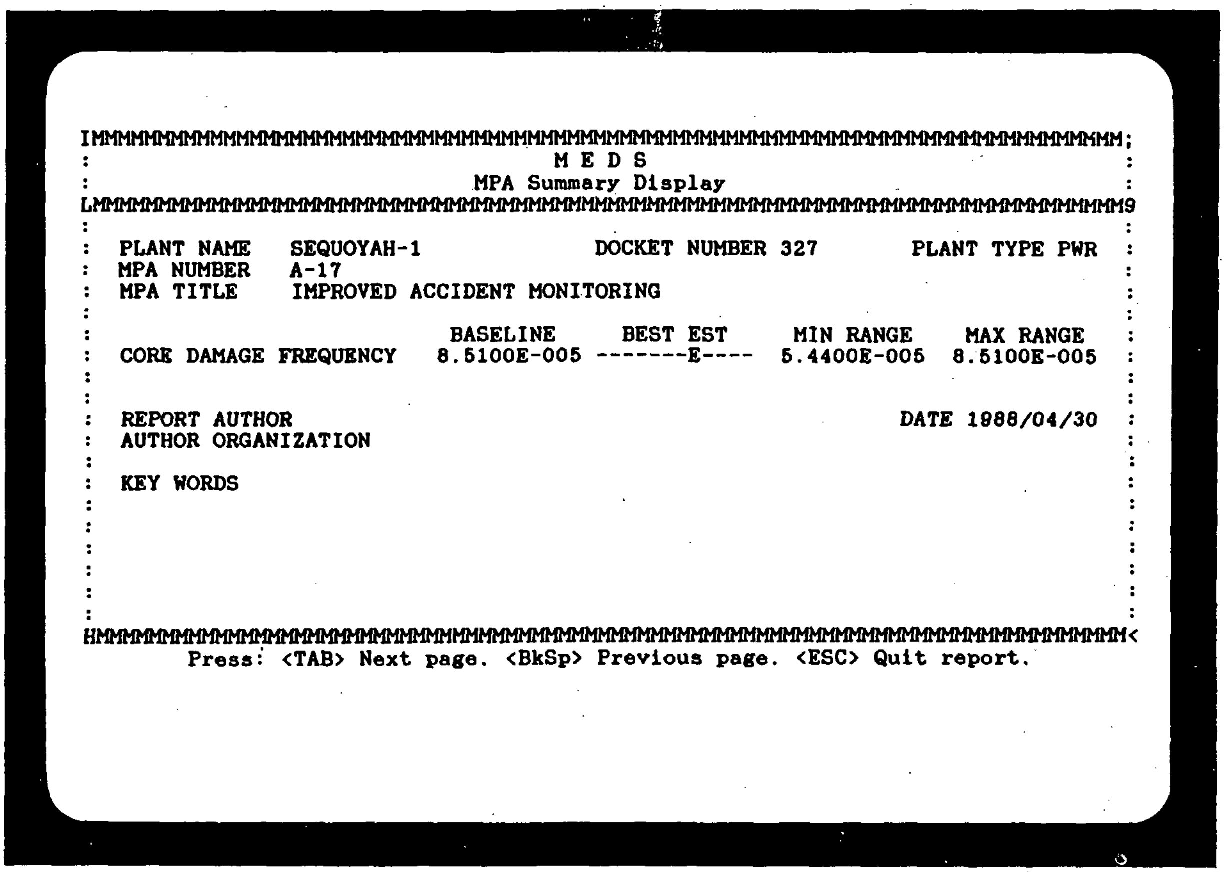
SUMMARY OF MPA EVALUATIONS FOR
SEQUOYAH-1

Review of the NRC Safety Issues Management System (SIMS) indicates there are ten MPAs that have been imposed but not yet implemented at Sequoyah-1:

MPA NO.	TITLE
A-17	Issue 67.3.3 - Improved Accident Monitoring
A-20	10 CFR 50.62 - Operating Reactor Reviews
A-24	Qualification of Class 1E Safety-Related Equipment
B-80	Item 4.1 - Reactor Trip System Reliability - Vendor Related
B-83	Technical Specifications Covered by Generic Letters 83-36 and 83-37

PgUp>,<PgDn> scroll. <TAB> Next page. <BkSp> Previous page. <ESC> Quit report

Figure 2. Screen from MPA summary file.



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IMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMM;
:                                  M E D S                                    :
:                            MPA Summary Display                              :
LMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMMM9
:                                                                             :
:  PLANT NAME    SEQUOYAH-1             DOCKET NUMBER 327       PLANT TYPE PWR :
:  MPA NUMBER    A-17                                                         :
:  MPA TITLE     IMPROVED ACCIDENT MONITORING                                 :
:                                                                             :
:                              BASELINE     BEST EST     MIN RANGE   MAX RANGE :
:  CORE DAMAGE FREQUENCY    8.5100E-005  -------E----  5.4400E-005 8.5100E-005 :
:                                                                             :
:                                                                             :
:  REPORT AUTHOR                                            DATE 1988/04/30   :
:  AUTHOR ORGANIZATION                                                        :
:                                                                             :
:  KEY WORDS                                                                  :
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        Press: <TAB> Next page. <BkSp> Previous page. <ESC> Quit report.
```

Figure 3. MPA summary display.

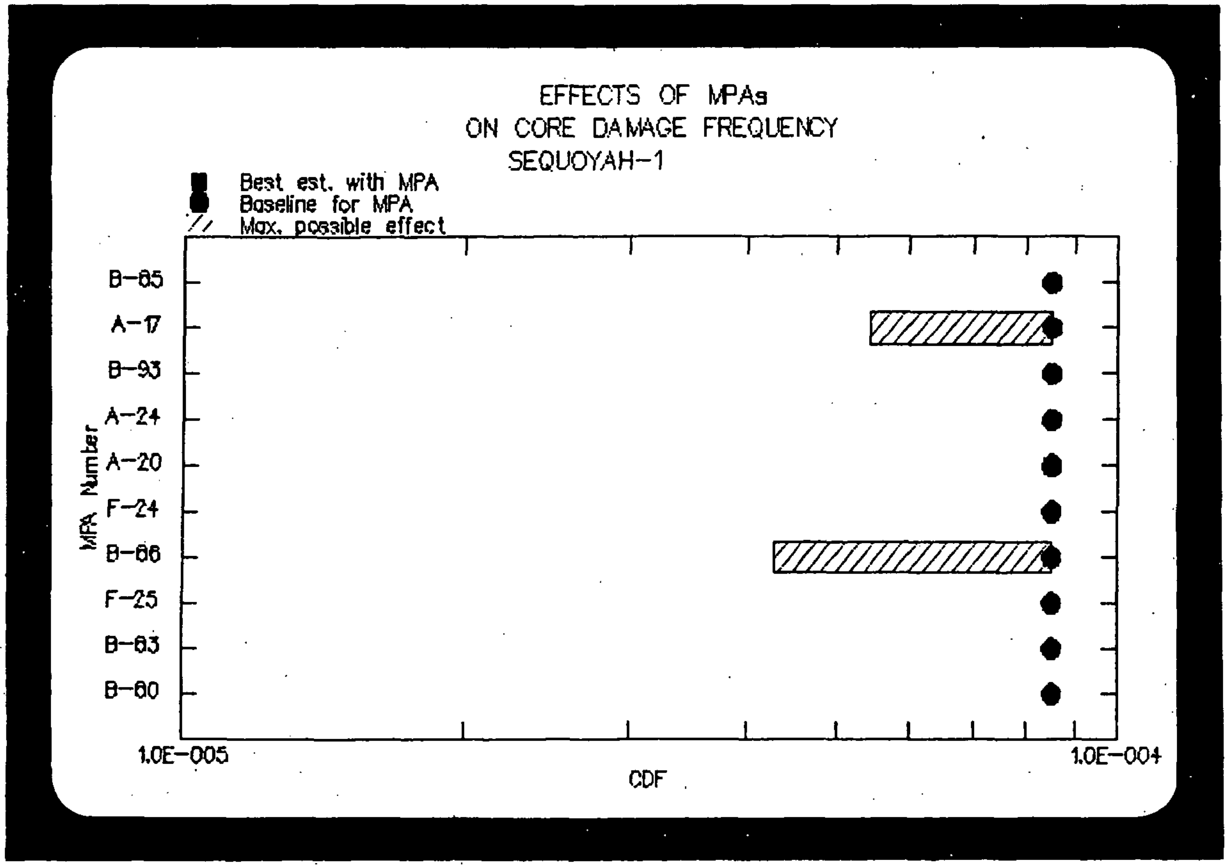

Figure 4. Bar graph for Sequoyah-1.

DISCUSSION OF FIRST YEAR'S EXPERIENCE

The results of the MPA Evaluation Program provide a perspective on the value of generic issue resolutions at operating nuclear power plants. They also provide a perspective on the usefulness of PRAs for such purposes. Experience has been acquired through the performance of MPA evaluations for the plants listed in Table 2. The MEDS contains the evaluations and is used by NRC.

TABLE 2. PLANTS FOR WHICH MPAs WERE EVALUATED IN FY-1988

Calvert Cliffs-1	Grand Gulf-1
Indian Point-2 and 3	Oconee-3
Peach Bottom-2	Sequoyah-1
Turkey Point-3 and 4	Zion-1

Table 3 summarizes the evaluation results for a typical plant. Some of the MPAs are characterized as closed, or nearly closed, according to documents found in the NUDOCS data base. Most of the other MPAs were estimated to have minor effects on the core damage frequency, or could be evaluated only qualitatively. MPA F-71 was estimated to have the most significant effect, but the estimate was a range rather than a best estimate.

Table 4 is a summary of the results. It shows how many MPAs have been looked at and how many have been evaluated by each kind of evaluation. Out of 119 outstanding MPAs, 52 appear to be closed or nearly closed based on documents found in NUDOCS or on consultation with the NRC project manager for the plant. Of the remaining MPAs, six could be evaluated only qualitatively, because they were administrative issues and/or involved accidents that were not within the scope of the available PRA. Thirty-three were evaluated quantitatively, but 26 could be evaluated only by a range calculation, because there was insufficient information available to develop a best estimate with the limited effort that was allotted to the evaluation of each MPA. A best estimate of the effect was provided for only seven of the MPAs.

Very few of those MPAs that were evaluated quantitatively appeared to have much of an effect on the CDF. It is not clear whether most MPAs indeed have little importance in terms of CDF, or whether the more important MPAs cannot readily be estimated by PRAs. The fact that only a range was obtained for most of the 33 MPAs that were estimated, does not facilitate comparison of the relative values of the MPAs. Comparison of two ranges is ambiguous; careful reading of the detailed evaluation reports for each MPA is required to obtain a feeling for what the range estimate means in each case.

It appears that few MPAs have much of an effect on the CDF.

TABLE 3. TABULATION OF EVALUATION RESULTS - ONE PLANT[a]

Plant Name: Example Plant		PRA: NUREG/CR-4550		
MPA Number	Title	CDF Before MPA	CDF After MPA	Delta CDF
A-04	Appendix J Containment Leak Testing	MPA completed per NUDOCS		
A-15	Quality Assurance Request Regarding Diesel Generator Fuel Oil	8.4E-06/yr	≥8.1E-06/yr	≤2.65E-07/yr
A-17	Improved Accident Monitoring	Combined with F-26		
A-20	Operating Reactor Reviews	1.0E-05/yr	8.2E-06/yr	1.7E-06/yr
B-22	Tech Spec Requirements for Mechanical Snubbers	MPA completed per NUDOCS; could not do anyway		
B-63	Emergency Training and Procedures for Station Blackout Events	MPA completed per NUDOCS		
B-77	Item 2.1 - Equipment Classification and Vendor Interface - RTS Components	8.2E-06/yr	≥7.2E-06/yr	≤1E-06/yr
B-85	Salem ATWS - 1.2 Data Capability	Qualitative - small effect		
B-87	Salem ATWS - 3.2.1 and 3.2.2 Safety-Related Components	8.2E-06/yr	≥7.3E-06/yr	≤9.2E-07/yr
D-01	Mark I Long Term Program	MPA completed per NUDOCS		
F-08	I.D.1.1 Detailed Control Room Design Review Program Plan	Combined with F-71		

TABLE 3. (continued)

Plant Name: Example Plant		PRA: NUREG/CR-4550		
MPA Number	Title	CDF Before MPA	CDF After MPA	Delta CDF
F-09	Plant Safety Parameter Display Console	3.7E-05/yr	≥2.7E-05/yr	≤9.7E-06/yr
F-26	Instrumentation for the Detection of Inadequate Core Cooling	8.4E-06/yr	≥7.9E-06/yr	≤5E-07/yr
F-71	Detailed Control Room Review	3.7E-05/yr	3.3E-05/yr	3.7E-06/yr

a. F-09 and F-71 were evaluated by NRC. A-20 is part of the base line. The ranges projected for B-77, B-87 and F-26 are not likely to be realized. A-15 is a reasonable expectation if in fact the DGs are not already better than assumed in the PRA.

TABLE 4. SUMMARY OF MPAs, FIRST TEN PLANTS[a]

	Plant-Name									
	CC-1	IP-2	IP-3	PB-2	SQ-1	TP-3&4	GG-1	Oco-3	Zion-1	Totals
Number of open MPAs (SIMS)	8	17	11	14	10	26	3	15	15	119
Number of closed NUDOCS	6	8	3	4	7	12	2	8	2	52
Could not evaluate	1	3	1	2	0	3	0	3	1	14
Combined with others	0	3	3	2	0	6	0	1	3	18
Evaluated qualitatively	0	0	0	2	0	0	0	2	2	6
Evaluated range	1	3	4	5	2	4	1	2	4	26
Evaluated best estimate	0	0	0	2	1	1	0	0	3	7

a. The columns may not add up, because some MPAs were counted in more than one category.

CONCLUSIONS

In conclusion, it is practical to obtain data on the detailed status of each Multiplant Action at each plant using a PC that accesses NRC data bases with a modem. The data thus obtained can be used with an available PRA to estimate the effect that completion of each open MPA will have on the core damage frequency of a plant. However, some MPAs do not lend themselves to evaluations by PRA methods, because they are administrative issues or they involve possible occurrences or failure modes that were not within the scope of the available PRA. For some MPAs, only a qualitative discussion can be provided. Best-estimate values for the projected changes in CDF can be obtained for some of the MPAs that can be addressed by the available PRA. But more often, the evaluation can develop only a maximum range of improvement that could possibly result from implementation of the MPA. Finally, it appears that very few MPAs have much of an effect on the CDF. This is due either to the small effect of the MPAs or inherent limitations of the PRA models to estimate the effects with only a limited effort.

NOTICE

SAND88-2942C

NRC RESEARCH IN COMMON-CAUSE FAILURES*

by

D. B. MITCHELL (SNL)
G. W. PARRY (NUS CORPORATION)
H. M. PAULA (JBF ASSOCIATES, INC)
D. W. WHITEHEAD (SNL)
D. M. RASMUSON (USNRC)

Abstract

The status and recent history of common-cause failure (CCF) research are briefly reviewed. Current NRC research in the area of CCFs is described with emphasis on the remaining problem areas. These include deficiencies in data and the need to more completely understand the characteristics of CCFs. The concepts and relationships of root cause, coupling mechanisms, and defensive mechanisms are discussed. Key definitions and some in-depth analysis of these concepts are included. An overview of the recent research to be published is presented. This research includes (1) the cause-defense methodology for analyzing CCFs, (2) guidelines for identifying potential CCFs as part of a nuclear power plant walkdown and procedures review, and (3) requirements for an industry-wide data base, including documentation of failure events and additional component failure and failure mode data requirements to support future PRAs.

1.0 INTRODUCTION

1.1 Background

The importance of common-cause failures as contributors to system unavailability in systems with redundant trains and as contributors to risk in probabilistic risk assessments (PRAs) has been demonstrated again and again. A recent report sponsored jointly by the Electric Power Research Institute (EPRI) and the United States Nuclear Regulatory Commission (USNRC),

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NUREG/CR-4780 [1] describes a procedural framework for common-cause failure analysis and gives specific examples of methods that can be used to perform a plant-specific analysis of common-cause failures. An important point emphasized in the EPRI/USNRC document is that there are not enough plant-specific CCF data to estimate CCF event probabilities; generic, industry-wide data must be used for that purpose. However, using information from a variety of plants requires (1) interpreting previous failure occurrences to identify the mechanisms involved in these events and (2) reinterpreting these occurrences in light of the design and operational features of a specific plant.

To date, this two-step procedure has been performed by examining individual failure events that have occurred in the industry and deciding whether those events apply or do not apply to a particular plant. The analyst may also postulate several different hypotheses regarding how the event would have occurred at a specific plant. For example, there could be three hypotheses regarding the applicability of an event involving the failure of two pumps: (1) the event does not apply to the plant in question, (2) the event applies and would have resulted in the failure of one pump at the plant in question, and (3) the event applies and would have resulted in the failure of both pumps at the plant in question. Based on assumptions regarding defenses that may or may not have been in place at the plant where the event actually occurred, and based on the analyst's understanding of the plant being analyzed, the analyst can assign probability values to each of the three hypotheses and evaluate the number of component failures that would be expected should the event occur at the subject plant. By performing this type of evaluation for a number of events, an analyst develops a specialized, or pseudo, data base for a particular plant.

A theme that occurs often in NUREG/CR-4780 is the interpretation that common-cause failure events result from some root cause of failure, a coupling mechanism or agent which results in more than one component failing, and inadequacies in defenses against the root cause and the coupling. These ideas are implicitly applied in the screening and reinterpretation of industry-wide event data that is the essence of the formation of a plant-specific "pseudo-data base."

It is apparent that there are problems resulting from the lack of good-quality, comprehensive data and the lack of methods for performing an analysis of that data that enables an analyst to apply his subjective assessments in a consistent and scrutable manner. While there are industry-wide data collections which can be referred to, such as the Licensee Event Reports (LERs) and Nuclear Plant Reliability Data System (NPRDS) [2], there is no consistent source that encompasses all the data needed to perform a comprehensive and consistent analysis of single and multiple failures. It is necessary for the further understanding of common-cause failures, and, in particular for future planning to mitigate against them, that a single data base be constructed which captures the necessary information to analyze single and multiple failure events consistently.

1.2 Summary of Current Efforts

Based on the concerns identified previously, the NRC has instituted a research program which contains three interrelated elements. The first element is the further evaluation of the conceptual framework introduced in [1] as the Cause- Defense Beta factor model. The original beta factor model was based on the premise that a fraction (β) of the failure rate of a component could be attributed to the common-cause events shared by one or more components in a group performing related functions. According to this model, whenever a common-cause event occurs, all components within the common-cause component group are assumed to fail. As with all other CCF techniques, the effect of methods to prevent failures, i.e., defenses, was not considered in the beta factor model. The cause-defense methodology, on the other hand, requires that plant-specific defenses against CCFs be identified and assessed as to their effect on specific causes and coupling mechanisms. Examples of defenses include such things as barriers, component diversity and training.

The quantitative CCF models were originally developed so that systems could be modeled with a reasonable number of basic events. One extreme in modeling would be to model explicitly all root causes and coupling mechanisms in the fault tree models, which obviates the need for common-cause failure basic events. Existing quantitative CCF models instead incorporate these different causes and couplings into one common-cause basic event. The Cause-Defense Beta Factor model evaluates the probability of the common-cause basic event using explicitly the concepts of root causes, coupling, and defensive mechanisms. The model was presented in [1] as a framework for the application of subjective judgment for the estimation of common-cause failure parameters in the absence of sufficient data to construct a plant-specific pseudo-data base. In another sense, it represents a framework for a more explicit representation of the engineering considerations which are part of the process of estimating the failure parameters. In the current phase of the work, the research that has been performed by EPRI, NRC, and others on cause classification schemes [1] and defensive strategies [3] is being reviewed to identify more clearly how to apply the model and to identify needs for specific information or improved methods. Clearly one of the major challenges will be to determine how to assess, in a consistent way, the strength of coupling mechanisms or the effectiveness of specific defenses.

In parallel with this activity, the NRC is pursuing research to construct a set of guidelines that can be used to perform a plant review to identify potential common-cause failures. In addition to physical walkdowns, the guidelines will cover the review of the normal and emergency operating procedures, maintenance and test procedures, and system descriptions. It is clear that in the context of the guidelines, the concept of a common-cause failure should be extended from the concept used in PRAs to include all non-hardwired dependencies. Consequently, there may be overlap with other NRC programs, and this will be discussed in the final report on the guidelines. The guidelines are intended to distill the knowledge of what has been learned concerning dependent failures into a form that will help a user phrase the right questions to perform a complete examination for common-cause vulnerability. The guidelines will not be a checklist based on specific historical events, but will extract from such examples general principles that define a philosophy or approach to a systematic examination that will maximize the chances of finding potential multiple failures before they occur. The guidelines are intended for use by a variety of people including PRA analysts, NRC inspectors, but most of all, utility staff.

The third element of the program is the writing of a document which motivates and defines the need for an improved industry-wide data base. The document will address the reporting and data requirements that would enable an analyst to perform a more reliable analysis of single and multiple failure events, and will demonstrate the importance of obtaining such data. Such a data base would not be used just for PRAs, but would also be directed at improving the state of knowledge of failure mechanisms, through more complete and comprehensive data reporting, so that multiple failures could be avoided.

The detailed research necessary to complete this program has only been partially completed. As a result, this paper primarily describes the results of the planning to complete the research in each of the three areas just discussed.

1.3 The Cause-Coupling-Defense View of Common-Cause Failures

An important observation made early in this effort was that some of the definitions and concepts used in describing CCF analysis and prevention techniques may not be sufficiently detailed and complete for the purposes of this program. Therefore, part of the current effort has been devoted to the development of appropriate definitions and concepts. This section describes and illustrates some of the definitions and concepts under consideration. Once they are firmly established, these definitions and concepts will be used in all three parts of this program.

In order to defend against common-cause failures it is essential to understand how components can fail and why more than one component can be susceptible to the same failure cause.

In NUREG/CR-4780 [1] the concept of looking at common-cause failures using the ideas of root cause, coupling mechanisms, and defensive mechanisms was highlighted. This results in a causal picture of failure with an identifiable root cause, a means by which the root cause is more likely to impact a number of components simultaneously (the coupling), and the failure of the defenses against such multiple failures.

The description of a failure in terms of a single root cause is, in reality, too simplistic. For some purposes, it may be quite adequate to identify that a pump failed because of high humidity. However, since we are interested in a detailed understanding of the potential for multiple failures, we need to identify further why the humidity was high and how it affected the pump. To discuss further the types of common-cause failure events, the following definitions are useful.

The cause of failure is a characterization of the condition which led to the failure, but it does not in itself necessarily provide a full understanding of why the failure occurred. To understand the failure it is useful to think in terms of conditioning events and trigger events. A conditioning event predisposes a component to failure but does not itself cause failure. For example, failing to provide adequate protection against high humidity. A trigger event activates a failure, such as an event which leads to high humidity. A trigger event may be an event internal or external to the component it affects. An event which led to high humidity in a room and a subsequent failure would be an external trigger event. An internal event which caused a short circuit in a component would be an example of an internal trigger event. It is not always necessary to identify separately conditioning and trigger events; the latter may be a sufficient description of the failure. An example is a design error that leads to a failure during a real as opposed to a test demand. The error could be regarded as a conditioning event with the trigger being the demand, or a trigger event since it puts the component in a failed state for the real demand.

A useful concept in discussing failures, and particularly defenses against them is the speed of the failure mechanism. This is a measure of the time between the trigger event and the actual failure for the case of impulsive (i.e., fast acting) triggers such as missiles, and the time scale for the development of degradation to the point of failure for the persistent (i.e. slow acting) trigger such as aging [5]. This is important from the point of view of detection. Defects which develop slowly with evidence of degradation have a greater chance of being discovered before they result in failure.

For failures to become multiple failures, the conditions have to be conducive to the trigger event and the conditioning events affecting all the components simultaneously. This leads to the concept of coupling factors. A coupling factor is a property of a group of components or piece parts that makes them susceptible to the same cause of failure. Such factors include similarities in design, environment, maintenance and test procedures. To understand how common-cause failures can arise it is important to understand how this common susceptibility is enhanced or activated to result in multiple failures. It must also be kept in mind that the coupling factors are a function of the causes, conditioning events, and trigger events.

The strength of the coupling is an important factor. Empirically it can be measured by the time between successive failures of the redundant components relative to their individual mean-time between failures (MTBF). A strong coupling means that the redundant component failures are likely to occur almost simultaneously. A weak coupling means only that there is an increased chance of simultaneous failure over the chance of purely independent failures.

EPRI NP-5777 [3] proposes a classification for defensive tactics that can be used to minimize the effects of CCFs. Defenses against CCF can be effective in many different ways. First, they can prevent the cause. An example would be to protect motor control centers (MCCs) against humidity by sealing them (not sealing the MCCs results in the existence of the conditioning event). This is equivalent to hardening the component, and the defense acts against the conditioning event. Another example is the training of maintenance staff to assure correct interpretation of procedures. This prevents potential trigger events due to misapplication of procedures.

Another defense is to decouple failures by effectively decreasing the similarity of components or the environments to which they are exposed to prevent a particular type of root cause from affecting all components simultaneously. This allows more opportunity for detecting failures before they appear in all components of the group. However, using dissimilar components must be approached with caution, since additional failure modes could be introduced.

The key to successful mitigation of potential common-cause failures is to understand how the defenses can fail. With this information, defenses which will prevent failures can usually be developed.

2.0 A CAUSE-DEFENSE METHODOLOGY FOR CCF ANALYSIS AND PREVENTION

Based on the discussion in Section 1.3, it is clear that CCFs can be prevented or mitigated by design and procedural defenses. From this it follows that reliability and safety analyses that properly address CCF events can aid in designing defenses to prevent or mitigate the occurrence of these events. The cause-defense methodology is a new CCF analysis technique that explicitly accounts for plant-specific defenses to reduce the likelihood of CCF events at nuclear power plants (NPP) [4].

The objective of the cause-defense methodology is to extend the state of the art of plant-specific CCF analysis by providing enough detail in reliability and safety analysis to (1) establish and assess plant-specific defense alternatives against CCFs and (2) improve the accuracy of plant-specific CCF analyses. This is accomplished by developing matrices that show the qualitative and quantitative impact of different plant-specific defenses on different categories of root causes and coupling factors associated with CCF events. The work to date has identified categories of potential root causes, characteristics of the associated coupling mechanisms, possible defenses, and characteristics of the matrices [4]. Some example matrices have also been developed, including one which shows the impact of defenses on selected causes of battery failure. The causes of failure, such as internal faults, are listed down the left side of the matrix. Across the top, selected defenses, such as inspection and testing, are listed. At the intersection of each cause and defense, the impact of the defense on the cause is specified. Depending on the type of matrix, the impact can be qualitative (e.g. weak defense or strong defense) or quantitative (a numerical value which indicates the strength of the defense). The items in the matrix can be broken down so the analysis can be done in as much detail as necessary. Once developed and reviewed, the matrices may be used by analysts to perform comprehensive analyses of CCFs for any NPP. The matrices will also be useful to power plant designers, inspectors, and operators.

The initial results from the development of the cause defense methodology have shown that the cause-defense matrices must be fairly detailed to allow for CCF analyses that truly account for design features, and operational and maintenance policies. Therefore, the matrices are being developed for each component type and will account for design variations as well as the way the equipment is tested, maintained, and operated. The cause-defense matrices will be developed and reviewed by experienced CCF analysts and by experts in design, operations, and maintenance of NPP equipment. The matrices will therefore reflect the consensus of the people involved in developing them, which is particularly desirable in areas where data are sparse. It should be stressed that in this context, the concept of cause is enlarged to include all elements of the causal chain of events that lead to multiple failures.

3.0 WALKDOWN AND PROCEDURES REVIEW GUIDELINES

Common-cause failures can be categorized from the point of view of a generic class of physical and administrative defenses. This provides a means of listing the types of common-cause failure mechanisms that could be identified by a physical walkdown of the plant. Primarily a walkdown can address mechanisms that are manifested by a detectable physical condition. Therefore, causes related to inadequate performance, procedures, and internal environmental conditions will largely be unaddressed by a walkdown.

Causes that are associated with external environmental effects on equipment and physical impulsive forces are addressed and will be discussed in some detail. Also, a walkdown is capable of identifying potential problem areas associated with physical limitations on the performance of tests, maintenance, or calibration acts. These causes and areas can be addressed in a walkdown at any stage of plant life.

In addition to the above, walkdowns are a normal part of plant operation. For instance, it may be necessary to check the positions of certain valves once per shift. This type of periodic walkdown is really part of the surveillance testing strategy. The key here is the search for changes in the physical condition of the plant, for example, the temperature in a room or of a piece of equipment or piping, or the position of a valve stem. This type of walkdown as a defense is effective against the effects of spurious failure of auxiliary equipment, and equipment failures which can only be indirectly detected by a physical effect on the system (e.g., failure of a check valve allows steam to enter a pipe and heats it up). In addition, it is effective as a defense against errors by plant personnel, for example, opening the wrong valve or propping open fire doors. The issues that should be addressed in such a walkdown are partly identifiable from a study of plant procedures. The completeness of the list is an issue that should be addressed by a review of those procedures. In addition, there are issues that come under the heading of general awareness of potential failure mechanisms, for example, when is the vibration of a pump becoming unacceptable, or when is the temperature in a room too high?

For either of the walkdowns discussed above to be successful and meaningful, the makeup of the team has to be carefully considered. For example, for the dynamic, periodic walkdown the main purpose is to identify changes, sometimes subtle ones, in the physical condition of the plant. Therefore, it is essential to conduct the walkdown with a person who has an intimate knowledge of the plant. For the special purpose walkdown, this is also highly desirable to increase the efficiency of the walkdown. However, it is possible to conduct such a walkdown if the team leader has access to someone with a knowledge of the layout of the plant. On the other hand, a human factors expert would be of value in reviewing those aspects of a walkdown related to physical limitations on maintenance and test personnel. For any of the walkdowns, one or more CCF experts would, of course, need to be on the team.

The walkdown guidelines will include procedures for identifying potential impulsive external mechanisms of common-cause failures. A defense against these events is the installation of barriers, and an assessment of the quality of the barriers is an integral part of the walkdown. This will include identification of type and location of the barriers relative to the vulnerable components, the type of agent to which the barrier is impermeable, the quality of its installation, and the quality of the administrative controls that maintain its integrity. This information must be coupled with an identification of potential trigger events for the various environmental agents in terms of their location and severity.

Some elements of a human factors-oriented walkdown will be discussed to address those issues related to the quality of the environment (in a general sense) for the plant operating staff to correctly perform what is required of them. As mentioned above, the document will address the issues associated with a periodic plant walkdown.

4.0 HISTORICAL DATA AND ITS ROLE

During the entire history of PRA the importance of adequate, reliable and well-interpreted data has been recognized. Without such data the quantification of accident risk is less accurate. The total amount of failure data is limited due to incomplete reporting and the high reliability of most plant safety equipment. This results in the use of failure data from several plants when performing a PRA, particularly in the realm of dependent failures. In turn, this increases the need for properly interpreting the data so that they can be applied to the plant being analyzed. One of the major problems in risk and reliability assessments has been the necessity of using data from different sources for independent and dependent failures. It is crucial that both types of failures are treated in a consistent manner.

In addition to PRA, failure data have many other uses. For example, knowledge about failure trends at a plant can provide insight for the operators on ways to improve the operation of a given system and thus reduce plant downtime. Failures at another plant can also alert the operators to potential problems at their plant. Plant inspections can be improved when the proper data are available, enabling the inspectors to concentrate on key areas. Because of the above factors, as well as the continuing need for better component failure data, a document which contains the recommendations for an industry-wide data base is being prepared. This document will tie the work described in sections 2.0 and 3.0 together to enhance the results of applying the guidelines and cause-defense methodology. The following are the objectives that led to this data requirements document:

1. Improve systems analysis by developing requirements for documentation of failure events so that an improved data base can be constructed.

2. Develop a list of components and component failure modes for which additional data are needed, using current PRAs as the primary source of information.

The document starts with a review of current failure reporting requirements and their limitations. A sample set of the existing data bases are assessed. Their limitations are described and recommendations for improvement are made. Finally, data needs identified from recent PRAs are discussed. This entire process has the objective of improving the quality of the results of the data classification and screening step in the common-cause analysis procedure [1]. The end result will be improved estimates of the parameters used in the CCF analysis model (e.g. the cause-defense methodology) and better predictions of system unavailability and event frequencies. Data bases that incorporate the recommendations proposed in this document will permit establishing defensive alternatives against CCFs and thus will be useful in improving and/or maintaining NPP availability and safety. *It should be noted that the scope of the document includes not only data that apply to CCFs but independent failures as well.*

5.0 SUMMARY AND CONCLUSIONS

The importance of CCFs as contributors to system unavailability and failure risk is well known. As a result, these events have been the subject of continuing research to develop better techniques for treating and analyzing these events. Current research in the treatment of CCF events is concentrating on two main problems: deficiencies in data and the need to more completely understand the characteristics and effects of CCFs. The first area makes quantification of risk in a PRA less certain and the second limits the quality of the analysis using the data which are available. To resolve these issues, the concepts of root cause, coupling mechanisms, and defensive mechanisms are currently being explored in three different areas.

The first area involves the further evaluation of the model for treating CCFs known as the Cause-Defense Methodology. As the name implies, this model considers the effects of defenses on CCFs as well as the root cause. This is done through the development of matrices which include these effects for a specific plant as a function of different categories of root causes, including the associated coupling factors.

The second area of research involves the development of guidelines for identifying potential CCFs as part of a physical walkdown and a procedures review of a nuclear power plant. These guidelines will focus on mechanisms that can be detected by observing an abnormal physical condition. Causes that are associated with external environments, physical limitations imposed on maintenance and test activities, and deviations from normal plant operation will be addressed by the guidelines. The assessment of defenses against these causes, such as barriers, will be included. Also, the composition of the walkdown team, as a function of the type of walkdown, will be documented.

In the last area, the problem of deficient and insufficient failure data is being addressed by an effort which has two parts. The first part will improve system analysis by developing failure reporting requirements which will provide an improved data base for the consistent estimation of the probabilities of independent and dependent failure events. The second part will develop a list of components for which additional data are needed for future PRAs.

Based on the results to date, the current NRC research in CCF is developing the techniques and procedures needed to treat these failures in a consistent, comprehensive and traceable manner. It is anticipated that the walkdown guidelines and data requirements specification will be completed in FY89. The Cause-Defense Methodology development will probably continue into FY90 in order to develop some of the cause-defense matrices.

6.0 REFERENCES

1. Mosleh, A., et al., Procedures for Treating Common-cause Failures in Safety and Reliability Studies, NUREG/CR-4780, EPRI NP-5613, Pickard, Lowe, and Garrick, Inc., January 1988.

2. Sinard, R. L., "Applications of NPRDS Data to Enhance Nuclear Plant Design and Operations", Proceedings: International Topical Meeting on Probabilistic Safety Methods and Applications Volume 1: Sessions 1-8, EPRI NP-3912-SR Volume 1, February 1985.

3. Crelin, G. L., et al., Defensive Strategies for Reducing Susceptibility to Common-Cause Failures", Volume 1: Defensive Strategies, EPRI NP-5777, Saratoga Engineering Consultants, June 1988.

4 Paula, H. M. and D. J. Campbell, A Cause-Defense Methodology for Common-cause Failure Analysis, JBF Associates, Inc., to be published as a NUREG/CR.

5. Lofgren, E. V., et al., Guidance for Using Reliability Programs to Defend Against Common-Cause Failures, Science Applications International Corporation, (Draft for Comment), June 1988.

U.S. DEPARTMENT OF ENERGY
NUCLEAR RESEARCH AND DEVELOPMENT PROGRAM

Advanced Reactor Program
Presentation to
Water Reactor Information Meeting

Jerry D. Griffith
Associate Deputy Assistant Secretary
for
Reactor Systems Development and Technology
Office of Nuclear Energy

October 1988

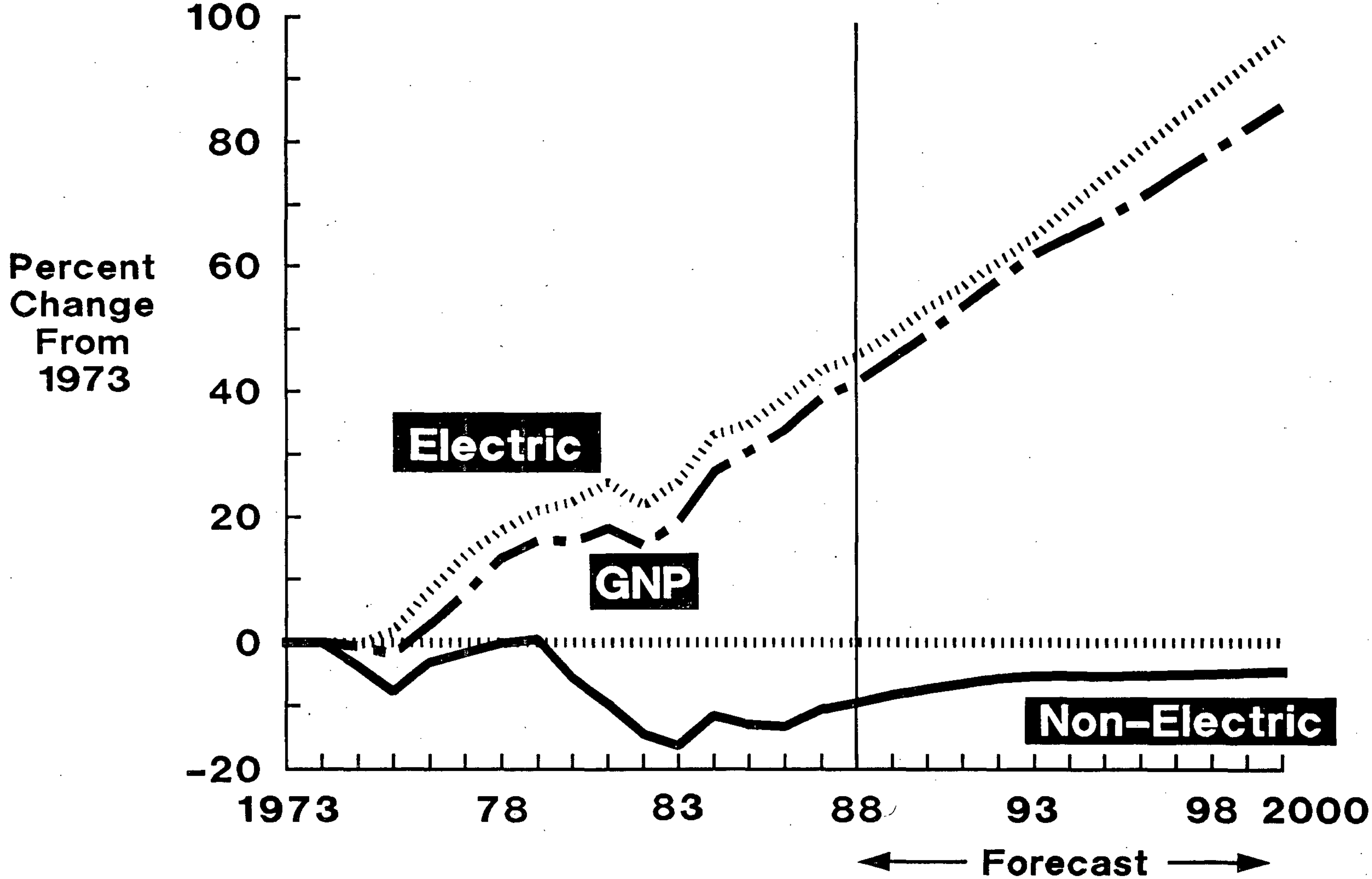
OUTLINE

- BACKGROUND
- ADVANCED REACTOR PROGRAM STRATEGY
- LMR PROGRAM
- MHTGR PROGRAM
- FACILITIES PROGRAM
- SUMMARY

9/22/88:ZM:OUTLIN

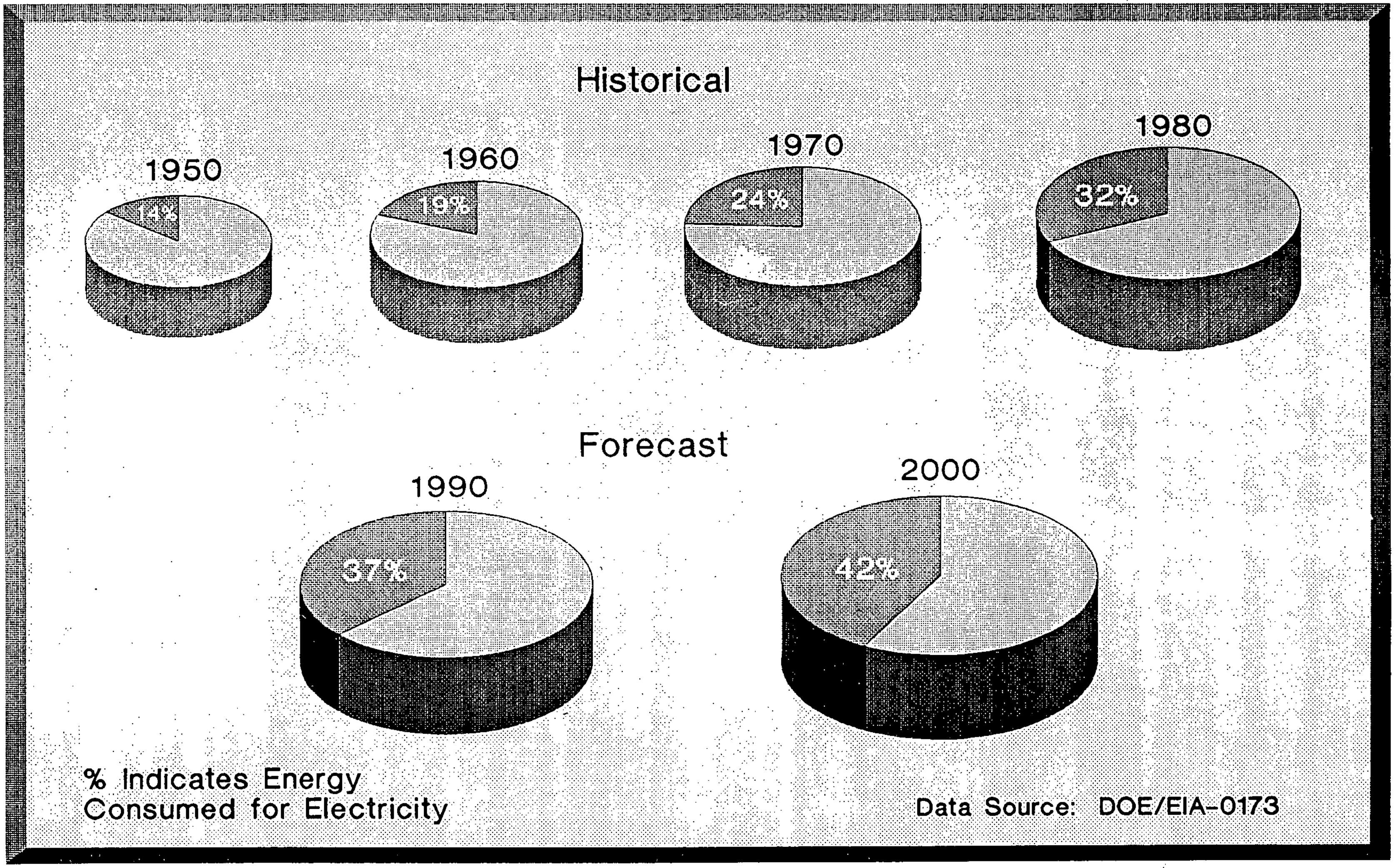
BACKGROUND

CHANGE IN ENERGY USE SINCE 1973
ELECTRIC VS. NON-ELECTRIC



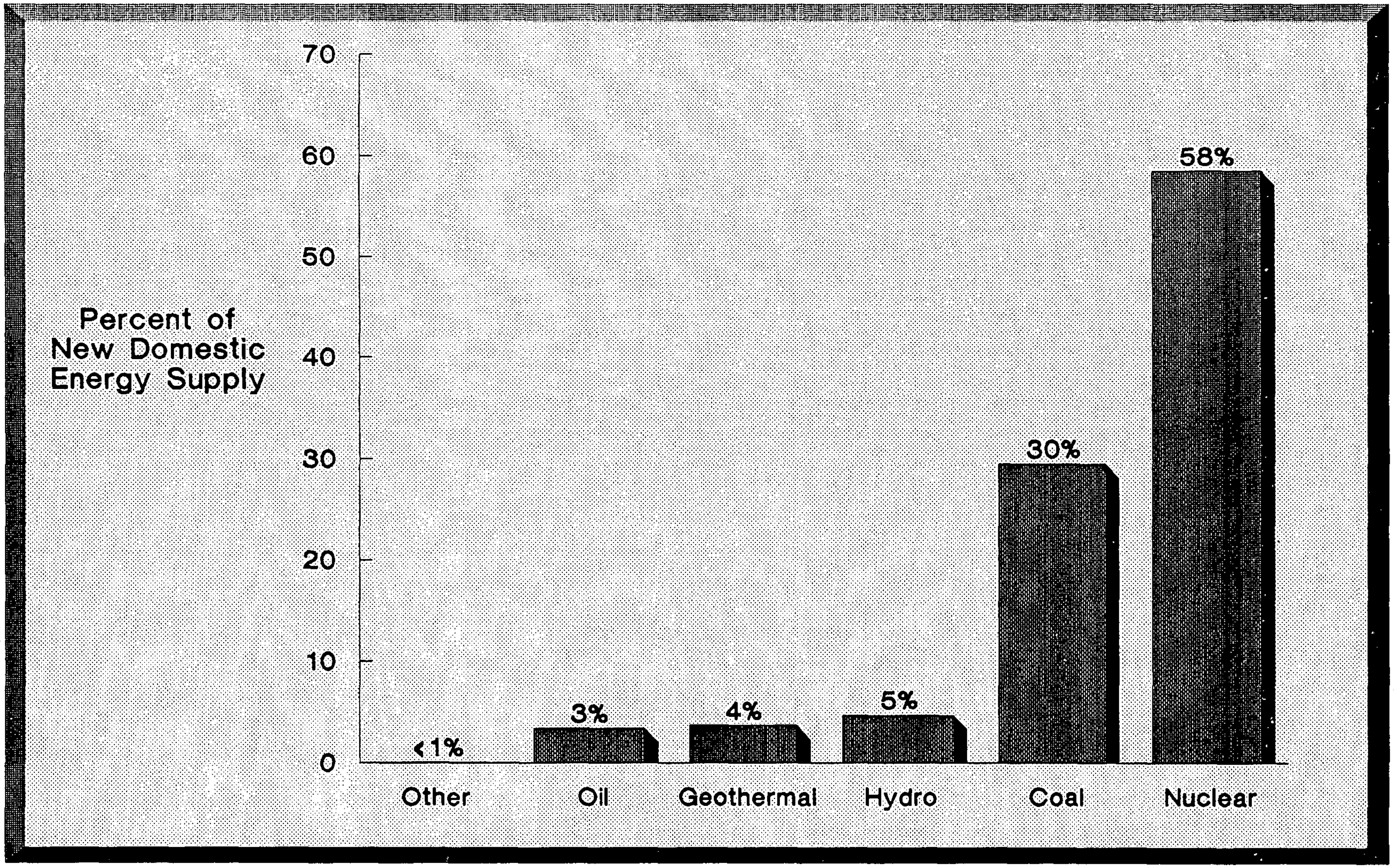
10/5/88:ZM:CHANGE

AN INCREASING SHARE OF U.S. ENERGY CONSUMPTION IS USED TO PRODUCE ELECTRICITY



10/22/88:ZM:6PIES

ADDITIONS TO DOMESTIC ENERGY SUPPLY: CONTRIBUTIONS FROM 1980-86



10/22/88:ZM:ADDITION

NUCLEAR DICHOTOMY IN THE UNITED STATES

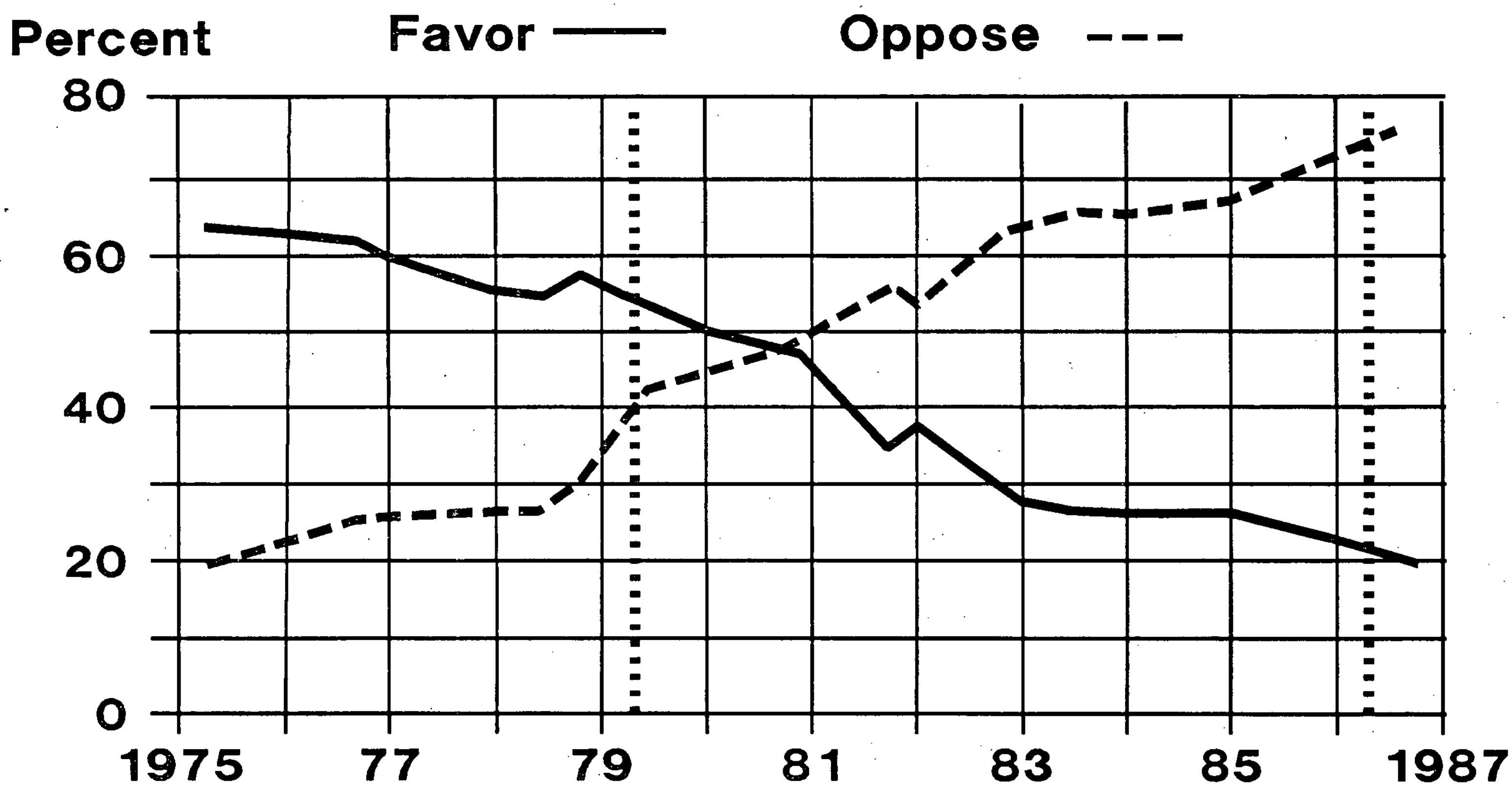
- Performance
  - 109 plants licensed to operate
  - Generating capacity of 97,539 MWe (18 percent of U.S. capacity)
  - Exemplary safety record

- No Plant Orders in 10 Years

9/20/88:ZM:NUDICH

PUBLIC ACCEPTANCE TOWARD NUCLEAR POWER

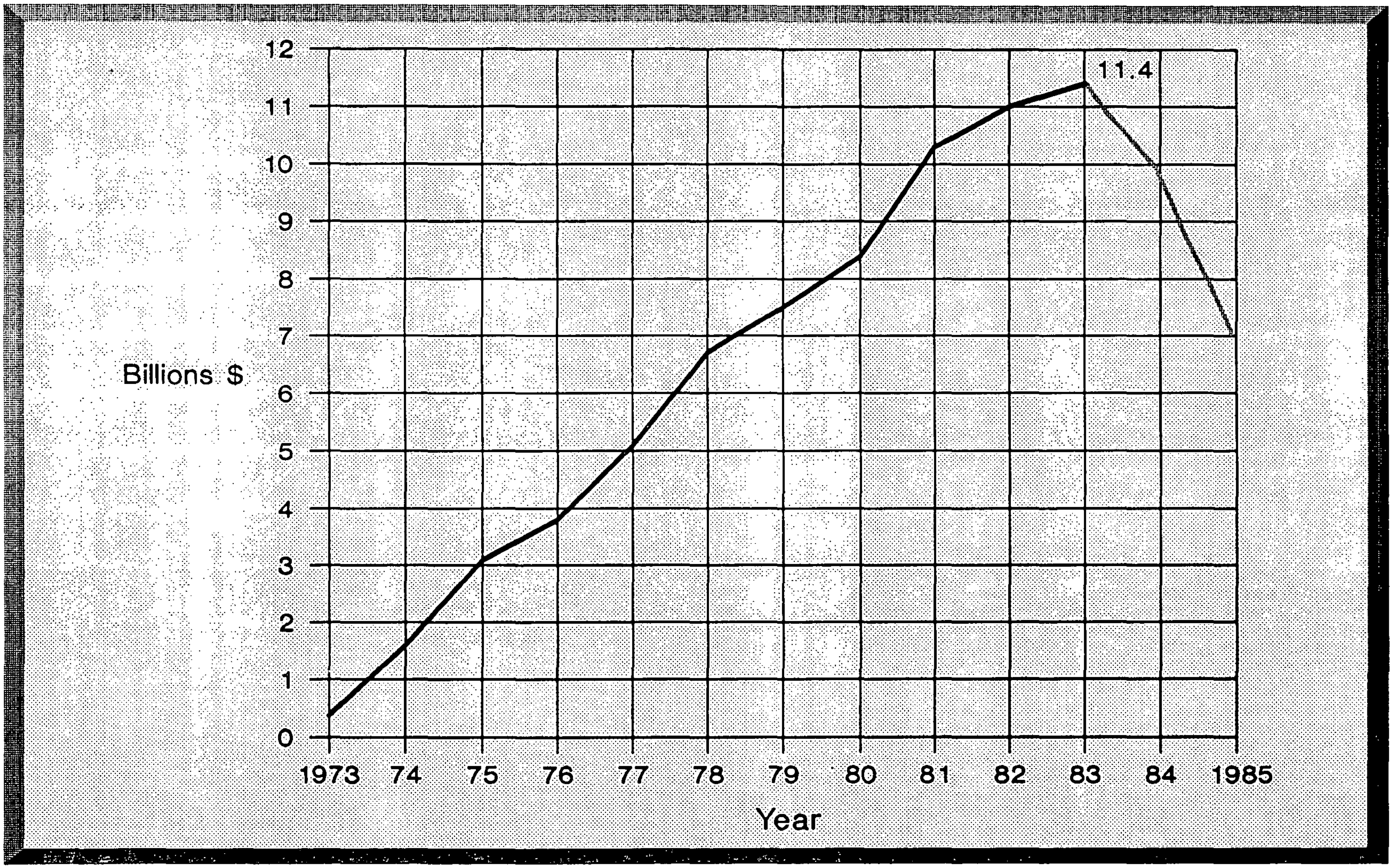
("...favor or oppose nuclear power plants in the U.S.")



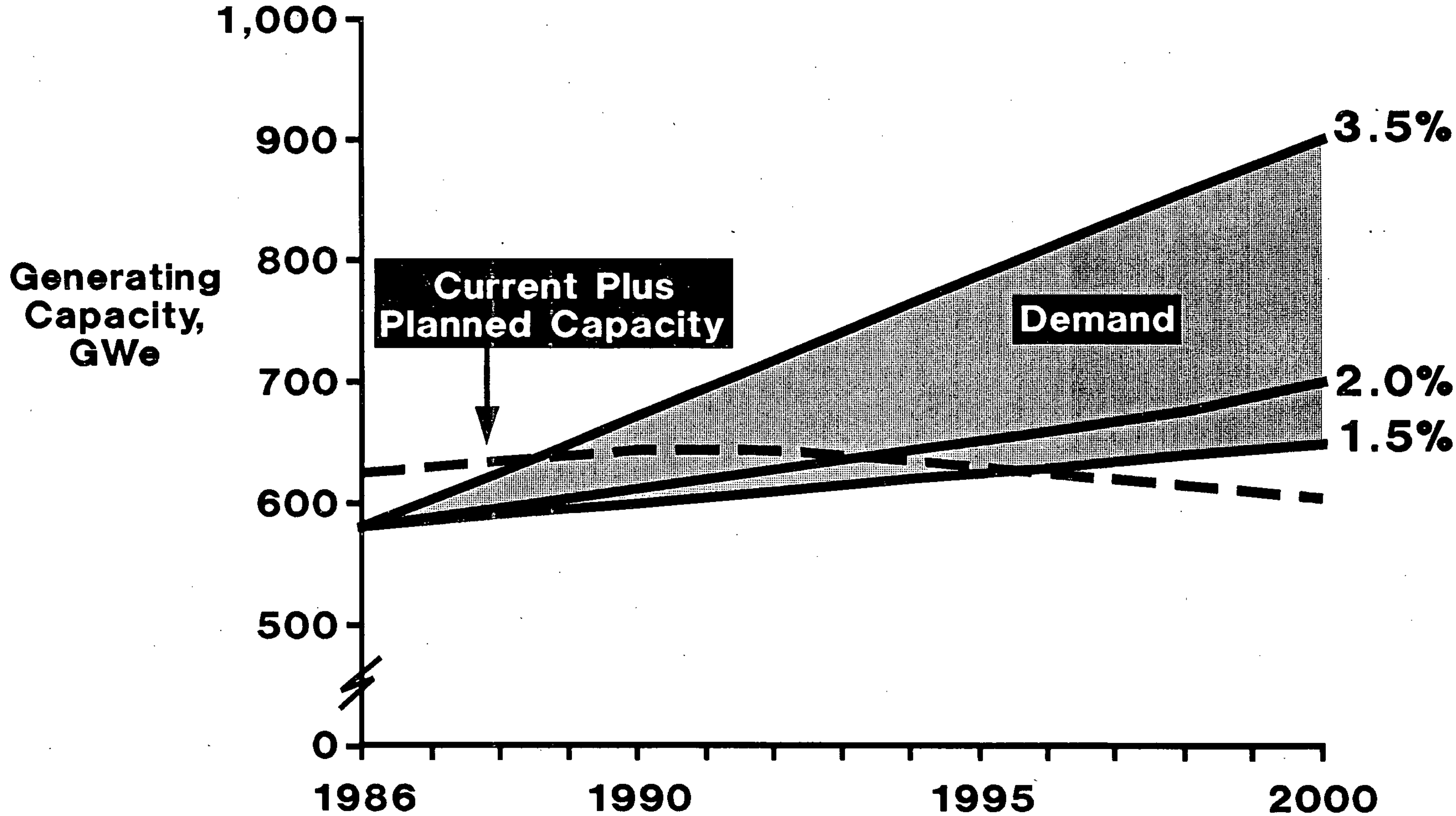
Source: Various Polls

10/17/88:ZM:PATNP

CUMULATIVE SAVINGS FROM NUCLEAR POWER



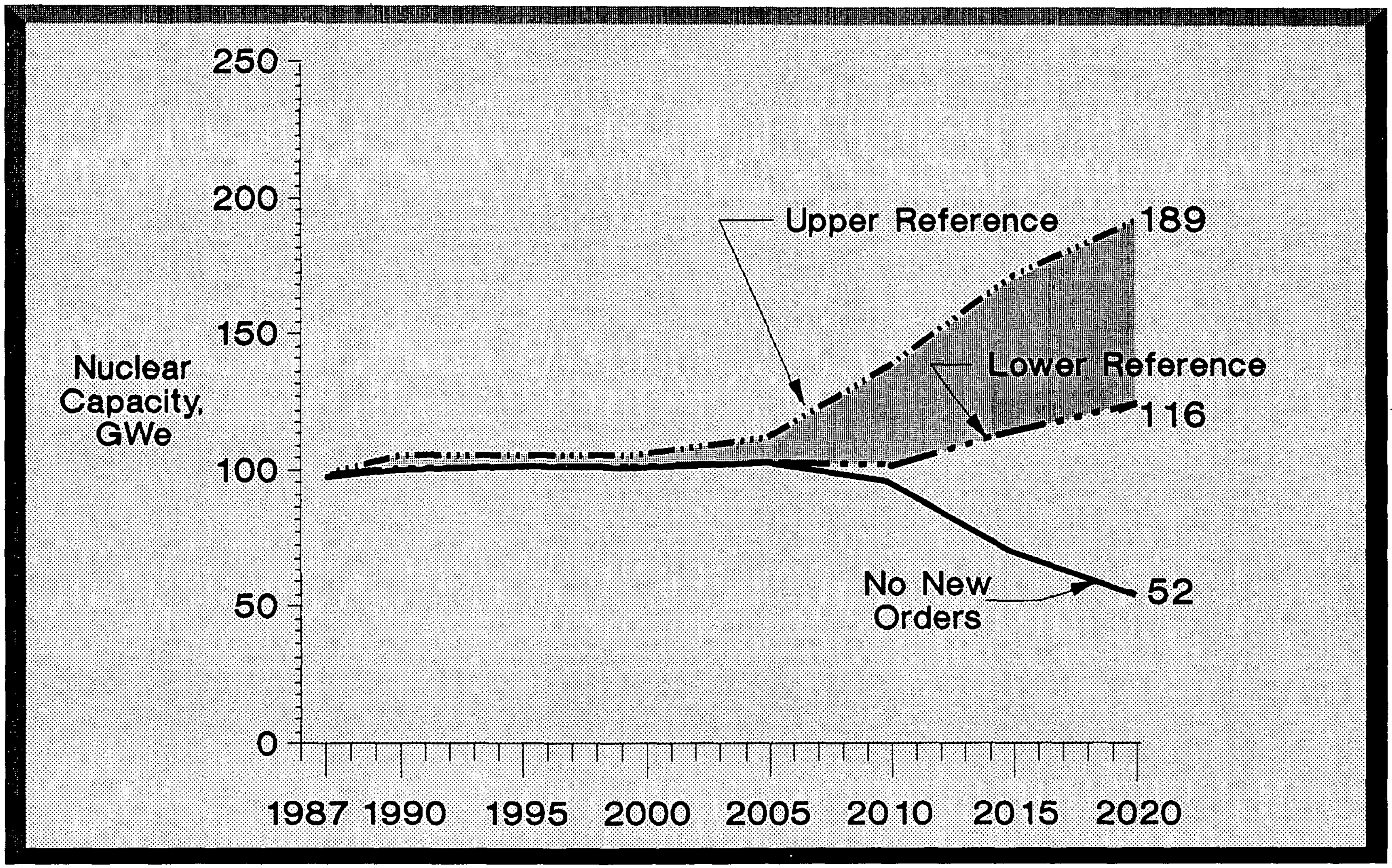
ELECTRICITY SUPPLY AND DEMAND 1986-2000



Sources: U.S. Department of Energy and Edison Electric Institute

10/17/88:ZM:ES&D

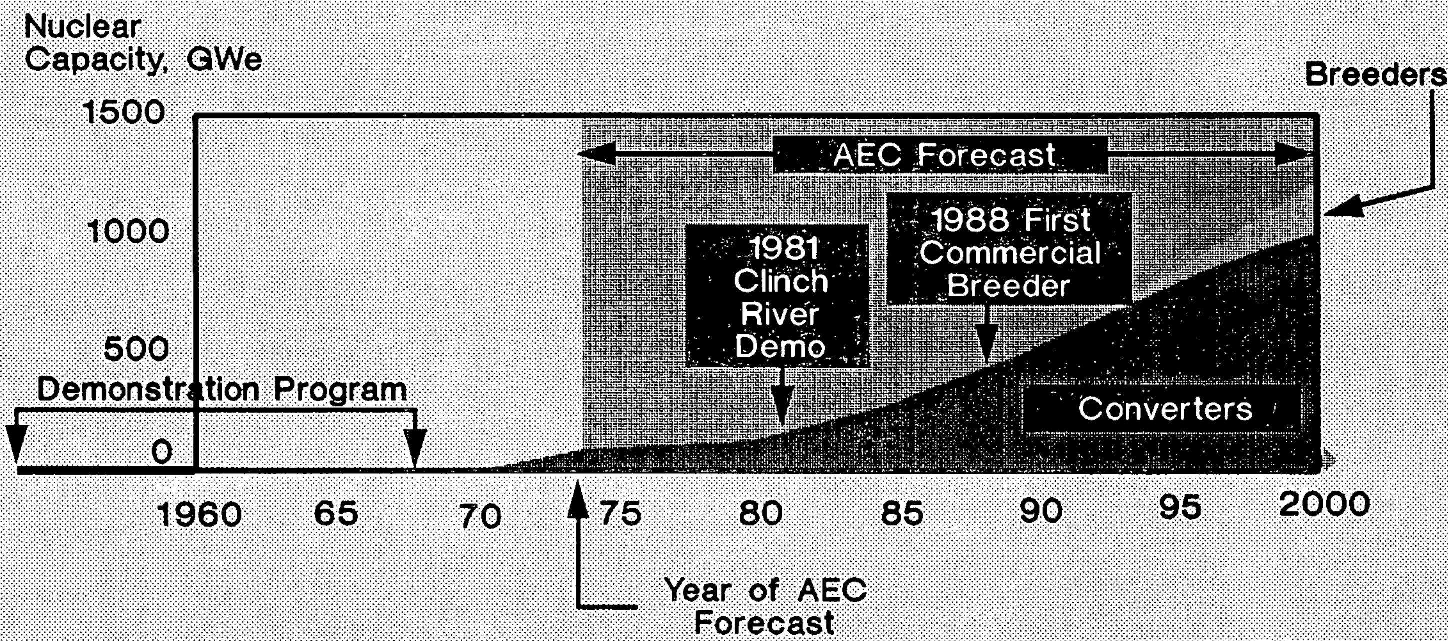
NUCLEAR GROWTH SCENARIOS



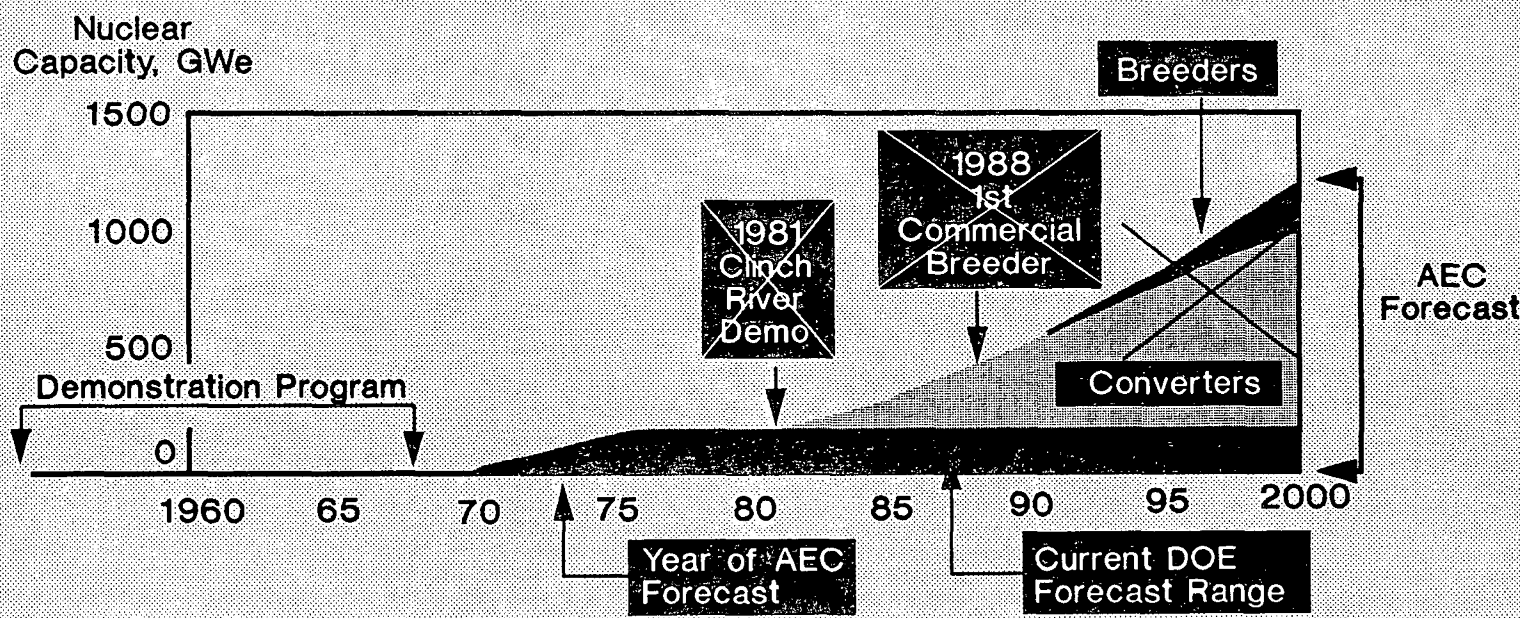
Source: 1988 EIA Forecast

10/22/88:ZM:NGS

EARLY ASSUMPTIONS FOR THE NUCLEAR R&D PROGRAM



REALITIES OF THE 1980s FOR THE NUCLEAR R&D PROGRAM



- 1972-77 36 Reactor Orders Cancelled
- 1973 Oil Embargo
- 1977-81 Double Digit Inflation
- 1977-83 75 Reactor Orders Cancelled
- 1979 TMI Accident
- 1981 1977 Ban on Reprocessing Rescinded
- 1982 Drop in Electricity Demand
- 1983 CRBR Cancelled
- 1986 Chernobyl Accident

10/22/88:ZM:REALIT

CHALLENGES SHAPING NUCLEAR R&D PROGRAM TO PROVIDE ECONOMIC POWER IN THE FUTURE

- Regulatory
- Public Attitudes
- Complexity
- Plant Life, Operations, and Maintenance
- Uncertain Load Growth
- Financing

10/22/88:ZM:CSNR&DBW

ADVANCED REACTOR PROGRAM APPROACH

CHALLENGES

- Regulatory
- Public Attitudes
- Complexity
- Plant Life, Operations and Maintenance
- Uncertain Load Growth
- Financing

TECHNICAL RESPONSES

- Passive Safety
- Modularity
- Standardization
- Improved Waste Management

10/22/88:ZM:ARPA

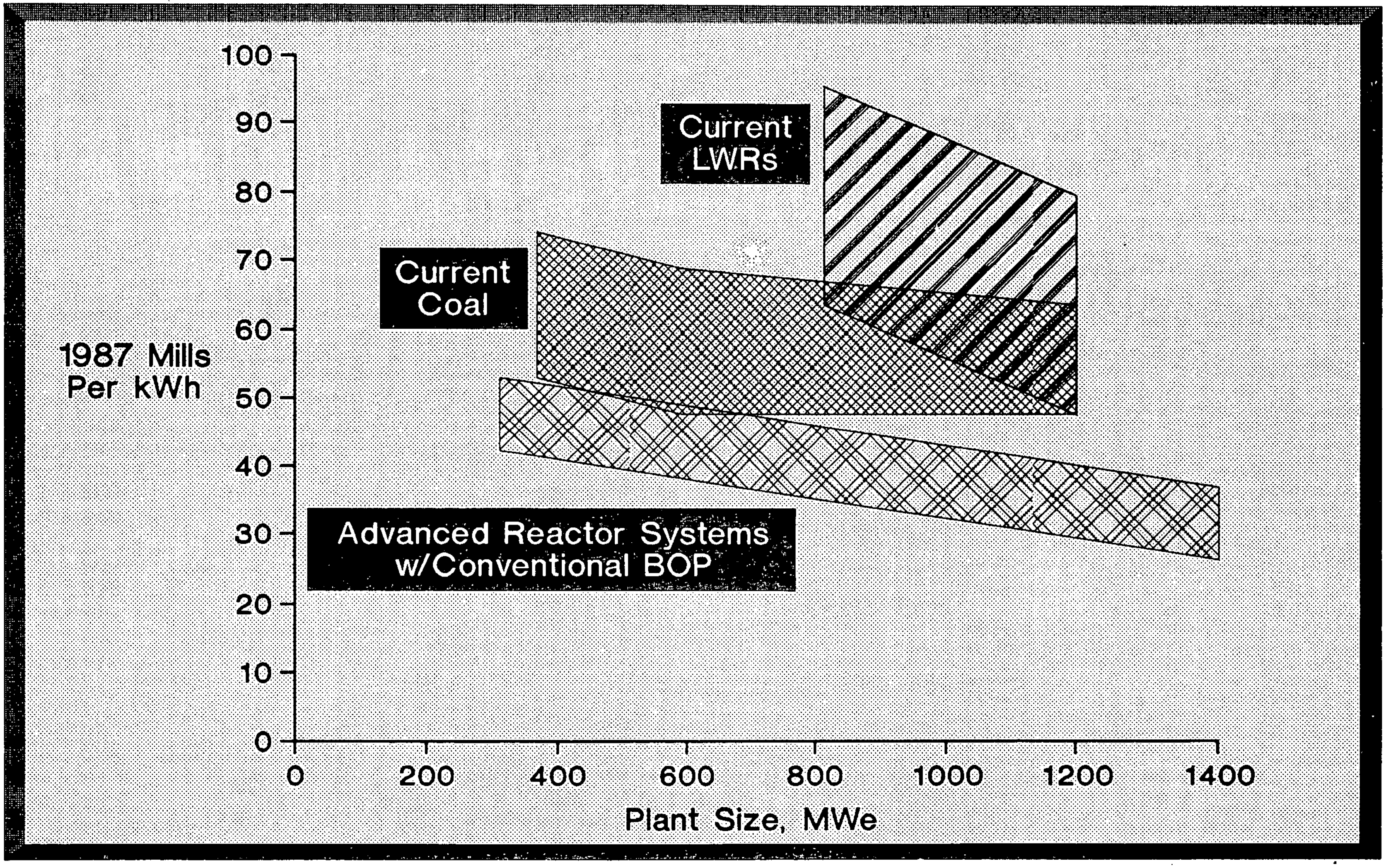
WHY PASSIVE AND INHERENT SAFETY

- Improves Safety Margin
- Requires No Operator Action or External Power
- Leads to Simpler Plants and Simpler Operations
- Risk of Core Melt Accidents is Insignificant
- Aids in Plants Capital Investment Protection
- Aids in Public Perception of Safety

ALL OF THE ABOVE LEAD TO
PLANT COST REDUCTION

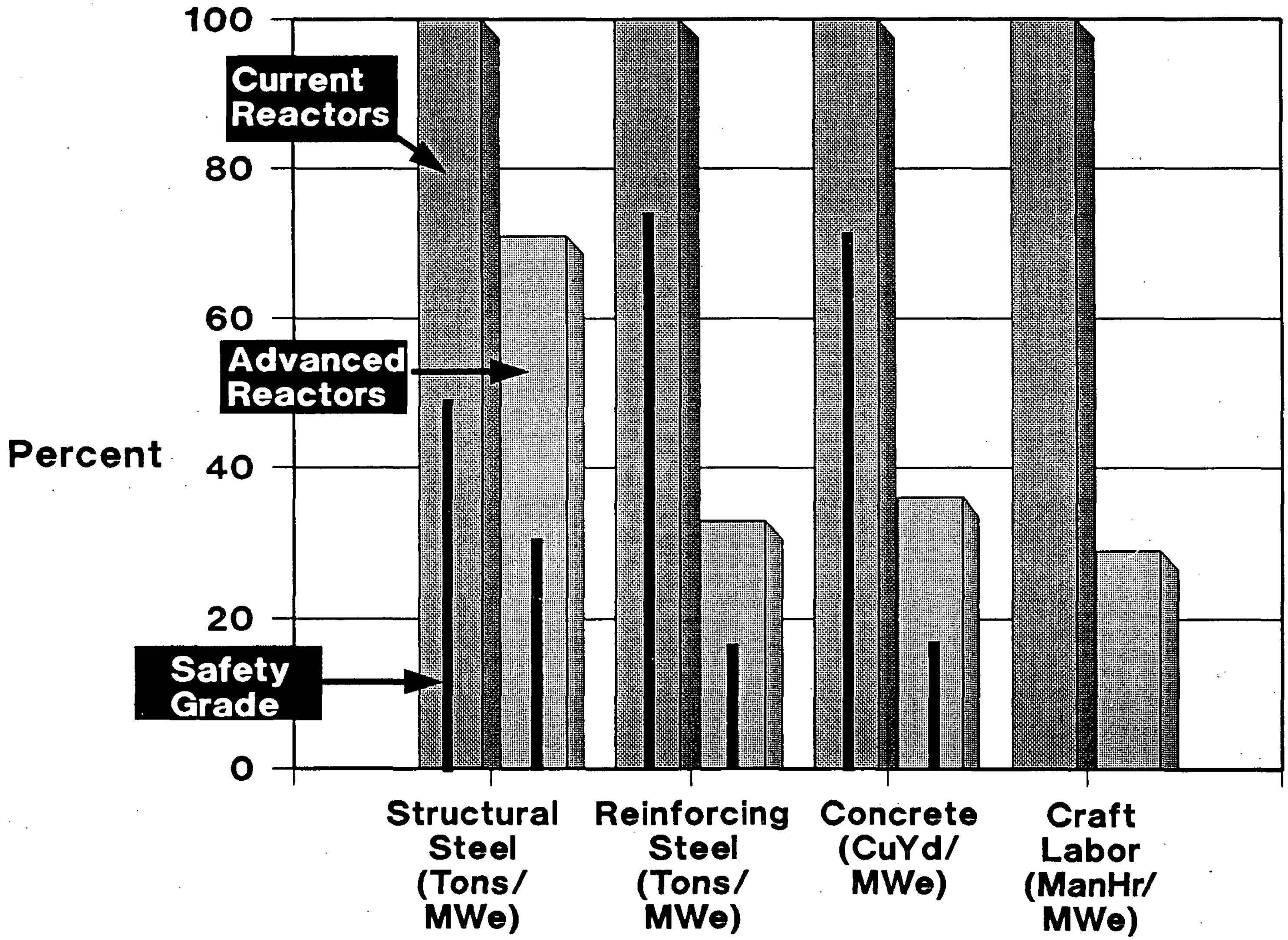
10/22/88:ZM:WHYPAS

POTENTIAL ECONOMIC BENEFITS FROM ADVANCED REACTORS



8/29/88:ZM:PEBFAR

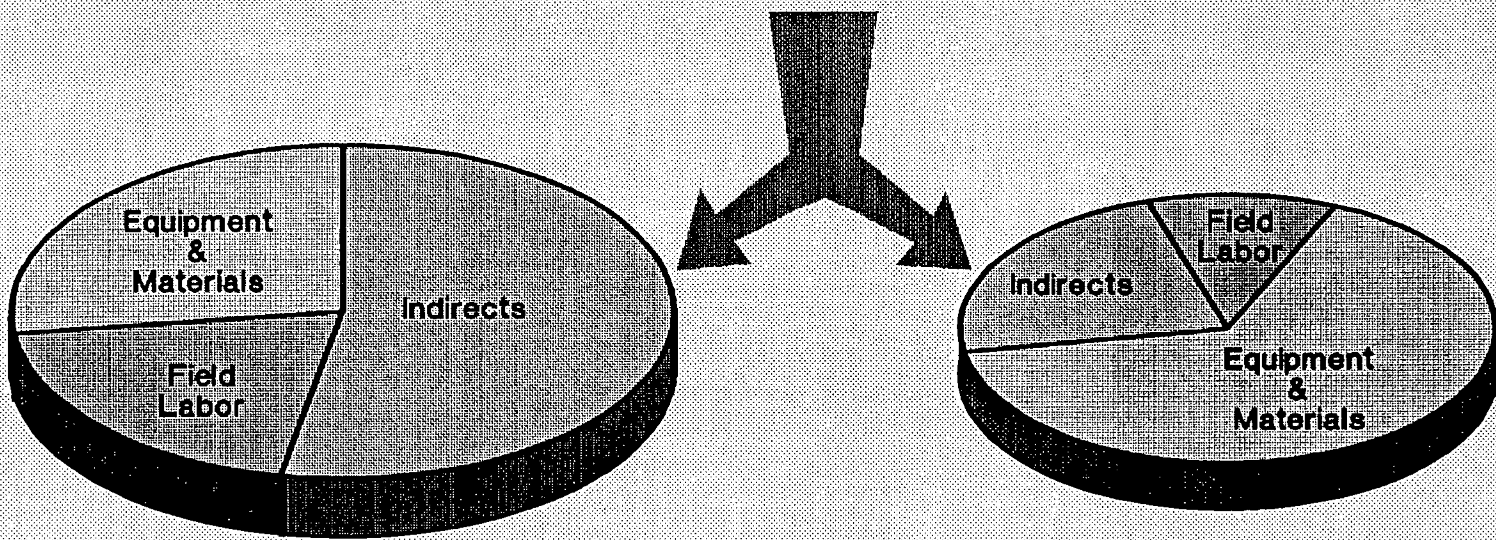
COMPARISON OF NUCLEAR PLANT COMMODITIES



10/20/88:ZM:CNPC

NUCLEAR PLANT COSTS CAN BE REDUCED

Largest Uncertainties



Current Technology

Advanced Reactors

10/22/88:ZM:NPCCBR

IMPROVED REACTOR DESIGNS

- Advanced Light Water Reactor
- Advanced Liquid Metal Reactor
- Modular High Temperature Gas Reactor

10/22/88:ZM:IMPREA

LWR PROGRAM

9/20/88:ZM:LWRPRO

NATIONAL LIGHT WATER REACTOR STRATEGY

- Demonstrate Improved Regulatory Process by Certification of Standardized Certified ALWR's Meeting Utility Requirements by 1991
- Design, Develop, and Certify Standardized Simpler, More Passively Safe ALWR Plants by 1995
- Concentrate on Evolutionary Designs Not Requiring a Demonstration Plant
- Ensure National Resource of More Than 100 Operating LWR's Not Prematurely Lost Due to Arbitrary License Period
- In Parallel, Assist in Resolution of Safety and Economic Regulatory Issues

10/7/88:ZM:NLWRS

EXPECTED BENEFITS OF THE ALWR

- Based on Existing Commercialized Design -- Direct Application of Proven Technology
- Improved Designs
  - Greatly simplified
  - Incorporate lessons learned
  - Passive safety
  - Potential for smaller size
  - Potential for standardization
  - Potential for extended plant life
- No Prototype Required
- Protection of Utility Investment -- Licensability
- Cost Competitive with Other Power Generation Options

10/7/88:ZMEBALWR

LMR PROGRAM

ADVANCED LMR PROGRAM APPROACH

CHALLENGES

Regulatory

Public Attitudes

Complexity

Plant Life, Operations and Maintenance

Uncertain Load Growth

Financing

TECHNICAL RESPONSES

Passive Safety
- Metal Fuel
- IFR Technology

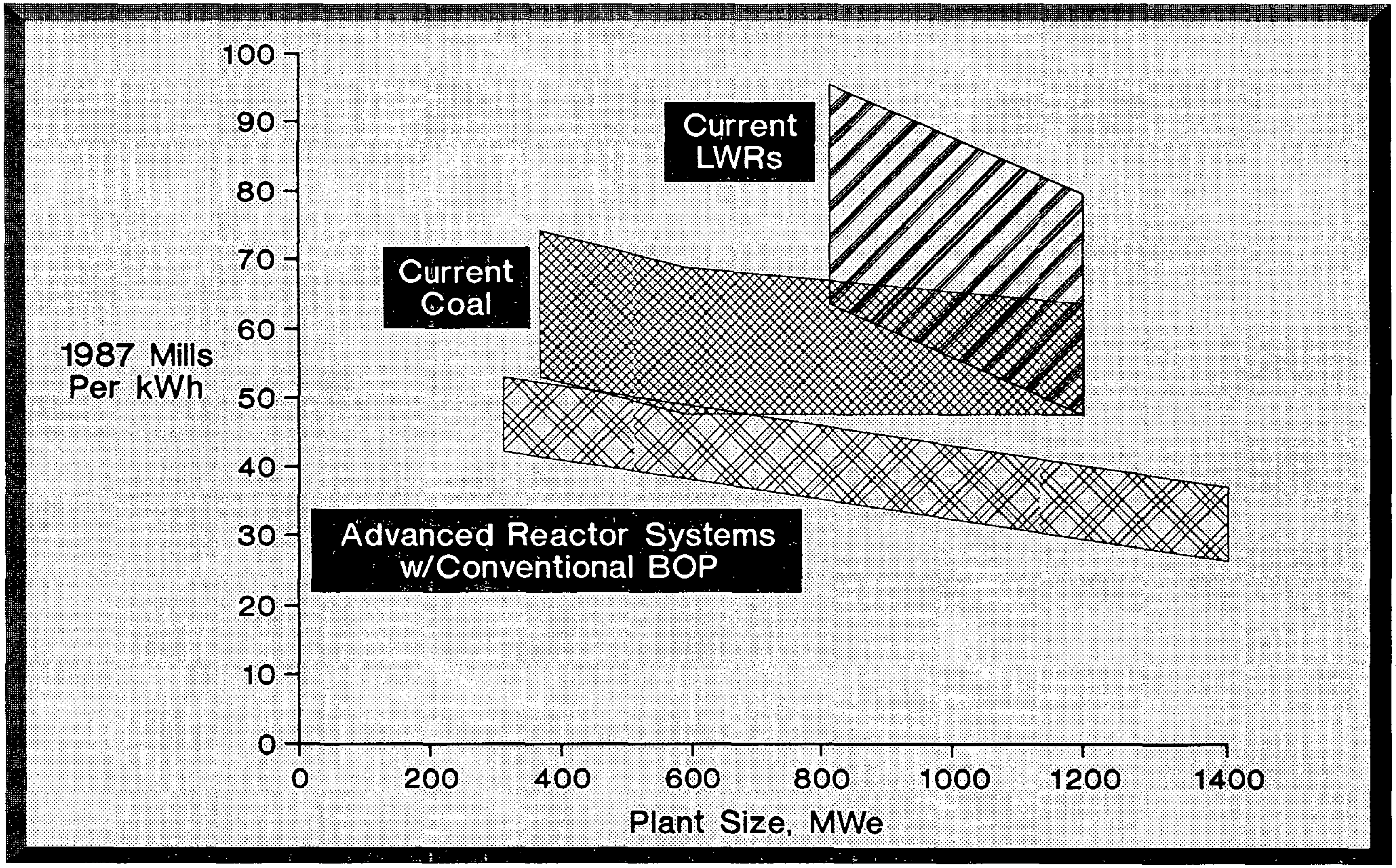
Modularity
- Factory Fabrication
- Advanced Instrumentation and Controls

Standardization
- Design Certification by NRC
- Component and System Reliability

Improved Waste Management
- IFR Fuel Cycle
- Actinide Burning

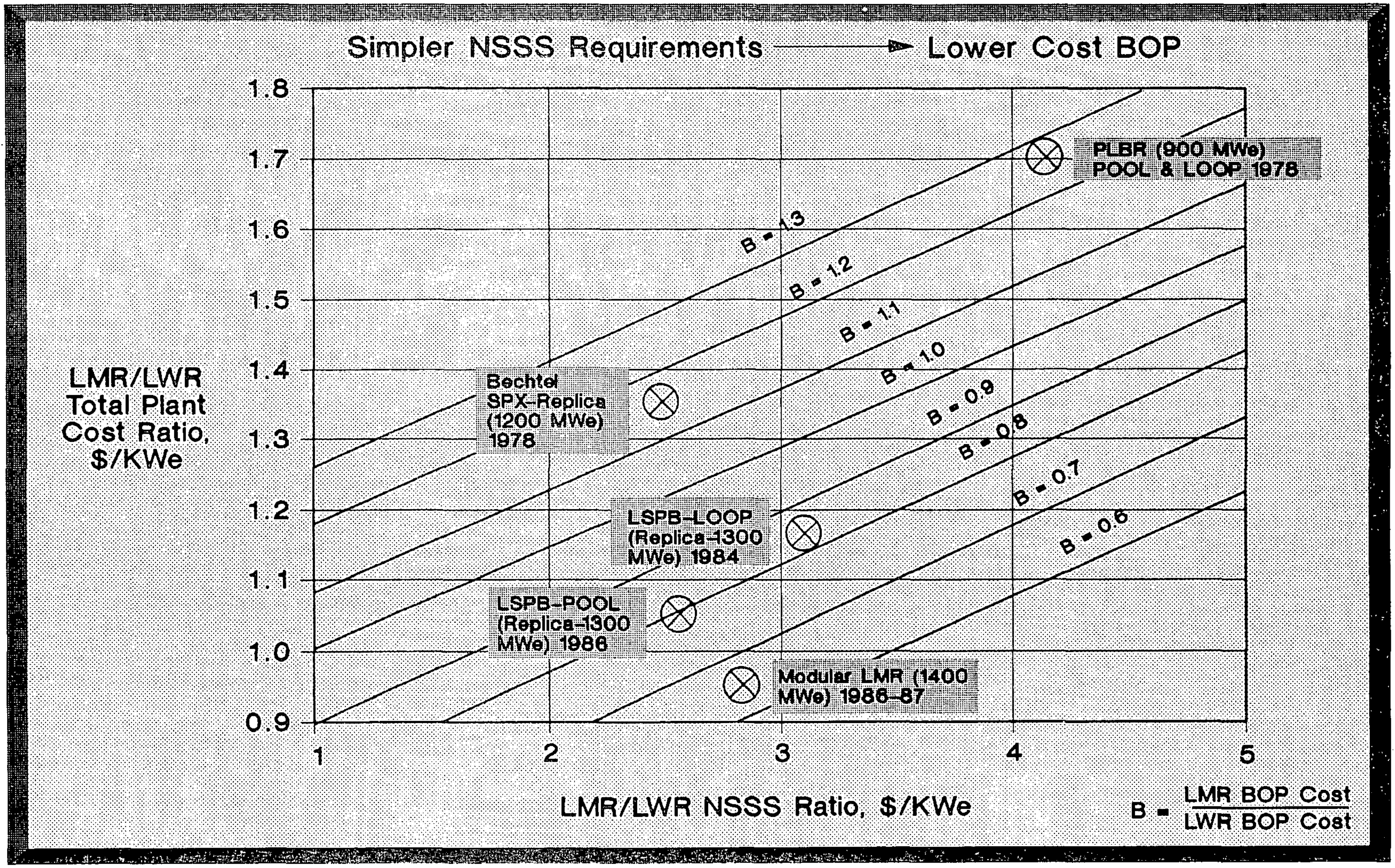
10/18/88:ZM:ALMRPA

POTENTIAL ECONOMIC BENEFITS FROM ADVANCED REACTORS



8/29/88:ZM:PEBFAR

ADVANCED LMR PLANT COST TRENDS



8/29/88:ZM:ALMRPCT

ADVANCED DESIGN INNOVATIVE FEATURE EXAMPLE: BOTTOM SUPPORT VERSUS TOP SUPPORT REACTOR VESSEL

	Potential Bottom Support Cost Savings (% of Top Support Costs)
Operating Floor and Top Closure	49%
Reactor Vessel	46%
Guard Vessel	73%
Total (Concept Sensitive Portions Only)	58% = $30.4 Million Savings★

★ Bechtel Corporation Analysis of Advanced LSPB MWe LMR Design

10/23/88:ZM:ADVDES

METAL FUEL CYCLE POTENTIAL BENEFITS

- Improved Fuel Performance
  - Greater safety margins
  - Smaller core
  - Fewer control rods
  - Improved breeding potential

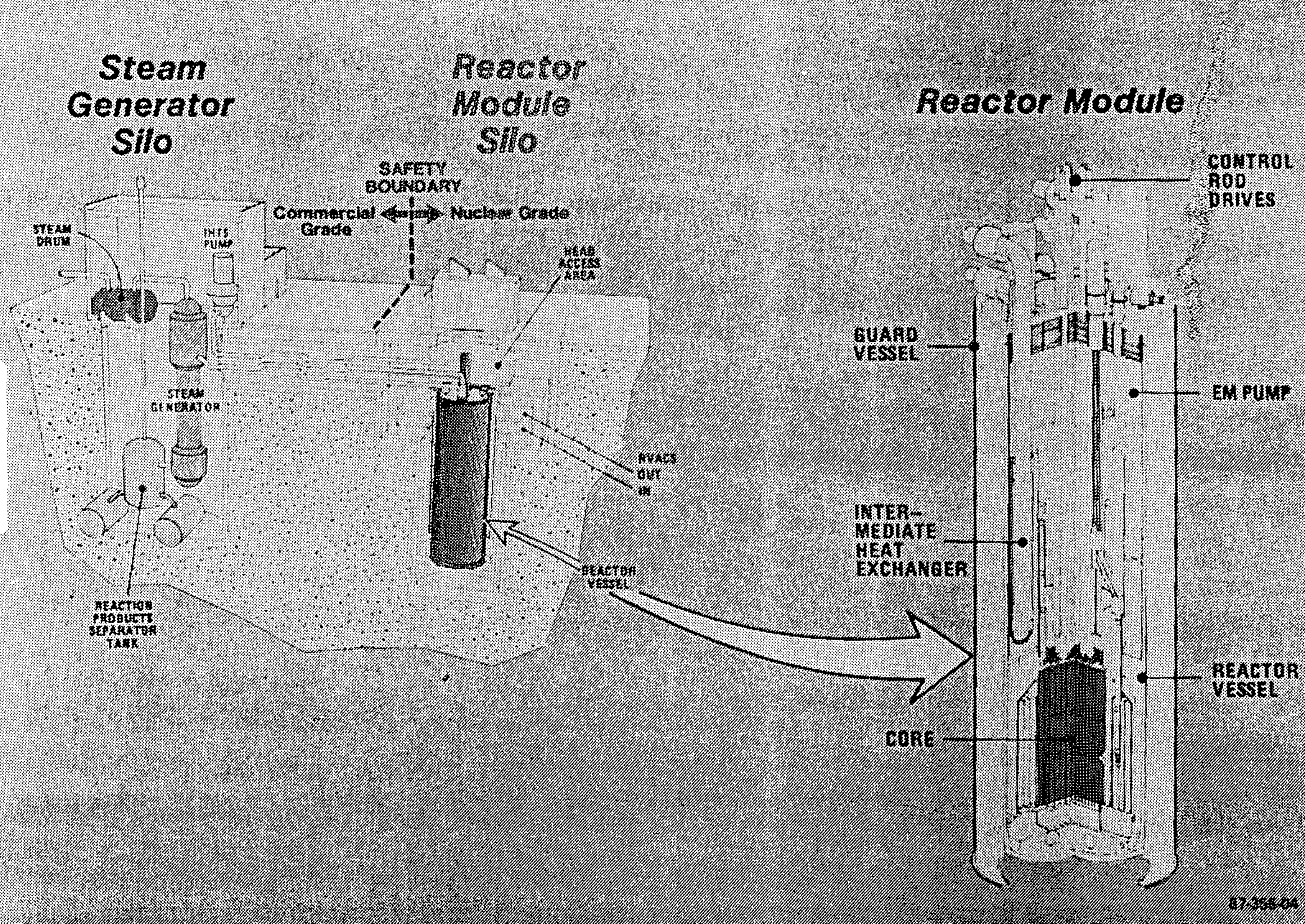
- Improved Fuel Cycle Economics
  - Simple and economic fuel fabrication
  - Inexpensive and compact fuel recycle

- Improved Waste Management
  - Smaller low-level waste volumes
  - Inactivation of actinides

- Deployment Flexibility and Low Cost Demonstration

10/23/88:ZM:MFCPB

PRISM NUCLEAR STEAM SUPPLY SYSTEM



LMR DESIGN DESCRIPTION

- Typical Power Unit Includes Three Nuclear Reactor Modules, Each 471 MWt
- Each Power Unit is Headered to One Turbine Generator
- Typical Plant Includes Three Power Units, Totalling 1395 MWe
- Reactor Modules are Located Below Grade

10/20/88:ZM:LMRDD

LMR PROGRAM PARTICIPANTS

- LMR Program is Funded by U.S. DOE with Limited Cost Sharing by Contractors

- The Design Team Includes
  - General Electric
  - Babcock & Wilcox/Foster Wheeler
  - Bechtel National, Inc.
  - Stearns-Roger
  - United Engineers & Constructors

- Argonne National Laboratory is Lead Laboratory for Technology

10/20/88:ZM:LMRPPC

WHAT ARE WE TRYING TO DO? LMR PROGRAM

- **Consolidate LMR Development Activities Around the Integral Fast Reactor Program**
  - Complete oxide fuel R&D (international collaboration)
  - IFR fuel cycle program (ANL)
  - Advanced IFR design (PRISM)
  - Advanced IFR technology development

- **Develop Breakthrough in Waste Technology**
  - Actinide burner (LMR) may significantly reduce waste management cost and public concerns (i.e., high-level waste reduced to low-level waste)
  - Establish synergistic fuel cycle/waste management relationship between LMR and/or LWR and HTGR

10/11/88:ZM:WAWLMR

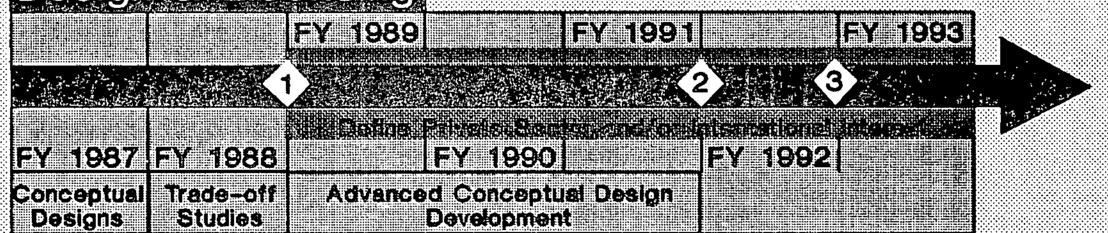
ADVANCED LMR PROGRAM SCHEDULE

Fuel Cycle R&D

IFR Fuel Cycle
- Fuel Development
- Core Design
- Safety Tests & Analysis
- Pyroprocess Development
- Recycle Demonstration

Oxide Fuel Cycle
- International Collaboration to Complete R&D

Design & Licensing



Technology Development

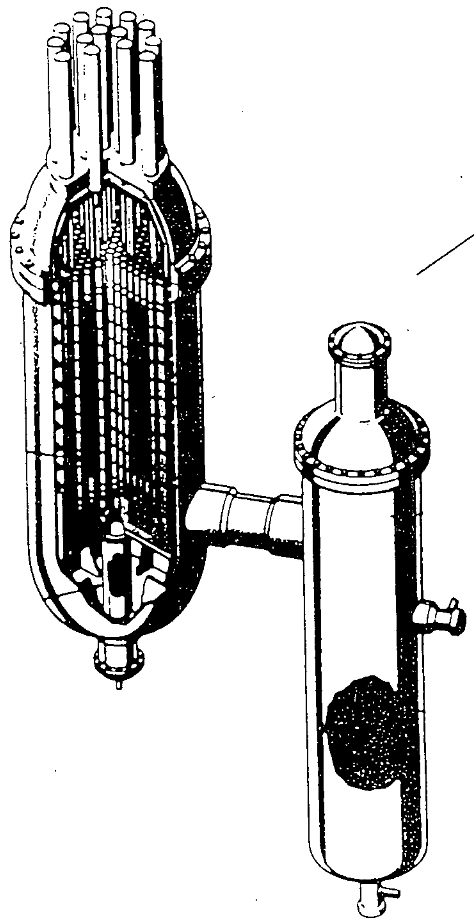
- Advanced Steam Generator
- IFR Technology
  - Control and monitoring systems
  - Advanced materials
  - Other

1. Select Reference Concept
2. Decide Whether to Further Develop Reference Concept
3. Define Future Needs for Advanced Reactor Development and Demonstration

8/29/88:ZM:ALMRPS

MHTGR PROGRAM

HIGH TEMPERATURE GAS REACTOR
POTENTIAL APPLICATIONS



COAL GASIFICATION

OIL SHALE RECOVERY

HEAVY OIL RECOVERY

TAR SANDS RECOVERY & UPGRADING

REPOWERING OF FOSSIL PLANTS

SMALL POWER PLANTS FOR U.S. & EXPORT

COGENERATION

PETROLEUM REFINING & PETROCHEMICALS

HIGH TEMPERATURE GAS REACTOR (HTGR) HIGHLIGHTS

- Peach Bottom 1 (40 MWe) Placed in Commercial Operation by Philadelphia Electric Company from 1967 to 1974
- Public Service Company of Colorado, AEC, and General Atomics Efforts Led to Construction of Fort St. Vrain (330 MWe), Which Began Generating Power in 1976
- Utility Interest in HTGR Expressed Through Gas-Cooled Reactor Associates, Representing 30 Utilities

10/11/88:ZM:HTGRHI

ADVANCED MHTGR PROGRAM APPROACH

CHALLENGES

Regulatory

Public Attitudes

Complexity

Plant Life, Operations and Maintenance

Uncertain Load Growth

Financing

TECHNICAL RESPONSES

Passive Safety

- Inert Helium Coolant
- Large Heat Capacity
- High Temperature Coated Fuel Particles
- Passive Heat Removal

Modularity

- Factory Fabrication
- Electricity Generation or Process Heat

Standardization

- Design Certification by NRC
- Component and System Reliability

10/18/88:ZM:AMHTGR

MHTGR PROGRAM PARTICIPANTS

- MHTGR Program is Funded by U.S. DOE in Cooperation with Gas-Cooled Reactor Associates (GCRA)

- The Design Team Includes
  - General Atomics
  - Bechtel National
  - Combustion Engineering
  - Stone & Webster Engineering

- Oak Ridge National Laboratory is Lead Laboratory for Technology

10/20/88:ZM:MHTGRPP

MHTGR SYSTEM REQUIREMENTS

- Plant Has Been Designed by a Top–Down Functional Analysis Approach
- Minimal Reliance is Placed on Active Component Features or Operator Action
- Top–Level Regulatory and User Requirements are Met Without Requiring a Conventional Containment Vessel

10/20/88:ZM:MHTGRSR

MHTGR DESIGN DESCRIPTION

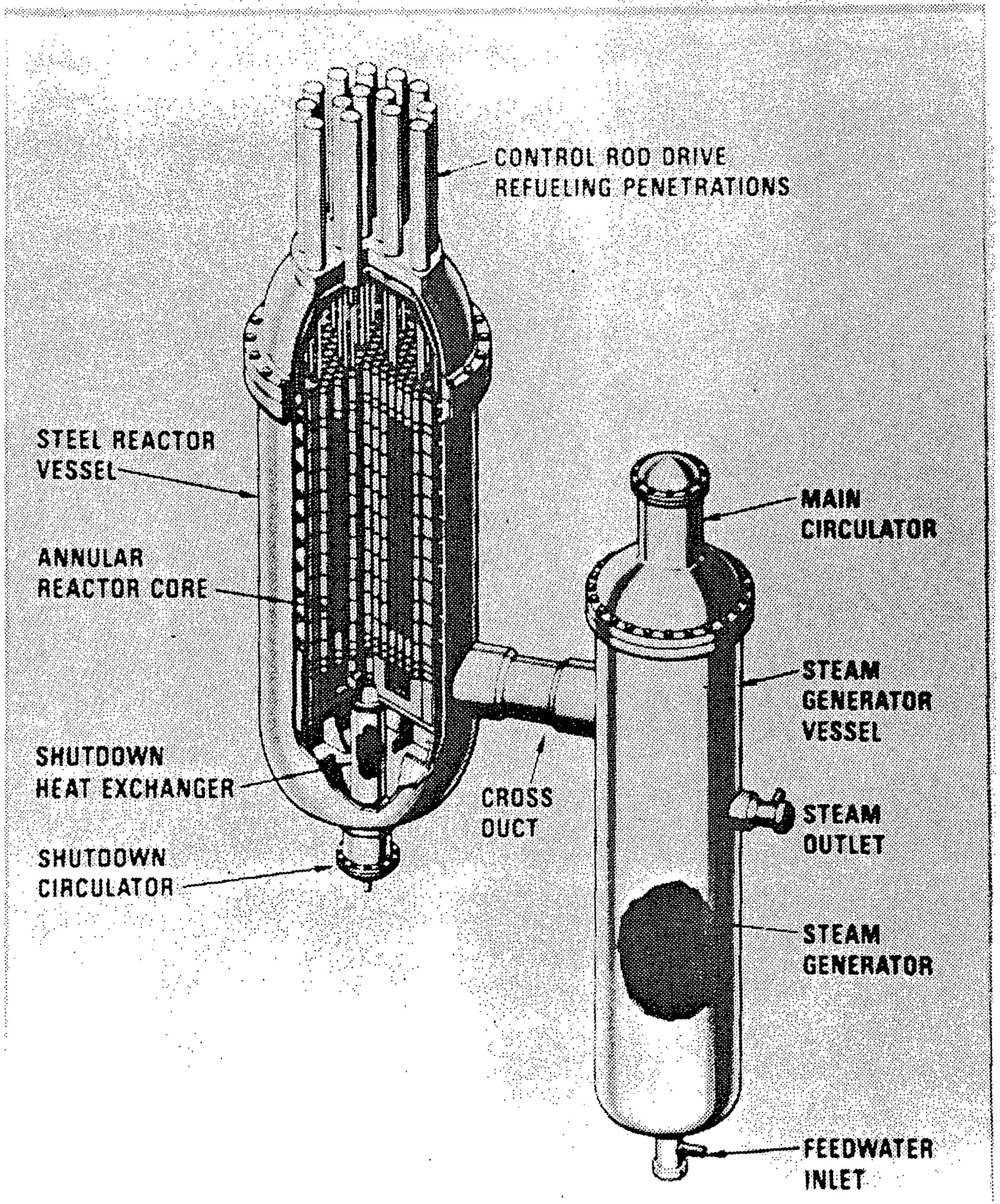
- Typical Plant Includes Four Reactor Modules, Each 350 MWt Capacity

- The Four Modules are Headered to Two Turbine Generators, to Produce 550 MWe

- Reactor Modules are Located Below Grade

10/21/88:ZM:MHTGRDD

MODULAR HTGR



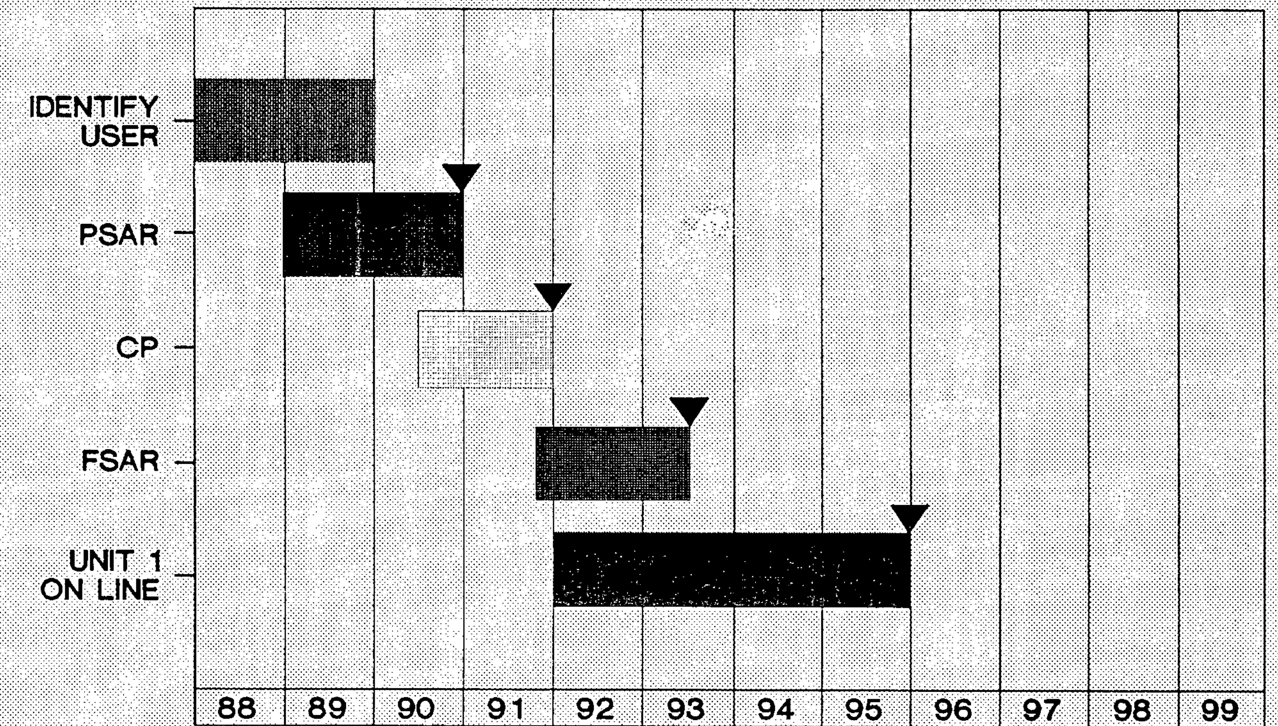
WHAT ARE WE TRYING TO DO?
MHTGR PROGRAM

- Head Towards an HTGR Project With Private Sector and International Cost Sharing
  - Establish technology and licensing bases
  - Develop cost-sharing support

- Integrate HTGR Development Efforts With NPR Requirements

10/11/88:ZM:WAWHTG

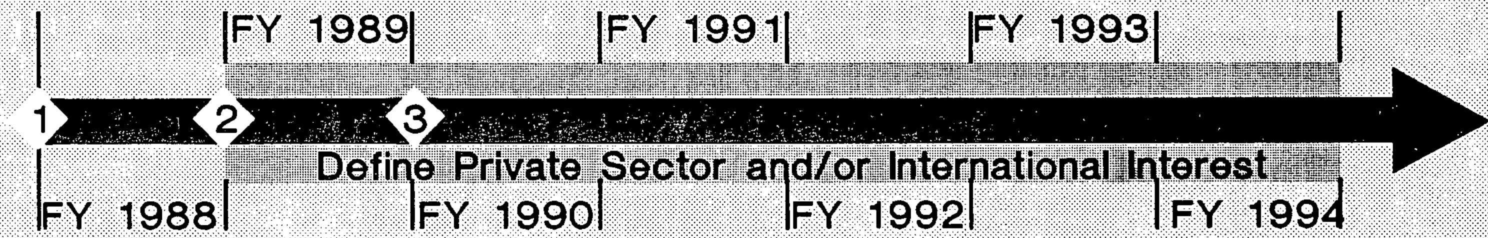
MHTGR INITIATIVE SCHEDULE



FOAK Funds: (1990 $M)											Total
DOE	30	50	85	85	75	50	25	0	0	0	400
Private & Foreign	8	10	32	35	32	26	30	27	0	0	200
First Plant	0	30	70	150	165	160	115	110	290	110	1200
	38	90	187	270	272	236	170	137	290	110	1800

7/19/88:ZM:MHTGR1

ADVANCED MHTGR PROGRAM SCHEDULE

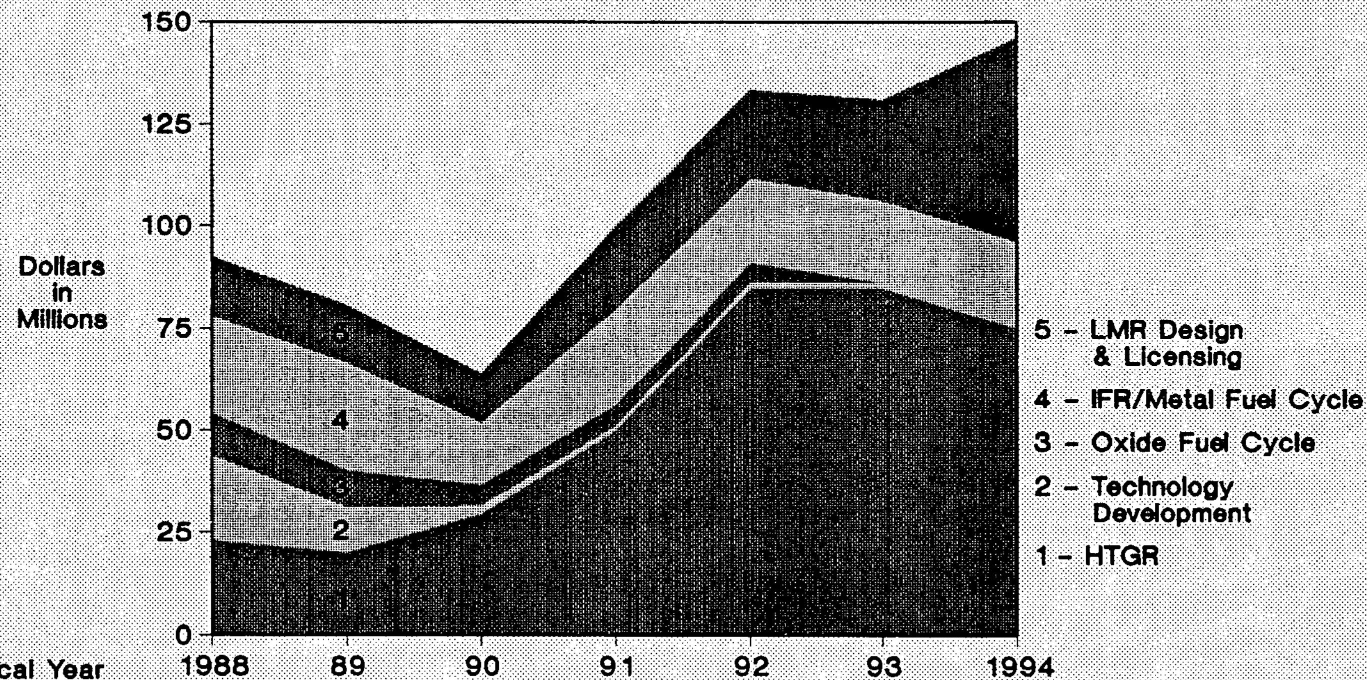


1. Start Title I Preliminary Design
2. NRC Issues Safety Evaluation Report
3. Assess Potential Private Sector Support for Detailed Design

9/2/88:ZM:AMHTPS

TOTAL ESTIMATED R&D FUNDING REQUIREMENTS (ADVANCED REACTOR R&D PROGRAM)

Year of Expenditure Dollars



Fiscal Year	1988	89	90	91	92	93	1994
LMR Design & Licensing	14.2	14.0	12.0	20.5	22.0	25.0	50.0
IFR/Metal Fuel Cycle	23.8	26.2	15.0	23.5	20.5	20.0	20.0
Oxide Fuel Cycle	10.1	8.9	5.0	5.0	5.0	0.0	0.0
Technology Development	20.7	11.4	2.0	1.0	1.0	1.0	1.0
HTGR	22.7	20.0	29.9	50.0	85.0	85.0	75.0
Subtotal	91.5	80.5	63.9	100	133.5	131	146

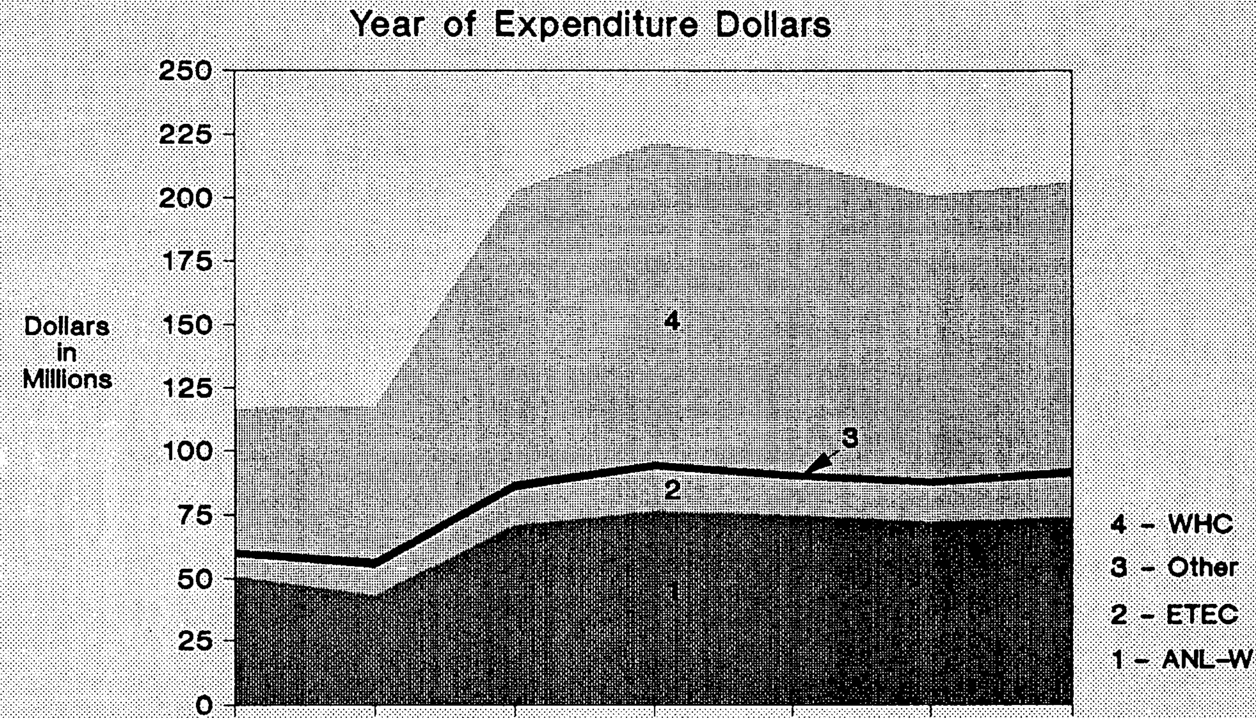
9/7/88:ZM:ADVREA

FACILITIES PROGRAM

FACILITIES UTILIZATION

EBR-II	Engineering Irradiation Tests with Rapid Turnaround, IFR Data Base
FFTF	Full-Scale Prototypic Irradiation, Pu-238 and Other Isotope Production, Materials Testing, Power Addition
FMEF	Pu-238 Target Fabrication and Recovery, Medical and Industrial Isotope Recovery
TREAT	Transient Tests
ZPPR	Critical Core Assemblies for Design Verification
ETEC	Non-Nuclear (e.g., Steam Generators) Component Performance Tests

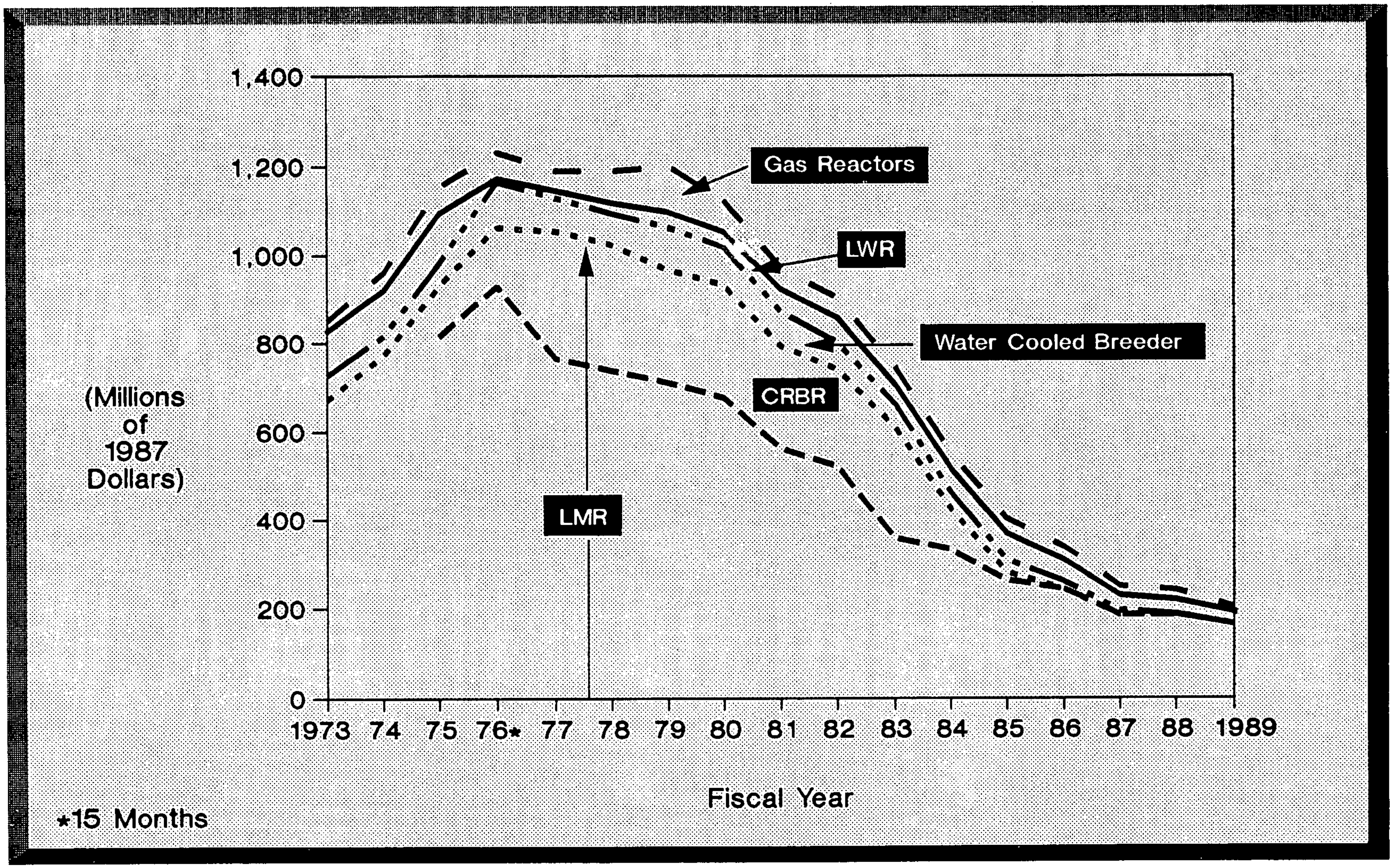
TOTAL ESTIMATED R&D FUNDING REQUIREMENTS (FACILITIES PROGRAM)



Fiscal Year	1988	89	90	91	92	93	1994
WHC	55.7	60.5	115.4	125.6	123.1	111.2	113.3
Other	2.6	3.5	2.5	2.8	2.6	2.4	2.4
ETEC	7.2	10.4	14.2	16.8	14.3	15.4	16.0
ANL-W	51.5	43.8	71.0	76.0	75.1	71.9	74.3
Facilities Total	117.0	118.2	203.1	221.2	215.1	200.9	206.0

SUMMARY

ANNUAL EXPENDITURES FOR CIVILIAN NUCLEAR R&D



8/8/88:ZM:AECNRD

THE ALWR - REGULATORY STABILIZATION THROUGH SIMPLICITY, MARGIN, AND IMPROVED SAFETY

GARY VINE
SUSAN GRAY

ADVANCED LWR PROGRAM

16TH WATER REACTOR SAFETY RESEARCH MEETING
OCTOBER 27, 1988

WRSRM

EPRI ALWR Program

ALWR FUNDAMENTAL ACCEPTANCE CRITERIA

- **Technical excellence**
  The ALWR must be a good power plant - safe, user-friendly, efficient, compatible with the environment
- **Economic advantage**
  The ALWR must be economically competitive with other power generation options, considering life cycle cost and first cost
- **Investment protection**
  The ALWR must provide very high protection of the utility investment, particularly in terms of:
  - very low susceptibility to major accident
  - assured licensibility
  - predictable and controllable construction schedule
  - predictable and controllable plant availability

WRSRM

EPRI ALWR Program

PROGRAM APPROACH

- Ensure utility focus, leadership
- Examine experience, build on success
- Involve NSSS vendors and A-Es, apply their talents; incorporate their best products and ideas
- Develop specific design and performance requirements for the ALWR ("The Requirements Document")
- Work with NRC in a constructive, nonconfrontational environment to resolve outstanding nuclear safety issues
- Establish a sensible starting point for standardization

WRSRM T0-5 *EPRI ALWR Program*

USC TECHNICAL GUIDANCE

In establishing ALWR requirements, Utility Steering Committee has placed very high emphasis on:

- Nuclear safety
- Simplification
- Margin
- Proven technology
- Human factors
- Regulatory Stability

WRSRM T0-6 *EPRI ALWR Program*

ALWR PROGRAM - SAFETY PRINCIPLES

The ALWR will achieve its safety criteria (including severe accident protection) by:

- Reliance on fundamentals - simple and rugged design
- Defense in-depth
- Balance between prevention and mitigation
- Reasonable consideration of severe accident events outside of the licensing design basis

WRSRM

EPRI ALWR Program

THE ALWR UTILITY REQUIREMENTS DOCUMENT

The Requirements Document is the primary work product in this phase of the ALWR Program

- It establishes top-tier, functional and system/component design requirements for evolutionary and passive plants, both PWR and BWR.
- It articulates policy and principles which define the ALWR program approach to nuclear safety, including severe accident protection.
- It incorporates resolutions of generic safety issues and optimization issues
- It reflects industry and NRC concensus on the principal safety, performance, and design requirements of the ALWR.

WRSRM

EPRI ALWR Program

ALWR Defense in Depth

Accident Resistant Designs

ALWR Design Features Which Provide Intrinsic Safety

- Design Margins
- Simplicity
- Best Materials
- Extended Operator Response Times

Chapters 3 & 4 - RCS and Reactor Systems

Core Damage Prevention

Systems Which Prevent Initiating Events from Progressing to the Point of Core Damage

Mitigation

Systems to Contain Fission Products Released as a Result of Core Damage Accidents

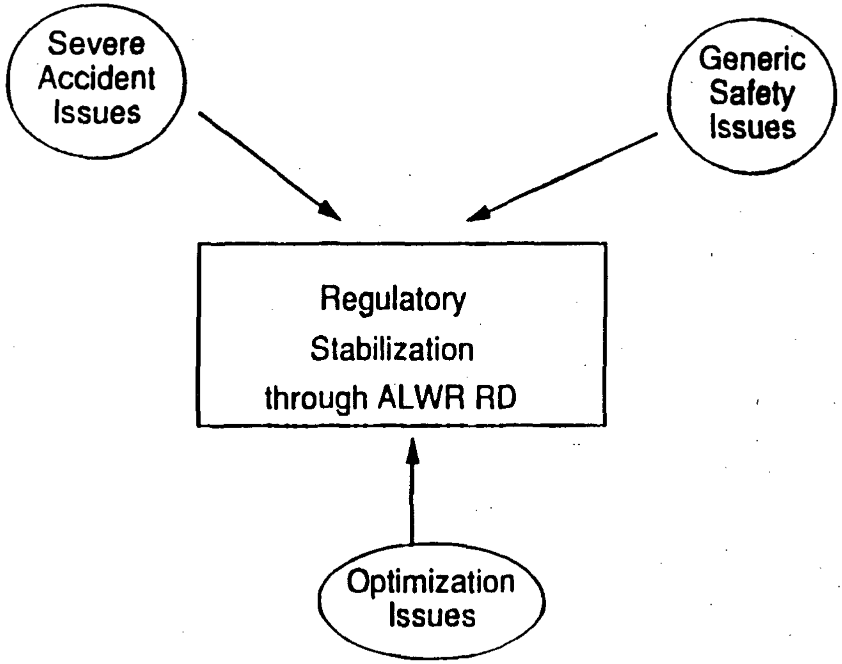
Chapter 5 - Engineered Safety Systems

Illustration of the Three Parts That Make Up the ALWR Design Bases

	ALWR Design Bases		
	Licensing Design Basis	**Risk Evaluation Basis**	**Performance Evaluation Basis**
Events	Safety Events (as defined in Chapter 1)	Core Damage Events	Performance Events (Chapter 1, Table 3-6) Investment Protection Events (Chapter 1, Table 3-8)
Analysis Methods	Conservative, NRC-approved Methods	Probabilistic Risk Assesment (PRA) Methods	Designer-selected Methods and Margins

-474-

CLOSURE OF ISSUES IS VITAL TO THE REGULATORY STABILIZATION EFFORT



Improved ALWR safety features as required by ALWR Requirements Document is a mechanism to achieve resolution of issues

CLOSURE OF ISSUES IS VITAL TO THE REGULATORY STABILIZATION EFFORT

INSERT DRAWING HERE WHICH SHOWS ISSUES. SEE 7/88 PRESENTATION NOW ON POWER POINT 2.

Improved ALWR safety features as required by ALWR Requirements Document is a mechanism to achieve resolution of issues

WRSRM

EPRI ALWR Program

GENERIC SAFETY ISSUE RESOLUTION

727 issues identified (prior to 7/1/86); of these:

- 450 deleted, as non-applicable or low priority
- 208 resolved via RD commitment to comply
- 69 to be resolved by study, topic paper deployment and treatment in RD; of these 13 have been resolved to date.

WRSRM

EPRI ALWR Program

ALWR TREATMENT OF GENERIC SAFETY ISSUES
Example: Station Blackout (GSI A-44)

ELEMENTS OF SAFETY ISSUE RESOLUTION INCLUDE:

Core Damage Frequency requirement < 1 X 10(-5) /year

Major Improvements in AC and DC power system reliability

Requirement to cope with SBO for specified period based on realistic evaluation (8 hours)

Requirement for large, diverse on-site alternate AC power source*

PWR RCP seals required to limit leakage on loss of all cooling such that core uncovering does not occur within 8 hours

PWR Emergency Feedwater System required to contain a turbine driven pump in addition to the motor driven pump in each of two trains

BWR required to have three divisions of core cooling, each with an independent AC and DC supply; BWR required to have one additional turbine driven isolation cooling pump

* Current Draft Chapter 11 direction

WRSRM

EPRI ALWR Program

ALWR TREATMENT OF ACCIDENTS INCLUDING SEVERE ACCIDENTS

- Design for Licensing Design Basis Events - this imparts substantial margin to the design
- Add margin and features for significant further prevention of core damage; include limited set of prudent mitigation features
- Evaluate dominant severe accident scenarios on a realistic basis; show sufficient margin to ultimate plant capability
  - demonstrate by PRA that top-tier ALWR requirements* have been met
  - account for:
    - - ALWR system/component requirements provided for accident prevention and mitigation
    - - other, specified ALWR features which enhance severe accident capability

* CDF < 1 X 10 (-5) / year; Frequency < 1 X 10 (-6) / year for site boundary dose greater than 25 rem whole body

WRSRM

EPRI ALWR Program

FIRM CLOSURE OF SEVERE ACCIDENT ISSUES IN ALWR

- Significant design features (sufficient for closure) in Requirements Document
- Requirements Document SER is the vehicle for documenting regulatory concurrence in design basis for severe accident closure; Key issues include:
  - definition of core debris coolability [.02 m_2/ MW_T + water on debris]
  - definition of acceptable H^2control [75% reacted clad; H^2<13%]
  - PWR safety depressurization system + cavity configuration *resolves direct containment heating*
- Staff appreciation for major improvements in Severe Accident prevention is essential

WRSRM

EPRI ALWR Program

EXAMPLES OF PWR FEATURES FOR CORE DAMAGE PREVENTION

- Rugged, high margin ALWR design (e.g., 15% thermal margin, 600 °Th, larger pressurizer); more tolerant to upset conditions
- Decay heat removal by:
  - improved EFW when steam generators are available
  - improved RHR, once partial cooldown achieved, during normal and safe cold shutdowns
  - SDVS/SIS (in bleed-and-feed mode) if steam generators not available
- Core coolant inventory control and diverse reactivity control by SIS
- RCS gas venting and depressurization by SDVS
- Improved AC and DC power system reliability

WRSRM

EPRI ALWR Program

EXAMPLES OF BWR FEATURES FOR CORE DAMAGE PREVENTION

- Rugged, high margin ALWR design, less vulnerable to upset conditions (e.g., 15% thermal margin, no recirc loops, improved CRDM)
- Decay heat removed by:
  - steam bypass to main condenser when main steam lines are available
  - decay heat removal heat exchangers once partial cooldown achieved, during normal and safe cold shutdowns
  - *safety/relief valves and decay heat removal heat exchangers operating on suppression pool if main steam lines not available*
- Core coolant inventory control by high pressure injection, reactor core inventory control, or automatic depressurization and decay heat removal flooding
- Reactor coolant system pressure control by safety/relief valves

WRSRM

EPRI ALWR Program

EXAMPLES OF FEATURES FOR MITIGATION

PWR

- Minimize fission product leakage (steel containment with annulus)
- Containment cooling and in-containment fission product control by CSS
- Cavity flooding capability from IRWST

BWR:

- Suppression pool retention of fission products
- Containment spray to reduce pressure and sweep fission products to suppression pool
- Drywell flooding capability

WRSRM

EPRI ALWR Program

ALWR PROGRAM - WHAT'S AHEAD?

Evolutionary Plant

- Complete Requirements Document Chapters 6-13
- Interact with NRC staff in review, comment and resolution of safety issues
- Proceed with integration and rollup phase; develop complete Requirements Document endorsed by NRC SER

Passive Plant

- Develop parallel, 13 chapter Requirements Document and design concepts for passive plant
- Interact with NRC in establishing regulatory foundation for passive safety concept; review Passive Plant Requirements chapters
- Plan for detailed design and NRC certification in 1990-1994

EPRI ALWR Program

U.S. ADVANCED LIGHT WATER REACTOR PROGRAM

OVERALL OBJECTIVE

N. KLUG
U. S. DEPARTMENT OF ENERGY

PERFORM COORDINATED PROGRAMS OF THE NUCLEAR INDUSTRY AND DOE TO INSURE THE AVAILABILITY OF LICENSED, IMPROVED AND SIMPLIFIED LIGHT WATER REACTOR STANDARD PLANT DESIGNS THAT MAY BE ORDERED IN THE 1990'S TO HELP MEET THE U.S. ELECTRICAL POWER DEMAND

U.S. ADVANCED LIGHT WATER REACTOR PROGRAM

PLANS TO MEET PROGRAM OBJECTIVES

CONDUCT COORDINATED PROGRAMS OF UTILITY INDUSTRY, REACTOR SUPPLIERS, DOE, AND NRC

- LICENSE ALWR STANDARD LARGE PLANT DESIGNS MEETING UTILITY REQUIREMENTS AND BASED ON LATEST TECHNOLOGY BY 1991

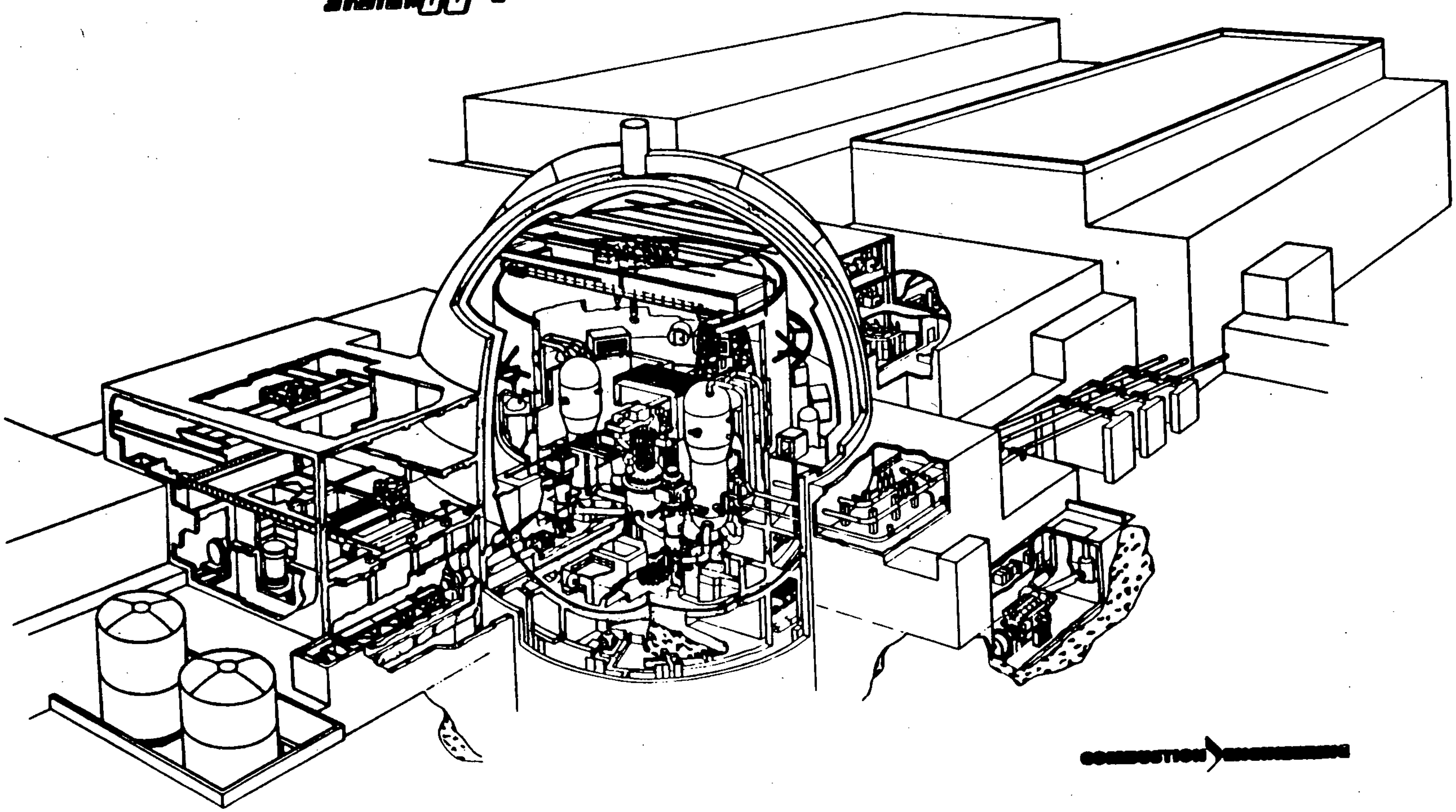
- DEVELOP SIMPLIFIED MID-SIZE PLANT DESIGNS, USING MAINLY PASSIVE SAFETY FEATURES:
  - DETERMINE TECHNICAL AND ECONOMIC ADVANTAGES BY 1989
  - IF FAVORABLE, LICENSE THESE DESIGNS BY 1995

- ESTABLISH TECHNICAL BASIS TO EXTEND LIFETIME OF LWR OPERATING PLANTS AND TO OBTAIN LICENSE RENEWALS BY NRC. SUPPORT FIRST RENEWAL APPLICATION IN EARLY 1990'S

DESIGN CERTIFICATION PROGRAM

COMBUSTION ENGINEERING SYSTEM 80+

GENERAL ELECTRIC ABWR



SYSTEM 80+™

COMBUSTION ENGINEERING

SYSTEM 80+ DESIGN CERTIFICATION PROGRAM

- Use System 80/CESSAR as Starting Point

    - Proven, Standard Design
    - Shown to Meet Current NRC Requirements
    - Complete Design Detail Available
    - Focuses Attention on Design Improvements and New NRC Requirements

- Address EPRI ALWR Requirements Document

- Address NRC Severe Accident Policy:

    - Current regulations
    - Probabilistic Risk Assessment and Evaluation
    - Resolve NRC's Unresolved Safety Issues
    - Evaluate Degraded Core Issues

- Apply for Design Certification under new NRC Standardization Policy.

MAJOR SYSTEM 80+ DESIGN ENHANCEMENTS

- GREATER PLANT MARGINS
- LONGER OPERATOR RESPONSE TIMES
- DESIGNED TO INCLUDE CONSIDERATION OF SEVERE ACCIDENTS
- GREATER REDUNDANCY IN SAFETY SYSTEMS
- SEPARATION OF SAFETY AND NON-SAFETY FUNCTIONS
- IMPROVED MAN-MACHINE INTERFACE
- ADDITIONAL SAFETY SYSTEM
- SIMPLIFICATION OF DESIGN
- IMPROVED OPERABILITY AND MAINTAINABILITY
- SCOPE OF DESIGN EXPANDED TO INCLUDE SYSTEMS MOST IMPORTANT TO SAFETY
- INCREASED PLANT AND COMPONENT LIFETIMES

SYSTEM 80+

DESIGN CERTIFICATION PROGRAM

EXPANDED SCOPE

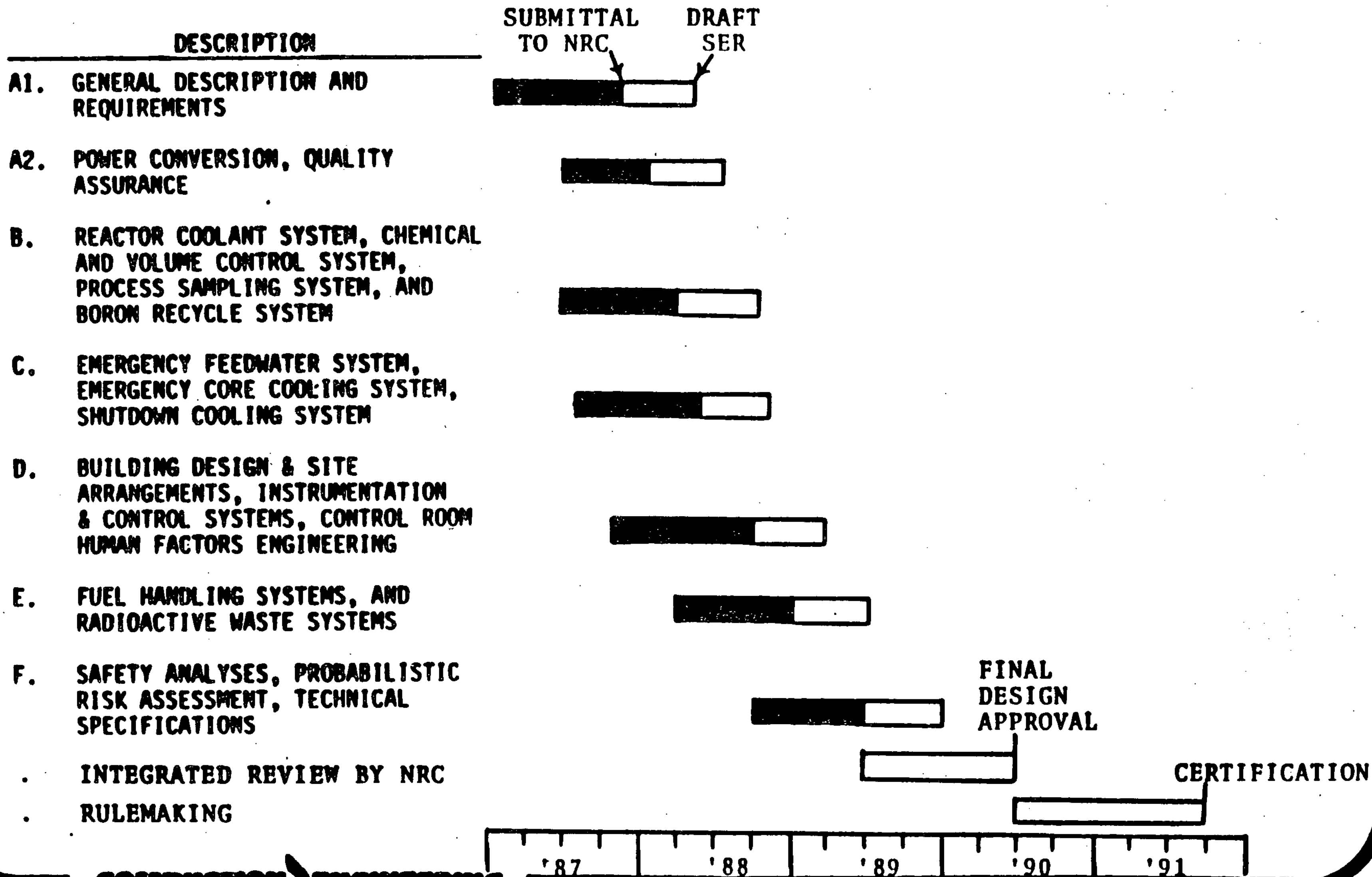
- o Current System 80 Design Includes Nuclear Steam Supply System (NSSS)

- o System 80+ Design Includes

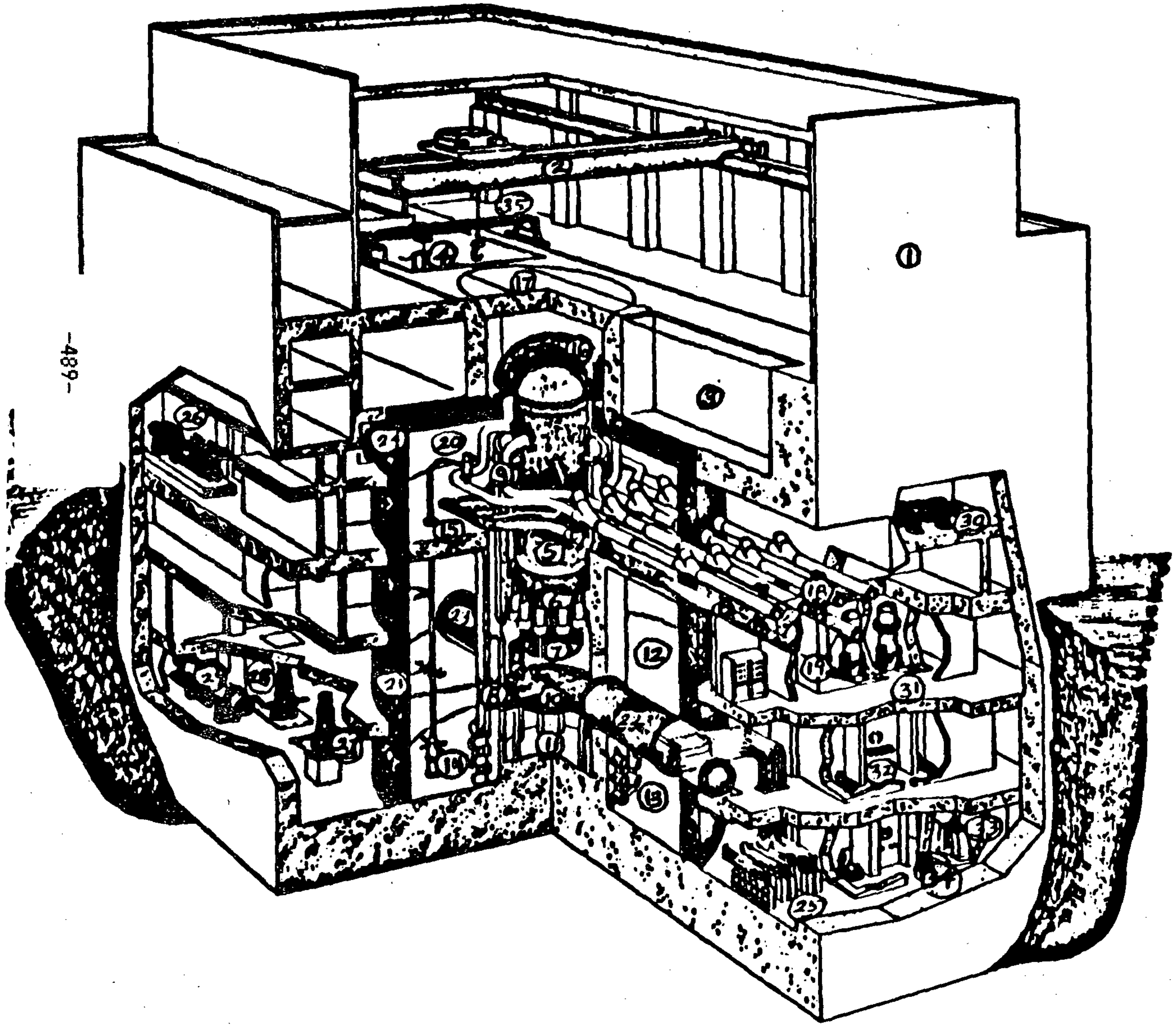
  - NSSS
  - Containment
  - Emergency Feedwater System
  - Advanced Control Center (NUPLEX 80+)
  - Standardized Functional Descriptions for Remainder of Plant

SYSTEM 80+

SYSTEM 80+
CERTIFICATION SCHEDULE



COMBUSTION ENGINEERING



ABWR
(Advanced Boiling Water Reactor)

REACTOR BUILDING

1 REACTOR BUILDING
2 BRIDGE CRANE
3 STEAM DRYER AND SEPARATOR STORAGE POOL
4 SPENT FUEL STORAGE POOL
5 REACTOR PRESSURE VESSEL
6 REACTOR INTERNAL PUMPS
7 FINE MOTION CONTROL ROD DRIVES
8 REACTOR PEDESTAL
9 REACTOR SHIELD WALL
10 LOWER DRYWELL EQUIPMENT PLATFORM
11 LOWER DRYWELL
12 SUPPRESSION POOL
13 HORIZONTAL VENTS
14 SRV QUENCHERS
15 UPPER DRYWELL
16 DRYWELL HEAD
17 SHIELD BLOCKS
18 MAIN STEAM LINES
19 FEEDWATER LINES
20 SAFETY/RELIEF VALVES
21 PRIMARY CONTAINMENT VESSEL
22 LOWER DRYWELL PERSONNEL LOCK
23 LOWER DRYWELL EQUIPMENT HATCH
24 UPPER DRYWELL EQUIPMENT HATCH
25 HYDRAULIC CONTROL UNITS
26 DIESEL GENERATOR
27 HPCS- PUMP
28 RHR- PUMP
29 RHR- HEAT EXCHANGER
30 FPC- HEAT EXCHANGER
31 RWCU- FILTER DEMINERALIZER
32 RWCU- HOLDING PUMP AND OPERATION ROOM
33 RWCU- PUMPS
34 RWCU/SPCU- BACKWASH PUMP AND OPERATION ROOM
35 REFUELING PLATFORM

ABWR CERTIFICATION PROGRAM TASKS

- O LICENSING BASIS
  - DEVELOP ACCEPTANCE BASES FOR REVIEW
  - ESTABLISH REVIEW PROCEDURES, SCHEDULES AND INTERFACES

- O PREPARATION AND SUBMITTAL OF SSAR
  - PREPARE AND SUBMIT SSAR
  - RESPOND TO NRC QUESTIONS
  - PARTICIPATE IN ACRS MEETING
  - OBTAIN FDA

- O DESIGN CERTIFICATION
  - PARTICIPATE IN RULEMAKING PROCEEDING
  - OBTAIN CERTIFICATION

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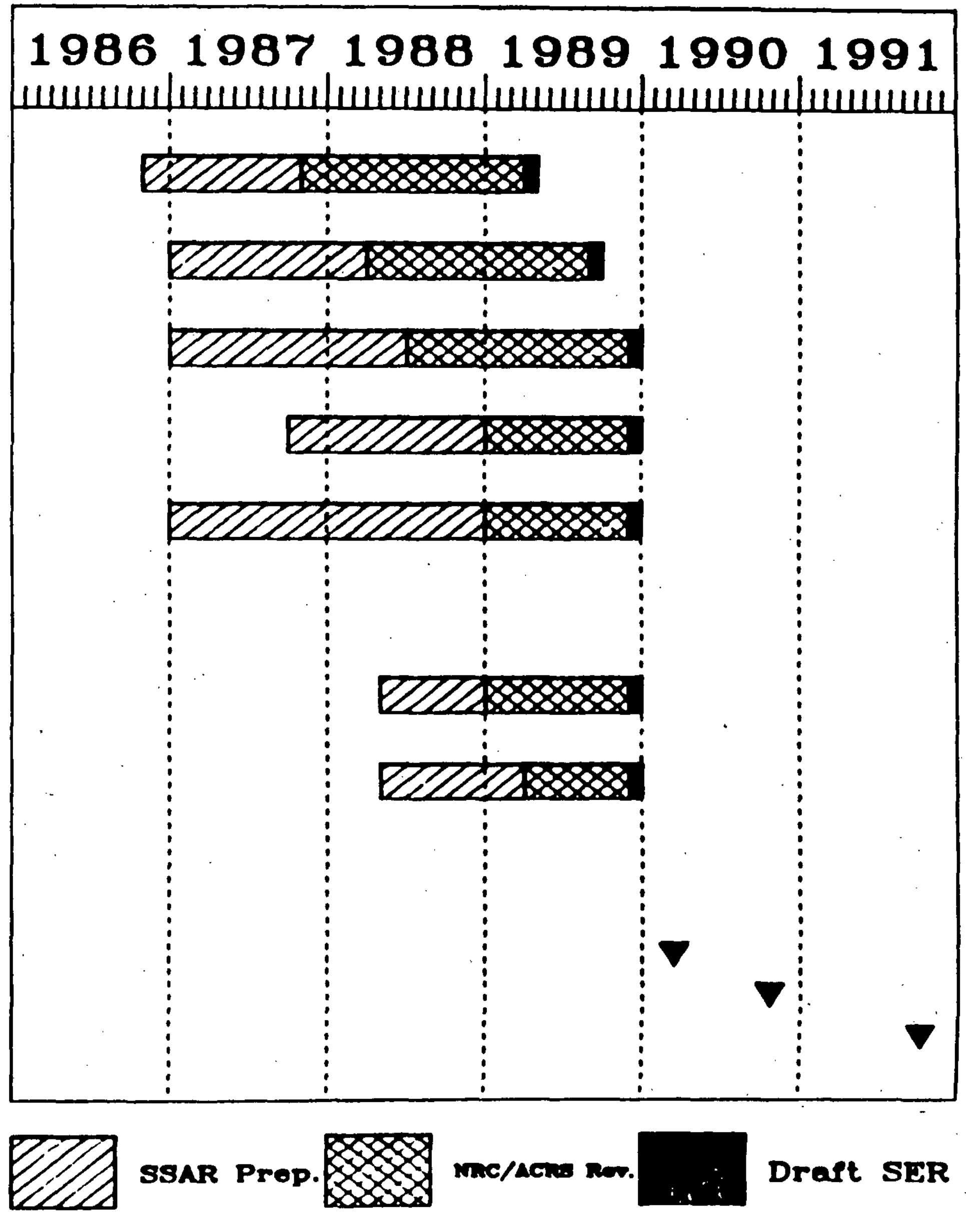
10/12/88

ABWR Certification Program
Scope and Schedule



Nuclear Island

- Reactor & Safety Systems
  - Chapters 4, 5, 6 & 15
- Plant Arrangement
  - Chapters 1, 2 & 3
- I&C, Auxiliary Systems & QA
  - Ch's 7-9, 11-14, 17
- Tech Specs & Emerg. Proc.
  - Chapters 16 & 18
- Severe Accidents
  - Chapter 19

Remainder of Plant

- Turbine Island
  - Ch. 10, Parts of other Chs
- Radwaste Facility
  - Ch. 11, Parts of other Chs

Key Milestones

- Final SER Issued
- FDA Issued
- Certification Issued

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SEVERE ACCIDENTS

- O US-NRC POLICY REQUIRES FUTURE PLANTS BE SAFER THAN CURRENT PLANTS
  - BALANCE BETWEEN PREVENTION AND MITIGATION
  - SEVERE ACCIDENT CONSEQUENCE ANALYSIS REQUIRED
    - -- CONTAINMENT FAILURES MODES KEY
  - SEVERE ACCIDENT RULEMAKING AHEAD

- O ACRS REQUIRES FUTURE PLANTS ADDRESS
  - CONTAINMENT MITIGATION OF SEVERE ACCIDENTS
  - PROTECTION AGAINST SABOTAGE
  - STATION BLACKOUT
  - EXTERNAL EVENTS
  - CONTROL ROOM PROTECTION FOR SEVERE ACCIDENTS

- O INCREASED EUROPEAN ATTENTION TO SEVERE ACCIDENT ISSUES
  - SWEDEN/GERMANY/FRANCE ADOPTING FILTERED VENTS
  - ASEA BWR/90 ADOPTING CORE CATCHERS
  - FINLAND ADOPTING CONTAINMENT FLOODING SCHEMES

- O EPRI SEVERE ACCIDENT GOALS
  - CAPABILITY TO PREVENT/MITIGATE SEVERE ACCIDENTS
    - -- $<10^{-5}$/YR CDF (INTERNAL/EXTERNAL)
    - -- <25 REM FOR EVENTS $>10^{-6}$/YR

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10/12/88

STATUS OF CERTIFICATION PROGRAM

LICENSING REVIEW BASES

- O ISSUED BY THE NRC STAFF 8/7/87
  - ACCEPTANCE CRITERIA DEFINED
  - ALLOWS FOR NEW REQUIREMENTS THAT HAVE BEEN PROMULGATED BY THE NRC

SSAR CHAPTERS SUBMITTED

O	REACTORS & SAFETY SYSTEMS: CHAPTERS 4,5,6 AND 15	9/29/87
O	PLANT ARRANGEMENT & CRITERIA: CHAPTERS 1,2 AND 3	3/30/88
O	I&C, AUXILIARY SYSTEM AND QA: CHAPTERS 7-9, 11*-14 & 17	6/30/88

NRC REVIEW

O	FIRST GROUP OF CHAPTER 4-6 & 15 NRC QUESTIONS RESPONDED BY BY GE	4/29/88
O	SECOND GROUP OF CHAPTER 4-6 & 15 NRC QUESTIONS RESPONDED TO BY GE	9/14/88

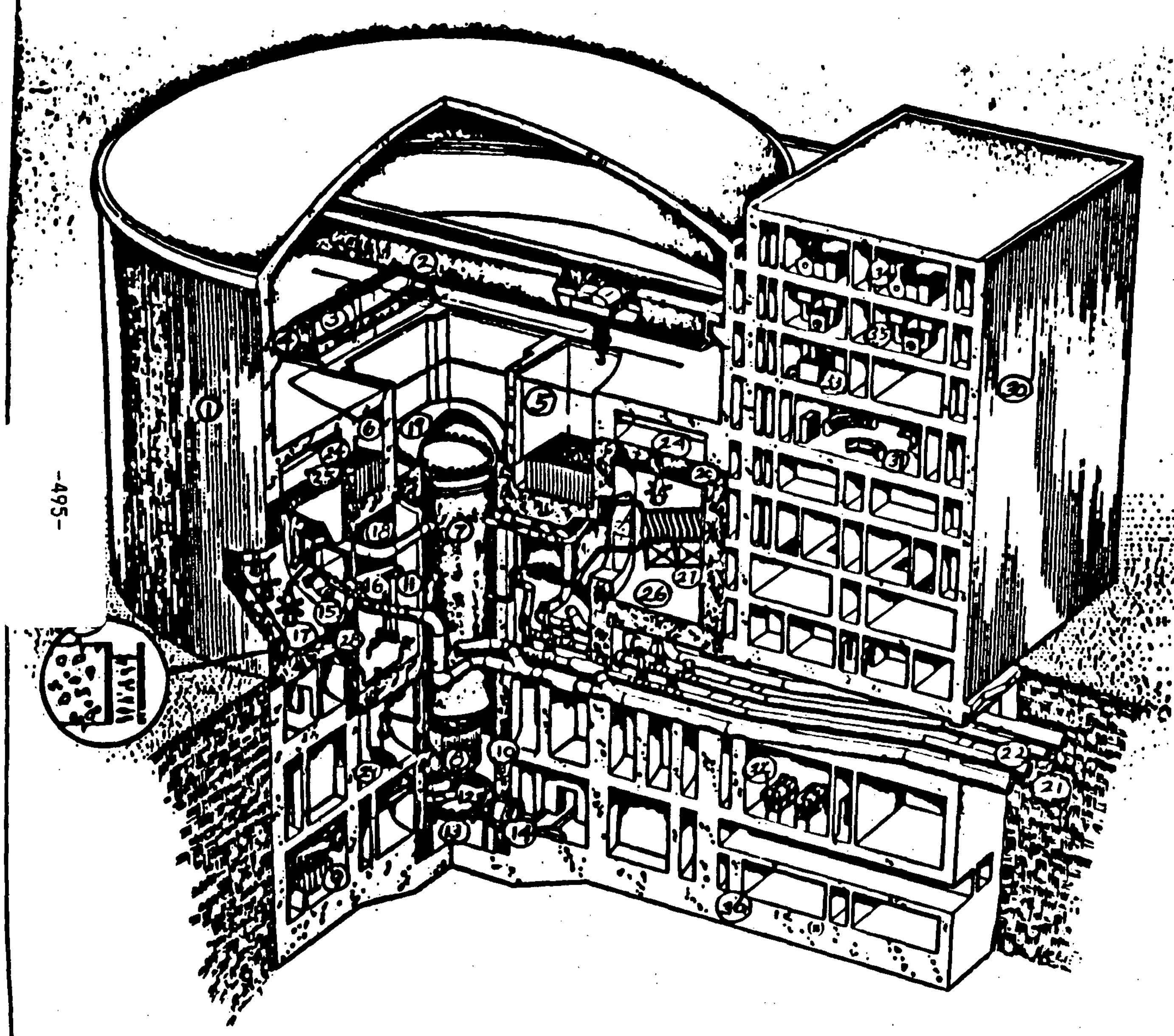
*NUCLEAR ISLAND PORTION OF CHAPTER 11.

SIMPLIFIED MID-SIZE (600 MWe) PLANT DEVELOPMENT

DEVELOP CONCEPTUAL DESIGNS, CONFIGURATIONS, ARRANGEMENTS, CONSTRUCTION METHODS/PLANS, AND PROOF TEST KEY DESIGN FEATURES:

- GENERAL ELECTRIC ASBWR — DOE CURRENTLY SUPPORTING:
  - GRAVITY DRIVEN COOLING SYSTEM TESTING
  - STEAM INJECTOR SYSTEM TESTING
  - DEPRESSURIZATION VALVE DEVELOPMENT/TESTING
  - ADVANCED CONTAINMENT DESIGN
  - CONSTRUCTION PLAN

- WESTINGHOUSE AP600 — DOE CURRENTLY SUPPORTING:
  - REACTOR COOLANT SYSTEM CONFIGURATION
  - PLANT SAFEGUARDS AND AUXILIARY SYSTEMS
  - PLANT ARRANGEMENTS AND CONSTRUCTION



ASBWR

Advanced Simplified Boiling Water Reactor

REACTOR ISLAND

1 REACTOR BUILDING
2 BRIDGE CRANE
3 REFUELING BRIDGE
4 STEAM DRYER AND SEPARATOR STORAGE POOL
5 SPENT FUEL STORAGE POOL
6 NEW FUEL STORAGE
7 REACTOR PRESSURE VESSEL
8 FINE MOTION CONTROL ROD DRIVES
9 FMCRD HYDRAULIC UNITS
10 REACTOR PEDESTAL
11 REACTOR SHIELD WALL
12 EQUIPMENT PLATFORM
13 LOWER DRYWELL
14 EQUIPMENT HATCH
15 HORIZONTAL VENTS
16 DEPRESSURIZATION (DPV) AND SAFETY (SV) VALVES
17 DPV AND SV QUENCHERS
18 UPPER DRYWELL
19 DRYWELL HEAD
20 PRIMARY CONTAINMENT VESSEL
21 MAIN STEAM LINES
22 FEEDWATER LINES
23 PASSIVE CONTAINMENT COOLING (WATERWALL)
24 WATERWALL REFILL POOL
25 WATERWALL REFILL LINE
26 SUPPRESSION POOL
27 ISOLATION CONDENSER
28 GRAVITY DRIVEN CORE COOLING INLET
29 STEAM INJECTOR
30 SERVICE BUILDING
31 CONTROL ROOM
32 EMERGENCY BATTERIES
33 CONTROL AREA EMERGENCY HABITABILITY SYSTEM
34 REACTOR AND TURBINE ISLAND HVAC SUPPLY
35 REACTOR AND TURBINE ISLAND HVAC EXHAUST
36 COMMON BASEMAT

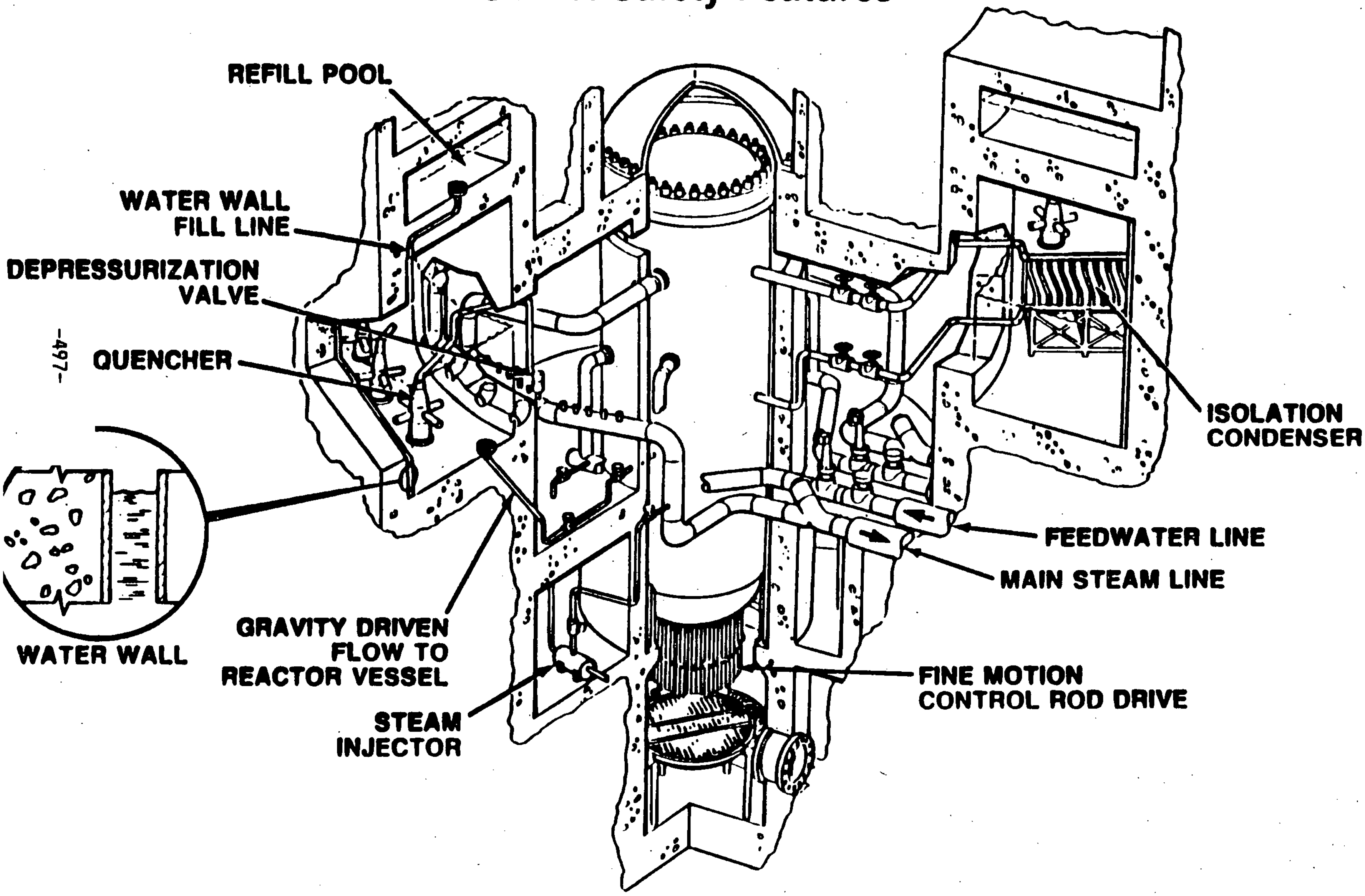
GENERAL ELECTRIC

Summary of Major New Features

FEATURES	POTENTIAL ADVANTAGES
Natural Circulation	Lower Costs, Simpler System
Isolation Condenser	Overpressure Protection with No Fluid Discharge
Steam Injector	Passive Protection for Small Break
Gravity Driven Core Cooling (GDCS)	Passive Safety, Reduced Cost of Safety Equipment
Passive Containment Cooling (PCCS)	No Short Term Operator Action
Tandem Compound Two Flow Turbine	Reduce Cost
Simplified Turbine Island Features	Reduce Cost
Passive HVAC for Control Areas	With GDCS, PCCS, Allows Elimination of Safety Grade Diesels

Description of SBWR Features

SBWR Safety Features



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GENERAL ELECTRIC

70357-24

SBWR PROGRAM
DEPARTMENT OF ENERGY

PROGRAM	MAJOR BENEFITS
o GRAVITY DRIVEN CORE COOLING	o PASSIVE SAFETY o EQUIPMENT ELIMINATION
o DEPRESSURIZATION VALVE	o FAIL OPEN DEPRESSURIZATION VALVE FOR GRAVITY COOLING
o PASSIVE CONTAINMENT SYSTEM	o ZERO POST SEVERE ACCIDENT RELEASE
o STEAM INJECTOR SYSTEM	o INCREASED SAFETY/ RELIABILITY (PASSIVE)
o CONSTRUCTION PLANNING	o DEMONSTRATE MINIMUM AND CREDIBLE CONSTRUCTION TIMES

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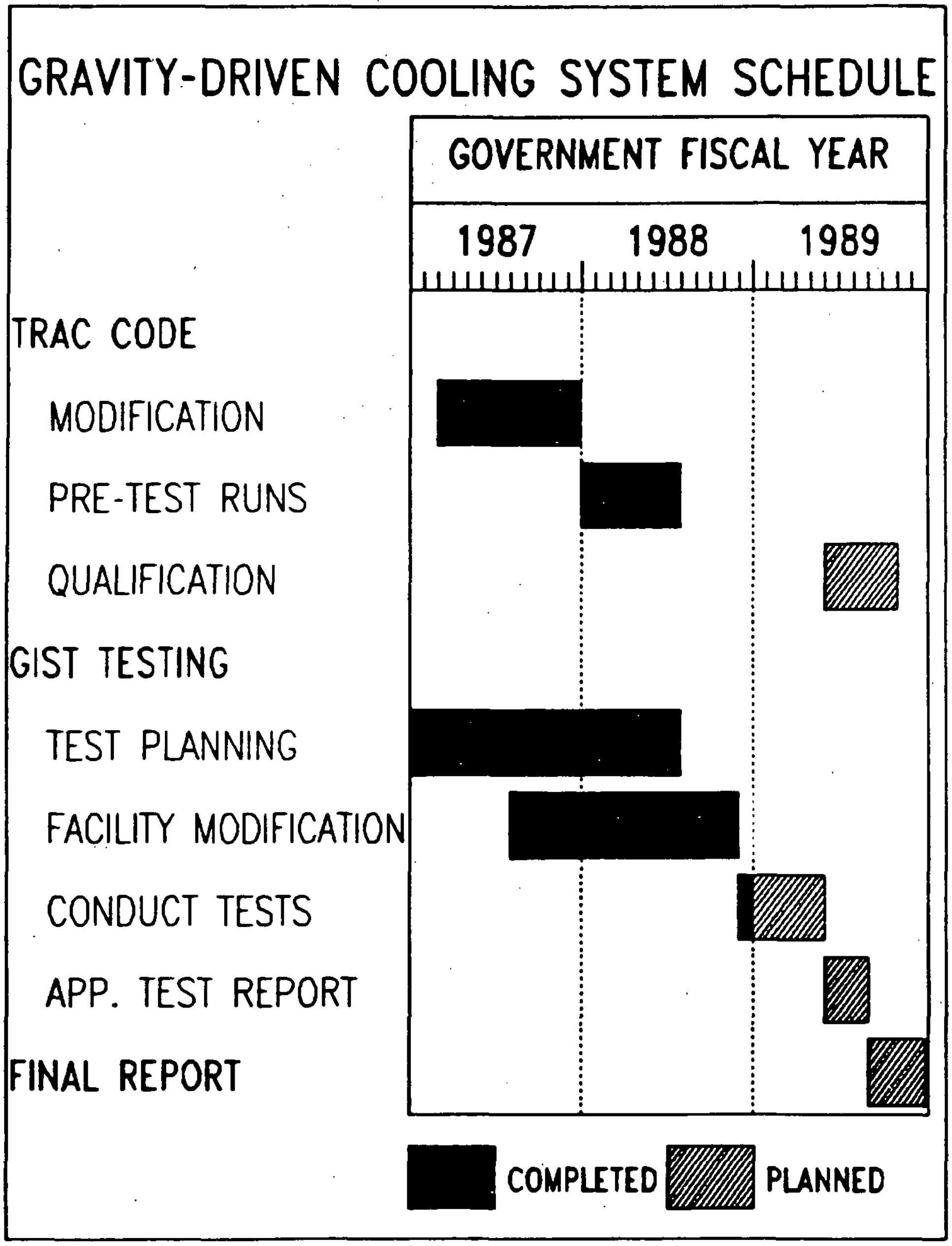
GRAVITY-DRIVEN COOLING SYSTEM

TASK STRUCTURE & SCHEDULE



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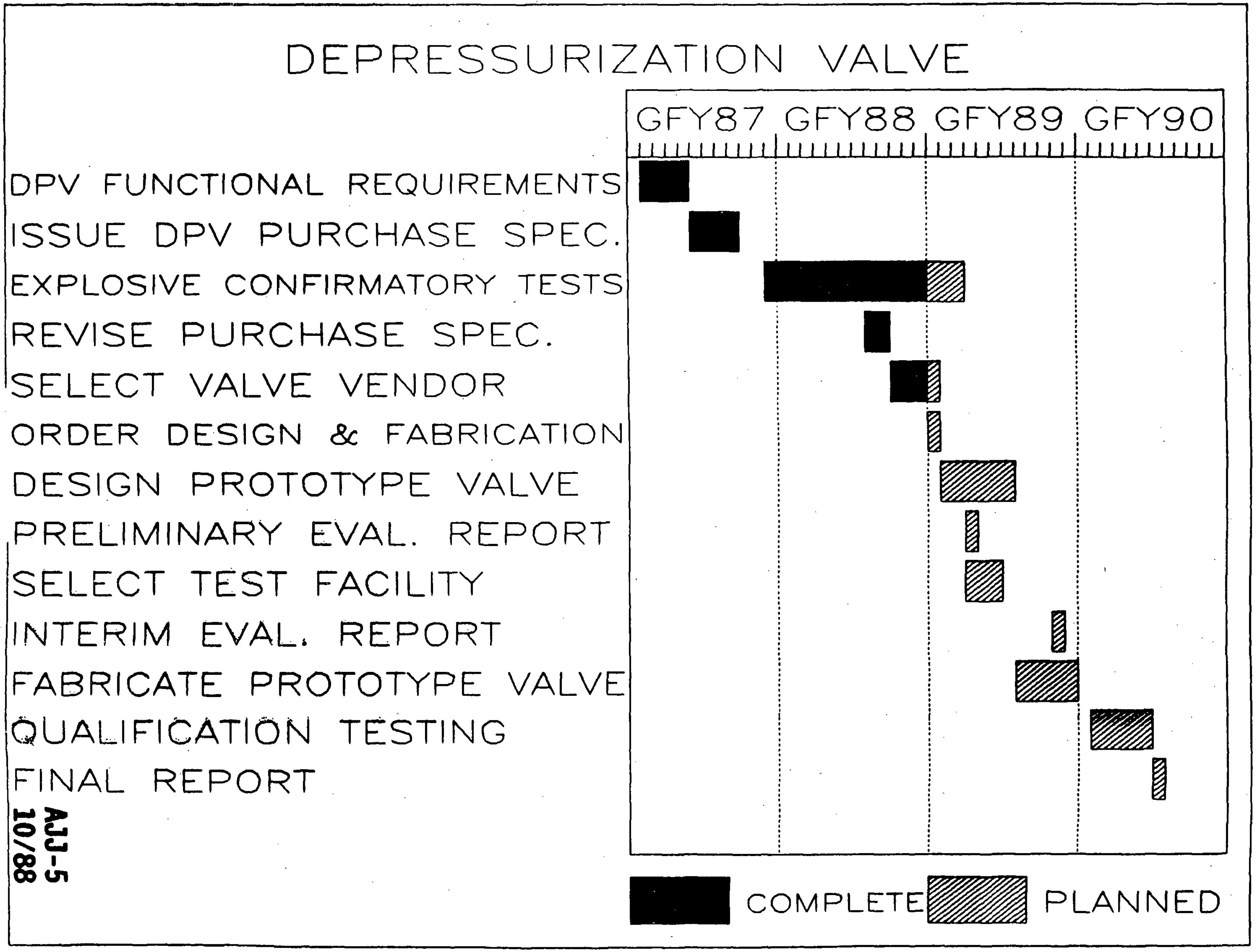
10/88


DEPRESSURIZATION VALVE
GFY87 GFY88 GFY89 GFY90
DPV FUNCTIONAL REQUIREMENTS
ISSUE DPV PURCHASE SPEC.
EXPLOSIVE CONFIRMATORY TESTS
REVISE PURCHASE SPEC.
SELECT VALVE VENDOR
ORDER DESIGN & FABRICATION
DESIGN PROTOTYPE VALVE
PRELIMINARY EVAL. REPORT
SELECT TEST FACILITY
INTERIM EVAL. REPORT
FABRICATE PROTOTYPE VALVE
QUALIFICATION TESTING
FINAL REPORT
COMPLETE
PLANNED
AJJ-5
10/88

SBWR DESIGN OVERVIEW

o NATURAL CIRCULATION	- CONFIRMED - DODEWAARD RECORD EXCELLENT - FORCED CIRCULATION BWRs TESTED AT 50% POWER
o GRAVITY DRIVEN COOLING	- CONFIRMED BY TEST
o STEAM INJECTOR SYSTEM	- SMALL SCALE TEST SUCCESS AT HIGH PRESSURE
o PASSIVE CONTAINMENT COOLING	- HEAT TRANSFER BASICS ESTABLISHED
o DEPRESSURIZATION VALVE	- CHARGE TESTING SUCCESSFUL
o CONSTRUCTION PLAN	- 30 MONTH CONSTRUCTION SCHEDULE POSSIBLE
o DESIGN OPTIMIZATION	- GOOD PROGRESS - GE/HITACHI/TOSHIBA/KEMA/ ANSALDO/BECHTEL/MIT/ BERKELEY - COMPLETE NOVEMBER 1988

CONCEPT ESTABLISHED
CONFIRMATION PROGRESS EXCELLENT
OPTIMIZATION UNDERWAY

M:RJM310053:TJM

ADVANCED CONTAINMENT DESIGN

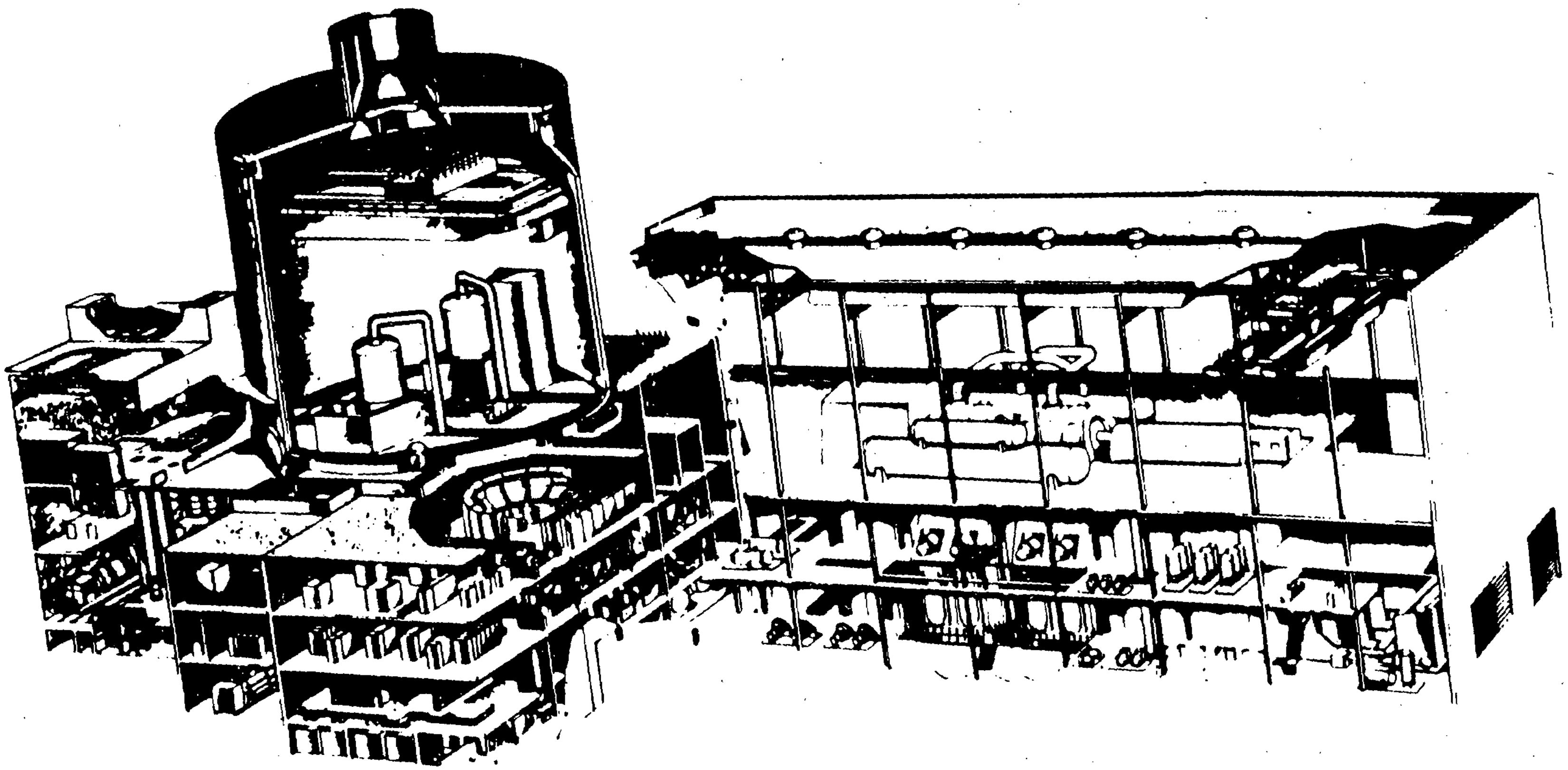
MAJOR 1988 ACCOMPLISHMENTS

- COMPLETED DESIGN/EVALUATION OF 3 PASSIVE CONTAINMENT COOLING SYSTEM (PCCS) CONCEPTS
  - INTERIOR WATER WALL MODULE ASSEMBLY
  - MODULAR COOLER
  - STEEL CONTAINMENT WITH AIR COOLING
- COMPLETED PRELIMINARY STRUCTURAL DESIGN OF REFERENCE REINFORCED CONCRETE CONTAINMENT
- OUTLINED A PCCS DEMONSTRATION TEST PROGRAM
- ISSUED ADVANCED CONTAINMENT DESIGN FINAL REPORT

LCH-6
10/88

Westinghouse

AP600



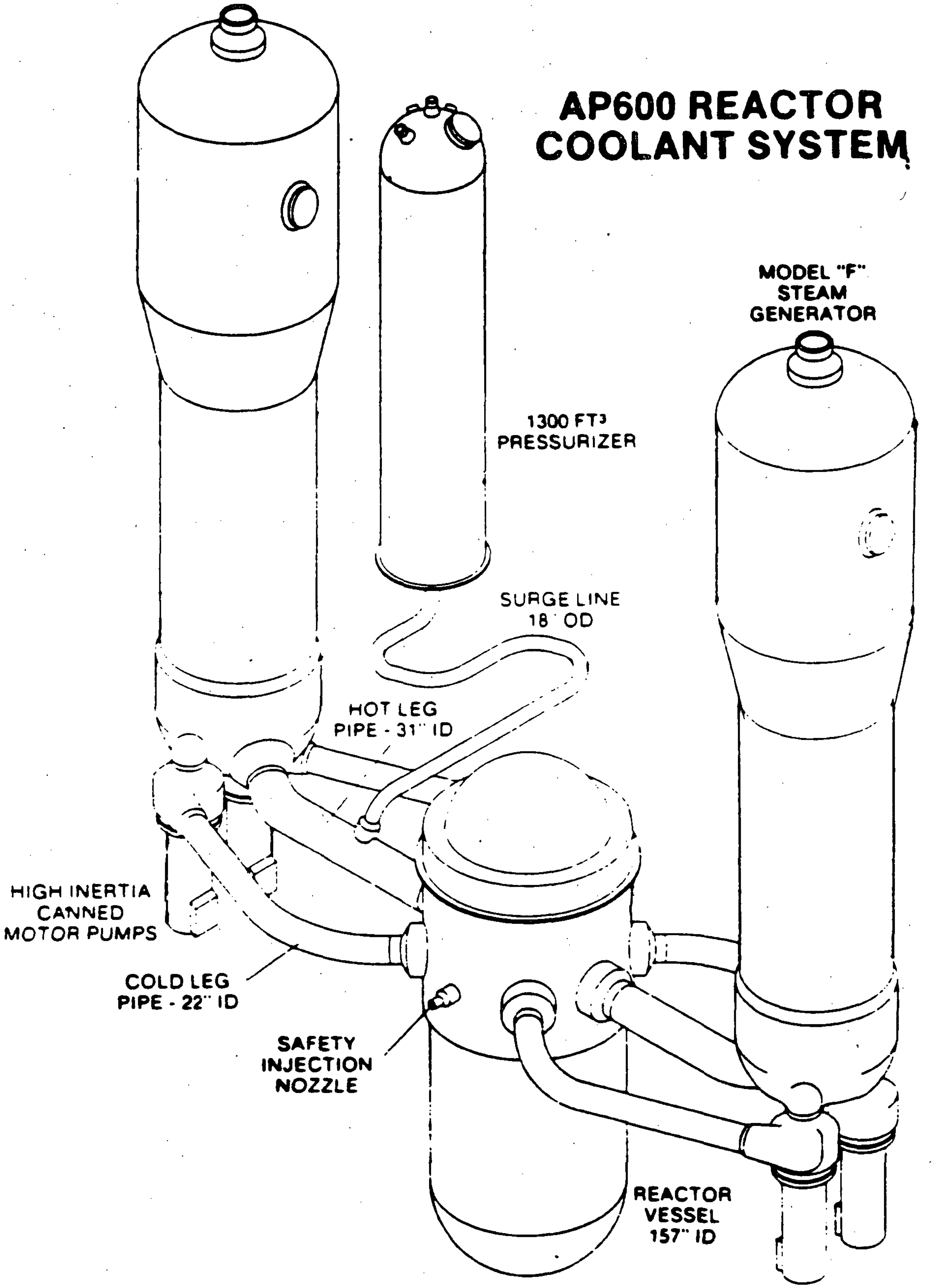
Mid-Size PWR Development

AP600

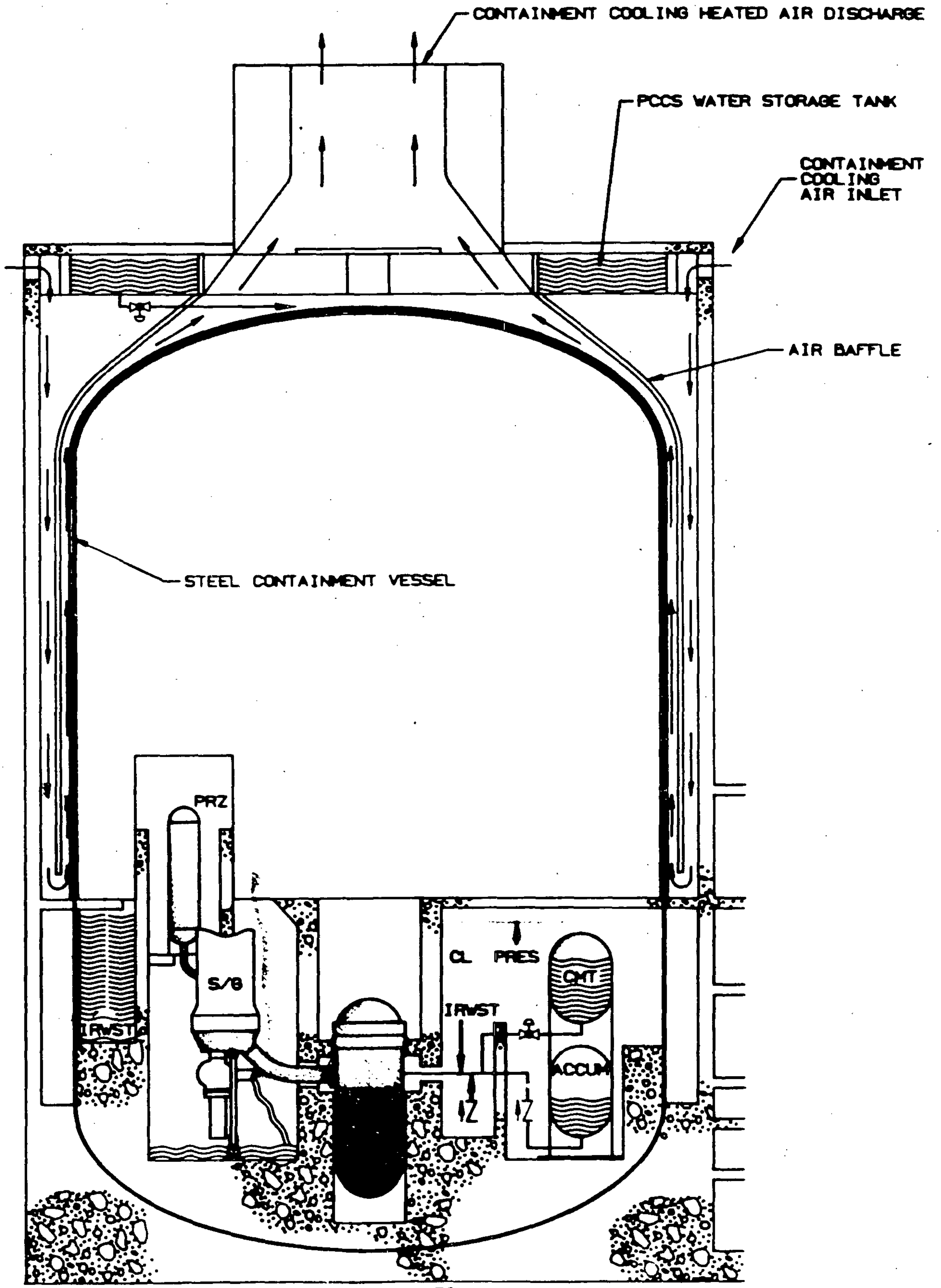
Key Design Features

- Simplified reactor coolant loop with canned motor pumps
- Low power density core
- Passive safety systems
- Simplified systems throughout the plant
- Plant arrangement based on integrated consideration for construction, operation, maintenance, safety, and capital cost
- Use of modular construction

1209 D24398.004


AP600 REACTOR COOLANT SYSTEM
MODEL "F" STEAM GENERATOR
1300 FT3 PRESSURIZER
SURGE LINE 18" OD
HOT LEG PIPE - 31" ID
HIGH INERTIA CANNED MOTOR PUMPS
COLD LEG PIPE - 22" ID
SAFETY INJECTION NOZZLE
REACTOR VESSEL 157" ID

AP600 PASSIVE SAFETY INJECTION
LOCA INITIATION



WESTINGHOUSE AP-600

DOE PROGRAM TASKS

- **REACTOR COOLANT SYSTEM CONFIGURATION DEVELOPMENT:**
  - REACTOR COOLANT SYSTEM DESIGN
  - PUMP (HIGH) INERTIA SYSTEM
  - ALTERNATE IN-CORE INSTRUMENT (ICI) SYSTEM
- **PLANT SAFETY AND AUXILIARY SYSTEMS DEVELOPMENT:**
  - SIMPLIFIED SYSTEMS DESIGNS
  - PLANT TRANSIENT & ACCIDENT ANALYSES, AND RISK EVALUATION
- **PLANT ARRANGEMENT AND CONSTRUCTION METHODS:**
  - OPTIMIZE PLANT ARRANGEMENT
  - MODULAR CONSTRUCTION METHODS
  - BUILDING & MODULE DESIGN
  - CONSTRUCTION PLAN & COST ESTIMATE

AP600 DESIGN DEVELOPMENT STATUS

A SUBSTANTIAL AMOUNT OF DESIGN, ANALYSIS AND TEST INFORMATION IS IN HAND:

- ENGINEERING FLOW DIAGRAMS/SYSTEM DESCRIPTIONS FOR ALL MAJOR SYSTEMS - 28 NEW SYSTEM DESIGNS/42 BASED ON MODIFIED EXISTING DESIGNS
- RCS DESIGN - BASIC STUDIES COMPLETED ON CORE, VESSEL, INTERNALS, PIPING, STEAM GENERATOR, CANNED MOTOR PUMP, PRESSURIZER
- COMPLETE SET OF TRANSIENT AND ACCIDENT DESIGN BASIS ANALYSES
- INITIAL CORE MELT FREQUENCY AND SEVERE ACCIDENT EVALUATIONS
- TESTING - PASSIVE CONTAINMENT COOLING TESTS PARTIALLY COMPLETE, HIGH INERTIA PUMP ROTOR TEST AND HYDROBALL ICIS TEST IN WORK
- PLANT GENERAL ARRANGEMENT ESTABLISHED, EQUIPMENT LOCATED, PRELIMINARY ROUTING OF PIPE, CABLE TRAY, DUCT
- MODULAR CONSTRUCTION APPROACH ESTABLISHED - DETAIL DRAWINGS OF REPRESENTATIVE EQUIPMENT MODULES AND CONTAINMENT INTERIOR STRUCTURAL MODULES
- PRELIMINARY PLANT COST ESTIMATE AND CONSTRUCTION SCHEDULE

AP600 - DESIGN SIMPLIFICATION

PLANT FEATURES	STD 2 LOOP	AP600
FUEL ASSEMBLIES	121	145
CRDM - SHUT / CONTROL	33	45
- GRAY	0	12
PRESSURIZER	1000 FT3	1300 FT3
RC PUMPS	2 SHAFT SEALED	4 CANNED
PUMPS - SAFETY	25	0
- NNS	188	139
HVAC FANS	52	27
HVAC FILTER UNITS	16	7
VALVES - NSSS (>2")	512	215
- BOP (>2")	2041	1530
PIPE - NSSS (>2")	44,300 FT	11,042 FT
- BOP (>2")	97,000 FT	67,000 FT
EVAPORATORS	2	0
DIESEL GENERATORS	2 (SC)	1 (NNS)
BLDG. VOL.- CONTAINMENT	2.7 MIL FT3	3.0 MIL FT3
- SEISMIC	6.7 MIL FT3	1.6 MIL FT3
- NON SEISMIC	6.2 MIL FT3	6.1 MIL FT3

AP600 CONSTRUCTION PLAN

I PLANT ARRANGEMENT CONSTRUCTABILITY

- o SMALLER, COMPACT NUCLEAR AND TURBINE ISLANDS
- o REDUCTION OF VOLUME AND QUANTITIES
- o FEWER MANHOURS AND SHORTER ACTIVITY DURATIONS
- o SHORTER CONSTRUCTION SCHEDULE
  - 36 MONTHS FROM FIRST CONCRETE TO FUEL LOAD

U.S. ADVANCED LIGHT WATER REACTOR PROGRAM

PRINCIPAL MILESTONES

LICENSING OF LARGE STANDARD PLANTS

• DESIGN CERTIFICATION OF G.E. ABWR AND C.E. SYSTEM 80+	1991

SIMPLIFIED MID-SIZE PLANT DEVELOPMENT

• SUBSTANTIAL CONCEPTUAL DESIGN INITIAL FEATURES TESTS COMPLETED TECHNICAL AND ECONOMIC ADVANTAGES	1989
• BASED ON FAVORABLE RESULTS, LICENSE ASBWR AND AP600 STANDARD DESIGNS	1995

PLANT LIFETIME IMPROVEMENT

• LEAD PLANT LICENSE RENEWAL BY NRC	1993

The Integral Fast Reactor

Yoon I. Chang
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

Abstract

The Integral Fast Reactor (IFR) is an innovative liquid metal reactor concept being developed at Argonne National Laboratory. It seeks to specifically exploit the inherent properties of liquid metal cooling and metallic fuel in a way that leads to substantial improvements in the characteristics of the complete reactor system. This paper describes the key features and potential advantages of the IFR concept, with emphasis on its safety characteristics.

The Integral Fast Reactor (IFR) is a generic reactor concept based on four technical features: (1) liquid sodium cooling, (2) pool-type reactor configuration, (3) metallic fuel, and (4) an integral fuel cycle, based on pyrometallurgical processing and injection-cast fuel fabrication, with the fuel cycle facility collocated with the reactor, if so desired. Much of the technology for the IFR is based on EBR-II. EBR-II was the first pool-type liquid metal reactor. Metallic fuel was successfully developed as the driver fuel in EBR-II. During 1964-1969, about 35,000 fuel pins were reprocessed and refabricated in the EBR-II Fuel Cycle Facility, which was based on an early pyroprocess with some characteristics similar to that now proposed for the IFR.

The IFR concept has a number of specific technical advantages that collectively satisfy all fundamental requirements demanded on the next generation reactor. Recent debates on the greenhouse effect reinforce the need to develop an advanced next generation reactor concept that can contribute significantly toward substituting the fossil-based energy generation. If nuclear is to make a significant contribution, breeding is a fundamental requirement, so that the uranium resources can be extended by two orders of magnitude, making nuclear essentially a renewable energy source. In addition to breeding, there are two other fundamental requirements that the next generation reactor should address. Safety and waste are two key factors that influence the public acceptance of nuclear power and, hence, determine the extent to which nuclear power contributes to meet the long-term energy substitution as well as future demand growth.

For the discussion of high-level waste management, it is convenient to categorize the nuclear waste constituents into two parts: fission products comprised of hundreds of various isotopes, and actinides comprised of uranium and the transuranic elements--neptunium, plutonium, americium, curium, etc. Fission products are produced by fissioning of heavy atoms, and transuranics are produced as a result of neutron capture reactions.

Most of fission products decay in relatively much shorter time periods than actinides. In the order of 200 years, the fission products decay to a sufficiently low level so that their radiological risk factor drops below the cancer risk level due to their original uranium ore. Actinides, on the other hand, have longer half-lifes and their radiological risk factor remains orders of magnitude higher than that due to fission products for millions of years. The relative radiological risk factors for fission products and actinides are presented in Figure 1 for the LWR spent fuel[1].

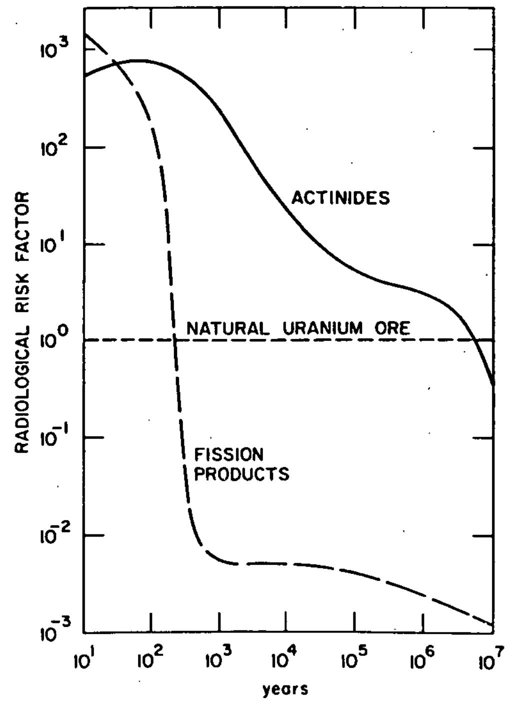


Figure 1. Relative Radiological Risk Factor of Fission Products and Actinides in the LWR Spent Fuel, Normalized to Their Original Uranium Ore (Data Source: Ref. 1).

There is a strong incentive to separate actinides and recycle them back into the reactor for in-situ burning. The benefit of the actinide recycling is in the fact that the effective lifetime of the nuclear waste is reduced from millions of years to a few hundred years. This would have an enormous impact on assuring the integrity of high-level waste over its lifetime and should ultimately be helpful in public acceptance of the nuclear power. However, even if the actinides are removed and the lifetime of the high-level waste is reduced to hundreds of years, this does not mean that actinide recycling could vitiate the need for a geological repository. A geological repository would still be necessary to store high-level wastes regardless of the actinide contents.

The IFR pyroprocessing has unique technical features that make the actinide separation more practical than it is in conventional PUREX processing. In the IFR process, most of the actinide elements accompany the plutonium product stream, and furthermore, the ability of pyrochemical process to separate rare earths from actinides, which is very difficult in the PUREX processing, is remarkable. The hard IFR neutron spectrum is better for actinide burning than that of any other reactor type. The prospects of the IFR concept for actinide recycling are excellent. Further research and development is needed to fully establish feasibility, but the main lines of the necessary development are easily defined and should be straightforward to carry out.

The IFR metallic fuel promises a higher degree of inherent safety than the conventional oxide fuel, and better or equal safety characteristics across the entire spectrum from normal behavior to postulated severe accidents. Although the metallic fuel melting temperature is much lower than that of oxide fuel, it is also much more difficult to raise the fuel temperature because of the high thermal conductivity (~20 W/m K for metal vs ~2 W/m K for oxide). As a result, operating margins in terms of power can, in fact, be greater for metal than for oxide cores. Typical metal core design parameters are presented in Table I. The TREAT experiments performed to date[2] indicate that the margin to fuel pin failure during transient overpower conditions is greater for metal than oxide fuel. However, it is in the inherent safety characteristics under the generic anticipated-transient-without-scram (ATWS) events, such as loss-of-flow without scram (LOFWS), loss-of-heat-sink without scram (LOHSWS), and transient overpower without scram (TOPWS), that the metallic fuel shows its greatest advantages over oxide fuel.

Table I. Typical Metal Core Design Parameters

Fuel Materials	U-Pu-10% Zr, U-10% Zr
Fuel Smear Density	75%
Pin Diameter	7.6 mm (0.3 in.)
Cladding Thickness	0.46 mm (0.018 in.)
Peak Linear Power	50 kW/m (15 kW/ft)
Peak Discharge Burnup	150 MWd/kg

In an LOFWS event, the coolant temperatures increase as flow reduces rapidly. The increased coolant temperature results in the thermal expansion of core assemblies, which provides a negative reactivity feedback and starts a power rundown. During this initial period, it is important to maintain a reasonable flow coastdown in order to avoid immediate sodium boiling. This requirement can be met with normal mechanical pump inertia, characterized by a flow halving time of the order of 5 seconds.

The characteristics of the negative reactivity feedback caused by the coolant temperature increase determines the reactor response. The most important factor differentiating the LOFWS and LOHSWS responses in metal and oxide fuels is the difference in stored Doppler reactivity between the two fuels. As the power is reduced, the stored Doppler reactivity comes back as a positive contribution tending to cancel the negative feedback due to the coolant temperature rise. The high thermal conductivity of the metallic fuel and consequent low fuel operating temperatures give a stored Doppler reactivity that is only a small fraction of overall negative reactivity feedback. As a result, the power is reduced rapidly. In contrast, oxide fuel has a much greater stored Doppler reactivity (primarily due to the higher fuel temperatures rather than the difference in the Doppler coefficient itself), and the power does not decrease rapidly during the LOFWS or LOHSWS event. And when the power has been reduced to decay power levels, in order to counter the stored Doppler reactivity, the coolant temperature maintains a much higher value in an oxide core. A typical comparison of LOFWS between the metal and oxide is illustrated in Figure 2. Both the LOFWS and LOHSWS accidents are perfectly benign in a properly designed IFR.

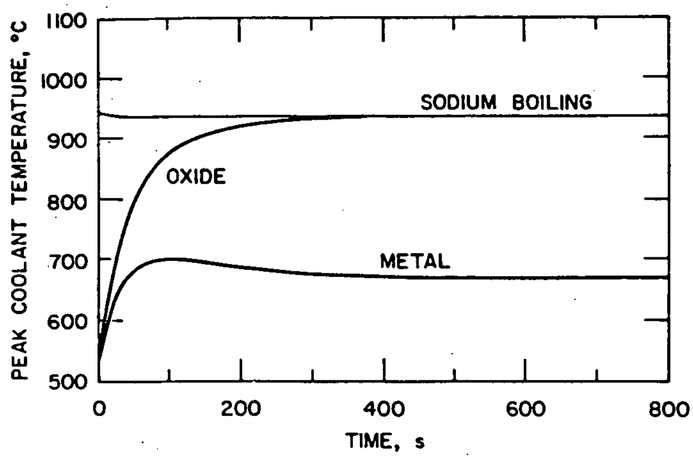


Figure 2. Loss-of-Flow Without Scram for Large Reactors (1350 MWe).

The inherent safety potential of the metallic fuel was demonstrated by two landmark tests conducted in EBR-II on April 3, 1986. The first test was loss-of-flow without scram and the other loss-of-heat-sink without scram. These tests demonstrated that the unique combination of the high heat conductivity of metallic fuel and the thermal inertia of the large sodium pool can shut the reactor down during these potentially very severe accident situations without depending on human intervention or operation of active, engineered components. The coolant temperature responses during these two tests are presented in Figures 3 and 4. More detailed data can be found in a collection of papers prepared for these tests[3]. The EBR-II tests demonstrated in a very concrete way what is possible with liquid metal cooling and metallic fuel in achieving wide-ranging inherently safe characteristics.

The superior neutronics performance characteristics of metallic fuel allows core designs with minimum burnup reactivity swing even for small modular core designs. Advantage can be taken of this in reducing the TOPWS

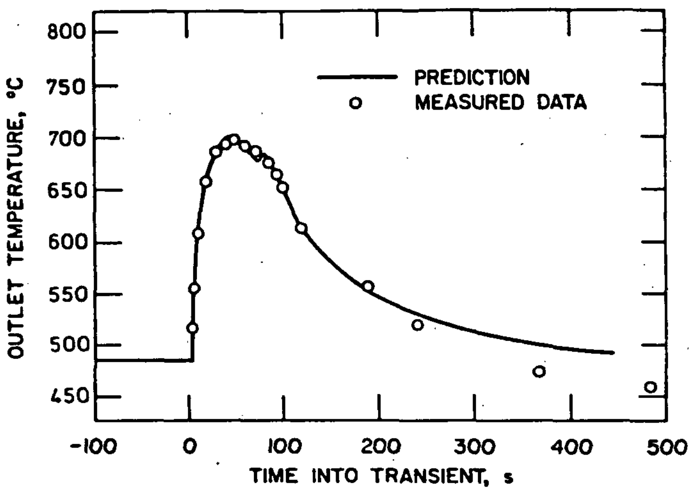


Figure 3. Loss-of-Flow Without Scram Test in EBR-II.

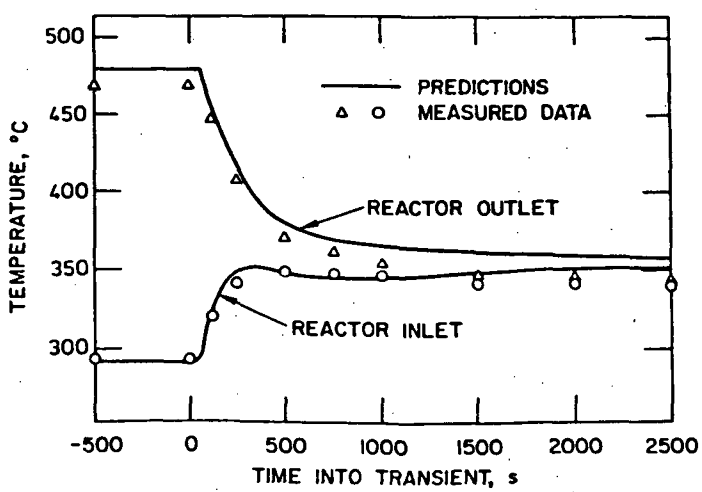


Figure 4. Loss-of-Heat-Sink Without Scram Test in EBR-II.

initiator caused by an unprotected control rod runout. In addition, TREAT tests performed to date have demonstrated, first, a larger margin to cladding failure threshold for the metallic fuel, and second, that fission gas driven axial expansion of fuel within the clad before failure provides an intrinsic and favorable negative reactivity feedback in the metal fuel that has no parallel in oxide. Thus, there are a number of factors that suggest that metallic cores can be designed for benign TOPWS responses.

The inherent safety characteristics of metallic fuel under generic ATWS events reduce the core disruption probability to an exceptionally low value. Furthermore, metallic fuel disruption characteristics are also superior to those of oxide fuel. Initial out-of-pile experiments indicate that no fuel-coolant-interaction (FCI) events occurred when molten fuel contacted flowing sodium. These results, along with physical arguments ruling out extremely high molten fuel temperatures, support the case for the exclusion of significant fuel coolant interactions. The absence of FCI events when molten fuel contacted sodium is in contrast to typical results with oxide fuel where FCI events are observed and, while not energetic, can void the channel of sodium. Also, out-of-pile tests showed that metallic fuel debris beds were characteristically in the form of large filaments and sheets, and, hence, are more coolable than oxide beds.

It is worth stressing again that the sharply improved performance characteristics of the metallic cores for the unprotected LOF, LOHS, and TOP events are directly traceable to the basic properties of the fuel, and not to engineered features of any kind. Designs must simply take advantage of these properties.

As discussed above, the IFR concept has a potential of satisfying all fundamental requirements for the next generation reactor--breeding, waste treatment and safety. Several aspects of the IFR concept require further proof, and development programs on each are underway at Argonne. The major areas are demonstration of the performance of the IFR U-Pu-Zr ternary alloy metallic fuel, development of the new pyroprocesses of electrorefining, and development of the new pyroprocesses based on electrorefining, and demonstration of the inherent safety characteristics. IFR development, which was initiated in the latter part of FY 1984, is proceeding rapidly. Results from experimental, analytical, design and hardware programs in all areas are accumulating daily and substantial progress has been made to date.

The key next step is to demonstrate the practicality of the entire fuel cycle using the EBR-II reactor and a refurbished EBR-II Fuel Cycle Facility. The EBR-II Fuel Cycle Facility, now called HFEF/S, has been decontaminated and is ready for the new equipment. As the necessary facilities are already in place, the total cost will be modest.

Modifications to the EBR-II complex will take IFR demonstration through the pilot plant stage. The crucial facilities are EBR-II (for tests and demonstration), TREAT (for transient, accident-simulation fuel tests), ZPPR (for the new metallic core neutronic properties), HFEF/N (for destructive fuel examinations), and HFEF/S (for fuel cycle demonstration). EBR-II is the natural prototype of the IFR. It was the first prototype of the pool concept. Gradual substitution of IFR fuel in EBR-II will lead to whole-core IFR-fueled operation. Modifications to the HFEF/S facility will equip the system with plant-scale metallic processing and fabrication modules. In this way, a complete prototype IFR can be operational in three years. EBR-II will then be in full operation as a complete prototype, with fuel at target burnup levels and fuel being processed, fabricated, and returned to the reactor.

References

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2. J. F. Marchaterre, J. E. Cahalan, R. H. Sevy, and A. E. Wright, "Safety Characteristics of the Integral Fast Reactor Concept," Proc. ASME Winter Meeting, 85-WA/NE-14, Miami Beach, FL (November 1985).

3. S. H. Fistedis, ed., "The Experimental Breeder Reactor-II Inherent Safety Demonstration," Vol. 101, No. 1 (1987).

ORNL R&D ON ADVANCED SMALL AND MEDIUM POWER REACTORS: SELECTED TOPICS*

James D. White
and
Donald B. Trauger

Oak Ridge National Laboratory†
Building 3500
P.O. Box 2008
Oak Ridge, Tennessee 37831-6010

ABSTRACT

From 1984-1985, ORNL studied several innovative small and medium power nuclear concepts with respect to viability. Criteria for assessment of market attractiveness were developed and are described here. Using these criteria and descriptions of selected advanced reactor concepts, an assessment of their projected market viability in the time period 2000-2010 was made. All of these selected concepts could be considered as having the potential for meeting the criteria but, in most cases, considerable R&D would be required to reduce uncertainties. This work and later studies of safety and licensing of advanced, passively safe reactor concepts by ORNL are described. The results of these studies are taken into account in most of the current (FY 1989) work at ORNL on advanced reactors. A brief outline of this current work is given. One of the current R&D efforts at ORNL which addresses the operability and safety of advanced reactors is the Advanced Controls Program. Selected topics from this Program are described.

NUCLEAR POWER OPTIONS VIABILITY STUDY

The objective of the Nuclear Power Options Viability Study (NPOVS)[1,2] was to explore the viabilities of several nuclear electric power generation options for this country in the 2000-2010 time frame. Innovative concepts were identified for which proponents claimed marketability at this time when studies indicated an expected increased demand for new electrical energy capacity. Criteria for future market viability were developed and used for

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assessment of these advanced small and medium power reactors. As shown later, these criteria (and related design characteristics) emphasized safety features, cost, operability, constructability, regulation, research needs, and market acceptance.

GROUND RULES AND THEIR SIGNIFICANCE

To facilitate useful study, NPOVS concentrated on a carefully selected, limited group of concepts by developing and applying ground rules. The following three ground rules were selected:

1. The nuclear plant design option should be developed sufficiently that an order could be placed in the 2000-2010 time period.

2. The design option should be economically competitive with environmentally acceptable coal-fired plants.

3. The design option should possess a high degree of passive safety to protect the public health and property and the owner's investment.

Ground Rule 1 determines the time period of interest. It was assumed that, if orders of additional nuclear power are placed before the year 2000, they will be filled by current or slightly modified designs, primarily of LWRs. By the turn of the millennium, the anticipated demand for power may permit consideration of advanced reactor concepts and their associated advantages. For the present the concept must be supported by an active and capable industrial proponent with a current program. It was considered very difficult, perhaps even impossible, for a proponent to obtain funds, complete a design, conduct R&D, build a demonstration plant or its equivalent, and demonstrate satisfactory operation by 2010, unless design work is already underway. The concept must have no major feasibility problems or major questions that must be resolved by long-term, high-risk R&D prior to commercial acceptance.

Ground Rule 2 stated that for a concept to be viable, it must be economically competitive. The measure chosen is the most likely and perhaps the only major alternative, the coal-fired power plant. Since the cost of coal and its transportation vary widely with location, this ground rule is somewhat site dependent. The most favorable situations for coal might eliminate some or all of the nuclear concepts. However, other problems such as mining, acid rain, and carbon dioxide buildup could become dominant by the time period considered in this study.

Ground Rule 3: Although licensable plants are considered adequately safe by NRC and the nuclear industry, passive safety provides additional protection that is independent from engineered devices and from human intervention or management. The added protection of the owner's investment through passively safe designs may enhance the acceptability of advanced concepts as viable power options. Passive safety should help overcome intervenor's objections, public apprehension, and utility hesitation. These features also may simplify plant operation for the owner-operator. Thus, passive safety, enhanced beyond that of the present safety philosophy of primarily diverse and redundant engineered (active) systems, may provide an ingredient to help revitalize the nuclear industry.

The concepts selected are advanced and have various degrees of innovation when compared with current concepts. For convenience, the selected concepts were classified in the traditional way by their coolants and respective generic names:

- Light-water reactor (LWR)
  - -- PIUS (process inherent ultimate safety). Promoted by Asea-Atom of Sweden
  - -- Small BWR (boiling-water reactor). Promoted by General Electric Company (GE)
- Liquid-metal reactor (LMR)
  - -- PRISM (power reactor inherently safe module). GE advanced concept supported by DOE
  - -- SAFR (sodium advanced fast reactor). Rockwell International Corporation advanced concept supported by DOE
  - -- LSPB (large-scale prototype breeder). Electric Power Research Institute-Consolidated Management Office (EPRI-CoMO) concept supported by DOE
- High-temperature reactor (HTR)
  - -- Side-by-side modular concept (small helium-cooled reactor concept that has the core and steam generator in separate steel vessels in a side-by-side configuration). Supported by DOE and promoted by Gas-Cooled Reactor Associates and industrial firms.

CRITERIA FOR VIABILITY

The criteria developed by the NPOVS study were as follows: (Some elaboration is offered for Criteria 1, 5, and 7 since they relate specifically to safety and licensing.)

1. <u>Public Risk</u> - The calculated risk to the public due to accidents is less than or equal to the calculated risk associated with the best modern Light Water Reactors (LWRs).

This is a fundamental public safety criterion. To implement it strictly, a probabilistic risk assessment (PRA) employing acceptable methods and data bases would be necessary for each new concept and for the "best modern LWRs." However, other approaches based on judgment can be useful. Compliance with this criterion is essentially a prerequisite for licensing.

2. Investment Protection - The probability of events leading to loss of investment is less than or equal to 10^{-4} per year (based on plant cost).

3. Economics - The economic performance of the nuclear plant is at least equivalent to that for coal-fired plants.

4. Design - The design of each plant is complete enough for analysis to show that the probability of significant cost/schedule overruns is acceptably low.

5. Certification - Official approval of a plant design must be given by the U.S. Nuclear Regulatory Commission to assure the investor and the public of a high probability that the plant will be licensed on a timely basis if constructed in accordance with the approved design.

This criterion addresses concern for delays and associated risk for fully designed or replica plants. Its prime concern is with the licensing process, including potential further changes in requirements and regulations. Today's cumulative experience with licensing is extensive and should be sufficient to permit the introduction of one-step licensing at the completion of design. Verification of quality control during construction, of course, would be required.

6. Marketability - For a new concept to become attractive in the marketplace, demonstration of its readiness to be designed, built, and licensed and begin operations on time and at projected cost is necessary.

7. Competence of Owner/Operator - The design should include only those nuclear technologies for which the prospective owner/operator has demonstrated competence or can acquire competent managers and operators.

For the operation of a new or substantially different concept to be satisfactory, utility plant managers and operators must have acquired an adequate background and experience with the technology, equipment, maintenance, and plant surveillance. For operation, simulator training has proven effective for current power plants, and simulators would be necessary tools for new concepts. Where the concept, such as the small BWR, derives from a prior system, this criterion should be relatively easy to meet.

Characteristics which augment the criteria and provide further guidance to designers are divided into two categories, essential characteristics and desirable characteristics. The essential characteristics involve construction costs and lifetime projections, investment risk, cost for reliable and safe operation, availability of financing and other resources, and public acceptance. The desirable characteristics that are related but not readily determined quantitatively are: practical research, development, and demonstration requirements; ease of siting; load-following capability; resistance to sabotage; ease of waste handling and disposal; good fuel utilization; ease of fuel recycle; technology applicable to breeder reactors; high thermal efficiency; low radiation exposure to workers; high versatility relative to applications; resistance to nuclear fuel diversion and proliferation; on-line refueling; ease of decommissioning; and low visual profile.

The logic for using a level of safety equivalent to that of Light Water Reactors (LWRs) as the standard of Public Risk (Criterion 1) was twofold. First, we considered conventional LWRs to be safe. Second, we observed that a different concept could be compared with an LWR on the basis of specific properties or components. Such comparisons include reactivity effects, stored energy, thermal capacity to absorb decay heat, temperature limits for fuel and cladding, and security of primary systems containment. Although such comparisons are not a substitute for a probabilistic risk analysis (PRA), they can provide quantitative means for comparative evaluation until the data base for component reliability and system integrity are adequate to perform an effective PRA.

The Competence of Owner/Operator (Criterion 7) is important, as illustrated by the Three Mile Island 2 and Chernobyl accidents as well as by the long, costly outages experienced by many nuclear plants. Errors in management and by operators will always be of concern since the human factor cannot be totally eliminated.

Small and medium power reactors offer benefits of potentially simpler systems and greater automation. The latter is particularly important since parallel units in a common facility are postulated to require automation. Judgments about licensability will focus on the design and safety features of these small, multi-unit reactors. As the overall complexity of the reactor station is reduced by smaller and more passive designs, the designer needs to ensure that the operational problems will be correspondingly reduced.

Multi-unit plants require standardization within the station complex. Extension of standardization to all station units of a given concept offers advantages for both licensing and factory assembly. The U.S. Nuclear Regulatory Commission [3], in their 1985 policy and planning guidance on advanced nuclear power plants have addressed this issue.

A companion to the reactor standardization policy is a preapproved siting policy. The time gained in the construction schedule by referencing an approved standard design could well be lost in a dispute over the adequacy of a proposed site. It should be possible to gain site approval in advance of applying for a construction permit; some regulations governing early site reviews have been adopted by the U.S. Nuclear Regulatory Commission.

The advent of smaller units has led to a concept of licensing by demonstration; i.e., to subject the first reactor of a concept to unusual stress and thus show its capability to accommodate potential accident initiators. There is great merit in demonstration units to identify problems in design and construction as well as to obtain licensing experience. However, there are limitations to licensing based primarily or totally on demonstration.[4] Not all safety claims or hypothetical accident sequences can be demonstrated; substantial analysis will still be required. Also, a license may be required for the test prototype. The demonstration tests would be complex and expensive. The test module may have to be sited remotely because of the potential risk of test failures. Savings in analysis may be minimal or even negative once the design needs for a successful set of tests are defined.

At the time the NPOVS report was prepared, little information was available from PRA studies for small and medium power reactors. We noted, however, that the designers of passively safe concepts responded to safety considerations in the following ways:

- Many small and medium power reactor concept proponents, relying as they do on passive safety features to prevent adverse effects of accidents, claim that nuclear safety-grade equipment can be limited to the nuclear island.

- Proponents of some of the concepts believe that minimal or no containment can be justified because of a lack of credible severe accident sequences. In fact, some of the proposed passive decay-heat-removal systems would be precluded by the use of conventional containments.

- Some proponents believe that a safety demonstration plant would greatly facilitate licensing.

- There is considerable support for the proposition that very rare accident precursors, with frequency below some particular value such as 10^{-7} per reactor year, need not be considered as design basis events. However, current experience and PRA methods may not be adequate to establish such values.

The use of performance-based regulation to replace the present prescriptive systems should be considered as a long-term objective. The concept can contribute to plant simplification (and reduced cost) while retaining a high degree of protection against public risk. The objectives are similar to those for many small and medium power reactor designs. Several of the following actions could be included in such an initiative:

- Adoption of passive safety systems to replace or supplement active safety systems. The use of passive systems makes verification simpler in that safety becomes more deterministic and less probabilistic.

- Performance standards can be applied to the plant's response to certain accident initiators such as an earthquake of a specified intensity or a pipe break of a particular timing and size. A combination of test and analysis can then be used to determine that a severe accident will not result.

- As experience is gained with the application of performance standards of limited scope and in the use of PRA, greater weight can be placed on the use of PRA to verify the overall achievement of safety goals.

- The response of plants to actual challenges to safety systems (Licensee Event Reports) can be analyzed to verify that the PRA is soundly based.

Current nuclear regulations require that there be a containment system, independent of reactor design, to mitigate the release of an arbitrary fraction of the reactor's fission products independent of reactor design. It is noted that this fraction is probably much greater than the actual release that would be experienced in most accidents; however, the regulation is intended to be conservative. Containment features for confinement of small and medium power reactors, particularly those designed without leak-tight containments, may require extensive research and demonstration to convince regulatory bodies that the proposed safety measures are adequate. At the time of the NPOVS, the following research and development areas were identified:

- Development of quantitative risk criteria for advanced reactors.

- Consideration of the significance of passive safety features to risk reduction.

- Determination of the frequency of rare events that would constitute a lower limit for design basis.

- Appropriate treatment of source term and containment for very safe designs.

- Appropriate focus on safety and risk reduction in the development and application of standard designs.

The NPOVS also addressed the question of market acceptability for new nuclear technologies. Case studies and interviews with public utilities, public utility commissions (which regulate electric rates), and interest groups were utilized to explore the market acceptability for new technology. From this research, a set of major issues was identified that is likely to be at the core of the acceptability question for new reactor technologies. It was concluded that for a new technology to be acceptable in the U.S.A. after the turn of the century, three necessary but not sufficient conditions would be required; these are:

- A projected need for new baseload capacity

- A narrowing of the gap in construction costs between environmentally acceptable fossil and nuclear plants; and

- The absence of a third, more environmentally and economically acceptable option for baseload power to compete with nuclear.

Even if all three necessary conditions are satisfied, there is no guarantee that nuclear options will be chosen. There is a further set of facilitating conditions that would substantially improve the position of nuclear technologies within the market. These include improvements in the following areas:

- Stability of the regulatory environment
- Improved accuracy and reliability of load-forecasting techniques
- Improved cost controls in nuclear construction and operation, including standardized or turnkey plants; and
- Demonstrated technical feasibility of new nuclear reactors.

Stability of the regulatory environment has been identified as important, but, in general, we received the impression that if economic incentives are strong enough, regulatory difficulties will eventually be overcome.

In summary, the features of advanced small and medium power reactors offer substantial potential advantages in safety and licensing. Smaller amounts of decay heat per reactor core, when combined with fuels and structures having higher temperature capability and improved ability to dissipate thermal energy in upset or accident conditions, make possible the innovative designs for current High Temperature Gas-Cooled Reactor (HTGR) and Liquid Metal Reactor (LMR) concepts. These designs include methods for dissipation of decay heat more directly to the atmosphere or to the earth and, in the case of Boiling Water Reactors (BWRs) and Process Inherent Ultimately Safe Reactors (PIUS), to large bodies of water. The safety features of these designs permit the following changes: eliminating conventional containments and engineered safety systems, clustering modules, using common control and power-generating units, and extensive shop fabrication, all of which result in cost savings. Some of these cost-saving changes partially offset the increased cost of the smaller-sized units that results from loss of the economy of scale enjoyed by larger units. Except for shop fabrication, the changes entail new issues that must be resolved in licensing a lead plant. Although the proposed designs may increase the margin of safety, they will require new methods for review at the outset. Licensing revision or reform appears necessary if these otherwise attractive units are to be economically competitive in the near term.

CURRENT DOE-SPONSORED R&D ON ADVANCED SMALL AND MEDIUM POWER REACTORS AT ORNL

The lessons learned from the NPOVS and later studies are integrated to the extent possible into the current R&D programs at ORNL.

We have four areas of activity funded by the Department of Energy in Advanced Reactor Technology: (1) Modular High-Temperature Gas-Cooled Reactor (MHTGR) Technology, (2) Strategic Technologies for Advanced Liquid Metal Reactor (ALMR) Concepts, (3) R&D on advanced LWRs, and (4) Advanced Fuel Cycle Technology. The first three of these areas are outlined here to give the reader a perspective of the kinds of work involved.

MHTGR TECHNOLOGY DEVELOPMENT

With regard to MHTGR technology development, extensive planning has taken place over the past several years culminating in a technology development program covering fuel, fission products, graphite, and metals behavior. These plans have been established by DOE, ORNL, and industry in direct response to specific MHTGR design and licensing requirements. Careful prioritization of technology development needs has led to rigorous cost control. The required experimental technology development program is now well-defined and underway.

Fuel and fission products technology

Fission product retention within the fuel coatings is the key ingredient to achieving passive safety in the MHTGR, and although results to data have been favorable, additional data are needed to demonstrate and validate that expected performance is achieved. Fuel fabrication process development is required to confirm that economic fuel fabrication processes will produce quality coatings which retain their integrity during fabrication and reactor irradiation. Fission product transport studies are needed to conclusively demonstrate adequate retention of fission products within the reactor system during postulated conditions.

Current work in fuel and fission product behavior is designed to update and validate behavior models. This work uses data from several national and international facilities to update/validate models of failure rate for standard and defective fuel particles under normal and accident conditions; fractional release of metallic and gaseous fission products under normal and accident conditions; plateout of metallic fission products on graphite and metallic components; and liftoff/washoff under accident conditions.

Fission product behavior work includes bench scale testing to evaluate: chemical forms of fission products, effects of alloys and surface conditions on plateout and reentrainment, and characterize dust and effect of dust on fission product transport behavior. In pressurized loop facilities, work is being done to evaluate: effect of shear ratio and of differential pressure during depressurization, coolant chemistry/moisture effects, effects of blowdown duration, steam quality and Reynolds number on fission product behavior. Models will be updated based on this bench scale and pressurized loop testing. Then the updated models will be validated in the French COMEDIE loop.

Graphite technology

Graphite constitutes a major volume fraction of the MHTGR core region. The graphite technology program effort emphasizes obtaining physical and mechanical property data under all postulated conditions so as to provide for reliable component design. Important determinations for design and licensing support are irradiation creep effects and oxidation characteristics.

Key areas of current work are physical and mechanical properties, dimensional stability under irradiation, fracture mechanics and steam corrosion. Work in these areas is needed to provide design engineers with better data for design, to provide safety engineers with better statistical properties for PRA analysis, and to provide quality assurance data sufficient for licensing purposes. Dimensional stability and creep data come from irradiation capsule experiments in the ORNL High Flux Isotope Reactor (HFIR) and HFR-Petten. Corrosion data come from bench scale tests and pressurized loop experiments.

Metals technology

The metals technology program for the MHTGR relates primarily to mechanical property data required for reliable component design and to ensure licensability. With regard to the reactor vessel, emphasis is on irradiation effects under normal MHTGR conditions and on material property information pertinent to accident conditions. Long-term testing is needed to complete the materials property data base for the steam generator, hot duct, and core internals; key measurements involve fracture mechanics, mechanical properties, radiation effects, environmental and corrosion effects, and weld properties.

Physics validation

This work demonstrates the capability of physics methods and codes to predict important characteristics: core criticality, temperature coefficient of reactivity, control rod worth, power distribution, moisture worth, burnup swing, Pu buildup and B_4C worth. Data from previous experiments are being analyzed and will be supplemented with new data from the AVR. Supplemental experiments in other facilities may be used.

Shielding analysis

Primary tasks are: (1) calculation of maintenance dose rates after shutdown; and (2) calculation of neutron fluence to reactor vessel system. The first task is to be done by Bechtel. The second task is being done by ORNL. Experimental work is planned to supplement calculations. The fluence to the vessel head by neutron streaming is very difficult to treat analytically. Experimental work in this area will reduce uncertainties.

STRATEGIC TECHNOLOGIES FOR ADVANCED LMR CONCEPTS

Work in several strategic technologies is funded by DOE with emphasis on ALMRs. Most of this work, however, is applicable to advanced concepts of gas-cooled and light-water cooled designs.

Advanced instrumentation

Advanced instruments under development at ORNL include an improved neutron flux monitoring system featuring high-temperature operation and advanced electronics. A prototype system is now undergoing testing. An automated noise diagnostics system has also been developed an is undergoing prototype testing in the Fast Flux Test Facility (FFTF).

Shielding

The present shielding program is a cooperative and co-funded effort with Japan. The cost of shielding materials can be reduced by as much as a factor of 5 by using boron carbide (B_4C), reducing design margins due to improved analysis techniques and uncertainty predictions, and optimizing localized shielding for individual components and work areas. Particularly for modular designs, feasibility of design options can hinge on the use of advanced shielding techniques and materials. Specific feasibility issues include:

- Interim storage of fuel within the vessel
- Vessel diameter reduction to allow barge/rail shipment
- Dose criteria due to activation of secondary heat-removal system
- Location of in-vessel nuclear instrumentation

Verification of these improved shielding materials and development of methods is underway. A four-year joint Japanese/American Shielding Program (JASPER) to accomplish these objectives is being carried out at ORNL's Tower Shielding Facility. Six major experiments are planned, with the first two having been completed. These technology improvements have broad application in all advanced reactor systems as well as space nuclear systems.

In addition to this major program, ORNL provides direct design support in shielding design to the PRISM design teams.

High-temperature materials and structural design

ORNL has been and continues to be DOE's leading center of excellence in Materials and Structural Design Technology. To build on this expertise and transfer the benefits of this technology to industry, DOE has recently added a new $19 million High-Temperature Materials Laboratory at ORNL.

A major element of ORNL's LMR Materials and Structural Design program over the last decade has been the development of a new alloy, modified 9Cr-1Mo steel. The development is now almost 90% complete. Modified 9Cr-1Mo is approved for application under ASME Section VIII, Div. 2 (fossil and non-safety-grade nuclear application); and approval under ASME Section III, N-47 (Nuclear Applications) has been requested. Modified 9Cr-1Mo is already manufactured in the U.S., Japan, and France. International acceptance of this alloy has been excellent; it is now in service in fossil plants in Europe and Japan, as well as in the U.S. Substitution of modified 9Cr-1Mo for alternate materials in the SAFR steam generator and intermediate heat exchanger was estimated by Rockwell International to save more than $20M per 350 MW(e) module. Remaining activities for modified 9Cr-1Mo include: (1) development of test data to optimize weld procedures and establish weldment properties; (2) completion of mechanical properties tests; and (3) development and validation of design methods, rules, and criteria for application in the LMR service environment. The level of effort required to complete these activities over the next four years is more modest than that carried out in previous years, but is essential to achieve the benefit of the technology. The Japanese Atomic Power Company (JAPC) expressed an interest in supporting this work and providing their database on modified 9Cr-1Mo in return for the U.S. database. Negotiations between JAPC and DOE have led to a three-year program to begin in FY 1989. In addition, modified 9Cr-1Mo has potential for application in space nuclear power systems.

ORNL's role in materials and structural design involves more than just new alloy development. A number of critical materials and structural design issues remain, involving traditional materials and design methods. Structural failures resulting from creep and fatigue have occurred in both fossil and earlier LMR plants. Current LMR design methods and criteria must be improved to preclude such failures in the future.

Nondestructive testing technology is under development at ORNL for remote in-service inspection of components in a sodium environment. Inspection of components such as steam generators and intermediate heat exchangers in situ can save downtime and enhance the system reliability and availability.

Robotics

A primary motivation for utilizing robots in nuclear power plants is the reduction of personnel radiation exposure to "as low as reasonably achievable" (ALARA) as recommended by the NRC. The exploitation of advanced robots for hazardous operations will contribute substantially to achieving this goal. Robotic access reducing the number of human entries to sensitive locations has the potential for an increased frequency of monitoring and inspection, resulting in enhanced overall safety and reduced personnel exposure. Furthermore, the development and utilization of robots can potentially improve plant availability by permitting some maintenance tasks within containment to be done remotely during power production. It is believed that the number of tasks which could be accomplished this way is large enough to substantially reduce outage durations.

ORNL is leading and coordinating a team effort to pursue the development and deployment of advanced robotic systems capable of performing surveillance, maintenance, and repair tasks in nuclear energy facilities. A cooperative five-year plan is being pursued. In addition to ORNL, the team involves four major universities and their respective industrial partners:

- University of Florida, Odetics
- University of Michigan
- University of Tennessee, Combustion Engineering, Remotec
- University of Texas, Martin Marietta Aerospace

Annual joint demonstrations of the team's progress are planned, with the first demonstration to take place in December 1988 at ORNL.

Reliability, availability, maintainability (RAM) data

This program has also become international with the addition of 50% Japanese funding and participation under the Centralized Reliability Data Organization (CREDO). CREDO gathers (1) detailed engineering data on components, (2) failure data, and (3) operating data from U.S. and Japanese facilities.

The purpose is to establish and maintain a well-documented, centralized, comprehensive source of RAM data for use by LMR designers in probabilistic risk assessments and other analyses in support of design and licensing and in establishing inspection and maintenance practices to achieve high plant reliability and availability. The evolving data base provides information needed to properly design and license LMR systems. Presently the data base includes data on 20,000 components, 1,800 events/incidents, and 1.2 billion component operating hours. Space nuclear power systems will also benefit from the data base.

ADVANCED CONTROLS PROGRAM

Modern automated reactor control systems can enhance the economic competitiveness of advanced reactors by increasing their operational reliability, availability, and maintainability; improving their safety and licensability by reducing challenges to the plant protection systems; and significantly reducing the manpower needed to operate the plant.

Through on-line monitoring and surveillance of equipment performance, rapid detection and response to equipment malfunction, and reduction of operator errors, automation can significantly reduce the occurrence of abnormalities in plant operation and prevent abnormalities from becoming accidents.

Automation is critical to the economic competitiveness of multimodular plants. Next-generation plants must adopt modern digital control technology to make use of these new automated and intelligent control strategies and systems. Today's power plants are based on time-tested analog equipment and controls. Thus the new advanced control systems and strategies must first be demonstrated and tested to provide the necessary confidence to regulators, manufacturers, and power plant operators.

To achieve a practical design for an automated control system for advanced nuclear power plants, ORNL is integrating activities in: (1) control system design, architecture, and components; (2) artificial intelligence for adaptive, predictive, and self-learning control strategies and expert systems for operator support; (3) integrated human-system engineering for allocating responsibilities between humans and computers, and for optimal control complex design; and (4) plant simulation, software development and validation, and system reliability improvements.

These integrated activities are being conducted in the Advanced Controls (ACTO) Program. The goal is to provide a national center of excellence in research, development, and testing of nuclear control systems. This program

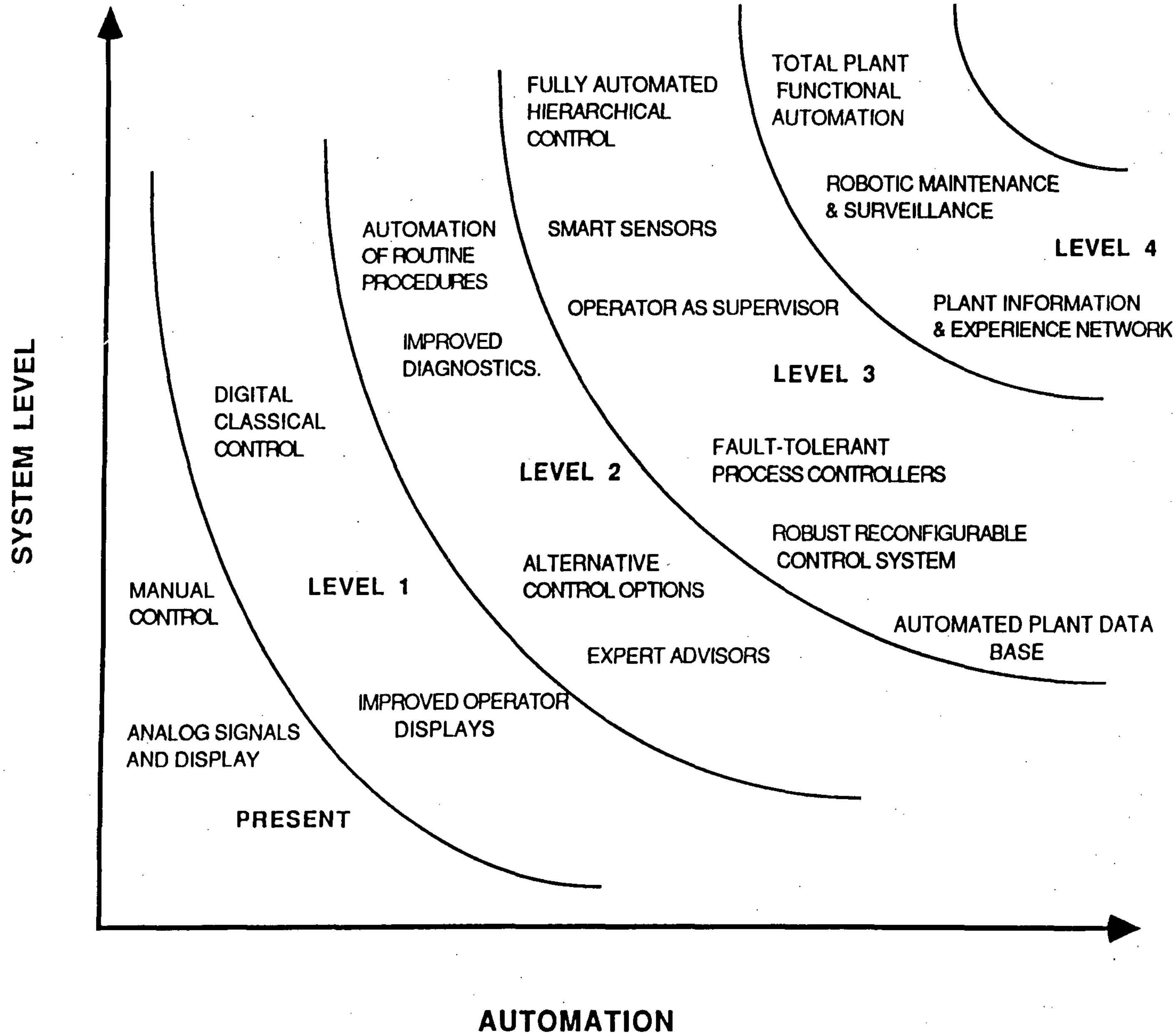

Fig. 1. Nuclear plant control automation with evolving technology.

will provide an integrated environment to support the rapid, confident design and testing of advanced control systems that will assure improved operability, reliability, and safety for advanced reactors.

The transition from today's nuclear control systems (with some analog control at the subsystem level and significant operator integration) to future designs (with complete automation under human supervision) will occur in phases. The transition may be described in terms of four levels as shown in Fig. 1. The first phase of improvement is automated data management at a plant, which is now actually occurring to a limited extent in U.S. LWRs and is under study in the U.S. LMR Program. Also in this level will be some replacement of today's analog controllers with more reliable digital controllers performing basic proportional-integral-differential control. EPRI is sponsoring some of this work at existing LWR sites.

Level 2 will be automation of routine procedures like startup, shutdown, refueling, load changes and certain emergency response procedures. Significant assistance will be given to the operator in the form of expert systems and control room displays of plant status. Control strategies will be predetermined choices selected from hierarchical, optimal, robust, multivariate options. Advanced LMR concepts being studied fit within this phase.

Level 3 is a significant advance toward automation with capability for full automation of all hierarchical levels of control. The operator's role will be to interact with and monitor the intelligent, adaptive supervisory control system. Smart sensors will validate their own signals and communicate with robust, fault-tolerant process controllers. The process controllers will be able to reconfigure the control logic to meet the operational objectives selected by the supervisory control system. Control strategies will be adaptive, uncompromised by nonlinear effects in the processes, and very robust to off-normal conditions. Plant designs will be completely automated, with plant data bases available to the control system and the operator. Operational experience of all plant systems and components will be tracked in an automated data base, and the control system will recommend maintenance schedules and outages to the operator. Good human performance modeling will permit optimum allocation of function decisions so as to keep the operator motivated and informed about plant status.

Level 4 is total functional automation of the plant through an intelligent control system aware of the entire operational status and in interactive communication with the operator to keep him apprised concerning operational status, any degraded conditions, likely consequences of degradations, and possible (recommended) strategies for minimizing deleterious consequences. By this time plant designs will have many automated and robotized functions including maintenance and security surveillance. The control system will be an integral part of not only the total plant design but also the national network of commercial power plants. The control system computer will learn from the network information concerning other plant and component operational experience and will alert the operator if that experience is relevant to current operations.

To provide the necessary national leadership in the design of advanced control systems, this program will support four major kinds of activities:

- Demonstrations of advanced control system design features using current developments in control theory, automation, artificial intelligence, information management, modeling and simulation, and man-machine interaction research.

  -- The Advanced Controls Program will provide national leadership in control system design by demonstrating examples of advanced control system designs for nuclear reactors. These demonstrations will be carefully designed to show how state-of-the-art research can be used to help accelerate the transition to fully automated control. For example, we developed and demonstrated in a prototype design this year a new, promising (easier, faster) technique for designing a hierarchical, distributed control system using multivariate optimal control theories for non-linear systems with uncertain dynamics.[5,6] Preliminary testing of the prototype design indicates that this technique leads to improved ability to: (1) adapt to changes in plant performance, (2) accommodate plant component modifications (as in plant aging or component replacement), and (3) perform well even in the case of noisy plant signals. The next demonstration will occur late this year and will be a prototype advanced, automated control system design for a feedwater system for an advanced LMR. The feedwater train is a complex system that is the origin of incidents causing a significant fraction of lost plant availability in conventional LWRs.

  These and other demonstrations in following years will help transfer to the reactor industry the benefits of the latest proven advances in control systems strategy, control system and whole-plant simulation, computer-aided software engineering for control systems design, man-machine interaction modeling and analysis, and the other technologies being used within the program.

- Establishment of a design environment that allows designers to formulate and test various control strategies for the plant of interest quickly and economically.

  -- The program will provide a centrally located, user-friendly design environment for control system designers within the DOE community. The environment will consist of four parts: (1) networked, intelligent, computer workstations into which have been integrated software tools, graphics capabilities, on-line design guidance, on-line documentation and interfaces to the large plant simulation capability at ORNL; (2) plant/component models and databases useful for control system design and plant simulation; (3) information resources concerning advanced control system strategies for automated control; and (4) man-machine interaction models and guidelines for designing control system interfaces with operators. This year, we developed and demonstrated a prototype of a unique intelligent workstation for control system design featuring artificial intelligence and graphical interfaces.[7,8,9] The prototype provides the ability to easily, rapidly produce, simulate and evaluate certain ALMR control system strategies and techniques.

Also during the year, we developed and linked a prototype model of a human operator to an ARIES-P simulation of PRISM in FY 1988.[10,11,12] This model enables a rough simulation of operator performance during ALMR operational upsets. The model accounts for cognitive functions of the operator as well as training, timing, and probable error rate.

In the area of advanced controls R&D, we developed and evaluated (using EBR-II data) several advanced multivariate optimal control strategies for ALMR systems.[13] Showed that certain advanced control techniques are better than classical proportional-integral techniques (more robust with respect to changes in plant performance due to fouling, etc.).

- Testing and validation of advanced control system designs by simulation.

  -- The ability to simulate an entire plant in real-time is critical to the design of a fully automated plant. The program will provide this simulation capability to the technical community. State-of-the-art advances in computer architectures, software engineering, very high-level languages, area networking, artificial intelligence, and database management will be integrated into a whole-plant, real-time nuclear power plant simulation capability. In FY 1988, we procured and installed a parallel processor and several computer workstations in the program computer laboratory to enhance our testing and simulation capability.

- Guidance in control software and hardware specifications.

  -- The program will provide standards, guidelines, and specifications for control software and hardware. ORNL will acquire and develop tools and methods for generation of large, standardized software programs needed for automation of nuclear reactors. Methods for locating errors in software programs will be acquired and developed, and software verification and validation procedures will be utilized.

R&D ON ADVANCED LWRs

Two efforts are underway at ORNL on Advanced LWRs. These are concerned primarily with requirements development and review for a large ALWR and preliminary investigations on more advanced developmental LWRs. In the EPRI requirement development work, emphasis is on man-machine interface systems (including control systems) for a large ALWR in the DOE/EPRI ALWR Program--but the work probably will be applicable to smaller ALWR concepts. The ORNL work consists of team participation in the development and review of requirements, review of codes and standards for digital control and protection systems and development of software verification and validation plans. The developmental LWR activity is directed to investigate advanced passive LWR features beyond those currently being considered in the ALWR program.

In a related activity, ORNL has the lead role in the Advanced Neutron Source (ANS) design. ANS is a major new research reactor project directed at establishing the world's leading center for neutron scattering research. The ANS work is funded by the DOE Office of Basic Energy Sciences. The facility will have very high thermal and cold neutron fluxes, state-of-the-art neutron-scattering facilities, isotope production capabilities, and materials irradiation positions.

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USE OF ARTIFICIAL INTELLIGENCE TO ENHANCE THE SAFETY OF NUCLEAR POWER PLANTS

Robert E. Uhrig
Oak Ridge National Laboratory*

ABSTRACT

In the operation of a nuclear power plant, the sheer magnitude of the number of process parameters and systems interactions poses difficulties for the operators, particularly during abnormal or emergency situations. Recovery from an upset situation depends upon the facility with which the available raw data can be converted into and assimilated as meaningful knowledge. Plant personnel are sometimes affected by stress and emotion, which may have varying degrees of influence on their performance. Expert systems can take some of the uncertainty and guesswork out of their decisions by providing expert advice and rapid access to a large information base. Application of artificial intelligence technologies, particularly expert systems, to control room activities in a nuclear power plant has the potential to reduce operator error and improve power plant safety and reliability.

Artificial intelligence (AI) burst on the scientific scene about 30 years ago with much fanfare and promise. Recognition that computer symbols could represent characteristics of the real world, and that computer programs could relate these features, provided the means by which computers could be used to simulate certain important aspects of intelligence and provided an information-processing model of the human mind. It is ironic that as progress floundered in the use of AI to increase the intellectual understanding of the workings of the human mind, certain practical applications of this information-processing model spawned whole new technologies that promise to revolutionize the way both business and industrial organizations operate. Expert systems (see Appendix A), probably the most commercially successful product of AI research, can be used to improve engineering, management, and operation of nuclear power plants in the United States.[1,2]

In the operation of a nuclear power plant, great quantities of numeric, symbolic, and quantitative information are handled by the reactor operators even during routine operation. The sheer magnitude of the number of process parameters and systems interactions poses difficulties for the operators, particularly during abnormal or emergency situations. Recovery from an upset situation depends upon the facility with which the available raw data can be converted into and assimilated as meaningful knowledge. Plant personnel are sometimes affected by stress and emotion which may have varying degrees of influence on their performance. Expert systems can take some of the

*Operated by Martin Marietta Energy Systems, Inc., for the U.S. Department of Energy under Contract No. DE-AC05-84OR21400.

uncertainty and guesswork out of their decisions by rapidly providing expert advice and access to a large information base. The application of AI technologies, particularly expert systems, to control room activities in a nuclear power plant can reduce operator error and improve plant safety and reliability. Furthermore, there are a large number of nonoperating activities (e.g., testing, routine maintenance, outage planning, equipment diagnostics, fuel management, etc.) in which expert systems can increase the efficiency and effectiveness of overall plant operation.

In the United States, development of expert systems in the nuclear power field is being carried out by a wide spectrum of organizations including nuclear equipment vendors, architect-engineer firms, universities, national laboratories, federal agencies, the electric power utility industry, and small entrepreneurial groups. The examples of application of expert systems in the nuclear power field cited here are typical of those being developed in the United States. They constitute only a small fraction of those being developed, although few systems are actually in use in nuclear plants today.

The most coherent of these efforts is the program undertaken in 1983 by the Electric Power Research Institute (EPRI) to demonstrate the usefulness of AI in a number of areas including augmenting plant automation activities. EPRI also has a program to transfer the technology of NASA's multiyear "AI Core Technology in Systems Automation" to the nuclear power industry.

One of the first EPRI projects in expert systems was REALM (Reactor Emergency Alarm Level Monitor), which was developed by Technology Applications, Inc.[3] The NRC has about 20 pages of guidance on classifying an emergency as an unusual event, an alert, a site area emergency, or a general emergency. Each level of emergency has a specific set of responses that the utility must undertake. The decision as to the level of the emergency must be made rapidly, sometimes in a time frame in which the true nature of the event is not yet clear. While many sensory and manual observations are available, certain needed data may be missing, ambiguous, or even conflicting. Judgment is required for proper interpretation in such situations, and REALM is designed to operate in a real-time process environment. It incorporates what might be called a "first-level" diagnostic system that readily identifies the cause of the emergency based on comparison of the symptoms observed and the events that are possible in a nuclear power plant. In addition, REALM provides a rationale as to why it recommends a particular classification. It then carries out a "vulnerability analysis," telling the operators which events would lead to a higher emergency level and what needs to be done to get to the next lower level. REALM was developed for Indian Point-2 in cooperation with Consolidated Edison of New York, and it performed well when operated in parallel with normal plant operating procedures during the two most recent plant emergency drills.

EPRI is also developing a computerized tracking system for emergency operating procedures.[4] This expert system is co-resident on the safety parameter display system computer and is presently being tested on the Kuosheng Nuclear Power Plant, a BWR-6 nuclear reactor in Taiwan. The emergency operating procedures are written in about 250 rules that can be evaluated in less than

1 second. Conclusions as to the steps that should be taken are available within seconds after a parameter change. Its inference engine looks for pattern matches between the rule premises and the operating conditions, which then lead to the recommendation of action to be taken. It is an on-line system requiring no input from the operators, and explanations for its conclusions are available to the operators.

Westinghouse Hanford Company has developed two expert systems that are "clones" of human experts at the Hanford Engineering Development Laboratory (HEDL) and the Fast Flux Test Facility reactor (FFTF).[5] Both expert systems have direct applicability to commercial nuclear power plants. CLEO (Clone of Leo, an expert on refueling the FFTF) is an expert system capable of generating a list of necessary refueling moves in less than 30 s, given the present and future core configuration of the FFTF. CRAW (Clone of Rawley, an expert in diagnosing fuel cladding failures in FFTF), interprets indications of fuel failure (i.e., tag gas detection). Rapid expert diagnosis shortly after detection is required 24 hours a day; this expert system is an effective substitute when the resident expert is not available.

Middle South Utilities has developed TRIBES (Trip Buffer Expert System).[6] TRIBES analyzes trips caused by the core protection calculator and the control element assembly calculator, which monitor nuclear power plant parameters and control element assembly positions respectively. These core protection systems will initiate a trip to prevent violation of fuel design limits (i.e., kilowatts per foot, DNBR limits, rate of power increase, etc.). Analysis of the computer output is required to establish the cause of the trip before the plant can be restarted.

Stone and Webster Engineering Company has developed an expert system to analyze the limiting conditions of operation (LCOs) and technical specifications in a nuclear power plant.[7] These limitations are imposed by regulation, and violations can result in regulatory action that may include civil penalties as well as shutdown of the plant. One of the uses of this system is to assess the effect of both operational changes and the removal of equipment from service to determine whether either of these activities will lead to a violation of LCOs or technical specifications. The program has a "what if" mode that allows the operators to determine the impact of the proposed maintenance actions and operational changes before they are authorized. This mode is used to detect the subtle interactions that might otherwise go undetected and inadvertently cause a trip of the plant or a violation of the LCOs or technical specifications.

Southern California Edison has developed TAGS (Tagout Administration and Generation System) for their San Onofre nuclear power plant.[8] TAGS is a conventional computer program to administer the safety tagout process. It has been integrated with an expert system in the form of an intelligent work station using PLEXSYS (plant expert system), which was developed by EPRI. PLEXSYS will present piping and instrumentation drawings (P&IDs) and one-line electrical schematics for the systems of interest. When the components to be tested are selected, PLEXSYS and TAGS recommend a "safety tagout boundary" that allows maintenance to be performed without danger of tripping the plant.

Texas Utilities and Westinghouse jointly have developed GenAID™, an on-line generator diagnostic system,[9] to diagnose hundreds of conditions with damage potential to the electrical generator and to recommend corrective action for each condition. Special monitors are attached to the generators located in Texas and are coupled to computer terminals continuously linked via phone lines to the Westinghouse Diagnostics Center in Orlando, Florida. Diagnoses and recommendations are based on the knowledge of the best experts (designers, service engineers, field engineers, operators, etc.). GenAID is now in operation and has proven to be an effective tool in reducing the risks of error in human judgment, thereby improving plant productivity and availability.

Westinghouse is also using its Intelligent Eddy Current Data Analysis (IEDA) expert system[10] to analyze eddy current data from the 45 miles of tubing in a typical nuclear plant steam generator tube bundle. The analysis typically requires 60,000 judgments, some extremely difficult. IEDA is based on a set of highly defined rules (developed from an "expert model data analyst") to which the eddy current data are compared. Incorporated into the system is a versatile and user-friendly operating mode that allows manual evaluations of signals the computer cannot categorize properly.

The Duane Arnold Energy Center and Iowa State University have initiated an advisory expert system called MOVES (Motor-Operated Valve Expert System) for valve maintenance planning.[11] The data base contains ~117 safety-related motor-operated valves. Maintenance encompasses diagnosis of operational symptoms, prescription of corrective maintenance, determination of procedural requirements, and identification of required postmaintenance testing. The importance of valve maintenance is indicated by industry estimates that valve-related problems cost U.S. utilities about $100 Million per year in lost plant availability and up to 30% of the industry's annual maintenance budget.

Other reported applications of expert systems in various stages of development include outage planning, heat rate improvement, alarm filtering, sequencing and suppression, diagnostics for instruments and equipment, welding rod selection advisor, generating welder procedure specifications that comply with regulatory codes, signal validation, disturbance analyses, condensate feedwater monitor, radwaste processing system advisor, bypass-inoperable status indicator system, sequencing BWR control rods after maneuvering, water chemistry control, pressure-temperature control during startup (to avoid pressurized thermal shock problems), real-time emergency evacuation planning, and real-time radiation exposure management.

The fundamental and synergistic relationship between training and expert systems offers a unique opportunity to improve the training of nuclear power plant personnel. One of the features that makes an expert system so compatible with diagnostics in nuclear plants is its ability to explain its reasoning and its conclusions for the postulated or real conditions given to it. All supporting evidence for machine opinions about systems or events can be cited for final evaluation and decision by human operators. As the operators work with an expert system, there is constant exposure to the bases, limits, and nature of system interrelations. Recent work at The University of Tennessee[12] has dealt with the symbiotic relationship between diagnostics and training. Indeed, the understanding gained in developing and encoding the knowledge base on the operation of a nuclear power plant into an expert system may be as important as (if not more important than) the use of that system in actual plant operation. This effort further enhances the quality of nuclear personnel training.

The utilities are introducing expert systems into nuclear power plants very slowly, possibly because they are reluctant to submit this new technology involving uncertainties to regulatory review until they are convinced that the benefits gained will warrant the effort required. Perhaps regulators' principal concern with expert systems is the ability to encode expertise properly, particularly the fine nuances and shades of meaning, into the knowledge base of an expert system so that it can emulate human expertise with fidelity. Another major concern is the narrow scope of the expertise and the associated limited area of applicability of expert systems. Two of the consequences of these limitations are the inability of an expert system to exhibit common sense and its limited ability to recognize when it is operating outside its field of knowledge. Researchers have sought to minimize the impact of these limitations by building "robustness" into expert systems (i.e., the ability to fail gradually and predictably when it gets outside its operating regime). These limitations, as well as the lower confidence associated with answers when data are missing or have low certainty factors, may be of concern to regulators when expert systems are introduced into the safety-related systems of nuclear power plants.

Demands by the safety and environmental regulatory authorities for increased safety margins and lower environmental impacts and those by the economic regulatory authorities and the financial community for increased efficiency in operation (e.g., fewer trips, higher availability, plant investment protection, etc.) inevitably lead to more sophisticated plants with additional systems that must be controlled and/or automated. Digital systems inevitably will totally dominate the control systems of the next generation of nuclear power plants unless they are specifically forbidden by regulatory authorities. Indeed, the integration of expert systems into the safety, control, and management systems of power plants is an integral part of the automation process.

In summary, a nuclear power plant is too complex a system to be managed or operated by anyone's "gut feeling." An expert system can be the ever alert, knowledgeable assistant to the operators as well as a valuable tool for plant management. Demands for increased safety margins, lower environmental impacts, increased performance, and greater investment protection will inevitably lead to automation of most functions of nuclear power plants. In turn, automation will be paced by the ability to develop efficiently the needed software through the use of modern computer science brought about by AI programming techniques. The regulators and the public must be assured that these plants are properly designed, properly built, properly operated, and properly maintained. Artificial intelligence and expert systems can and must play a major role in providing this assurance.

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APPENDIX A
WHAT ARE EXPERT SYSTEMS?

Expert systems can be defined as "computerized processes or programs that attempt to emulate the human thought processes associated with the application of expertise to problems." As expert systems have evolved over the past decade, they have typically consisted of two separate components, an "inference engine" (i.e., an information processor) and a knowledge base. The inference engine gathers the information needed, guides the search process in accordance with the strategy programmed into it, uses rules of logic to draw inferences about the processes involved, and presents conclusions (when warranted) along with an explanation of the bases for the conclusions. The inference engine may use either "forward chaining" (i.e., forward reasoning), in which it starts with the given data and proceeds toward a solution, or "backward chaining," in which it assumes a conclusion and then looks for evidence to support that conclusion. Since the inference engine and the knowledge base are entirely separate, changes in the knowledge base can be made easily without any influence on the inference engine.

Generally, the knowledge base of the expert systems relies on the expertise of experts or expert knowledge that has been codified in publications, books, or regulations to provide advice under a wide variety of conditions. When the data and/or information in the knowledge base are specific and precise, expert systems give results that are unambiguous. However, when the needed information is imprecise or "fuzzy," incomplete, missing, or even conflicting, expert systems can still reach a rational conclusion or solution through the use of confidence factors or probabilities. Under these conditions, an expert system will give the "most probable" solution or the "best" solution in a statistical sense. For this reason expert systems usually identify alternative or less probable solutions along with the associated probabilities or confidence factors. This characteristic of expert systems is one of their greatest advantages, although it may be of concern to regulatory authorities when these systems are installed in nuclear power plants.

OVERVIEW OF THE CONTAIN LMR CODE*

D. E. Carroll
Sandia National Laboratories
Albuquerque, New Mexico 87185

ABSTRACT

The use of containments in proposed U.S. Liquid Metal Reactors (LMR) will require that a computational tool be available for safety analysis. As a result of an international cooperative effort, such a tool has been produced in the CONTAIN-LMR computer code. This is a version of the CONTAIN code that has been enhanced with extra capabilities for LMR applications. CONTAIN is the NRC's best-estimate code for the evaluation of the conditions that may exist inside a reactor containment building during a severe accident. Included in the phenomena modeled are thermal-hydraulics, radiant and convective heat transfer, aerosol loading and transient response, fission product transport and heating effects, and interactions of sodium and corium with the containment atmosphere and structures. CONTAIN-LMR includes models for sodium-concrete interactions, debris bed phenomena and other LMR-specific models in an integrated manner. This paper summarizes the current state of CONTAIN-LMR. A brief description of the physical models is presented. As an example of model integration in CONTAIN-LMR, recent work on the debris bed models will be discussed. Recent applications of CONTAIN-LMR are discussed.

I. INTRODUCTION

An important thrust in liquid metal reactor design in the U.S. is the emphasis on passive safety utilizing inherent safety mechanisms of the reactor.[1] These mechanisms are the self-regulating features of the reactor that passively limit damage to the core in the event of an accident. Some have carried this emphasis to the extent of stating that containment structures will not be required for such reactors. This paper will not discuss the pros and cons of such a position, but will rather address the situation in which conventional containment structures will be utilized.

Liquid metal reactor designs in Japan and Europe have included robust, traditional containment features. Should the U.S. find itself in a similar position, a computational tool for severe accident containment analysis is

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available. This tool, CONTAIN-LMR, has been developed as a collaborative effort between the U.S., West Germany, and Japan.

CONTAIN was developed by Sandia National Laboratories (SNL) for the U.S. Nuclear Regulatory Commission (NRC) for both LMR and Light Water Reactor (LWR) analysis. NRC support of the LMR aspects of the code ended with the demise of the Clinch River Breeder Reactor Project. However, since that time the CONTAIN code has been used by workers in both Japan and Germany to investigate the effects of severe LMR accidents. The code is employed in the assessment of internal threats to containment integrity and the radiological source term in the event of containment failure. In addition to using the code, workers in Japan and Germany have participated in model development specifically aimed at improving the capability of CONTAIN to analyze LMR conditions. These efforts, together with those of workers at Sandia, have recently been combined into one new package called CONTAIN-LMR/1A. This is a derivative of the standard released version of the CONTAIN code used for LWR accident analysis and includes the LMR improvements that have resulted from international collaboration. This paper presents an overview of these capabilities.

II. THE CONTAIN CODE

The CONTAIN code is the NRC's best-estimate tool for predicting the physical and radiological conditions that may exist in reactor containment buildings in the event of a severe accident. CONTAIN offers a broad variety of models to the analyst in a system-level computational structure, which allows for the complex interactions and feedback among the diverse phenomena to be treated. Among models available are those for:

- intercell gas flow, including natural circulation effects,
- two-phase atmospheric thermodynamics,
- conduction in structures,
- convective and radiant heat transfer,
- condensation/evaporation at structure and pool surfaces
- hydrogen combustion,
- multicomponent aerosol processes,
- the transport and decay heating effects of fission products, and
- ablation of concrete by core debris.

A more complete description of CONTAIN is available in the CONTAIN User's Manual.[2]

The CONTAIN code is widely distributed to reactor safety researchers in both the United States and around the world. The code is currently in use at 43 research organizations in 12 countries. As testimony to its portability, CONTAIN is being used on some 13 different computer systems. CONTAIN has become a widely accepted tool for containment analysis in severe accident conditions.

III. THE CONTAIN-LMR CODE

CONTAIN-LMR is a special version of the CONTAIN code that has been provided with extra capabilities to model LMR applications. The code is derived from the CONTAIN code by means of the application of several update sets. The development of a new LMR version is not linked to the production of a new released version of the CONTAIN code. Thus the LMR code may be derived from a CONTAIN version which predates the most recently released code. This is the current case where CONTAIN-LMR/1A is built upon the 1.06 version of CONTAIN. Version 1.10 is the most recent release of the general code. When the same capabilities that are currently available in the 1A LMR version are made compatible with version 1.10, the new LMR code will be known as CONTAIN-LMR/1B.

Some of the sodium specific features described here are actually part of the standard version of CONTAIN. Their presence in the code dates back to the original versions, which provided both LMR and LWR analysis capability. This dual capability of standard CONTAIN has been maintained, and modifications to these LMR-specific models developed in the course of LMR applications have been incorporated into the standard code when this could be done easily. In this way the overhead of maintaining a separate set of updates for the LMR version is reduced. However, there will always be some capabilities that will be unique to the LMR version of the code. Examples are sodium-concrete interactions and the current set of debris bed models.

A brief description is given here of the sodium-LMR-specific models in CONTAIN-LMR/1A. The debris bed models will be described in somewhat more detail to demonstrate the type of integrated model application that is a primary goal in the development of CONTAIN-LMR.

A. Sodium Chemistry

The effects of chemical reactions of oxygen and water vapor with sodium vapor and other airborne sodium compounds are modeled. Sodium aerosols are also allowed to react with any water vapor present. Sodium spray fires are treated using algorithms obtained from the NACOM[3] code and improved by workers at Power Reactor and Nuclear Fuel Development Corporation (PNC) in Japan and Kernforschungszentrum Karlsruhe (KfK) in Germany. Also modeled are the effects of sodium pool fires using the model present in the SOFIRE II[4] computer code. Energy addition to both the atmosphere and the sodium pool and aerosol production may result from the application of this model. Several user-specified parameters control the model's operation thus accounting for the uncertainty in some of the physical processes. Finally, chemical reactions for gases that enter the sodium pool from either sodium-concrete or debris-concrete interactions are modeled.

B. Cavity Phenomena

Models employed in the code to account for the presence of liquids and solids at the bottom of a compartment are collectively referred to as lower cell models. The lower cell is modeled as a series of one-dimensional layers of concrete, metals and/or oxides and coolant materials.

Phenomena modeled in the lower cell include heat transfer among the various layers, sodium pool boiling, condensation and evaporation at the sodium pool surface, radiative energy exchange between the contents of the cavity and the atmosphere and structures in a compartment, debris-concrete interactions and sodium-concrete interactions. The latter two models are discussed next.

C. <u>Debris-Concrete Interactions</u>

The interactions of molten core debris with concrete following a core melt accident with primary vessel breach are modeled with an imbedded version of the CORCON-MOD2[5] code. CORCON is based on a two-dimensional axisymmetric model of the thermal attack of molten corium on the concrete surfaces of the reactor cavity, including the evolution of gases from the ablated concrete and reactions of these gases with metallic species in the melt. Some of these gases are combustible and represent a potentially significant threat to containment integrity. Heat transfer from the top surface of the corium to an overlying sodium pool is modeled with a set of sodium-specific boiling curve correlations. Integration of the CORCON model into CONTAIN's system level treatment has allowed for the possibility of feedback effects and more consistent analysis than is the case when using side calculations with the stand-alone CORCON code.

As it is currently written, CORCON is most directly applicable to LWR scenarios. There is no provision for sodium chemistry or for any of the by-products of such reactions. The presence of some metallic components of LMR cores is not treated in the CORCON chemistry package. While more work remains to be done in this area, CORCON represents the best available tool that currently exists.

D. <u>Sodium-Concrete Interactions</u>

Sodium-concrete interactions consist of ablation, caused by chemical attack on the concrete by sodium, and concrete outgassing of evaporable water, chemically bound water and CO_2. The models used are those developed and verified in the SLAM[6] code. This code uses a one-dimensional, three-region treatment that includes wet (hydrated) and dry (dehydrated) concrete zones and a boundary layer at the interface of the ablating concrete and the sodium pool. One-dimensional mass, momentum and energy equations in each region are included, as well as a chemical reaction model to treat reactive species in the boundary layer and those introduced into the pool by either sodium ablation or debris-concrete interactions.

The incorporation of the SLAM model was a joint effort by Sandia and PNC and represented a major improvement in the capabilities of CONTAIN-LMR/1.

E. Debris Bed Models

Recently, improvements have been made by SNL to include debris bed models into CONTAIN-LMR/1A. These models allow the treatment of both coolable and uncoolable beds. Mechanistic detail is not the goal of these models. They are included in the LMR code in order to provide a transition through a major branch point in the evolution of a severe accident scenario. This point is the initial quench of a coolable bed or the melting of an uncoolable bed. While of great significance to the outcome of an accident, the event itself may represent a relatively short period of time in the entire history of the accident. For example, the entire accident analysis may be concerned with time scales on the order of weeks, while the quench of the bed may take place in minutes. The models in CONTAIN-LMR/1A are intended to provide the transition between the initial bed state and its final configuration in as smooth a manner as possible while not producing physically unreasonable results. Even given this limited goal, there are many possible scenarios that must be taken into account.

The debris bed coolability treatment is based on work of Lipinski.[7] The heat flux at the bed surface, defined as the ratio of the volumetric power developed in the bed to the bed cross-sectional area, is compared with the dryout heat flux computed with the Lipinski model. If this dryout heat flux is exceeded, the bed is treated as uncoolable. This approximation is done in the sake of simplicity and reflects an assumption that if any portion of the bed dries out, the entire bed will remelt on a time scale which is small compared to the entire accident duration.

After a determination that the bed is uncoolable, control is passed to the CORCON model. This transition is simplified by the fact that CORCON property modules are employed in the bed model, thus eliminating a transition between two material data bases. CORCON may also take control of the bed if the computed average bed temperature exceeds a user-specified value, or if the problem time has exceeded a user-specified transition time. Once CORCON obtains control of the material, the bed model is removed from the modeling sequence. The temperature and composition of the materials passed to CORCON may be such that solid layers are formed. Melting of these layers due to decay heating is treated by the CORCON model.

For coolable debris beds, a bed quench model is included to allow for the transition from a dry bed to the quenched state. Only top down quenching is treated in this simple model. The downward quench velocity is determined by the excess of the dryout heat flux over the actual bed surface heat flux.[8] This excess is used to quench material at the moving front. Sodium boiloff is allowed with possible uncovering of the bed as the pool level drops.

During quench separate temperatures are kept for the dry region of the bed, the wet region, and any upper area which has been exposed to the atmosphere by a dropping pool level. Heat transfer is computed between the sodium pool and the wet region of the bed. Also, any exposed region at the top of the bed exchanges heat with the atmosphere. Restrictions in the CONTAIN version in which the LMR/1A model is implemented prevent modeling of heat transfer between the bed and the concrete layer and among the bed regions themselves. This limitation will be removed when the transition to CONTAIN 1.10 is made and the new lower cell heat conduction algorithm is available.

Figure 1 presents a stylized picture of the debris bed model. The bed is completely submerged by the coolant pool and the quench front has progressed about halfway down through the bed. While the bed model is operating, the SLAM sodium concrete ablation model is eroding the concrete region not covered by the bed. Gases from this interaction may undergo chemical reaction in the sodium pool. Figure 2 presents the most likely outcome of the bed quench scenario. In this case, the bed is entirely quenched and the decay heat developed in the bed contributes to boiling sodium from the pool. The pool and the bed are at the saturation temperature, which may be changing slowly due to atmospheric pressurization. As the temperature of the pool correspondingly changes, some of the bed decay energy must be used to heat the bed along with the pool.

In Figure 3 the pool surface level has dropped due to removal of sodium from the cavity cell system by some mechanism. This could perhaps be the sodium-concrete interaction, which results in the conversion of liquid sodium metal to other compounds. The exposed bed region above the pool level exchanges heat with the atmosphere. Figure 4 shows the situation just prior to bed dryout. After the bed is completely devoid of sodium, decay heat acts to raise the temperature. At the remelt point, determined by the user-specified criteria, the entire bed contents are passed to the CORCON model.

The situation with CORCON active is shown in Figure 5. The material is assumed to occupy the full diameter of the cavity. The picture in Figure 5 shows that some ablation has taken place, eroding the bottom and sides of the cavity. Should it be desired, the sodium-concrete interaction can still be modeled at the walls of the cavity. Also shown in Figure 5 is a sodium coolant pool overlying the corium layers. Heat transfer to this pool is modeled as well.

The most important aspect of the incorporation of the models discussed above into CONTAIN is the simultaneous and coordinated nature of their operation. While debris bed processes or debris-concrete interactions are taking place, the ablation of cavity surfaces by sodium may still be active. The models make use of common modules whenever possible and are effectively subcomponents of a coherent picture of cavity phenomena in LMR severe accidents.

The coupling of pool boiling processes and condensation on structures provides a specific example of the advantages of model integration. Sodium vapor introduced into the compartment atmosphere from the boiling pool will condense on structures and reenter the pool due to drainage. This sodium liquid will have a lower temperature than the pool and thus will cool the pool-bed system. This cycling of sodium between the pool and structures may represent an important mechanism for removing heat from the debris bed.

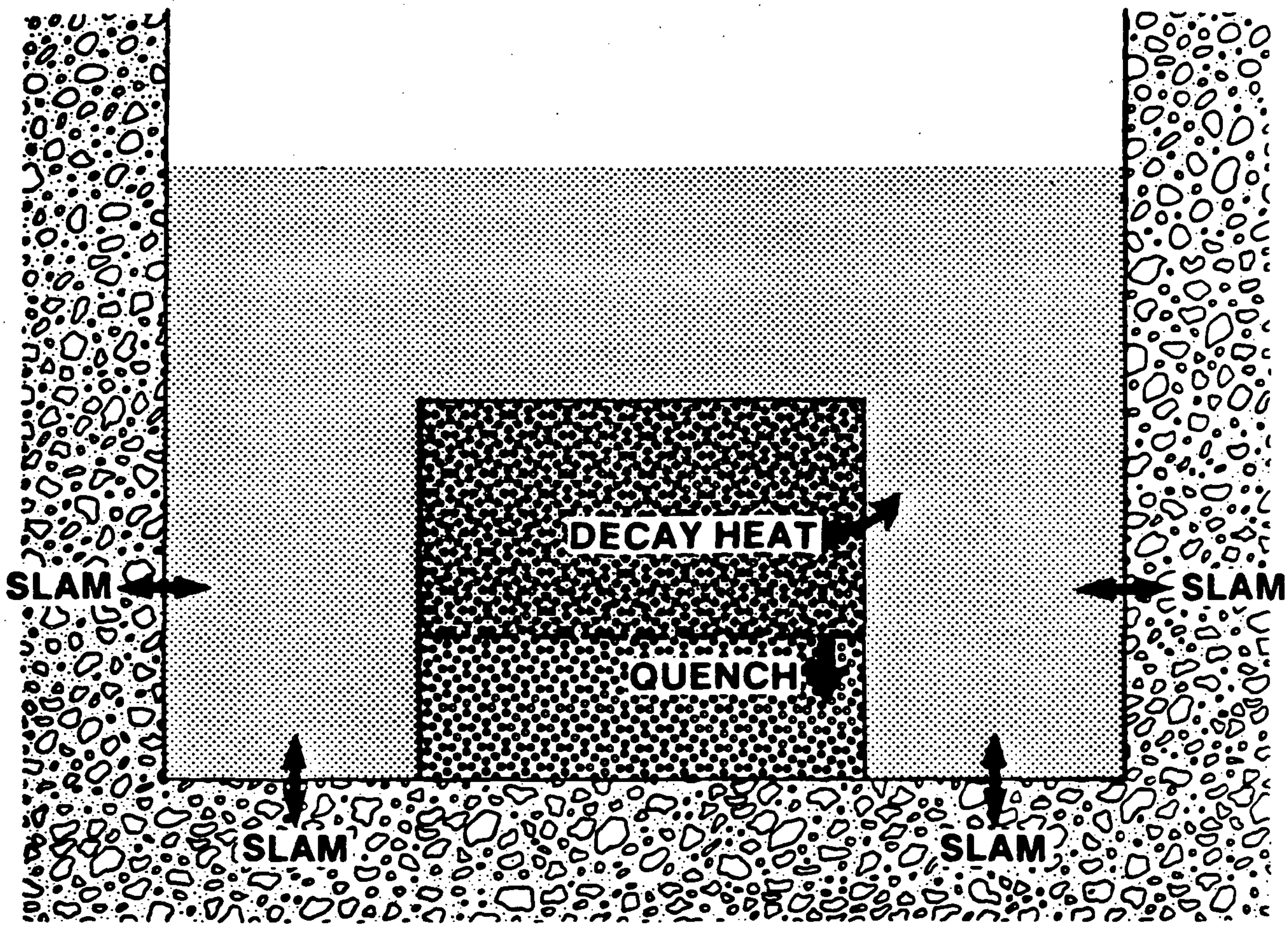

Figure 1. CONTAIN-LMR debris bed configuration during the initial quench of a dry bed. Sodium-concrete interactions are modeled on those cavity surfaces not covered by the debris bed. Light stipling indicates the sodium pool inventory.

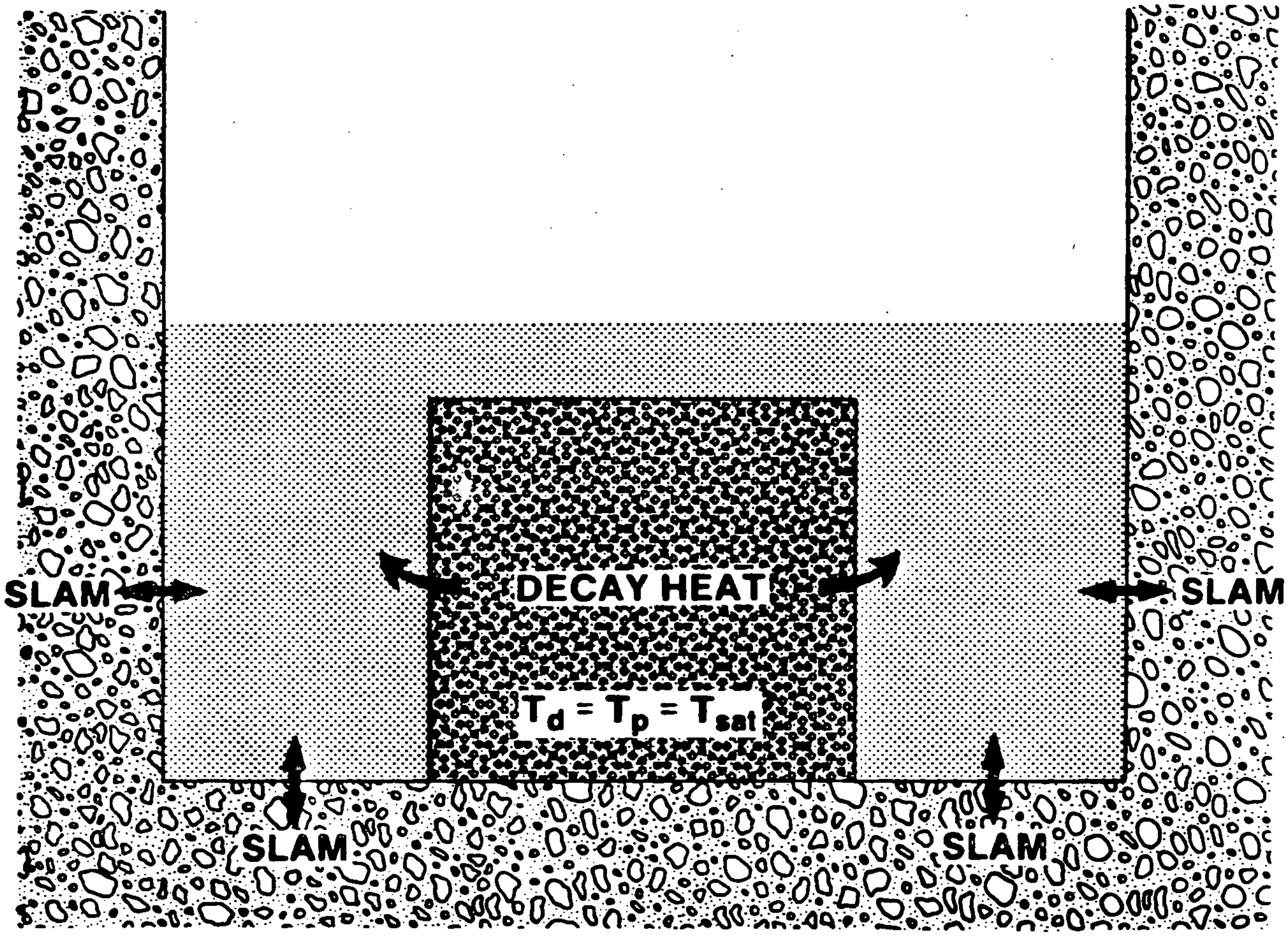

Figure 2. Debris bed configuration after complete quench. T_d is the temperature of the wet region of the bed, T_p is the pool temperature, and T_{sat} is the saturation temperature for sodium vapor.

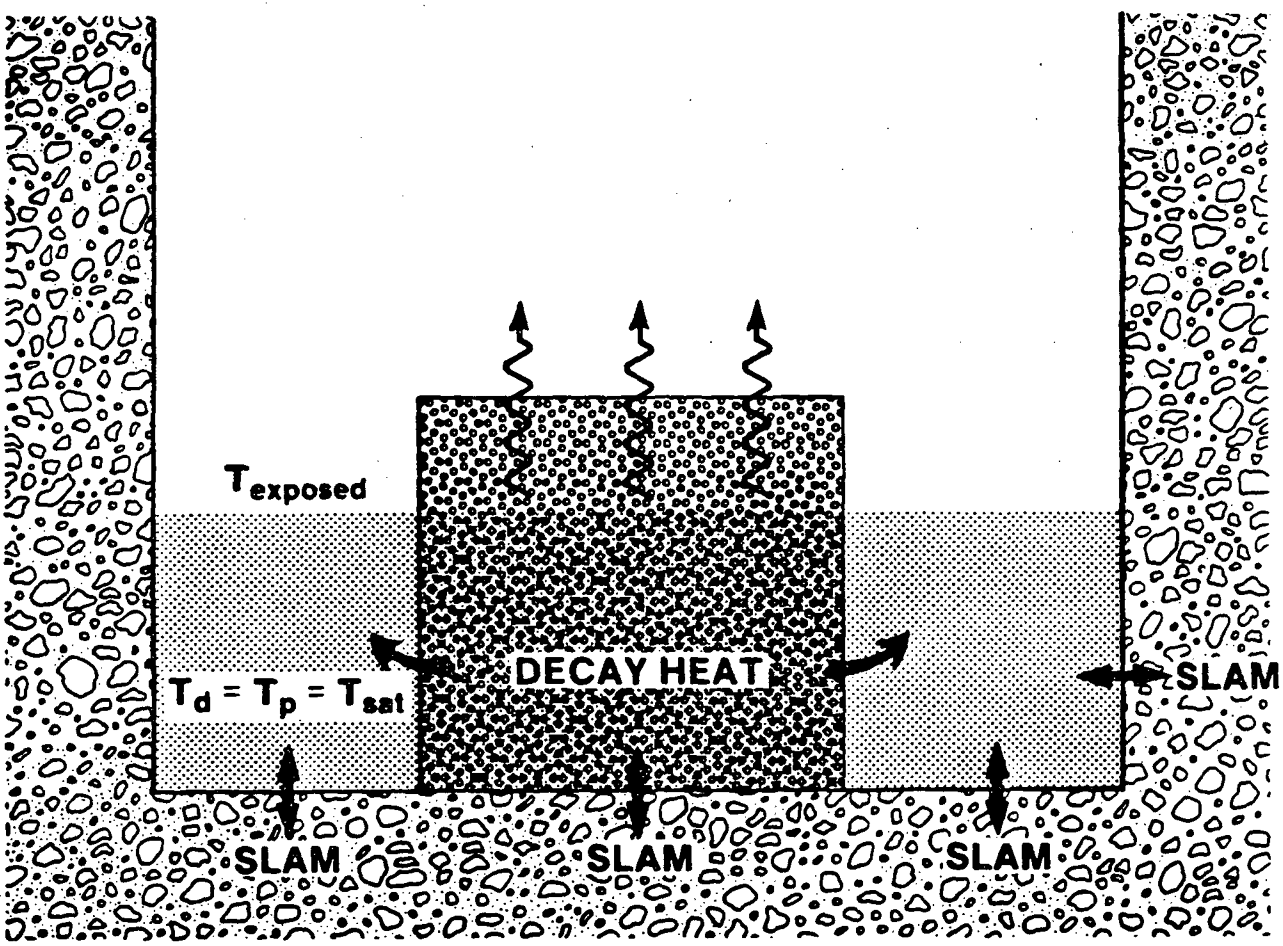

Figure 3. Debris bed configuration after the formation of an exposed region at the top of the bed, due to a decreasing sodium pool level. The temperature of this region is $T_{exposed}$. The arrows in the exposed bed region represent sodium vapor rising through the bed.

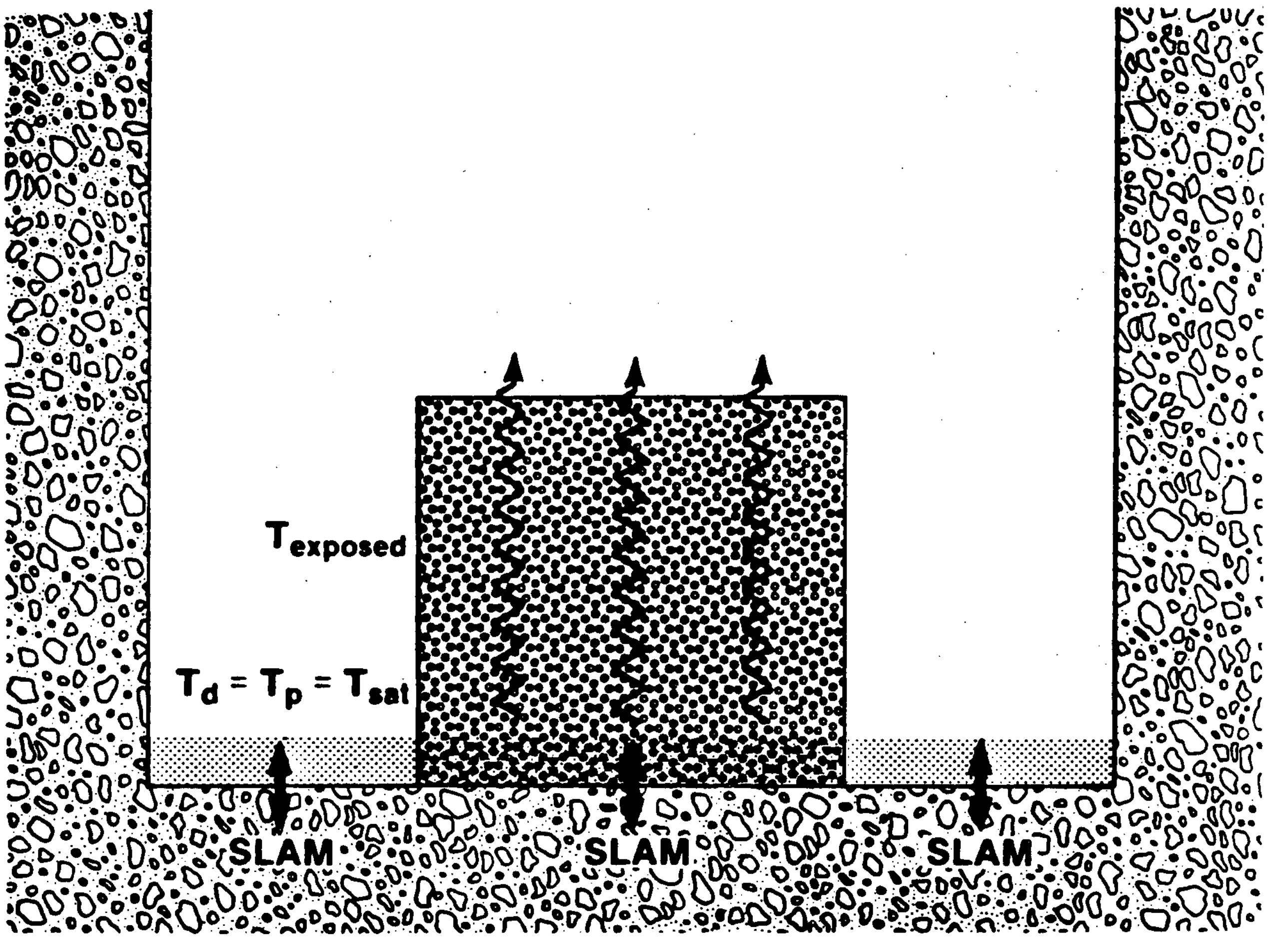

Figure 4. Debris bed configuration immediately before the final dryout of the sodium pool.

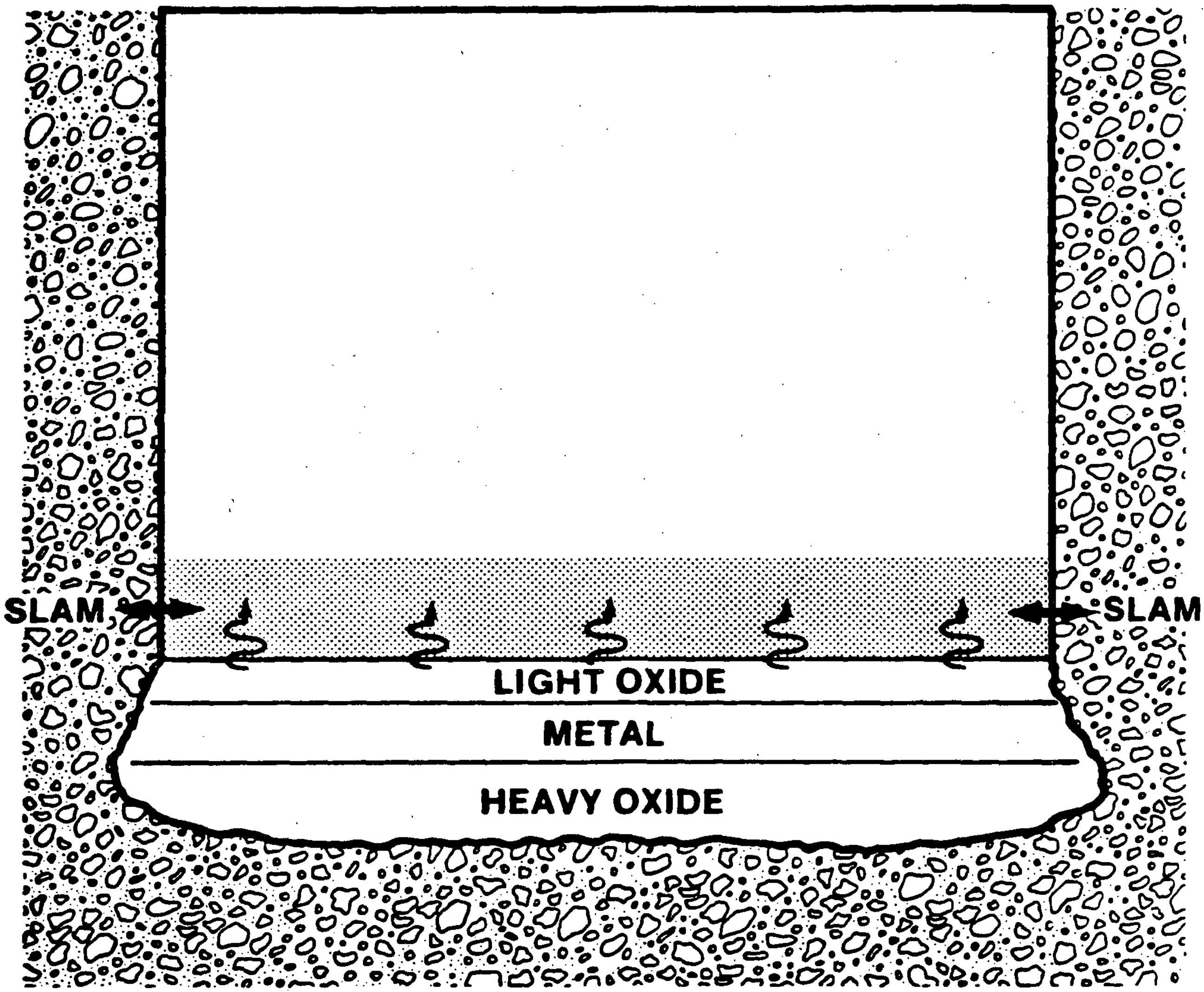

Figure 5. CONTAIN-LMR cavity model for debris-concrete interactions with the CORCON model. Some ablation has taken place and sodium-concrete interactions are proceeding at the cavity sidewalls. A sodium pool is present above the corium material. Arrows in the pool represent gases resulting from ablation.

III. APPLICATIONS OF CONTAIN-LMR/1A

The integrated nature and the broad spectrum of models available make CONTAIN-LMR/1A well suited for analysis of accidents, ranging from relatively benign scenarios to severe core-melt accidents involving release of radioactive materials to the environment. While being most directly applicable for those situations where a traditional containment is utilized, CONTAIN-LMR may also be useful for applications in which there is no conventional containment building, per se. This is so because the phenomena of sodium fires, aerosol transport and deposition, and natural convection are critical to the source term at any time the pathway to the environment is indirect (e.g., through reactor buildings). For example, in many cases, best-estimate analyses with CONTAIN have shown that even after the containment has failed, deposition processes in the containment and connected buildings can have large decontaminating effects on the source term. Experience with CONTAIN on LWR accident analysis has shown that the synergism among the phenomena can lead to results that would be difficult to predict with nonintegrated analysis tools.

CONTAIN is being used in Germany in an application with a prototype LMR plant. Using only the sodium-specific features in the standard version of the code, and some local modifications to model plant specific features, a calculation has been performed by workers at KfK for a complete 12-day accident scenario.[9] Similar work is taking place in Japan, where CONTAIN-LMR/1A is being used in analysis of the Monju reactor plant by workers at PNC. Implementation of the new capabilities in CONTAIN-LMR (e.g., debris bed models and debris-concrete interactions) allows more mechanistic treatments of many accidents than has been possible in the past.

V. CONCLUSION

If U.S. LMR plants of the future are to utilize containment features, a tool will be needed for severe accident analysis. Such a tool has been developed in the CONTAIN-LMR code as a result of international collaboration. This tool is based upon the containment analysis code which has been widely accepted in the worldwide reactor safety community. CONTAIN-LMR provides an integrated treatment of the phenomena that may occur within the containment during a severe accident and includes several sodium-specific models that can not be found in any other integrated safety analysis tool.

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This five-volume report contains 141 papers out of the 175 that were presented at the Sixteenth Water Reactor Safety Information Meeting held at the National Institute of Standards and Technology, Gaithersburg, Maryland, during the week of October 24-27, 1988. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included twenty different papers presented by researchers from Germany, Italy, Japan, Sweden, Switzerland, Taiwan and the United Kingdom. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

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