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Proceedings of the Twenty-Eighth Water Reactor Safety Information Meeting

Held at
Bethesda Marriott Hotel
Bethesda, Maryland
October 23–25, 2000

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research

Proceedings prepared by
Brookhaven National Laboratory

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S. Nesmith, NRC Project Manager

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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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ABSTRACT

This report contains papers presented at the 28th Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel in Bethesda, Maryland, October 23-25, 2000. The papers for the Plenary Sessions are included first, followed by the papers presented in each of the eight breakout sessions conducted over the course of the three days. They describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from France, Germany, Japan, and Norway.

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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REGISTERED ATTENDEES
28TH WATER REACTOR SAFETY INFORMATION MEETING

E. ABELQUIST
ORISE
112 LOCUST LN.
OAK RIDGE, TN 37830
USA
Phone: 865-576-3740
Fax: 865-241-3497
E-Mail: abelquie@ornl.com

N.S. AHN
KEPRI
103-16 MUNJI-DONG YUSUNG-KU
TAEJON, 305-380
KOREA
Phone: 8242 865 5670
Fax: 82 42 865 5504
E-Mail: nsahn@kepri.re.kr

K. AINGER
COMMONWEALTH EDISON
1400 OPUS PLACE, SUITE 500
DOWNERS GROVE, IL 60515
USA
Phone: 630 663 7350
Fax: 630 663 7155
E-Mail: kenneth.a.ainger@ucm.com

A. ALAPOUR
SOUTHERN NUCLEAR
2087 STONE BROOK DR.
BIRMINGHAM, AL 35242
USA
Phone: 205-992-7608
Fax: 205-992-5536
E-Mail: aalapour@southernco.com

T. ALLEN
ARGONNE NATIONAL LABORATORY WEST
19901 GERMANTOWN RD
GERMANTOWN, MD 20874
USA
Phone: 301 903 1647
Fax: 301 903 5057
E-Mail: todd.allen@hq.doe.gov

A. ALONSO
NUCLEAR SAFETY COUNCIL
JUSTO DORADO 11
MADRID, 28040
SPAIN
Phone: 34 91 346 0334
Fax: 34 91 346 0378
E-Mail: aas@csn.es

T. ANDREYCHEK
WESTINGHOUSE
PO BOX 355
PITTSBURGH, PA 15230
USA
Phone: 412 374 6246
Fax: 412 374 5099
E-Mail: andreyts@westinghouse.com

T. ANGERHOFER
NAVAL REACTORS
JEFFERSON DAVIS HIGHWAY
ARLINGTON, VA
USA
Phone: 703 602 8995
Fax: 703 602 7197
E-Mail: angerhoferte@navsea.navy.mil

S. ANGHAIE
UNIV. OF FLORIDA
202 NSC
GAINESVILLE, FL 32611-8300
USA
Phone: 352 392 8653
Fax: 352 392 8656
E-Mail: anghaie@ufl.edu

A. ANKRUM
PACIFIC NORTHWEST NATIONAL LAB
PO BOX 999
RICHLAND, WA 99352
USA
Phone: 509 372 4095
Fax: 509 372 6242
E-Mail: alvin.ankrum@pnl.gov

G. APOSTOLAKIS
MASS. INST. OF TECHNOLOGY
ROOM 24-221
CAMBRIDGE, MA 02139-4307
USA
Phone: 617 252 1570
Fax: 617 258 8863
E-Mail: apostola@mit.edu

V. ASMOLOV
RRC, KURCHATOV INSTITUTE
KURCHATOV SQ. 1
MOSCOW, 123182
RUSSIA
Phone: 196 9320
Fax: 196 1702
E-Mail: asmolov@nsi.kiae.ru

G.S. AUH
KOREA INSTITUE OF NUCLEAR SAFETY
PO BOX 114
YUSONG, TAEJON 305-600
KOREA
Phone: 82 42 868 0466
Fax: 82 42 861 2535
E-Mail: k332ags@kins.re.kr

K. BAHR
WESTINGHOUSE
PO BOX 355
PITTSBURGH, PA 15230
USA
Phone: 412 374 2149
Fax: 412 374 2045
E-Mail: bahrke@westinghouse.com

R. BARI
BROOKHAVEN NATIONAL LABORATORY
PO BOX 5000
UPTON, NY 11973-5000
USA
Phone: 631 344 2629
Fax: 631 344 1430
E-Mail: bari@bnl.gov

R. BEEDLE
NUCLEAR ENERGY INSTITUTE
1776 I ST., NW
WASHINGTON, DC 20006-3708
USA
Phone: 202 739 8088
Fax:
E-Mail: rb@nei.org

E. BEK
PJSC RUSSIA
KARL MARX ST. 12
ELECTROSTAL, MOS. ZEG. 144001
RUSSIA
Phone: 095 702 97 31
Fax: 095 702 98 66
E-Mail: e1msz@do1.ru

R. BENNETT
INEEL
PO BOX 1625
IDAHO FALLS, ID 83415-3860
USA
Phone: 208 526 7708
Fax:
E-Mail: rcb@inel.gov

V. BESSIRON
IPSN
CE CADARACHE, BAT 702
ST PAUL LEZ DURANCE, 13108
FRANCE
Phone: 33 44225766
Fax: 33 442256143
E-Mail: vincent.bessiron@ipsn.fr

C. BEYER
PACIFIC NORTHWEST NATIONAL LAB
PO BOX 999
RICHLAND, WA 99352
USA
Phone: 509-372-4605
Fax: 509-372-6485
E-Mail:

M. BILLONE
ARGONNE NATIONAL LABORATORY
BLDG 212, 9700 S. CASS AVE.
ARGONNE, IL 60439
USA
Phone: 630 252 7146
Fax: 630 252 9232
E-Mail: billone@anl.gov

A. BIRKHOFER
GRS
FORSCHUNGSGELAENDE
GARCHING, BAVARIA 85748
GERMANY
Phone: 49 89 32 00 41 00
Fax: 49 89 32 00 45 00
E-Mail: bir@grs.de

N. BIXLER
SANDIA NATIONAL LABS
PO BOX 5800
ALBUQUERQUE, NM 87185
USA
Phone: 505-845-3144
Fax: 505-844-8709
E-Mail: nbixler@sandia.gov

T. BLEJWAS
SANDIA NATIONAL LABS
PO BOX 5800
ALBUQUERQUE, NM 87185
USA
Phone: 505 844 6005
Fax: 505 844 9066
E-Mail: teblejwas@snl.gov

B. BOYACK
LOS ALAMOS NATIONAL LAB
PO BOX 1663
LOS ALAMOS, NM 37545
USA
Phone: 505 667 2023
Fax: 505 665 2897
E-Mail: bboyack@lanl.gov

B. BRAND
SIEMENS NUCLEAR POWER GmbH
FREYESLEBEN STR. 1
ERLANGEN, BAYERN D91058
GERMANY
Phone: 49 9131 18 2593
Fax: 49 9131 18 2851
E-Mail: Bernhard.Brand@erls11.siemens.de

J. BRANDES
81 IRVING AVE
ATHERTON, CA 94027
USA
Phone: 650 324 0321
Fax:
E-Mail:

T. BROOKMIRE
DOMINION GENERATION
5000 DOMINION BLVD.
GLEN ALLEN, VA 23060
USA
Phone: 804 273 3243
Fax: 804 273 3543
E-Mail: tom_brookmire@dom.com

G. BROWN
AEA TECHNOLOGY
THOMSON HOUSE, RISLEY
WARRINGTON, CHESHIRE,
UK
Phone: 44 1925 254473
Fax: 44 1925 254536
E-Mail: geoff.brown@aeat.co.uk

R. BUDNITZ
FUTURE RESOURCES ASSOCIATES, INC
2039 SHATTUCK AVE., SUITE 402
BERKELEY, CA 94704
USA
Phone: 510 644 2700
Fax: 510 644 1117
E-Mail: budnitz@PACBELL.net

J. BUTRAGUENO
NUCLEAR SAFETY COUNCIL
C/JUSTO DORADO, 11
MADRID, 28040
SPAIN
Phone: 34 91 3460 407
Fax: 34 91 3460 588
E-Mail: jlbc@csn.es

M. CALLAHAN
GSI
12 4TH ST. SE
WASHINGTON, DC 20004
USA
Phone: 202-544-4522
Fax: 202-544-7522
E-Mail:

A. CAMP
SANDIA NATIONAL LABS
PO BOX 5800
ALBUQUERQUE, NM 87185-0747
USA
Phone: 505 844 5960
Fax: 505 844 1648
E-Mail: alcamp@sandia.gov

N. CANTORO
FABRICATZIONI NUCLEARI SPA.
55-35 BIS DEI GLOVI - KM 15 - BOSCO
HDRENGO
ACESSANDRIA, ITALLY
ITALY
Phone: 0131-247223
Fax: 0131-247315
E-Mail: cantopo.nicola@libero.it

J. CAVALLO
CORROSION CONTROL LABS
PO BOX 226
ELIOT, ME 03903
USA
Phone: 207-439-2464
Fax: 207-439-2969
E-Mail: jrcpe@aol.com

B. CAZALIS
IPSN
CE CADARACHE, BAT 702
ST PAUL LEZ DURANCE, 13108
FRANCE
Phone: 33 442256516
Fax: 33 442256143
E-Mail:

Y.S. CHANG
KOREA POWER ENGINEERING CO.
360-9, MABUK-RI, JUSONG-MYON
YONGIN-SHI, KYUNGGI-DO 449-713
KOREA
Phone: 82 31 289 4283
Fax: 82 31 289 3189
E-Mail: cys@ns.kopec.co.kr

S.Y. CHEN
ARGONNE NATIONAL LABORATORY
EID-900, 9700 SO. CASS AVE.
ARGONNE, IL 60439
USA
Phone: 630 252 7695
Fax: 630 252 4611
E-Mail: sychen@anl.gov

W. CHOE
TXU ELECTRIC
1601 BRYAN ST.
DALLAS, TX 75201
USA
Phone: 214-812-4371
Fax: 214-812-8687
E-Mail: whee.choe@txu.com

H. CHUNG
ARGONNE NATIONAL LABORATORY
9700 S. CASS
ARGONNE, IL 60439
USA
Phone: 630 252 5111
Fax: 630 252 3604
E-Mail: heechung@anl.gov

J. COLVIN
NUCLEAR ENERGY INSTITUTE
1776 I ST., NW, STE 400
WASHINGTON, DC 20006-3708
USA
Phone: 202 739 8080
Fax: 202 533 0164
E-Mail: am@nei.org

J. CONDE
CONSEJO DE SEGURIDAD NUCLEAR
JUSTO DORADI, 11
MADRID, 28040
SPAIN
Phone: 34 91 3460253
Fax: 34 91 3460588
E-Mail: jmcl@csn.es

B. COPSEY
FRAMATOME
3315 OLD FOREST RD
LYNCHBURG, VA 24551
USA
Phone: 804 832 3176
Fax:
E-Mail: acopsey@framatech.com

J.L. CORTEZ
EMR&T
10101 GROSVENOR PL. #1516
N. BETHESDA, MD 20852
USA
Phone: 202-425-5435
Fax: 301-564-9515
E-Mail: jlmcortez@aol.com

S. CRAWFORD
6722 SHADOW OAK DRIVE
GAITHERSBURG, MD 20866
USA
Phone:
Fax:
E-Mail:

G. CUMMINGS
LAWRENCE LIVERMORE NATIONAL LAB
PO BOX 808, L-05
LIVERMORE, CA 94550
USA
Phone: 925 423 6268
Fax: 925 422 4116
E-Mail: cummings2@llnl.gov

M. CUNNINGHAM
PACIFIC NORTHWEST NATIONAL LAB
PO BOX 999
RICHLAND, WA 99352
USA
Phone: 509 372 4987
Fax: 509 372 4989
E-Mail: mitch.cunningham@pnl.gov

G. DARDEN
DOMINION GENERATION
5000 DOMINION BLVD 3SW
GLEN ALLEN, VA 23060
USA
Phone: 804 273 3497
Fax: 804 273 3543
E-Mail: gary_darden@dom.com

K. DAVIS
MPR ASSOCIATES
320 KING STREET
ALEXANDRIA, VA 22303
USA
Phone: 703-519-0260
Fax: 703-519-0224
E-Mail: kdavis@mpi.com

P. DE LOZIER
DE LOZIER CONSULTING, INC.
3202 GEMSTONE CT.
OAKTON, VA 22024
USA
Phone: 703 591 1684
Fax: 703 591 2292
E-Mail: nucexprt@laser.net

R. DENNING
BATTELLE/COLUMBUS
505 KING AVE.
COLUMBUS, OH 43201
USA
Phone: 614 424 7412
Fax: 614 424 3404
E-Mail: denning@battelle.org

D. DIAMOND
BROOKHAVEN NATIONAL LABORATORY
BUILDING 130, PO BOX 5000
UPTON, NY 11973-5000
USA
Phone: 631-344-2604
Fax: 631-344-3957
E-Mail: diamond@bnl.gov

B. DUNN
FRAMATOME TECHNOLOGIES INC
PO BOX 10935
LYNCHBURG, VA 24506-0935
USA
Phone: 804 832 2427
Fax: 804 832 2531
E-Mail: bdunn@framatech.com

M. DUPONT
WESTINGHOUSE
SAVANNAH RIVER SITE
AIKEN, SC 29808
USA
Phone: 803 725 0954
Fax:
E-Mail: mark.dupont@srs.gov

N. EISENBERG
UNIV. OF MARYLAND
1208 HARDING LANE
SILVER SPRING, MD 20505
USA
Phone: 301-384-6507
Fax: 301-314-9601
E-Mail: nae@bellatlantic.net

M. EL-HAWARY
CANADIAN NUCLEAR SAFETY COMMISSION
280 SLATER ST
OTTAWA, ONTARIO K1P5S9
CANADA
Phone: 613 943 0131
Fax: 613 943 1292
E-Mail: elhawarym@cnsc-ccsn.gc.ca

T. FABIAN
NUCLEAR WASTE NEWS
8737 COLESVILLE RD, SUITE 1100
SILVER SPRING, MD 20910
USA
Phone: 301-587-6300 X362
Fax: 301-587-1081
E-Mail: tfabian@pbinews.com

M. FAURY
IPSN
BAT 721, CEN CADARACHE
ST PAUL LEZ DURANCE, 13108
FRANCE
Phone: 33 44 225 4124
Fax: 33 44 225 3555
E-Mail: maria.faury@ipsn.fr

M. FELTUS
U.S. DEPT. OF ENERGY
1000 INDEPENDENCE AVE
WASHINGTON, DC 20585
USA
Phone: 202 586 0843
Fax:
E-Mail: madeline.feltus@hq.doe.gov

D. FITZGERALD
CONSULTANT
3601 CONNECTICUT AVE NW #510
WASHINGTON, DC 20008
USA
Phone: 202 237 2332
Fax: 202 237 2333
E-Mail: DFitzgrld@aol.com

S. FLOYD
NUCLEAR ENERGY INSTITUTE
1776 I ST., NW, STE 400
WASHINGTON, DC 20006-3708
USA
Phone:
Fax:
E-Mail:

S. FRANKL
NAC INTERNATIONAL / SNF
485 WASHINGTON AVE
PLEASANTVILLE, NY 10570
USA
Phone: 914 741 1200
Fax: 914 741 2093
E-Mail: ifrankl@nacintl.com

M. FUJITA
KANSAI ELECTRIC POWER
2001 L ST., NW, SUITE 801
WASHINGTON, DC 20036
USA
Phone: 202 659 1138
Fax: 202 457 0272
E-Mail: mfujita@kansai.com

T. FUKETA
JAERI
TOKAI-MURA
IBARAKI-KEN, 319-1195
JAPAN
Phone: 81 29 282 6386
Fax: 81 29 282 5429
E-Mail: toyo@nsrr.tokai.jaeri.go.jp

F. GANTENBEIN
IPSN
BP 6
FONTENAY AUX ROSES, 92265
FRANCE
Phone: 33 1 46548177
Fax: 33 1 4654 8925
E-Mail: francoise.gantenbein@ipsn.fr

J. GARRICK
GARRICK CONSULTING
221 CRESCENT BAY DR.
LAGUNA BEACH, CA 92651
USA
Phone: 949 497 6802
Fax: 949 497 6072
E-Mail: bjgarrick@aol.com

R. GAUNTT
SANDIA NATIONAL LABS
13501 GRANADA HILLS DR. NE
ALBUQUERQUE, NM 87123
USA
Phone: 505-284-3989
Fax:
E-Mail: rogaunt@sandia.gov

G. GAYDOS
BECHTEL POWER CORP.
5275 WESTVIEW DR.
FREDERICK, MD 21703
USA
Phone: 301 228 6534
Fax: 301 631 0936
E-Mail: ggaydos@bechtel.com

P. GENOA
NUCLEAR ENERGY INSTITUTE
1776 I ST., NW, SUITE 400
WASHINGTON, DC 20006
USA
Phone: 202 739 8034
Fax: 202 785 1898
E-Mail: phg@nei.org

J. GILMER
BECHTEL POWER CORP.
5325 SPECTRUM DR.
FREDERICK, MD 21703
USA
Phone: 301 228 6422
Fax: 301 631 0841
E-Mail: jgilmer@bechtel.com

C. GOGOLAK
U.S. DEPT OF ENERGY
201 VARICK ST.
NEW YORK, NY 10014
USA
Phone: 212-620-3635
Fax: 212-620-3600
E-Mail: cvg@eml.doe.gov

W. GOLL
SIEMENS NUCLEAR POWER / NBTW
PO BOX 3220
ERLANGEN, D-91050
GERMANY
Phone: 49 9131 184974
Fax: 49 9131 185751
E-Mail: wolfgang.goll@erl19.siemens.de

C. GRANDJEAN
IPSN
DRS/SEMAR, CE CADARACHE
ST PAUL LEZ DURANCE, 13108
FRANCE
Phone: 33 4 4225 4480
Fax: 33 4 4225 6143
E-Mail: claude.grandjean@ipsn.fr

W. GRANT
CANADIAN NUCLEAR SAFETY COMMISSION
280 SLATER, PO BOX 1046 STN. "B"
OTTAWA, ONTARIO K1P5S9
CANADA
Phone: 613-943-5885
Fax: 613-995-5086
E-Mail: grantw@cnsc.ccsn.gc.ca

C. GUEY
FPL
ENG / JB. 700 UNIVERSE BLVD.
JUNO BEACH, FL 33408
USA
Phone: 561-694-3137
Fax: 561-694-5090
E-Mail: ching.guey@fpl.com

B. GUILLEMARD
IPSN
BP 6
FONTENAY-AUX-ROSES, 92265
FRANCE
Phone: 33 1 4654 7172
Fax: 33 1 4654 8925
E-Mail: bernard.quillemard@ipsn.fr

A. GUPTA
NORTH CAROLINA STATE UNIV.
410 MANN HALL, CAMPUS BOX 7908
RALEIGH, NC 27695-7908
USA
Phone: 919 515 7338
Fax: 919 515 5301
E-Mail: gupta@eos.ncsu.edu

J. GUTH
SANDIA NATIONAL LABS
PO BOX 5800, DEPT 6401/MS0742
ALBUQUERQUE, NM 87185-0742
USA
Phone: 505 845 8791
Fax: 505 844 3321
E-Mail: jrguth@sandia.gov

G. HACHE
IPSN
DRS-BATIMENT 250-CADARACHE
ST PAUL LEZ DURANCE CEDEX, 13108
FRANCE
Phone: 33 442252055
Fax: 33 442257679
E-Mail: georges.hache@ipsn.fr

R. HARDIES
CCNPPI
1650 CALVERT CLIFFS PARKWAY
LUSBY, MD 20657
USA
Phone: 410-495-6577
Fax: 410-495-6937
E-Mail: robert.o.hardies@ccnppi.com

C. HARRINGTON
TXU/COMANCHE PEAK
PO BOX 1002
GLEN ROSE, TX 76043
USA
Phone: 254 897 6705
Fax: 254 897 0530
E-Mail: charrin1@txu.com

D. HARRISON
U.S. DEPT. OF ENERGY
19901 GERMANTOWN ROAD
GERMANTOWN, MD 20874
USA
Phone: 301-903-2884
Fax: 301-903-5057
E-Mail: dennis.harrison@hq.doe.gov

S. HAYES
DUKE POWER CO.
526 SO CHURCH ST., MC 3C08F
CHARLOTTE, NC 28201-1006
USA
Phone: 704 382 6786
Fax: 704 382 7852
E-Mail: sjhayes@duke-energy.com

L. HENDRICKS
NUCLEAR ENERGY INSTITUTE
1176 "EYE" ST. NW
WASHINGTON, DC 20006
USA
Phone: 202-739-8109
Fax: 202-785-1898
E-Mail: lxh@nei.org

A. HENRY
MPR ASSOCIATES
320 KING STREET
ALEXANDRIA, VA 22314
USA
Phone: 703-519-0200
Fax: 703-519-0224
E-Mail: ahenry@mpr.com

P. HIGHBERGER
J.A.I.
4103 CHAIN BRIDGE RD.
FAIRFAX, VA 22030
USA
Phone: 703-359-9355
Fax:
E-Mail:

E. HIRUO
McGRAW-HILL NUCLEAR PUBS
1200 G ST., NW, STE 1100
WASHINGTON, DC 20005
USA
Phone: 202 383 2163
Fax: 202 383 2125
E-Mail: ehiru@mh.com

D. HOFFMAN
EXCEL SERVICES CORP
11921 ROCKVILLE PIKE, #100
ROCKVILLE, MD 20852
USA
Phone: 301 984 4400
Fax: 301 984 7600
E-Mail: donaldh@excelservices.com

J. HOLM
SIEMENS POWER CORP.
2101 HORN RAPIDS RD
RICHLAND, WA 99352
USA
Phone: 509 375 8142
Fax: 509 375 8775
E-Mail: JerryS_Holm@nfuel.com

W. HOPKINS
BECHTEL POWER CORP.
5275 WESTVIEW DRIVE
FREDERICK, MD 21703-8306
USA
Phone: 301-228-6436
Fax: 301-631-0849
E-Mail: whopkins@bechtel.com

H. HUNTER
OAK RIDGE NATIONAL LAB
PO BOX 2008
OAK RIDGE, TN 37831-6362
USA
Phone: 865 574 6297
Fax:
E-Mail: h30@ornl.gov

J. IN de BETOU
SWEDISH NUCLEAR POWER INSPECTORATE
STOCKHOLM, SE 10658
SWEDEN
Phone: 46 8 698 8459
Fax:
E-Mail: jan@ski.se

R. JANATI
PA DEPT. OF ENVIRONMENTAL PROTECTION
PO BOX 8469, 400 MARKET ST
HARRISBURG, PA 17105
USA
Phone: 717 787 2163
Fax: 717 783 8965
E-Mail: janati.rich@dept.state.pa.us

C. JANG
KEPRI
103-16 MUNJI-DONG, YUSONG-GU
TAEJON, 305-380
REP. OF KOREA
Phone: 8242 865 5515
Fax: 8242 865 5514
E-Mail: chjang@kepri.re.kr

J. JANSKY
BTB JANSKY
71229 LEONBERG, GERLINGER STR. 151
LEONBERG,
GERMANY
Phone: 07 152 41058
Fax: 07152 73868
E-Mail: btb@btbjansky.com

S. JENSEN
SIEMENS POWER CORP.
2101 HORN RAPIDS RD
RICHLAND, WA 99352
USA
Phone: 509 375 8477
Fax: 509 375 8402
E-Mail: sej@nfuel.com

G. JOBSON
GNSI
COLUMBIA, SC
USA
Phone: 803 214 5878
Fax: 803 214 5801
E-Mail: gjobson@gnsi-gnb.com

B. JOHNSON
UNIV. OF VIRGINIA
CHARLOTTESVILLE, VA 22903
USA
Phone:
Fax:
E-Mail:

M. KAMITSUMA
JAPAN NUS CO. LTD
LOOP-x BLDG 7F, 3-9-15 KAIGAN, MINATO-KU
TOKYO, 1080022
JAPAN
Phone: 81 3 5440 1861
Fax:
E-Mail: mkamitsuma@janus.co.jp

S.C. KANG
KOREA INSTITUTE OF NUCLEAR SAFETY
19 GUSUNGDONG, YUSONGGU
TAEJON, 305-338
KOREA
Phone: 8242 868 0185
Fax: 8242 861 9945
E-Mail: kang@kins.re.kr

E. KAPLAR
RRC, KURCHATOV INSTITUTE
KURCHATOV SQ. 1
MOSCOW, 123182
RUSSIA
Phone: 196 9725
Fax: 196 1702
E-Mail: kaplar@nsi.kiae.ru

S. KAUFFMAN
U.S. DEPT OF ENERGY
NR-1
WASHINGTON, DC 20585
USA
Phone: 703 602 8995
Fax: 703 602 7197
E-Mail: kauffmanSR1@navsea.navy.mil

K. KEITHLINE
US NAVY
6313 NO. 29TH ST
ARLINGTON, VA 22027
USA
Phone: 703 602 8995
Fax:
E-Mail: keithline@alum.mit.edu

S. KELPPE
VTT ENERGY
PO BOX 1604
, FIN 02044 VTT
FINLAND
Phone: 358 9 456 5026
Fax: 358 9 456 5000
E-Mail: seppo.kelppe@vtt.fi

W. KEMPER
BALTIMORE GAS AND ELECTRIC
1650 CALVERT CLIFFS PARKWAY
LUSBY, MD 20657
USA
Phone:
Fax:
E-Mail:

M. KHATIB-RAHBAR
ENERGY RESEARCH, INC.
PO BOX 2034
ROCKVILLE, MD 20847
USA
Phone: 301 881 0866
Fax: 301 881 0867
E-Mail: mkr-eri@radix.net

H. KIM
COMMONWEALTH EDISON CO.
1400 OPUS DR., SUITE 400
DOWNERS GROVE, IL 60517
USA
Phone: 630 663 3072
Fax: 630 663 7181
E-Mail: Hak-Soo.Kim@ucm.com

K. KITANO
JAERI
TOKAI-MURA
NAKA-GUN, IBARAKI-KEN 319-1195
JAPAN
Phone: 81 29 282 5954
Fax: 81 29 282 5429
E-Mail: kitano@nsrr.tokai.jaeri.go.jp

M. KNAPIK
McGRAW-HILL
1200 G STREET, NW, SUITE #1100
WASHINGTON, DC 20005-3802
USA
Phone: 202-383-2167
Fax: 202-383-2125
E-Mail: mknap@mh.com

J.S. KOH
KOREA INSTITUTE OF NUCLEAR SAFETY
YOUSUNG,
KOREA
Phone: 865 576 8275
Fax:
E-Mail: kohj@ornl.gov

E. KOLSTAD
OECD HALDEN
PO BOX 173
HALDEN, N-1751
NORWAY
Phone: 47 69212200
Fax: 47 69212201
E-Mail: erik.kolstad@hrp.no

K. KUSSMAUL
MPA STUTTGART
32 PFAFFENWALDRING
STUTTGART, D70569
GERMANY
Phone: 49 711 6853582
Fax: 49 711 685-2698
E-Mail: Kussmaul@mpa.uni-stuttgart.de

J. LAAKSONEN
STUK
LAIPPATIE 4, PO BOX 14
HELSINKI, FIN 00881
FINLAND
Phone: 358 9 7598 8200
Fax: 358 9 7598 8216
E-Mail: jukka.laaksonen@stuk.fi

J. LANG
EPRI
1300 HARRIS BLVD.
CHARLOTTE, NC 28262
USA
Phone: 704-547-6066
Fax: 704-547-6168
E-Mail: jlang@epri.com

S. LANGENBUCH
GRS
FORSCHUNGSGELANDE
GARCHING, 85748
GERMANY
Phone: 893200442
Fax: 8932004599
E-Mail: lab.grs.de

D. LANNING
PACIFIC NORTHWEST NATIONAL LAB
PO BOX 999
RICHLAND, WA 99352
USA
Phone: 509 372 4181
Fax: 509 372 6240
E-Mail: dd_lanning@pnl.gov

C. LECOMTE
CEA/IPSN/PG
PB6
FONTENAY AUX ROSES, 92265
FRANCE
Phone: 146547736
Fax: 146567971
E-Mail: catherine.lecomte@ipsn.fr

B. LETELLIER
LOS ALAMOS NATIONAL LAB
PO BOX 1663, MS K557
LOS ALAMOS, NM 87545
USA
Phone: 505 665 5188
Fax: 505 667 5531
E-Mail: bcl@lanl.gov

H. LEWIS
UCSB
4184 GRESLA AVE.
SANTA BARBARA, CA 93110
USA
Phone:
Fax: 805-563-1066
E-Mail: hlewis@physics.ucsb.edu

M. LIVOLANT
IPSN
BP 6
FONTENAY AUX ROSES, CEDEX 92265
FRANCE
Phone: 33 1 46567173
Fax: 33 1 46569511
E-Mail: michel.livolant@ipsn.fr

D. LOCHBAUM
UNION OF CONCERNED SCIENTISTS
1616 P ST., NW, SUITE 310
WASHINGTON, DC 20036
USA
Phone: 202 332 0900
Fax: 202 332 0905
E-Mail: dlochbaum@ucsusa.org

F. LOPEZ
BECHTEL POWER CORP.
5325 SPECTRUM DR.
FREDERICK, MD 21703-8388
USA
Phone: 301 228 6322
Fax: 301 631 0859
E-Mail: fxlopez@bechtel.com

G. LOWENHIELM
SWEDISH NUCLEAR POWER INSPECTORATE
STOCKHOLM, SE 10658
SWEDEN
Phone: 46 8 698 8496
Fax: 46 8 661 9086
E-Mail: gustaf.lowenhielm@ski.se

E. LYMAN
NUCLEAR CONTROL INSTITUTE
1000 CONNECTICUT AVE, NW STE 410
WASHINGTON, DC 20036
USA
Phone: 202 822 6594
Fax: 202 452 0892
E-Mail: lyman@nci.org

A. MACHIELS
EPRI
PO BOX 10412
PALO ALTO, CA 94303
USA
Phone: 650-855-2054
Fax: 650-855-2002
E-Mail: amachiel@epri.com

A. MAILLIAT
IPSN
CE CADARACHE BP 1
ST PAUL LEZ DURANCE, 13108
FRANCE
Phone: 33 44 225 3637
Fax: 33 44 225 2929
E-Mail: alain.mailliat@ipsn.fr

V. MALOFEEV
RRC, KURCHATOV INSTITUTE
KURCHATOV SQ. 1
MOSCOW, 123182
RUSSIA
Phone: 196 7466
Fax: 196 1702
E-Mail: malofeev@nsi.kiae.ru

C. MARQUIE
IPSN
CE CADARACHE
ST PAUL LEZ DURANCE, 13108
FRANCE
Phone: 33 442262572
Fax: 33 442253555
E-Mail: marquie@ipsn.fr

T. MARSTON
EPRI
3412 HILLVIEW AVE
PALO ALTO, CA 94303
USA
Phone: 650 855 2997
Fax: 650 855 1026
E-Mail: tmarston@epri.com

M. MASSOUD
CALVERT CLIFFS NPP
1650 CALVERT CLIFFS PARKWAY
LUSBY, MD 20657
USA
Phone: 410 495 6522
Fax: 410 495 4498
E-Mail: mahmoud.massoud@bge.com

R. McCOLLUM
BETTIS LABORATORY
PO BOX 79
WEST MIFFLIN, PA 15122
USA
Phone: 412-476-6356
Fax: 412-476-6664
E-Mail: mccollum@bettis.gov

D. McPHERSON
REACTOR SAFETY INC.
6636 KIRBY COURT
FALLS CHURCH, VA 22043
USA
Phone: 703-536-6954
Fax: 703-536-6954
E-Mail: gdm@kreative.net

C. MEINHOLD
NCRP
7910 WOODMONT AVE, SUITE 900
BETHESDA, MD 20814
USA
Phone: 301-657-2652
Fax:
E-Mail: ncrp.exec@ncrp.com

J.C. MELIS
IPSN
CE CADARACHE BP 1
ST PAUL LEZ DURANCE, 13108
FRANCE
Phone: 33 44 225 3722
Fax: 33 44 225 2971
E-Mail: jean-claude.melis@ipsn.fr

J. MEYER
INFORMATION SYSTEMS LABORATORIES
11140 ROCKVILLE PIKE, STE 500
ROCKVILLE, MD 20852
USA
Phone: 301 468 6425
Fax: 301 468 0883
E-Mail: jmeyer@islinc.com

S. MIMURA
NUCLEAR POWER ENGINEERING CORP.
TORANOMON BLDG, 6F 17-1, 3-CHOME
TRANOMON
TOKYO, 105-0001
JAPAN
Phone: 81-3 4512 2650
Fax: 81 3 3435 3413
E-Mail: mimura@nupec.or.jp

S. MITANI
NUCLEAR POWER ENGINEERING CORP.
17-1, 3-CHOUME, TRANOMON, MINATO-KU
TOKYO, 105-0001
JAPAN
Phone: 81 3 3435 3427
Fax: 81 3 3435 3428
E-Mail: mitani@nupec.or.jp

D. MITCHELL
WESTINGHOUSE
5801 BLUFF RD
COLUMBIA, SC 29250
USA
Phone: 803 647 3611
Fax:
E-Mail: mitched@westinghouse.com

N. MIYAZAWA
JEPIC WASHINGTON
1120 CONNECTICUT AVE, SUITE 1070
WASHINGTON, DC 20036
USA
Phone: 202 955 5610
Fax: 202 955 5612
E-Mail: genden@jepic.com

M. MODARRES
UNIV. OF MARYLAND
2100 MARIE MOUNT HALL
COLLEGE PRAK, MD 20742
USA
Phone: 301-405-5226
Fax:
E-Mail: modarres@ens.umd.edu

D. MODEEN
NUCLEAR ENERGY INSTITUTE
1776 I ST., NW, SUITE 400
WASHINGTON, DC 20006
USA
Phone: 202 739 8084
Fax: 202 533 0170
E-Mail: djm@nei.org

E. MOELLER
SWEDISH POWER
PO BOX 527
STOCKHOLM, SE-16216
SWEDEN
Phone: 46 70 539 7433
Fax: 46 8 739 6900
E-Mail: erik.moeller@swedpower.vattenfall.se

S. MONTELEONE
BROOKHAVEN NATIONAL LABORATORY
32 LEWIS ROAD
UPTON, NY 11973-5000
USA
Phone: 631-344-7235
Fax: 631-344-3957
E-Mail: susanm@bnl.gov

R. MONTGOMERY
ANATECH CORP.
5435 OBERLIN DRIVE
SAN DIEGO, CA 92121
USA
Phone: 858-455-6350
Fax: 858-455-1094
E-Mail: rob@anatech.com

R. NANSTAD
OAK RIDGE NATIONAL LAB
PO BOX 2008
OAK RIDGE, TN 37831-6151
USA
Phone: 865 574 4471
Fax: 865 574 5118
E-Mail: nanstadrk@ornl.gov

J. NASER
EPRI
3412 HILLVIEW AVE.
PALO ALTO, CA 94304
USA
Phone: 650-855-2107
Fax: 650-855-2090
E-Mail: jnaser@epri.com

M.A. NATISHON
PHOENIX ENGINEERING ASSOC., INC.
3300 ROYALE GLEN AVE.
DAVIDSONVILLE, MD 21035
USA
Phone: 410-798-0070
Fax: 301-314-9477
E-Mail: natishon@peai.net

L. OTT
OAK RIDGE NATIONAL LAB
PO BOX 2009
OAK RIDGE, TN 37831-8057
USA
Phone: 865 574 0324
Fax: 865 574 8216
E-Mail: ottlj@ornl.gov

O. OZER
EPRI
PO BOX 10412
PALO ALTO, CA 94303
USA
Phone: 650 855 2089
Fax: 650 855 1026
E-Mail: oozer@epri.com

J. PAPIN
IPSN
CE CADARACHE BAT 702
ST PAUL LEZ DURANCE, 13108
FRANCE
Phone: 33 44 225 3463
Fax: 33 44 225 6143
E-Mail: joelle.papin@ipsn.fr

Y.I. PARK
MINISTRY OF SCIENCE & TECHNOLOGY
GOVERNMENT COMPLEX, KWACHON
KWACHON, KYUNGGI-DO 427-760
KOREA
Phone: 8242 868 0009
Fax: 8242 861 1700
E-Mail: yipark@most.go.kr

M. PARKER
DNS-ILLINOIS
1035 OUTER PARK DR
SPRINGFIELD, IL 62704
USA
Phone: 217 785 9854
Fax: 217 524 5671
E-Mail: parker@idns.state.il.us

C. PARKS
OAK RIDGE NATIONAL LAB
PO BOX 2008
OAK RIDGE, TN 37831
USA
Phone: 865-574-5280
Fax: 865-576-3513
E-Mail: parkscv@ornl.gov

L. PHILLIPS
NUCLEAR ENG'R CONSULTANT
118 MONROE ST #705
ROCKVILLE, MD 20850
USA
Phone: 301 294 3069
Fax: 301 294 7879
E-Mail: larryephillips55@cs.com

T. PIETRANGELO
NUCLEAR ENERGY INSTITUTE
1776 I ST., NW, SUITE 400
WASHINGTON, DC 20006
USA
Phone: 202 739 8081
Fax:
E-Mail: arp@nei.org

D. POWERS
ACRS-NRC
WASHINGTON, DC 20555
USA
Phone:
Fax:
E-Mail:

C. PUGH
OAK RIDGE NATIONAL LAB
PO BOX 2009
OAK RIDGE, TN 37831
USA
Phone: 865-574-0422
Fax: 865-241-5005
E-Mail: pughce@ornl.gov

M. QUECEDO
ENUSA
SANTIAGO RUSINOL 12
MADRID, 28040
SPAIN
Phone: 34 91 3474264
Fax: 34 91 3474215
E-Mail: MQG@enusa.es

E. QUINN
MDM SERVICES
23292 POMPEII DR
DANA POINT, CA 92677
USA
Phone: 949 365 1350
Fax: 949 365 1360
E-Mail: equinn@mdmcorp.com

G. QUINN
BECHTEL POWER CORP.
5275 WESTVIEW DR
FREDERICK, MD 21703-8306
USA
Phone: 301 228 6352
Fax: 301 662 0584
E-Mail: gquinn@bechtel.com

R. RANIERI
NATL. AGENCY FOR ENVIRONMENT PROTECTION
VIA VITALIANO BRANCATI, 48
ROME, 00144
ITALY
Phone: 39 06 50072150
Fax: 39 06 50072941
E-Mail: ranieri@anpa.it

J. RASHID
ANATECH CORP.
5435 OBERLIN DR.
SAN DIEGO, CA 92121
USA
Phone: 858 455 6350
Fax: 858 455 1094
E-Mail: joe@anatech.com

S. RAY
WESTINGHOUSE
5801 BLUFF ROAD
COLUMBIA, SC 29212
USA
Phone: 803-647-3787
Fax:
E-Mail: rays@westinghouse.com

C. REID
BECHTEL POWER CORP.
5325 SPECTRUM DR
FREDERICK, MD 21703
USA
Phone: 301 228 6533
Fax: 301 631 0936
E-Mail: creid@bechtel.com

W. RHODES
UNIV. OF PITTSBURGH
520 CASTLE DR., APT 109
BETHEL PARK, PA 15102
USA
Phone: 412 854 2702
Fax: 412 624 9639
E-Mail: wdrst1@pitt.edu

I. RICKARD
WESTINGHOUSE
2000 DAY HILL RD
WINDSOR, CT 06095-0500
USA
Phone: 860 285 9678
Fax: 860 285 4189
E-Mail: ian.c.rickard@us.westinghouse.com

T. RIECK
COMMONWEALTH EDISON CO.
1400 OPUS PL.
DOWNERS GROVE, IL 60515
USA
Phone: 630 663 7687
Fax: 630663 7181
E-Mail: terrance.a.rieck@ucm.com

T. RIEKERT
TUV NORD E.V.
BOX 54 02 20
HAMBURG, D-22502
GERMANY
Phone: 49 8557 2339
Fax: 49 8557 2958
E-Mail:

J. RILEY
NUCLEAR ENERGY INSTITUTE
1776 I ST., NW, SUITE 400
WASHINGTON, DC 20006
USA
Phone: 202 739 8137
Fax: 202 533 0193
E-Mail: jhr@nei.org

D. RISHER
WESTINGHOUSE
1039 BLACKRIDGE RD.
PITTSBURGH, PA 15235
USA
Phone: 412-374-5774
Fax:
E-Mail:

S. ROSINSKI
EPRI
1300 HARRIS BLVD.
CHARLOTTE, NC 28262
USA
Phone: 704 547 6123
Fax: 704 547 6035
E-Mail: strosins@epri.com

D. ROSS
CONSULTANT
7441 BEE BEE DR.
ROCKVILLE, MD 20855
USA
Phone: 301-869-9586
Fax: 301-869-9586
E-Mail: dfross@erols.com

F. ROSS
U.S. DEPT OF ENERGY
19901 GERMANTOWN RD
GERMANTOWN, MD 20874
USA
Phone: 301 903 4486
Fax: 301 903 5057
E-Mail: frank.ross@hq.doe.gov

K. ROSS
OMICRON SAFETY AND RISK
2500 LOUISIANA, STE 310
ALBUQUERQUE, NM 87110
USA
Phone: 505 883 0553
Fax: 505 883 0588
E-Mail: kwross2@attglobal.net

T. ROSSEEL
OAK RIDGE NATIONAL LAB
PO BOX 2008, MS 6138
OAK RIDGE, TN 37831-6138
USA
Phone: 865 574 5380
Fax: 865 241 3650
E-Mail: rosseeltm@ornl.gov

P. RUIZ LOPEZ
C.N.S.N.S.
DR. BARRAGAN 779 COL. NARVARTE
MEXICO CITY, MEXICO CITY 03020
MEXICO
Phone: 52 5 0 95 32 31
Fax: 52 5 0 95 32 00
E-Mail: pablorl59@mixmail.com

J. RUSSELL
CTR FOR NUCLEAR WASTE REG ANAL
12300 TWINBROOK PWY, STE 210
ROCKVILLE, MD 20852-1606
USA
Phone: 301-881-0289
Fax: 301-881-0294
E-Mail: jrussell@swri.org

J. SAEKI
GLOBAL NUCLEAR FUEL - AMERICAS
PO BOX 780, M/C A16
WILMINGTON, NC 28402
USA
Phone: 910 675 6719
Fax: 910 675 5764
E-Mail: jun.saeki@gnf.com

M. SATTISON
INEEL
PO BOX 1625, MS 3870
IDAHO FALLS, ID 83415
USA
Phone: 208 526 9626
Fax: 208 526 2930
E-Mail: sbm@inel.gov

P. SCHEINERT
BETTIS LABORATORY
PO BOX 79 (ZAP32B)
WEST MIFFLIN, PA 15122
USA
Phone: 412 476 5974
Fax: 412 476 7500
E-Mail: pscheinert@aol.com

R. SINDELAR
WESTINGHOUSE
SAVANNAH RIVER SITE
AIKEN, SC 29808
USA
Phone: 803 725 5298
Fax:
E-Mail: robert.sindelar@srs.gov

B. SINGH
JUPITER CORPORATION
2730 UNIVERSITY BLVD., W, STE 900
WHEATON, MD 20902
USA
Phone: 301 903 3741
Fax:
E-Mail: singhb6@asme.org

W. SLAGLE
WESTINGHOUSE
4350 NORTHERN PIKE
MONROEVILLE, PA 15146-2886
USA
Phone: 412 374 2088
Fax: 412 374 2045
E-Mail: slaglewh@westinghouse.com

C. SMIDTS
UNIV. OF MARYLAND
COLLEGE PARK, MD 20742-2115
USA
Phone:
Fax:
E-Mail:

A. SMIRNOV
RIAR FRD
DIMITROVGRAD, 433510
RUSSIA
Phone: 007 84235 35350
Fax: 007 84235 64163
E-Mail: adm@niiiar.ru

V. SMIRNOV
RIAR FRD
DIMITROVGRAD, 433510
RUSSIA
Phone: 007 84235 35350
Fax: 007 84235 64163
E-Mail: adm@niiiar.ru

C. SMITH
INFORMATION SYSTEMS LABORATORIES
11140 ROCKVILLE PIKE, SUITE 500
ROCKVILLE, MD 20252
USA
Phone: 301-255-2265
Fax: 301-468-0883
E-Mail: csmith@isline.com

J. STONE
BMS
4505 AMHERST LAANE
BETHESDA, MD 20814
USA
Phone: 301-564-1214
Fax:
E-Mail: jonstoneb@cs.com

P. STOREY
NUCLEAR SAFETY DIRECTORATE, HSE
NSD 1E, ROOM 314, SPM, BOOTLE
MERSEYSIDE, L20 3LZ
UK
Phone: 44 151 951 4172
Fax: 44 151 951 4163
E-Mail: peter.storey@hse.gsi.gov.uk

M. STRAKA
SCIENTECH
BOX 10261
ROCKVILLE, MD 20849
USA
Phone: 301-258-1866
Fax:
E-Mail:

G. STRAMBACK
GE NUCLEAR ENERGY
175 CURTNER AVE M/C 747
SAN JOSE, CA 95125
USA
Phone: 408 925 1913
Fax: 408 925 1490
E-Mail: george.stramback@gene.ge.com

M. STRAND
SCIENTECH
GAITHERSBURG, MD
USA
Phone: 301 258 1722
Fax:
E-Mail: mstrand@scientech.com

A. STRASSER
AQUARIUS SERVICES CORP.
17 POKAHOE DR.
SLEEPY HOLLOW, NY 10591
USA
Phone: 914 366 8875
Fax: 914 366 8876
E-Mail: Stras69@att.net

B. SUGNET
POLESTAR APPLIED TECHNOLOGY, INC.
9175 OAK LEAF WAY
GRANITE BAY, CA 95746
USA
Phone: 916 791 7846
Fax: 916 791 5908
E-Mail: bsugnet@polestar.com

M. SUMINI
BOLOGNA UNIVERSITY
VIA DEI COLLI, 16
BOLOGNA, 40136
ITALY
Phone: 39 05 16441711
Fax: 39 05 16441747
E-Mail: marco.sumini@mail.ing.unibo.it

T. TAKANO
JAERI
1120 CONNECTICUT AVE, NW
WASHINGTON, DC 22036
USA
Phone: 202 955 5610
Fax: 202 955 5612
E-Mail: tohuko@jepic.com

M. TAYLOR
PO BOX 426
STERLING, VA 20167-0426
USA
Phone: 703 437 1508
Fax:
E-Mail:

Y. UCHIYAMA
NUCLEAR POWER ENGINEERING CORP.
3-13, 4-CHOME TORANOMON, MINATO-KU
TOKYO, 105-0001
JAPAN
Phone: 81 3 4514 5651
Fax: 81 3 4514 5651
E-Mail: uchiyama@nupec.or.jp

R. UHRIG
ACRS-NRC
WASHINGTON, DC 20555
USA
Phone: 352 367 0384
Fax: 352 337 9106
E-Mail: ruhrig@utk.edu

K. VALTONEN
STUK
PO BOX 14
HELSINKI, 00881
FINLAND
Phone: 358 9 75988331
Fax: 358 9 75988382
E-Mail: keijo.valtonen@stuk.fi

W. van DOESBURG
SWISS FEDERAL NUCLEAR SAFETY INSP.
PO BOX 5232
VILLIGEN, 05232
SWITZERLAND
Phone: 41 56 3103862
Fax: 41 56 3104979
E-Mail: willem.vandoesburg@hsk.psi.ch

M. VIDARD
EDF/SEPTEN
12-14 AVE DUTRIEVOZ
VILLEURBANNE, 69628
FRANCE
Phone: 33 4 7282 7565
Fax: 33 4 7282 7701
E-Mail: michel.vidard@edf.fr

J. VILLADONIGA
CONSEJO DE SEGURIDAD NUCLEAR
JUSTO DORADO, 11
MADRID, 28040
SPAIN
Phone: 34 91 3460240
Fax: 34 91 3460588
E-Mail: jivt@csn.es

G. VINE
EPRI
2000 L ST., NW, SUITE 805
WASHINGTON, DC 20036
USA
Phone: 202 293 6347
Fax: 202 293 2697
E-Mail: gvine@epri.com

K. VINZENS
TUV SUDDEUTSCHLAND GmbH
WESTENDSTRASSE 199
MUNCHEN, BAVARIA 80686
GERMANY
Phone: 089 5791 2317
Fax: 089 5791 2605
E-Mail: kurt.vinzens@tuevs.de

D. VOGT
LAWRENCE LIVERMORE NATIONAL LAB
PO BOX 808, MS L-638
LIVERMORE, CA 94551
USA
Phone: 925 423 7798
Fax: 925 422 9913
E-Mail: vogt3@llnl.gov

N. WAECKEL
EDF/EPRI
3412 HILLVIEW AVE
PALO ALTO, CA 94304
USA
Phone: 650 855 86 88
Fax: 650 855 10 26
E-Mail: nwaeckel@epri.com

I. WALL
CONSULTANT
81 IRVING AVE
ATHERTON, CA 94027
USA
Phone: 650 324 0321
Fax: 650 325 0765
E-Mail: ianwall@alum.mit.edu

L. WARD
INFORMATION SYSTEMS LABORATORIES
11140 ROCKVILLE PIKE, SUITE 500
ROCKVILLE, MD 20852
USA
Phone: 301-255-2279
Fax: 301-468-0883
E-Mail: eward@islinc.com

R. WEINER
KW CONSULTING, INC.
PO BOX 101567
PITTSBURGH, PA 15237
USA
Phone: 412 422 7732
Fax: 412 687 3965
E-Mail: bob@kwconsulting.com

E. WIESER
BUSINESS PUBLISHERS, INC.
8737 COHESVILLE RD, SUITE 1100
SILVER SPRING, MD 20910
USA
Phone: 301-587-6300
Fax: 301-587-1081
E-Mail: ewieser@bpinews.com

L. WOLF
UNIV. OF MARYLAND
BLDG 090, DEPT. OF MATERIALS & NUC ENG
COLLEGE PARK, MD 20742-2115
USA
Phone: 301 405 0042
Fax: 301 314 9467
E-Mail: wolf@eng.umd.edu

R. WOOD
OAK RIDGE NATIONAL LAB
PO BOX 2008, BLDG. 3500 MS 6010
OAK RIDGE, TN 37831-6010
USA
Phone: 865 574 5578
Fax: 865 576 8380
E-Mail: woodrt@ornl.gov

R. YANG
EPRI
3412 HILLVIEW AVE
PALO ALTO, CA 94304
USA
Phone: 650 855 2481
Fax: 650 855 1026
E-Mail: ryang@epri.com

L. YEGOROVA
RRC, KURCHATOV INSTITUTE
KURCHATOV SQ. 1
MOSCOW, 123182
RUSSIA
Phone: 196 7283
Fax: 196 1702
E-Mail: asmolov@nsi.kiae.ru

R. YOUNGBLOOD
INFORMATION SYSTEMS LABORATORIES
11140 ROCKVILLE PIKE
ROCKVILLE, MD 20852
USA
Phone: 301-255-2270
Fax: 301-468-0883
E-Mail: ryoungblood@isline.com

D. ZANNONI
NEW JERSEY BNL
9 TUDOR CT.
EWING, NJ 08628
USA
Phone:
Fax:
E-Mail: dzannooni@dep.state.nj.us

P. ZAPP
WESTINGHOUSE
SAVANNAH RIVER SITE
AIKEN, SC 29808
USA
Phone: 803 725 2567
Fax:
E-Mail: philip.zapp@srs.gov

M. ZAVISCA
ENERGY RESEARCH, INC.
1649 E. JEFFERSON ST. #104
ROCKVILLE, MD 20852
USA
Phone: 301-881-0866
Fax: 301-881-0867
E-Mail: mjz@eri-world.com

G. ZIGLER
ITS CORP.
6000 UPTOWN BLVD. NE, SUITE 300
ALBUQUERQUE, NM 87123
USA
Phone: 505-872-1089
Fax: 505-872-0233
E-Mail: gzigler@itsc.com

P. ZMOLA
C&P ENGINEERING
5409 NEWINGTON RD
BETHESDA, MD 20816
USA
Phone: 301 656 5756
Fax: 301 656 5756
E-Mail: pzmola@attglobal.net

OPENING REMARKS

by

Ashok C. Thadani, Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Good morning. My name is Ashok Thadani and I am the Director of the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research.

As many of you know, it is the vision of the office to develop the technical bases for realistic safety decisions and to prepare the agency for the future by evaluating safety issues involving both current and new designs, as well as new technologies. This meeting with your participation is extremely important to achieving our vision, and in this light, I welcome you all to the NRC's 28th Annual Water Reactor Safety Meeting.

We have designed this meeting to provide a forum that will facilitate open, substantive and, we believe frank dialogue from both our domestic and our foreign participants on a range of important nuclear safety research issues, initiatives and results. With the rapid changes taking place, both inside, as well as external to the NRC, it is especially important that we effectively utilize this opportunity to continue discussions and obtain your views on key safety research issues that are being pursued.

The agenda for the first Water Reactor Safety Meeting in the new millennium brings together both international and domestic experts to report and discuss progress across a spectrum of diverse research topics that are the focus of today's meeting. In preparing the agenda, we anticipate dialogue and debate on the rigor and the completeness of the scientific knowledge and the technical basis for the findings and the conclusions that will be presented. I believe that the knowledge and the technical bases that will be presented and discussed in the technical sessions of this year's meeting are both important and challenging. This year the meeting focuses on ongoing research in areas such as risk-informed regulation, improving regulatory effectiveness, the integrity of the primary coolant pressure boundary, behavior and acceptance criteria for high burnup fuel, reactor and spent fuel thermal hydraulic and severe accident analysis, digital instrumentation and control systems, and reactor decommissioning.

I also am looking forward to the presentations and the discussions of the plenary session expert panels that we have organized for this year's meeting. I believe that the topics of these panels are most appropriate as part of this, our first meeting in the new millennium. They will explore the legacy and the lessons in the 25 years since the Reactor Safety Study, Challenges in the Future for Risk-Informed Regulation, and the Future Role of Nuclear Power and the Future Needs for Nuclear Regulatory Research. In organizing and composing these panels, we have succeeded in bringing together many distinguished and world class experts who will lead us in insightful and thought-provoking dialogue on these subject matters. I am also extremely pleased that this year each of the NRC Commissioners, including the Chairman, has found the time in their busy schedules to participate in the meeting either as a plenary session guest speaker or to chair a plenary session expert panel discussion. I am also delighted that we will have the opportunity to honor Professor Emeritus Norman C. Rasmussen, who lead the team

that produced the landmark Reactor Safety Study, and Saul Levine, who provided the critical day-to-day management of this pioneering effort.

I would, however, be remiss if I failed to also acknowledge the leadership and the pioneering role of NRC and the Office of Nuclear Regulatory Research in helping to get us to where we are today in the application of PRA for regulatory decision-making. As you know, many NRC regulations were developed without the benefit of quantitative assessment of risk. They have been based largely on deterministic engineering criteria. Since the pioneering work of WASH 1400, progress has been made to provide a technical basis for incorporating risk insights into the regulatory decision-making process. However, much work remains to fully utilize these techniques in revising our fundamental regulatory fabric.

Another area I would like to mention, and I am sure you all are well aware, is the continuing decline in resources that are available for nuclear safety research. This has regrettably led to the loss of research facilities, challenged our ability to maintain a strong technical capability, and added to the difficulty of sustaining effective research leadership. And so we have been challenged to find creative ways to efficiently invest our resources so as to be well prepared to successfully meet future challenges, challenges such as ensuring that a sound technical basis for assessing license renewal, power uprate requests, the use of higher burnup and possibly mixed oxide fuels, and the changeover from analog components to digital. In some areas, what we believe is proving to be one of the successful strategies is to make effective use of cooperative research agreements. I believe that cooperative agreements with nuclear safety organizations, both at the international level, as well as with organizations here within the United States, will be essential. In this regard, in addition to agreements with the Department of Energy, Electric Power Research Institute, and other domestic organizations, NRC's Research Office has entered into cooperative agreements with nuclear safety organizations in more than 20 countries. These agreements provide NRC with invaluable information and access to experimental facilities of the kind that would otherwise be foreclosed because the similar facilities in this country are now shut down. And so, with challenges such as license renewal and those that would emerge in the event of license applications for new plant designs, it is important that we pay close attention to identifying new and successful approaches for sustaining the kinds of capabilities that are embodied today in our people, our facilities and our analytical tools. Thus, I encourage you to carefully follow the panel discussion on the future role of nuclear power and the need for nuclear regulatory research.

I would now like to draw your attention to a few administrative matters about this year's meeting. This year, for the first time, our meeting is open to all who care to attend without a fee for registration. Registration fees were eliminated this year so as to make our meeting more accessible to all our stakeholders and to enhance public communication and involvement. Other decisions this year have necessitated that our traditional luncheons, as well as our evening reception mixer at the end of the first day have sponsors other than the NRC. In this regard, I would like to acknowledge and express my sincere thanks to the participating national laboratories which are sponsoring this evening's mixer, MIT's Department of Nuclear Engineering for organizing today's noontime lunchtime, and, finally, my thanks to Elsevier Science for sponsoring Tuesday's noontime luncheon. Because of the support of these organizations, we are able to continue the longstanding Water Reactor Safety Meeting tradition of enabling our participants to listen to distinguished luncheon speakers and to meet in a social setting that is conducive to the frank exchange of ideas and issues in areas of mutual professional interest.

Finally, I would like to mention that last year we implemented a number of significant changes to the format and the conduct of the meeting, and we evaluated the effectiveness and the receptiveness of the changes from our participants. The feedback we received was that the new approach was, in fact, well received, and that future meetings should seek to optimize the effectiveness or the implementation of these changes. And so for this meeting, we have endeavored to do just that. And as was the case last year, we once again look to each of you to help us evaluate the refinements we have made so as to successfully build upon last year's improvements.

Now, it is indeed my pleasure to introduce the NRC Chairman, Dr. Richard A. Meserve, our keynote speaker. Dr. Meserve, immediately prior to becoming Chairman, was a partner in the Washington, D.C. law firm of Covington & Burling, having joined the firm in 1981. From 1977 to 1981, Dr. Meserve served as legal counsel to the President's Science and Technology Advisor.

He has served on a variety of committees of National Academies of Sciences and Engineering. He served as chairman of committees that examined the safeguarding of nuclear weapons material in the former Soviet Union. He advised the Department of Energy on declassification of information. He examined environmental issues associated with the nuclear weapons complex, assessed technical and safety issues relating to certain DOE reactors, and examined the prospect for enhanced fuel economy in automobiles. We need that very much now.

Dr. Meserve is a Fellow of the American Academy of Arts and Sciences, the American Physical Society, and the American Association for the Advancement of Science, and serves on the Board of Carnegie Institution of Washington, and the Board of Overseers for Arts and Sciences of Tufts University. He formerly served as chairman of the Advisory Council of the Princeton Plasma Physics Laboratory as a Member of the Secretary of Energy Advisory Board. He has also been a law clerk to former Supreme Court Justice Harry A. Blackmun.

Chairman Meserve holds a B.A. from Tufts University, a Ph.D. in applied physics from Stanford University, and J.D. from the Harvard Law School. We are privileged to have Chairman Meserve with us today as our keynote speaker and to share his thoughts on The Role of Research in a Changing Environment.

"The Role of Research in a Changing Environment"

Remarks of

Dr. Richard A. Meserve
Chairman, U.S. Nuclear Regulatory Commission

at the

28th Water Reactor Safety Information Meeting
Bethesda, Maryland

Good morning. It gives me great pleasure to add my welcome to all of you. This is the 28th year that the Water Reactor Safety Information Meeting has been held, but it is the first that I have had the pleasure of attending. I am pleased to be able to address this opening session, particularly since the panel on the WASH-1400 study that follows this talk includes several friends. I am looking forward to hearing their reflections on that landmark effort.

The topic of my talk this morning is "The Role of Research in a Changing Environment." I hope to give you a sense of where I see the nuclear industry heading over the next several years, what the change means for the Nuclear Regulatory Commission, and the essential and vital role that research must play in ensuring that the NRC is equipped to deal with the challenges ahead.

The Changing Environment

The electric utility industry as a whole, and the nuclear sector of that industry in particular, is encountering a period of profound change. For the nuclear industry, the current turbulence is certainly greater than at any time since the Three Mile Island accident, and it may be unequaled in the history of civilian nuclear power electric production. The driving force for these changes is the deregulation of electricity pricing. In a competitive and deregulated market, the economics of generation is the essential consideration, and reliable nuclear power plants - particularly those for which the capital costs have been largely amortized - have become increasingly valuable assets. The changed view of nuclear generating assets is driving a number of initiatives: industry consolidation, plant sales, and license renewal. We are even beginning to see the first stirring of interest in construction of new nuclear power plants in the United States. These developments have significant implications for the NRC in general, and for our research program in particular.

The Role of Research in the Near Term

In the near term, NRC-sponsored research has a key role in developing the regulatory tools that the NRC will need to deal with the changing environment. The industry's focus on economics has a number of potential consequences. During a time of change, it is important to maintain vigilance so as to assure that safety is maintained. I am optimistic, however, that the changed economic circumstances could in fact lead to safety improvements. Industry consolidation has the potential to enhance nuclear plant safety as companies with many plants apply best practices and lessons learned across their entire fleets. Perhaps even more important is the reality that safe operation and economic operation should go hand-in-hand. A safe and well-run

plant is reliable, stays on-line, and is able to avoid extended shutdowns, either as a result of the need to fix problems or because of regulatory action on the NRC's part to address a significant safety deficiency.

How do these developments affect the NRC? The NRC's statutory mandate, and our foremost obligation, is to provide reasonable assurance of adequate protection of public health and safety and the environment. We must never allow economic considerations to compromise our commitment to fulfill that obligation. However, that does not mean that we should not strive to operate as efficiently and effectively as possible. The price deregulation of the electric generation business means that the cost of safety regulation - both direct, from fees charged to licensees to recover the cost of the NRC's operations, and indirect, from the costs of regulatory compliance - come directly off the bottom line. Just as we owe the public the assurance that their health and safety are protected, we owe our licensees the assurance that the regulatory obligations that we impose on them minimize unnecessary burdens. We must therefore sharpen our focus to those areas that are safety-significant.

As you are undoubtably aware, the NRC has embarked on a fundamental re-examination of our reactor regulations to consider risk explicitly. This move to risk-informed regulation builds on the foundation that has been established through NRC-sponsored research, beginning with the WASH-1400 study and continuing to the present day, to develop and apply quantitative methodologies for the assessment of reactor risk. The current focus of the agency's efforts in this area include risk-informing the technical bases of our reactor regulations and supporting the efforts to risk-inform the so-called "special treatment" requirements, such as quality assurance, environmental qualification, and technical specifications. We have also made substantial changes in our reactor oversight program, with a focus on safety and objectivity. Our research programs support these initiatives through evaluation of plant operational experience and development of risk-based performance indicators, thereby helping us to sharpen the safety focus of the oversight process.

The process of risk-informing our regulations requires that our tools for assessing technical issues be as realistic as possible. This move away from a traditional conservative, bounding approach has been made possible through a combination of operating experience, which now comprises more than 2000 reactor years in the U.S. alone, and experimental and analytical programs nurtured by NRC-sponsored research to develop better models of the behavior of a reactor during design-basis and beyond-design-basis accidents. One recent product of this research was an NRC-approved alternate source term for more realistic assessment of radiological consequences. Other ongoing research programs in this same vein include upgrading of the NRC's thermal-hydraulic codes to support review of industry-sponsored "best-estimate" accident analysis codes, and revisions to the pressurized thermal shock rule, based on a better understanding of radiation-induced embrittlement and fluid-structure interactions in reactors.

The drive for improved economic performance of operating plants is also manifesting itself in other ways. One outgrowth of the application of more realistic analyses is that the margins between calculated plant conditions and operational or regulatory safety limits are larger than previously demonstrated. Licensees are naturally inclined to make use of these additional margins in ways that allow improved economic performance, such as by increasing fuel burnups, changing core power distributions, and increasing reactor power. (We refer to these as power uprates.) The research program on high-burnup fuels, along with the improved

analytical techniques for accident analyses, are essential elements of the NRC's capability to review such initiatives. Licensees are also bringing on-line new technologies, such as digital I&C systems, that have the potential to increase plant reliability; the programs to assess the potential impacts of these new technologies are needed to ensure that the NRC is not an impediment to the appropriate deployment of these technologies.

The developments that I have just covered are extremely important both to the industry and to the NRC. However, I believe that the most significant near-term impact of the new environment is the widespread interest in nuclear plant license renewal. A few years ago, pundits claimed that a large number of nuclear plants would shut down prematurely. But the changed economic circumstances now make it worthwhile for a generating company to take steps to keep a plant operating beyond the term of the original 40-year license if the plant can operate safely and reliably for an extended period. As a result, we are seeing a strong interest in license renewal. We have renewed the licenses of two plants, Calvert Cliffs and Oconee, and are currently reviewing the applications for three other plants -- Hatch, ANO-1, and Turkey Point. Five more applications are expected in the current fiscal year, and the number in the years beyond 2001 continues to grow. About 40 percent of operating plants have indicated their intention to seek license renewal, and that fraction may ultimately reach 85 percent or more. If license renewal can appropriately be granted, nuclear power from existing plants will continue to make a significant contribution to our energy supply well into this century.

The core question is whether license renewal is appropriate. Fortunately, the NRC has been working on various aging-related issues for many years. As a direct consequence of these research programs, we have the technical bases to approach license renewal in a manner that focuses appropriately on the effects and management of aging. We were able to complete comprehensive assessments of the first two applications that we received for license renewal within the targeted schedule of 30 months. The challenge is to maintain this record as more applications are submitted. I believe we are up to the challenge, with the help of the tools that the NRC research program has helped to provide. As you may know, the NRC recently published its Generic Aging Lessons Learned, or GALL, report, reflecting insights gained as a result of our work to date on license renewal. (The report is available on the NRC's website.) There were many contributors to this important compilation of lessons learned, but a significant portion of the information is derived from reports prepared as part of our Nuclear Plant Aging Research Program. Without that technical foundation, I suspect that we would not be in the position to respond to the applications for license renewal with the depth of knowledge that we can now bring to bear.

Long-Term Developments and the Role of Anticipatory Research

I have concentrated thus far on areas that are of current or near-term interest to the industry and the NRC. Now, I would like to take out my crystal ball and speculate about what the future might hold for the industry, and discuss how the NRC's research programs with a longer-term focus support future NRC regulatory needs.

The overall environment for nuclear power is changing, in addition to the economic environment. Concern about global warming, for example, should focus attention on power technologies, such as nuclear, that minimize the emission of carbon dioxide and other potential "greenhouse gases." Similarly, consideration of energy security is seen to justify the support of

a portfolio of energy technologies. The renewed interest in such matters may bring about a national reconsideration of the role of nuclear technology.

Perhaps as a natural reflection of these changes, the Department of Energy has begun to increase its research expenditures for civilian nuclear power technology after a period of essentially zero funding. The current program has several components. The Nuclear Energy Plant Optimization program, or "NEPO," focuses on existing plants, with research projects to develop new technologies to increase reliability, availability, and efficiency. By contrast, the Nuclear Energy Research Initiative, or "NERI," is to overcome scientific and technical obstacles to the future use of nuclear energy in the U.S. Many of the projects in the NERI program involve what is referred to as "Generation IV" reactor designs -- plants that might offer improved safety, lower capital and operating costs, proliferation resistance, and reduced waste production. A separate Nuclear Engineering Education Research (NEER) Program has funds that are earmarked for university research; a number of the projects supported by this program also deal specifically with advanced reactor concepts and related technology.

What might all of this mean for the future use of nuclear power? Again, I must offer an impressionistic and distant view. The NRC does not have a promotional role, and must remain agnostic on the question of whether the nuclear path should be resuscitated. Nonetheless, we must watch developments so that our processes do not serve as a needless impediment. As I said earlier, we are beginning to see the first stirring of interest among our licensees in constructing new plants. Given these circumstances, the NRC must prepare to deal with future demands.

Several years ago, we developed a licensing process for standardized plant designs. The idea was to permit the certification of a design in a fashion in which many key technical issues could be resolved once and for all, thereby stabilizing and streamlining the plant licensing process. An application to build a plant based on a certified design would not require examining issues that had been resolved during the certification. Upon approval of such application, a single combined construction permit and operating license would be issued. We have certified three standardized plant designs: General Electric's Advanced Boiling Water Reactor, the System 80-plus design of Combustion Engineering, which is now under the BNFL umbrella, and Westinghouse's AP600 passive plant design, which is also now a BNFL product. We have recently begun a review of Westinghouse's AP1000 design for possible certification. We have not received any applications to build these plants in the U.S., but I must note that two ABWRs are operating in Japan, and several more are planned.

I would also like to mention that the confirmatory testing and analysis programs conducted by the Office of Research were a key element in the review of the AP600 design. While these projects were specific to the AP600 review, they also contributed to the more general objective of upgrading the NRC's thermal-hydraulics codes, and initiated development of advanced risk assessment techniques that should ultimately contribute to risk-informed regulation for both current and future plants.

Some longer-term needs have already been defined for us. The end of the Cold War and the move toward reductions in nuclear weapons stockpiles have resulted in the need to manage significant amounts of weapons-grade plutonium. The strategy selected for this task involves using a portion of that material to create mixed-oxide fuel to be burned in commercial nuclear power reactors. We have already begun to prepare for the licensing of a MOX fuel fabrication

plant, and have a research program to develop a technical basis for reviewing the license amendments that will be required to permit licensees to burn that fuel in their reactors. Other longer-term issues are perhaps not so clear cut. We are following DOE's work on NERI and Generation IV reactors, so that we can understand the primary features of potential advanced reactor concepts. We recognize that our current reactor regulations may not translate well to the licensing of new reactor designs, particularly if the new designs are not water-cooled. Some of these issues may be resolved by our efforts to risk-inform our regulations, but, in other cases, the best approach may well be to start with a clean sheet of paper. This challenge is clearly a considerable one, but we must ensure that our research program has adequate resources to prepare us for the future. If we do not start now, we may find it extremely difficult to respond when we are called upon to begin to review these advanced designs.

Resources and Other Research Issues

My reference to "adequate resources" brings me to my next topic: research funding within the NRC. This is a subject that tends to generate a significant amount of discussion, especially among our licensees, since their fees currently pay our costs, including those for research. Earlier this year, I spoke to a meeting of the Nuclear Energy Institute. The topic of the meeting was "change," and I stated that our research programs provide the basic technical capabilities that allow us to master change rather than to be its victim. I hope that I have conveyed throughout this talk how our research effort provides the technical "backbone" of the NRC's regulatory requirements. Our research program also plays a major role in maintaining the NRC's core technical competencies. This is essential not only from the standpoint of our relationship with our licensees, but also for developing and maintaining public confidence and trust in the NRC as a competent, technically knowledgeable regulator.

Despite the vital contributions of research to the NRC's activities, however, I must also acknowledge that over nearly the last two decades, the research budget has been significantly reduced. Accordingly, I - with the support of my colleagues on the Commission - have taken action to stabilize the budget to ensure that we have adequate resources for key research initiatives. I would also like to note that the bill containing the appropriation for the NRC's 2001 budget includes a provision to remove 10 percent of the NRC's total budget from our fee base, in 2 percent increments over a five-year period. We requested this provision in recognition that some of our activities, while valuable to the NRC's overall mission, do not directly affect the activities of our current licensees, but are of a more general benefit to the public. Instead of license fees, these funds would be supplied from general revenues. I am hopeful that this initiative will ease some of the pressure on our budget in future years.

The strain on the research budget is also occurring in other countries. Under such circumstances, international cooperation becomes essential so as to sustain major research initiatives that are beyond the means of any single country. We have many important international collaborations. I note that our international research partners are well-represented at this conference, and I would particularly like to acknowledge the contributions that you make to further our common understanding.

Our cooperative research efforts extend to the nuclear industry, as well. While we are mindful of the need to conduct independent assessments of important safety issues, there are times in which it is appropriate pool our resources and work with the industry to develop research programs. These include, for example, facility designs and test plans, with each party

performing an independent analysis of the results. We have developed memoranda of understanding on the conduct of cooperative research with both the Electric Power Research Institute and the Department of Energy. I would like to acknowledge the value of these programs, as well.

We are also taking other steps to address the issue of resources and the broader question of the direction of the research program. A few months ago, we convened a group of experts drawn from a wide range of disciplines - academia, the nuclear industry, the public, Congressional staff, and other government agencies - to review the research program and provide suggestions regarding the role, funding, and focus of the research program. The initial reports of the participants were recently submitted and I very much appreciate the group's efforts. I note that several of the members of this group will be participating in a panel session on Wednesday morning to discuss their views on these important questions.

I have been able to touch upon only a portion of the research-related activities that are underway. Fortunately, some of the matters that I did not have time to address are the subject of later sessions. For example, you will hear presentations dealing with reactor decommissioning, dry cask storage, the transportation of spent fuel, and PWR sump blockage issues. The fact that I was not able to discuss these programs, and many others, in the course of this talk, does not mean that I ascribe any less value to them. I hope you will take the opportunity to learn about them first-hand during the remainder of the meeting.

Conclusion

Let me conclude by emphasizing once again the crucial role that our research programs play in meeting our current regulatory challenges and in preparing the NRC to deal effectively and efficiently with issues that may confront us in the future. Whether we are considering operating plants, new reactor designs that may be deployed a few years down the road, or other aspects of the nuclear power enterprise, such as decommissioning and waste disposition, we depend on the results of our research to establish the technical foundation for our regulatory activities. The organizational agility and responsiveness demanded by the rapidly changing environment in the electric utility industry is possible only if we have that firm technical foundation. I am proud of the past record of NRC's research efforts and am committed to sustaining the program in the future.

Thank you.

Opening Remarks on
Twenty-Five Years Since the Reactor Safety Study -
The Legacy and the Lessons

by

Ashok C. Thadani, Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Well, good morning again, and welcome to this panel discussion of Twenty-Five Years Since the Reactor Safety Study - The Legacy and Lessons. The Reactor Safety Study, WASH 1400, was indeed a landmark report. It changed, I believe forever, how we thought of reactor safety. The Reactor Safety Study was done in response to a request from Senator John Pastore, Chairman of the Joint Committee on Atomic Energy. The objectives given to the study by the Atomic Energy Commission were as follows, and I want to read it to you.

> *"The principle objective of the study is to try to reach some meaningful conclusions about the risks of nuclear accidents using current technology. It is recognized, however, that the present state of knowledge probably will not permit a complete analysis of low probability accidents in nuclear plants with the precision that would be desirable. Where this is the case, the study will consider the uncertainty and present knowledge, and the consequence range and predictions, as well as delineating outstanding problems. In this way, any uncertainties in the results of the study can be placed in perspective. Thus, although the results of the study, of necessity, will be imprecise, in some ways, the study, nevertheless, will provide an important first step in the development of quantitative risk analysis methods."*

That was in the early '70s. As you well know, the team was led by Professor Norm Rasmussen from MIT, who reported directly to the Commission. The day-to-day direction of the study team was done by Saul Levine, the project staff director.

Prior to the study, most considered core damage accidents extremely rare events, but also expected they would have catastrophic consequences. WASH 1400 changed that.

In 1972, we believed the accident to be concerned with was a double ended large break in the reactor coolant system. By 1975, we were all aware of the importance of transients and small Loss of Coolant Accidents.

On a very personal note, the insights from the Reactor Safety Study had a great deal of influence on me. I had the fortune to take a look at the draft WASH 1400 study when it first came out, I believe it was August of 1974, and I had been working on issue called anticipated transients without scram. At the time we thought we should make sure the designs can cope with that accident and the frequency should be reduced to 1 in 10 million. Of course, looking at the Reactor Safety Study, it was clear that one needn't go that far. Similarly, it raised questions in my mind about why did we not consider station blackout in the design of these plants, because the frequency of station blackout was believed to be in the range of 10 to the minus 4 to 10 to11 the minus 5 per reactor year. And ever since that time, I have often paid attention to

the relative importance of issues, and I think risk analysis tools have been very, very valuable in that understanding. That certainly shaped my views about reactor safety.

Today we have assembled a panel of pioneers, and there are more in the audience. Before introducing the panel, I do want to recognize Matt Taylor, who led the event tree work, and Garth Cummings, who led the BWR fault tree work. I believe they are in the audience.

On the panel we have Professor George Apostolakis, who early in his career worked on the comparative risk portion of the Reactor Safety Study, and he is now a member of ACRS.

Professor Birkhofer, who is the Managing Director of GRS, and one of the leading experts on reactor safety in the world. He managed the German Risk Study, which was completed shortly after the Reactor Safety Study, and clearly is one of the pioneers in the development and use of PRA in Germany.

Bob Budnitz, a former Director of Research, and Vice Chairman of the Lewis Committee that performed a peer review. Bob has had a continuing role in promoting the use of PRA, particularly in analysis of external events.

Rich Denning is a Professor and Chair of Nuclear Engineering Graduate Studies Committee at Ohio State University. Rich is also a member of the Integrated Risk Management Staff at Battelle Columbus Lab. He was responsible for much of the Level 2 work done in the Reactor Safety Study.

John Garrick, who is the Chairman of ACNW now, but for many years he led one of the preeminent consulting companies in the world of risk analysis. His early efforts at Holmes & Narver predates the Reactor Safety Study. He remains one of the leading thinkers in the PRA profession.

Hal Lewis is a Professor Emeritus at the University of California, Santa Barbara. He is a former member of ACRS and led the Lewis Committee which conducted the review of the Reactor Safety Study. The committee was formed, in fact, in response to a request from Congressman Morris K. Udall and charged with performing independent review.

Ian Wall is a consultant. He was responsible for the Level 3 portion of WASH 1400. He later was Chief of the Probabilistic Analysis Staff at Research and then led the PRA effort at the Electric Power Research Institute for many years.

And then we have Joe Murphy. Joe was a member of the WASH 1400 team from its beginning and remains with the NRC as Special Assistant to the Director of Office of Nuclear Regulatory Research. He is currently Chair of the NRC's Committee to Review Generic Requirements and of OECD NEA's Working Group on Risk Assessment. He also serves on the IAEA's Nuclear Standards Committee. Joe is probably the most knowledgeable person from the inception of WASH 1400 to now. He has been actively engaged. So let me turn the podium over to Joe now.

**Twenty-Five Years Since the Reactor Safety Study -
The Legacy and the Lessons**

by

**Professor George Apostolakis
Massachusetts Institute of Technology**

In the effort to risk-inform the regulations, one major issue is the quality of PRA. And perhaps all of you know that ASME and ANS have been developing standards for PRA, several drafts have been on the street, and I think that eventually we'll have something. The industry also has the so-called certification process which has some interesting differences from the ASME work. I am not going to do justice to it in a few minutes, but basically what they do is, they review a particular PRA for a plant and then identify areas or methods or issues that, if corrected, if updated, then the PRA would be good enough for certain applications.

So the thought occurred to me, if the Reactor Safety Study went through this process, how well would it do? Very well, in my opinion. For a Level 1 PRA, there would be a few things that one would need to fix, but basically it would survive without much change. I think the event trees and the fault trees are still there, we're using them. The analysts certainly displayed the uncertainty in failure rates. They did not call their methodology Bayesian at the time. We took another ten years or so to do it, but they did a rigorous uncertainty analysis.

The human reliability analysis was pretty good. We've spent a lot of money since then, trying to develop better methods. I think we understand the issues better now, but I'm not sure that there is a huge gap between what those folks did and what we're doing now. Certainly, in the area of pre-initiating event human errors, where we are all using essentially the same source, the Human Reliability Handbook.

In the area of common cause failures, well, I'm not sure. The Study's model did not survive for too long, but I'll come back to that in a minute. Of course now we are also doing other things that are helping us in risk-informing the regulations. For example, importance measures all of a sudden seem to be at the center of every initiative the Agency is taking to risk-inform particular pieces of 10 CFR Part 50. The Reactor Safety Study did not use importance measures, but several studies after it did not do it either. It's only in the last few years that importance measures have become so important. But it's not really the methods, I think, that we should think of when the Reactor Safety Study comes to mind. I think a more fundamental impact that the study has had has been the change in culture in the reactor safety arena.

Joe Murphy has already mentioned that some people were surprised, if not shocked, that the core damage frequency was so high. It's probably the same people who thought, before the study was undertaken, that the risk from nuclear power plants could not be quantified. And the study, of course, changed that attitude in a fundamental way. Now we believe that -- well, we accept it as a routine matter -- that we can, in fact, quantify those risks, we can find reasonable estimates of the core damage frequency, large early release frequency, and, of course, the health consequences.

Returning to the common cause failure model, well, the Reactor Safety Study was done at a time when really the amount of research in the PRA area had been absent or minimal, not negligible, but minimal. And they had to improvise. So they came up with a model that didn't survive for long, but the important thing was that they introduced the idea that one had to worry about dependent failures, about common cause failures. And it's interesting to go back to papers that appeared at conferences before '75 and '76. People were doing things that are now simply unacceptable. There were so many papers that started out by saying "we will do a random independent failure analysis." If you said that now, you'd be laughed out of the city. And the number was typical, ten to the minus six for engineered systems. The unavailability was ten to the minus six. The Reactor Safety Study changed that, and now we know it's in the neighborhood of ten to the minus four.

So that was a major impact, in my view. It's not really the method they used, the square root method; it's the fact that they dared to go into this subject and to quantify dependencies, and after that, everybody realized that this was really where the action should be. They established the main framework for risk assessment. There have been various improvements, for example the Zion-Indian Point studies did a much more detailed analysis on seismic and fire risk.

But all these improvements have been within the basic framework of the Reactor Safety Study. Nobody has changed the fundamental approach that was established by its authors, by the pioneers. So, I think this is really the most important contribution of the Reactor Safety Study. It established a new culture for reactor safety, and like all pioneers, sometimes people take things for granted and they don't appreciate the original contribution, but all you have to do is go back to studies before 1975.

The other day, I was reading a book about the history of theater. And much to my surprise, I found out that there was a time when there were no actors on the stage or, there was only one actor on the stage, and it was a major revolution when someone introduced more than one actor. Now, how many people go to the theater these days not expecting to see a number of actors simulating an act? We take this for granted, and yet at some time, somebody revolutionized theater by introducing the idea of more than one actor. And I think you have a similar situation in some respects with the Reactor Safety Study. We take it for granted. You look at the study, nice blue covers, event trees, fault trees, and you say everybody's doing that. Well, there was a time where nobody was doing that, and those guys were the first ones to do it, and I think they deserve all the credit in the world for that.

Thank you very much.

Twenty-five Years Since the Reactor Safety Study
The Legacy and the Lessons

Prof. Dr. Adolf Birkhofer, Managing Director
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Germany

Thank you very much. I feel honored and somewhat uncomfortable to be the second on the panel. I should be the last. I have a minor portion of this.

Let me just review with you some of the work that has been done in Europe prior to this study. We started to cooperate within the framework of the European Nuclear Energy Agency, then called Nuclear Energy Agency. You remember the very intensive discussions about ECCS in the late '60s and the famous 1972 hearing on the effectiveness of emergency cooling in the U.S.. So it was natural that the Committee of Reactor Safety Technology (CREST) of the European Nuclear Energy Agency started to discuss nuclear safety methods, severe accidents and accident phenomena, especially on light water reactors. At that time Reg Farmer was the Chairman of CREST. He promoted the probabilistic approach very strongly. A working group was set up in 1968 to prepare a report on water cooled reactor safety, which got known as the *Yellow Report*. It was a kind of review on the state of knowledge in reactor safety. The report was not very widely distributed, it seemed to be too critical at that time for the Agency.

Then the reactor safety study, known as WASH-1400, came. I would just like to repeat some of the objectives of the study, because they are still very modern: To perform a more realistic assessment opposed to the "conservatively-oriented" safety approach taken in the licensing process for nuclear power plants; to identify areas where future safety research may be fruitfully directed; to provide an independent check of the effectiveness of reactor safety practices in the industry and the government. Concerning the results it was very important to see the significance of small leaks and transients. You may remember that the problem observed at TMI has been drawn up before in WASH-1400. I never forgot a small note that Saul Levine gave me when we had a discussion after TMI: He took the appropriate conditional probabilities out of the report - if I remember well, something like 10^{-2} or 10^{-3}.

So the findings of WASH-1400 promoted a very strong international discussion and cooperation, especially in Europe, but also within the OECD countries. The cooperation on severe accidents started after the publication of WASH-1400 and got certainly a strong emphasis after the TMI accident. Also, the discussion of "how safe is safe enough" was always embedded. Dave Okrent was the one who always pushed this question very strongly. And he got some answers after publishing of WASH-1400.

The first presentation of WASH-1400 in Europe was shortly after it's draft release in 1974. Norm and Saul Levine came to Paris to lecture for two days on the WASH-1400 report. Although it was a draft report, it was very impressive for the whole community to see the profound knowledge of both in going into details within their lectures. They didn't need assistance from their staff. So Pierre Tanguy wrote 1975 in his article "Que faut-il penser du rapport Rasmussen?" - "How do we feel about the Rasmussen report?": *"The Rasmussen report, even*

in its draft version, is a remarkable accomplishment which is supported by most of the experts in this area (...) a very big step has been done. This is a merit of Professor Rasmussen and Dr. Levine, the Atomic Energy Commission and all of their co-workers.". Because of the strong similarity of the first series of the French 900 MW reactors with those of the U.S. WASH-1400 example, the results were applicable to French plants. The later approach of the French licensing authorities regarding a probabilistic safety objective concerning the design basis accidents has been heavily influenced by the results of WASH-1400.

After the publication of the draft report, the German government asked us to perform a similar study, taking into account the different siting conditions, the different population and of course the different configuration of the German plants. As a reference plant we chose the BIBLIS nuclear power plant, a large pressurized water reactor. We could heavily use - as far as it was possible - the methods developed by WASH-1400. The biggest achievement was the event tree method, that really helped us. It would have been impossible to go only with fault trees. We certainly found some differences and I must also say, if we criticize ourselves, we didn't really address the hydrogen problem. We felt the temperature is high enough to burn the hydrogen immediately. Later on we had a very interesting and deep discussion in Germany, because probabilistic methods were not very much favored by engineers at that time, especially in the Reactor Safety Commission. They were traditional engineers and had some difficulties to accept the PSA methods. The discussion was helpful to some extent, since some of the findings went into design changes. In my impression this was an opportunity that has been taken much more aggressive in Germany than in the U.S.. If this is wrong, please tell me.

Let me come back to Norm. Norm visited Europe several times to lecture on the study. One famous series of lectures was in Denmark, Sweden and Norway. He always was called upon when there were debates to phase out nuclear power or not to enter nuclear power, as it was the debate in Denmark and in Sweden as well.

He received a Dr. Honoris Causa by the faculty of Engineering of the Catholic University of Leuven in February 1980, which is a very distinguished and old European university. It was a very memorable ceremony where I had the privilege to assist.

In 1987 Norm Rasmussen lectured in Zurich. Let me repeat some of his essential conclusions: *"I don't believe the methods are developed to the point that they should be the sole method for determining that plants are adequately safe."* He goes on, *"Nevertheless, they should be seen as an essential part of the process. I believe that eventually we will improve the methodology and our confidence in it to a point where it can be used to determine that an acceptable safety level has been achieved."*

I also found a note from Reg Farmer that he wrote to the CSNI - as it is called now - in 1990. I quote the final part of his note: *"I see danger in continuing the promotion of risk analysis several decades below the level which can be analyzed. If such numbers are issued by or supported by known authorities or people of recognized expertise, they will be believed and reused, even extended. Some opponents will not believe and will challenge. I am not an opponent. I encourage PSA but I don't like very small numbers"* - and I believe the same words could be spoken by Norm even now.

Thank you very much.

Twenty Five Years Since the Reactor Safety Study - The Legacy and the Lessons

by

Richard Denning, Battelle Columbus Laboratories
Professor and Chair
Nuclear Engineering Graduate Studies Committee
Ohio State University

Well, as you can see, I'm going to be talking about the history of severe accident analysis. And I couldn't actually make this lecture today without recognizing two people, Pete Cybulskis, who was in charge of the accident phenomenology work on WASH-1400 and still is doing accident phenomenology work at Battelle, and Bob Ritzman, who was in charge of the Fission Product Release and Transport work. Bob went on to EPRI, and is now retired in the Palo Alto area.

Well, in the beginning was WASH-1400. That's not totally true. In about 1968, the Atomic Energy Commission undertook some work to look at severe accident phenomena, and there were some experiments done on release of fission products from fuel. But in 1972, when we started WASH-1400, there really were not techniques to use in the analysis of severe accidents, and this was the beginning as far as an attempt to realistically estimate accident consequences.

These are some of the lessons that we learned from WASH-1400: First of all, we learned that severe accidents dominate nuclear power plant risk. That wasn't obvious before WASH-1400. There were many people that felt that the probability of severe accidents was so low that in a risk sense, they would not be risk-dominant. Also in safety analysis reports, the consequences of design basis accidents are significantly overestimated, so that it wasn't immediately obvious that severe accidents would dominate risk. But I think that WASH-1400 has shown that, and certainly has been upheld in subsequent analyses.

The key thing in a severe accident to reduce the consequences is to have the containment remain intact, and to remain intact for a significant period of time. So containment integrity is the focus of severe accident research or severe accident analysis.

Another thing that we found was that the design basis accidents for containments do not represent severe accident loads very well. So, when we look at containments that are designed for loss of coolant accidents, the suppression of steam from loss of coolant accidents, they may not behave very well with severe accident loads. Similarly, the source term that was used, that came out of the report, TID-14844, as the design basis accident source term, is not a very good representation of severe accident source terms. The TID-14844 source term emphasizes iodine in its elemental form, and gives very little treatment to the real problem in accidents, which is the large production of radioactive aerosols.

There were a number of containment challenges that were identified in WASH-1400. I'll point out hydrogen as perhaps one of the most important threats that was identified. Although we

undertook the analysis of severe accidents and tried to do it as realistically as possible, we really were not able to do an adequate job. The knowledge of severe accident processes at that point was really not adequate to examine all of the phenomena that affect the release and transport of radionuclides. In particular, in the accident progression area, most of the analyses in WASH-1400 are hand calculations that involve energy balances, mass balances; they don't consider many of the phenomena that actually occur in an accident, in detail. And as a result, the source terms in WASH-1400 are overestimated; the probability of early containment failure and containment failure in WASH-1400 is overestimated, and the timing of accident sequences is underestimated; that is, accidents in their progression, actually take longer than in WASH-1400.

At the conclusion of WASH-1400, it was recognized that there was a need for severe accident research to be able to better characterize these processes. The NRC undertook a comprehensive severe accident research program. Most of that work was done at Sandia, Idaho National Engineering Laboratories, Oak Ridge National Laboratories, although there were many other contributors, including Los Alamos.

Early on, however, the first uncontrolled severe accident experiment occurred at Three Mile Island, Unit II. As you know, there were two major accidents that occurred, and I won't talk about the insights that we gained from those accidents, though they were important. What I'll talk about are the political impact of those accidents. Prior to the Three Mile Island Unit II accident, WASH-1400 was a paper study. It was looked on by the industry as hypothetical. I don't think that the industry really believed the numbers, really believed in the credibility of severe accidents, but when Three Mile Island Unit II occurred, then the hypothetical became the credible, and certainly the Severe Accident Research Program would not have been undertaken with the magnitude of funds that it did, if there hadn't been a Three Mile Island Unit II accident.

At that time, the industry also, out of self defense, I think, began its own research program, more focused than the NRC program. In Europe and Japan, severe accident research programs were undertaken, and particularly in that time period, we had very high levels of interaction with theGerman research program. This was a time also of the great source term debates in which the contractors and laboratories of the Nuclear Regulatory Commission sat on one side of the table and the contractors of industry sat on the other side of the table, and there were debates on such issues as the magnitude of the peak pressure in high-pressure melt ejection accidents. Those were very stimulating interactions, and I think that they led to some general consensus, although it would be hard to believe that at the time, and they were obviously also very stimulating to both of the programs.

That period ended with two studies, the NUREG-0956 study that is the source term reassessment document that Joe mentioned, and also NUREG-1150.

Now, I'm going to very quickly run through some of the things that occurred in the severe accident work that was done, the severe accident developments. In the source term area, the most obvious thing is the recognition of cesium iodide as the principal chemical form of iodine released in severe accidents. That was actually recognized in WASH-1400, but it was too

controversial at that time for credit to be taken. Model development and experiments were undertaken in the areas of release from fuel, core/concrete interactions, transport and deposition of aerosols in the reactor coolant system, transport and deposition in containment.

In the melt progression area, the mechanics of the initiation of fuel damage, eutectic formation, slumping, vessel failure modes, debris coolability, melt progression in the concrete, and containment loads. The response of the containment was also studied looking at containment failure thresholds, locations, and mechanisms. The destructive testing of scaled models of containments also examined the potential for leak versus rupture. And then there was the development of a number of computer codes: the source term code package, which was the evolution and the documentation of the methods that were used in WASH-1400, that was soon replaced by the MELCOR Code developed at Sandia. On the industry side, the MAAP Code was developed, hence, through the world, MAAP and MELCOR are the primary codes now used for analyzing severe accidents for risk assessments.

We have mentioned that in NUREG 1150, we looked at the reassessment of the WASH-1400 plants with improved source terms. NUREG 1150 analysis of accident consequences for one of the -- for a very large release is shown in the viewgraph.

This is the source term distribution for bypass scenarios at Surrey. On the left we see the release fractions, the distributions for the different radionuclide groups from NUREG 1150. You can see very broad distributions. At the bottom, we have the fifth percentile of the distributions, and at the top, the 95th percentile. The upper bar that goes to the right is the mean of the distribution, and the lower one on the left is the median, and the little pluses that I put in there are the corresponding PWR-2 release category from WASH-1400. In general, the WASH-1400 value is at the 95th percentile level, about an order of magnitude above the best estimate median of the distribution. Similarly, for the Peach Bottom plant, the BWR-2 case has very similar results with the WASH-1400 results being systematically high at about the 95th percentile level.

The source term is only part of the question. The real question is when does the containment fail? And what's the frequency of early failures of containments? The next viewgraph from NUREG 1150 shows the conditional probability of early containment failure for a number of different accident scenarios.

The distributions you can see here are extremely broad, covering three to four orders of magnitude. The best estimates down here, the median is in the neighborhood of one times ten to the minus four. Up on the top, I've shown the WASH-1400 across all the accident scenarios, the probability of early containment failure in WASH-1400 was 20 percent. You can see three to four orders of magnitude higher than early containment failure probability in WASH-1400, although the difference between the WASH-1400 value and the mean is not as great. Thus there is a significant overestimation of early containment failure in WASH-1400, also in the timing. Everything happens a lot earlier in WASH-1400 than a mechanistic analysis shows.

The most important contribution of NUREG 1150 is a systematic approach to looking at severe accident uncertainties relative to the uncertainty treatment in WASH-1400 which was crude.

The other thing I'd like to point out is that NUREG 1150 shows that safety goals are a piece of cake. If you've got a plant with a ten to the minus four per year or better core melt frequency, you don't have to have a great containment design to be able to satisfy goals that were discussed.

Performing a risk analysis does not improve safety. The thing that improves safety is making a change in the configuration, or in the way that the plants operated. Let's look and see, based upon WASH-1400 and PRA insights, in the Level II area, how effective have we been in improving safety? And the answer is, only to limited extent.

Because of the expense of backfits, we have done limited upgrades but some important upgrades: Hydrogen control in the PWR ice condenser and in BWR Mark III designs, reduction of the interfacing system LOCA potential, wet well venting in BWR Mark I designs; some PWR reactor cavity reconfiguration. As Professor Birkhoffer pointed out in Germany, in Europe, they have gone a little bit further in some areas than we have in the United States. However, the place that Level II can really have an impact is on the design of future plants. If you look at the advanced light water reactor utility requirements document that was developed by EPRI for the industry, in there you see design criteria that not only relate to Level I, but also to Level II, and those designs. In the evolutionary designs that are now beginning to be implemented and in the passive reactor designs that hopefully will be implemented in the future, we have a much higher confidence of low risk than for the current designs of plants.

I'll point out another example here, deterministic severe accident design criteria, which were developed by Sandia for the heavy water new production reactor. These are deterministic criteria developed using risk insights.

Well, where do we go from here? I mentioned that there were two major severe accidents. The second was the Chernobyl IV. The big impact, politically, of Chernobyl IV was the public realization of the potential consequences of severe accidents in nuclear power plants.

I mentioned earlier that safety goals are a piece of cake. Well, it doesn't matter. As far as the public is concerned, they look at accident consequences; they don't look at accident frequencies. They have a hard time dealing with risks in a risk/benefit perspective. At the same time as there is this public distrust for nuclear power, the need is greater than ever before, and I don't have to tell you about what the need is today or what the need is going to be in 50 years for inexpensive sources of energy.

Will nuclear be able to play a role? I think that nuclear will be able to play a role, if and only if we can look the public in the eye and say a severe accident, a significant release of radionuclides cannot happen. The people in this room know that we live in a probabilistic world. We understand that. But we have to believe and be able to explain to the public, with their limited understanding of probability, that within limits of credibility, a major release of radioactivity can't happen.

Well, can we design a reactor that can't melt down? The answer is absolutely yes, we can. I think that as we look at some of the Generation IV designs, modular pebble bed reactor

designs, for example, we can design one where you can say really with a great deal of credibility, that it can't melt down. The question is, can we do it economically?

Okay, well, what are the possible futures? On the one hand, we have the open universe, and that is if we are able to develop new designs that, in my words, can't melt down; if we're able to then sell this to the public, then nuclear power is going to represent a major source of energy into the future of the world. If we can't do that we have a closed universe. Our current reactors will get to the end of their lifetimes, we will be in the middle of the 21st Century with an urgent need for moderately priced energy, but nuclear power is not going to be available to contribute.

Twenty Five Years Since the Reactor Safety Study - The Legacy and the Lessons

by

B. John Garrick, Chairman
Advisory Committee on Nuclear Waste
U.S. Nuclear Regulatory Commission

Hearing this long line of distinguished speakers on this panel with each succeeding speaker working diligently to say something different and interesting from the previous speakers reminds me of an observation made by one of my uncles at a recent family reunion. After visiting everybody and hearing what they had to say, he came up to me and said, "well, you know, you better be the last liar here or you won't have a chance." Now, I'm not suggesting that we're liars here, but the idea is that I think Hal Lewis being the last in line has a great opportunity.

Well, I have some good news and bad news: The bad news is, I have 17 viewgraphs that I prepared for my panel remarks; the good news is that I'm not going to show any of them. This way I hope I can avoid some repetition with the several panel speakers who have preceded me.

I had planned to comment on three eras of reactor safety, pre-Reactor Safety Study, the Reactor Safety Study itself, and finally, post-Reactor Safety Study activities. I'm now going to confine my remarks principally to the post-Reactor Safety Study era that was inspired by the Reactor Safety Study. The other panel members have done an excellent job of covering my other two topics.

Between 1976 and 1985, just following the completion of the Reactor Safety Study, there were some ten major risk studies performed by industry that greatly influenced future applications of probabilistic risk assessment. This was before the Nuclear Regulatory Commission had really become heavily involved in PRA applications through the Individual Plant Examination (IPE) program. I was fortunate enough to be the lead on about eight of those ten. These were very important studies because we were engaged in trying to specialize the methodology of WASH-1400 that was principally set up to answer a much bigger question than the question of the safety of a single nuclear power plant. As you recall, WASH-1400 was primarily attempting to answer the general question of the safety of a nuclear power industry. To be sure they used specific plants as surrogates to answer the bigger question and that did provide some clues on how to specialize the methodology.

We were faced with the task of taking the WASH-1400 methodology and specializing it to plant-specific and site-specific applications. The industry PRAs made contributions in two primary areas: One would be in the methods area, and here George Apostolakis is correct when he says that the fundamental methodology of WASH-1400 has been the basis for all of these studies. What we're talking about primarily are analytical tools within that fundamental framework for specializing some of those models and improving them for getting the kind of answers that were needed to quantify the risk of "specific plants." So, we're going to talk a little

bit about methods, and secondly, a little bit about results, the kinds of results that have impacted the PRAs that followed.

In each of the PRAs performed following WASH-1400, there were many new firsts. For example, the first commercially financed risk assessment was Oyster Creek, and we paid a great deal of attention to the Lewis report in terms of the criticism of WASH-1400 on such matters as the treatment of uncertainty and external phenomena. Thus, the Oyster Creek risk assessment was the first to attempt to propagate uncertainties through the logic diagrams and to systematically consider external events. That is, we included in the Oyster Creek study, as an integral part of the analysis, seismic and fire analyses. The studies that really resulted in some major analyses breakthroughs were the Zion and Indian Point studies. These studies were carried out by three major entities: my company, PLG; Westinghouse; and Fauske and Associates. The issue here was the nuclear plant sites close to large population centers. The focus was on accident progression following core damage and the effectiveness of the containment. In order to get some real insight into the effectiveness of containment, we had to go way beyond the analyses that had been performed to that point. And it was these studies that made considerable contributions to giving us insights as to the actual worth of some of the mitigating systems including containment. As you know, there was also a petition to shut down the plants close to population centers or to consider such extreme additional safeguards as filtered, vented containment systems, core catchers, and what have you.

One of the major outputs of the study was some very specific insights about the value received from constructing filtered, vented containments, and such other proposed safeguards as core catchers. Also, one result of the study was the importance of looking at alternatives. For example, we found that by going to a diesel-driven containment spray system, we could achieve essentially the same safety protection, at least for the plants we were looking at, as could be achieved from the filtered, vented containment system, at about one-tenth the cost. The bottom line is that these studies even in the face of challenges in the hearing room were effective in demonstrating the safety of the plants and avoiding such actions as shutting down the plants or requiring additional safeguards.

Another example was the Seabrook PRA. Among the firsts in this study was a much more detailed analysis than had been done before of close-in atmospheric dispersion and dose calculations following a core damage accident. The safety of the plant had been challenged by the neighboring state of Massachusetts in reference to the exclusion zone since the zone boundary went into Massachusetts. The analysis indicated that 95 to 98 percent of the early fatalities would occur within one to one and a half miles of the site, therefore providing a technical basis for a two-mile exclusion zone that would not extend into Massachusetts. Even though the analysis did not result in changing the exclusion zone boundary, it did satisfy a lot of concerns and avoided any further changes in the design of the plant or the boundaries of the site.

There were a number of other studies involving firsts in analytical methods supporting PRA. One of the most interesting applications involved a plant that was never completed, the Midland plant being constructed by Consumers Power in Michigan. For this plant the designer chose to incorporate the PRA thought process into the design process. This study gave us a great deal of confidence that PRA can make a major contribution to the design process. The benefit of the

approach is that it allowed through an iterative process a balancing of the design in terms of the contribution of the safety systems to the overall safety of the plant.

Another plant that was unique in many respects was Diablo Canyon. Diablo Canyon had a huge seismic program, and the question was, is it such a dominating contributor to risk that nothing else much matters? Well, the risk assessment put it in context, and while seismic was a major contributor to risk, it was not dominant in the sense that 90 percent of the scenarios that went to core damage were initiated by seismic events. It was more like 20 to 30 percent, which was quite a surprise to a lot of people.

The South Texas Project was a fascinating study because unlike most plants, South Texas was a three-train plant, that is, it has three safety trains. The question was, what do you get out of that third safety train? The PRA was extremely effective in quantifying the contribution to safety of the third safety train.

Now I would like to mention a few methodology contributions that came out of these studies.

One was the introduction of what we then called the scenario-based risk assessment model. It was also called the modular event tree model. The approach was developed primarily during the Zion/Indian Point studies and took advantage of the linear properties of event tress to unravel contributors to risk in a systematic and organized way. In particular, the event trees could be represented as linear operators and made possible the first formal ranking of such specific contributors to risk as initiating events, plant damage states, release categories, and different risk measures. The approach greatly facilitated the diagnoses of contributors to risk.

Another mathematical concept that has since become a widely accepted definition of risk is the "risk triplet." The concept is based on the idea that when one wants to know what the risk is, they are really asking three questions. What can go wrong? What is the likelihood? What are the consequences? I am pleased to say that the Nuclear Regulatory Commission has adopted the risk triplet definition of risk.

Many other new methods came out of the Zion/Indian Point studies performed in the early 1980s. One that the team was proud of had to do with the containment event tree. Unlike, the event trees for the plant model where the branch points mainly represented the state of such active systems as safety equipment, the containment event tree branch points were more related to phenomenological conditions of an accident. Examples are temperature, pressure, and other physical properties. The result was that it was now possible to model the entire system (plant, containment, site) in a consistent and quantitative logic model. It gave meaning to "full scope" risk assessments. Other innovations coming out of this and the other industry studies were the greatly improved treatments of uncertainty, more comprehensive models of common cause analysis, and a more explicit treatment of operator actions. The latter was achieved by putting operator actions into the event trees (making them top events) and representing the actual conditions of an accident sequence.

Perhaps one of the more important contributions that came out of the post-Reactor Safety Studies was the treatment of the data itself. If you're going to do uncertainty analysis, you need to have a convincing way to do it. The cornerstone of the uncertainty analysis was the application of the Bayesian methods of inferential logic.

Now, let me close by just commenting on some of the challenges and disappointments. Of course, those of us who are active in this business are always disappointed at the slowness with which risk assessment is accepted as a basis for making decisions about safety. As an analyst, I have always been amused by the level of confusion that exists on the difference between what I will call actuarial risk and assessed risk. There is the tendency for people to say you can't do risk assessment because you don't have the data. Well, risk assessment was invented because we didn't have the data. In particular, if you have lots of data, you don't need to do a risk assessment, and that's why I like to say that the need for a risk assessment is inversely proportional to the information you have, not directly. The real issue in risk assessment is mapping from the level of interest and the questions that you're trying to answer, down to a level about which you do have some information, and convincing yourself that that mapping is correct. Once you have done that, then, of course, you're on your way.

One other area that has always been of interest to me is the level of confusion on the meaning of quantification. Quantification is not the achievement of a precise number. Quantification is the full expression of your state of knowledge about something based on the supporting evidence. The analyst should employ whatever tools necessary to communicate his or her state of knowledge about what the risk is. Because there is always uncertainty about risk and the language of uncertainty is probability, it is natural that probability curves should play a major role in quantifying states of knowledge.

Finally, I would like to make a parting comment on the meaningless debate on the difference between deterministic and probabilistic reactor safety analysis. I maintain they are not competitive methods. One contains the other. In the spirit of the risk triplet, one could say that deterministic risk assessment is the "doublet", what can go wrong, and what is the consequence? All we should be doing when we do a quantitative risk assessment is adding "scope" to the problem, namely, an answer to the question, what is the likelihood?

Thank you.

PROBABILISTIC RISK ASSESSMENT DEVELOPMENT IN THE UNITED STATES 1972 - 1995

by

Joseph A. Murphy
Mark A. Cunningham
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission

INTRODUCTION

On the Tuesday after Labor Day, 1972, a disparate group gathered at the then AEC Headquarters in Germantown to discuss the formation of a new type of study of the safety of nuclear reactors, one that would estimate both the probability and consequences of severe accidents in U.S. nuclear power plants, and to meet Norm Rasmussen for the first time. Saul Levine, the project staff director, was a former regulator and known to at least the regulators. The group assembled consisted of personnel from what was then called "the Regulatory Staff" of the former AEC, Boeing Co. employees who had done quantitative analyses of the probability of inadvertent launch on the Minuteman project, personnel from the National Labs who had applied fault trees to studying the reliability of nuclear weapons and the estimate of inadvertent detonation, data analysts, personnel familiar with the severe accident experimentation that had been done to that date. And health physicists. An ambitious schedule was set. Initially, completion of a full risk assessment on two reactors was scheduled for the end of 1972. We only missed that three month schedule by three years.

We found there was much to be learned. We all had to change our basic approach to problem solving. The nuclear industry was still thriving. Plants were being built; regulatory tools such as the Standard Review Plan were just beginning to be considered for development. Most engineers thought in "success space", i.e., what was needed to meet established criteria. Risk assessment required thought in "failure space", i.e., what could cause an undesired event. Even our fault tree-ers had to re-calibrate. They were used to determining the likelihood of something operating when it wasn't supposed to. We were concerned about things failing to operate when they were supposed to. Our data analysts were hampered by the lack of a good data base, and, of course, by the fact that there were less than 100 reactor years of experience in the U.S. In some cases, data had to be inferred from other experiences. The large pipe break frequency analysis started with data from the large bore gas transmission pipes.

We started and rapidly came to the conclusion that using fault trees alone, coupled with an undesired event - unacceptable release from the plant - wouldn't cut it. There were two problems (1) the results were not parsed fine enough for consequence estimation, but, more importantly, it was almost impossible to represent all the conditionalities.

Let us explain. The probability of A and B occurring is the probability of A times the probability of B given A has occurred. Trying to represent these conditionalities in the fault tree eventually led to statements in the trees more than 100 words long, decipherable to no one, not even the author if he left the tree to go to lunch. In early 1973, Matt Taylor joined the team. He sat down with Norm and Saul and they discussed the possible use of decision trees from decision theory.

Matt went home, used up several sheets of paper, and came in the next morning, bleary-eyed, with the first event trees. Matt is really the father of the application of event trees in PRA.

Before going on, let us recognize those who went before us. While WASH-1400 was the first comprehensive analysis of a nuclear power plant there were earlier efforts that we relied upon. First and foremost, there was the father of the use of risk information in a regulatory context, Sir F. Reginald Farmer, known to all as Reg, who was with what is now called the Health and Safety Executive in the UK. Reg published a seminal paper in 1967. The Farmer paper proposed a systematic and comprehensive assessment of reactor risk by considering a complete set of initiating events and sequences proceeding from them. It includes consideration of structural failure in the reactor system, leading to release of radioactivity and the exposure of a population downwind of the accident. It appears that all of the elements of the modern PRA were envisioned, although not in the same form as employed today. He also proposed a means of judging acceptability which considered the estimated number of casualties as a result of release; and the increased risk incurred by an individual. He suggested one event in 1000 reactor years of operation as a starting point. Farmer further suggests a slope of -1.5 in the probability consequence curve, such that an increase of 100 fold in consequence is accomplished by a 1000 fold decrease in frequency.

Throughout the study, we received assistance from the UK. In particular, Eric Green, co-author of perhaps the first text on risk assessment, made several periodic visits to peer review our work and give advice. We also had the benefit of discussions with Prof. Birkhofer and his team doing the German Risk Study.

Modern fault trees were essentially invented by David Haasl, then of Boeing, for the Minuteman project. Dave provided training to those not already fault-tree-ers, and peer reviewed our work periodically. His focus on systems safety was contagious.

For solution of the Boolean equations associated with the fault trees, we employed tools that depended heavily on the early work of Fussell, Vesely, and Burdick at INEL and of Dick Worrell at Sandia.

The Reactor Safety Study was the first comprehensive assessment of the risks associated with nuclear power plants, examining the system design and operating practices at two specific plants, and using composite models to represent site meteorology and population density. The results were published in draft WASH-1400, Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, in August 1974. The draft report was widely circulated both internally and externally, and extensive comments were received. For example, the comments generated by NRC staff were almost as long as the report. One of the reviewers was Mr. Thadani. (They were, in the main, good comments and helped us improve our report, but they also gave us a good appreciation of how a license applicant must have felt at about the same time.)

Following modification in light of the comments received, the final version of WASH-1400 was issued in October 1975. It concluded, for the plants analyzed, that the risk of nuclear reactor accidents is much smaller than other man-made and natural events to which society is generally already exposed.

Probably as a consequence of this conclusion, WASH-1400 became controversial in the eyes of many people who regarded it as promotional of nuclear power rather than an objective safety study. The controversy led to a delay in the systematic application of PRA in the analysis of operating plants.

In time it was recognized that WASH-1400 provided new and important insights relative to reactor safety. Prior to the study, most emphasis by both government and industry was directed to protecting the plant against very large pipe breaks in the reactor coolant system. The general consensus within the reactor safety community was that the probability of a core melt was exceedingly low, but that the consequences would be disastrous, based on WASH-740 in 1959. In contrast WASH-1400 illustrated that the dominant contributors to the risk arose from small loss of coolant accidents and transients, and that while the likelihood of severe core damage was higher than earlier believed, the consequences were significantly smaller than earlier envisioned. It also highlighted the importance of support systems, such as the auxiliary feedwater system, which at that time was not regarded as a safety-grade system, and pointed out the significance of operator errors, e.g. such as the failure to manually transfer the emergency core cooling system to the recirculation mode, as required, once the refueling water storage tank was depleted, and the failure to restore manual valves to the correct position after routine periodic testing or maintenance.

Following the completion of WASH-1400, and similar efforts conducted in other countries (most notably, Phase A of the German Risk Study) (Reference 7), research efforts were initiated in several countries to develop advanced methods for assessing accident frequencies, improved means for collecting and analyzing operational plant data were put in place, methods were initiated to improve the ability to quantify the effects of human errors, and studies to better predict the nature and effect of common cause failures were begun. Further, the NRC initiated limited research on those key severe accident physical processes identified in the Reactor Safety Study, e.g., research on the likelihood and magnitude of steam explosions under various conditions, molten core-concrete interaction, and initial efforts to determine the release of radioactive material from molten fuel, including a better understanding of the basic physical and chemical properties of the materials that might be involved in a severe accident.

In parallel, the NRC staff began a gradual use of probabilistic risk analysis techniques to support the regulatory process. One of the first important use of such methods in the United States was the investigation of the risk impact of a broad range of generic safety issues to develop a list of the higher priority issues in 1978.

However, expanded application of probabilistic techniques was held back as a consequence of the interpretation of the peer review of the Reactor Safety Study performed by a special committee (the Risk Assessment Review Group) chaired by Prof. Harold Lewis, who is here today. Dr. Budnitz, also here, was also on the Committee. The Lewis Committee found that "...the fault-tree/event-tree methodology is sound, and both can and should be more widely used by NRC." However, the Committee also reported that WASH-1400 was inscrutable, and that it was very difficult to follow the detailed thread of calculation through the report. The committee also found that WASH-1400 was "a conscientious effort to apply the methods of fault-tree/event-tree analysis", but that there were a number of sources of both conservatism and nonconservatism. Thus, the committee concluded that "We are unable to define whether

the overall probability of a core melt given in WASH-1400 is high or low, but we are certain that the error bands are understated." At the time, as already noted, many people focused on the negative aspects of the Lewis committee findings. However, the committee also made several positive recommendations to the Commission for future efforts. These included the following: (1) re-evaluate the inspection and quality assurance systems and licencing criteria to reflect the lessons learned, (2) use PRA methods to guide the reactor safety research program, (3) use PRA methods to uncover the topology of accident sequences, even when there is inadequate data, but state the limits of knowledge, (4) avoid use of PRA methods for the determination of absolute risk values unless an adequate data base exists and the uncertainties can be quantified, (5) PRA methods should be used to deal with generic safety issues, to formulate new regulatory requirements, to assess and revalidate existing regulatory requirements, and to evaluate new designs.

On January 18, 1979, the Commission issued a policy statement on "Risk Assessment and the Reactor Safety Study report (WASH-1400) in Light of the Risk Assessment review group Report". Further, the Commission provided instructions to the staff in a memorandum from the secretary of the Commission to the EDO, Lee Gossick.

These instructions are three pages long, and will not be repeated here. They actually are quite reasonable, emphasizing the need for adequate data and for uncertainty analyses. They indicate quantitative techniques are better suited to relative instead of absolute comparisons. Examined in the light of current guidance in the current PRA Policy Statement and Regulatory Guide 1.174, they show great foresight. However, there was a requirement that the staff ". . . review the extent to which past and pending licensing or other regulatory actions . . . have relied on the risk assessment models and the risk estimates of the RSS. The Commission will examine the results of this review to determine whether the degree of reliance identified was and continues to be justified and to decide whether regulatory modifications are appropriate." This had a chilling effect. Needless to say, there are upwards of 20 memoranda in our electronic file which say basically I didn't use it, and I have no plans to.

The 1979 accident at Three Mile Island substantially changed the character of the analysis of severe accidents world-wide. Based, at least in part, on the comments and recommendations of the major investigations of that accident (Kemeny, et al.; Rogovin, et al.), a substantial research program on severe accident phenomenology was planned and initiated with international sponsorship. Both the Kemeny and the Rogovin reports also recommended that probabilistic risk analysis techniques be used to complement the traditional deterministic methods of analyzing nuclear plant safety, and that probabilistic safety goals be developed for nuclear plants. This revived PRA activities at NRC in both research and licensing activities.

The next major landmarks in probabilistic safety analyses were the Zion and Indian Point 2 Probabilistic Risk Analyses, published in 1981 and 1982, respectively, and led by Dr. Garrick, who is also here. These studies were important for several reasons:

(1) The methodology employed focused on the uncertainties involved in a more comprehensive manner than in any study performed earlier, using improvements that went beyond WASH-1400 methods, presenting results in the form of distributions, and employing the principles of Bayesian statistics in a more rigorous manner. The studies also explored

the risk significance of a variety of design options which had been suggested as possible ways to reduce risk, if such reduction were to be required.

(2) A detailed containment analysis was performed which indicated that in most cases a severe core damage accident did not lead to containment failure, whereas WASH-1400 considered that containment failure or leakage was likely.

(3) The two PSAs represented the first studies performed by the utility industry to respond to the specific regulatory concern, viz., the high population density near these sites, and the potentially high risks that might obtain if these plants had similar risk profiles to the WASH-1400 plants.

(4) The Indian Point 2 study was subjected to review in an extensive public hearing. This was the first time a probabilistic safety analysis was reviewed for acceptability in an adjudicatory process, and it stood this test.

Since the Zion and Indian Point studies, the use of probabilistic techniques by the industry steadily increased, both as a tool by industry to improve plant operation and safety, as well as to respond to specific regulatory concerns (e.g., the risk-based optimization of Technical Specifications). The NRC sponsored the RSSMAP and IREP studies and numerous probabilistic safety assessments were performed throughout the world. A number of the early studies, as well as compendia of results were examined for insights in NUREG-1050, Probabilistic Risk Assessment (PRA) Reference Document, published in 1984. NUREG-1050 was subjected to peer review by a committee formed under the aegis of the National Science Foundation by Dr. Vince Covello, with generally favorable results.

Many of the insights provided in NUREG-1050 remain still valid, but we will mention but two because of their global importance: (1) "The process of performing PRA studies yields extremely valuable engineering and safety insights. Conceptual insights are the most important benefits of PRAs, and the most general of these is the entirely new way of thinking about reactor safety in a logic structure that transcends normal design practices and regulatory processes. PRA methods introduce much-needed realism into safety evaluations, in contrast with more traditional licensing analyses that generally use a conservative, qualitative approach that can mask important matters," and (2) "While much attention has been placed on dominant accident sequences and ways to reduce risk even further, one of the most important insights gained from PRAs is the need to identify and maintain the reliability of risk-important systems and components at or near the levels now present. Degradation of such systems or components can sharply increase risk or the likelihood of core melt. A safety or reliability assurance program appears to be the desirable way to proceed. ...". The positive peer review of this document made possible the funding of the NRC's next major effort, NUREG-1150.

In 1986, the Commission issued the safety Goals for the Operation of Nuclear Power Plants; Policy Statement to establish goals that broadly define an acceptable level of radiological risk. Further direction and clarification to this Policy Statement was provided in a memorandum from the Secretary of the Commission to the EDO in 1990. This statement permitted improvements

in the Regulatory Analysis Guidelines which are used to evaluate potential modifications to regulations against the requirements of the Backfit rule, and gave guidance on how safe was safe enough.

Another important milestone in NRC's work was the 1986 publication of the staff's reassessment of containment performance and source terms. This work summarized progression severe accident research and described the embodiment of this progress in a new computer code, the Source Term Code Package. This milestone also had the indirect effect of initiating a study of the risk implications of this new source term technology. While originally envisioned as a short and relative simplified study, this risk assessment grew to what we now know as NUREG-1150. Early in this study, it was recognized that the Source Term Code Package, though state-of-the-art, did not address certain phenomena which we believed were real and simply could not be ignored in a document attempting to analyze the risk associated with a nuclear power plant (e.g., high pressure melt ejection, revolatilization of deposited material in the reactor coolant system after vessel breach). Thus, it was clear that it would be necessary to use expert opinion, because comprehensive models, accepted by consensus in the technical community, did not exist in many areas. In fact all earlier studies had made extensive use of expert judgement, but did not clearly identify it as such. Therefore, it was decided that NUREG-1150 would explicitly identify how expert judgement was utilized in the analysis. Also it would respond to the comments of the Lewis Committee to the effect that sufficient attention had not been given to the uncertainty associated with the earlier WASH-1400 study, considerable emphasis was given to the uncertainty analysis. It presented the perspectives gained with respect to severe accident frequencies, containment performance, and risks; risk significant uncertainties that merit future research; and comparisons with NRC safety goals. Finally, it provided a set of PRA models and results that could support the prioritization of potential safety issues and related research.

There was extensive peer review. This included review by two independent committees of experts sponsored by the NRC, one, chaired by H. Kouts focusing exclusively on the uncertainty methodology, published as NUREG/CR-5000; and a second, charged with performing a comprehensive review of the entire report and chaired by W. Kastenberg. In addition, a significant review was performed for the American Nuclear Society by a committee chaired by L. LeSage, and the team had the benefit of extensive public comments, suggestions and recommendations from colleagues both domestic and international, and discussions at an IAEA Workshop. After modification, there as another review by a special committee chaired by Dr. Herbert Kouts.

The peer review process on this scale was extremely time consuming and expensive, but, in retrospect, clearly worth the effort.

The next major milestone was the request for plants to conduct an Individual Plant Evaluation. In the Commission policy statement on severe accidents in nuclear power plants issued in 1985, the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there was no present basis for immediate action on generic rulemaking or other regulatory requirements for these plants. However, the Commission recognized, based on experience with plant-specific probabilistic risk assessments, that systematic examinations are beneficial in identifying plant-specific vulnerabilities to

severe accidents that could be fixed with low cost improvements. Therefore, each existing plant was requested to perform a systematic examination to identify any plant-Specific vulnerabilities to severe accidents and report the results to the Commission. The general purpose of this examination was for each utility (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant, (3) to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) if necessary, to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents. The Commission noted that the maximum benefit from the IPE would be realized if the licensee's staff were involved in all aspects of the examination to the degree that the knowledge gained from the examination becomes an integral part of plant procedures and training programs.

The effort involved was massive. Besides the appreciation for the risk profile of each plant gained by the operating organization, overall perspectives on reactor safety and plant performance were also gained. These are documented in NUREG-1560, itself a major milestone in permitting an understanding of the global state of the art and of the risk profiles of existing plants.

In 1991, as the use of PRA as a regulatory tool began to expand, the EDO established the PRA Working Group to improve consistency in staff use of PRA. The Working Group defined a set of basic principles for staff use of PRA , identified the need for improvements in guidance development, training enhancements and methods development. They developed guidance for use of PRA in screening safety issues and in analyzing issues in detail. It was published as NUREG-1489 in 1994.

Recognizing the need for overall policy guidance, the Commission issued the PRA Policy Statement in 1995. In this statement, the Commission expressed the following:

> (1) The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.
>
> (2) PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

(3) PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

(4) The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

At this point, we bring this history to a close. This, then is an overview of our legacy. Other panels today will discuss the present and future. Thank you

"You Can't Have One Without the Other"

The Honorable Nils J. Diaz
Commissioner
U.S. Nuclear Regulatory Commission

Remarks before the 28th Water Reactor Safety Meeting
October 23, 2000

Good afternoon. It is indeed a pleasure to be with you here today and address the Twenty-Eighth Water Reactor Safety Meeting. Many of you have dedicated your professional careers to researching and using tools that are important to protection of the health and safety of the public, in the United States of America and internationally. As a decision maker, I want you to know that I personally consider that well-executed, safety-focused research has played and will continue to play, a significant role in assuring safety. It is obvious to all nuclear practitioners that there is a tightly-knit light water reactor (LWR) research community working on important and closely-related issues, with many well-established feedback and communication mechanisms serving everyone. I believe it is important that we continue to work together to identify and support the research needed to better protect the workers and the public. Therefore, because you are in positions to identify, initiate, and conduct research, I consider this to be an ideal audience to present my views on our need to focus our research based on two fundamental regulatory principles: one a mature principle and the other much less so. The first regulatory principle has always been there but occasionally seems to get lost in the background: nuclear regulatory activities, and regulatory research, are conducted to reduce the potential for undue radiological exposures and/or to mitigate radiological consequences to the public from the peaceful uses of nuclear energy. For LWRs, radiological protection during normal operation, anticipated transients and accidents is the ultimate requisite that should drive regulatory research efforts.

Prioritization of regulatory research efforts needs to start at characterization and prevention of undue radiological risks to the public. I will discuss a few examples. Let me start at the very source: the fuel rods. During last year's meeting, this community discussed the Nuclear Regulatory Commission's (NRC's) regulatory criteria and research efforts to address the issues associated with high burnup fuels. There are emerging issues associated with high burnup fuel that the research community as a whole, including the industry and the regulator, needs to study for greater understanding. Whether it is research on the fuel and cladding behavior under high burnup conditions, the merit of new cladding alloys, or the analysis of radiological source terms from high burnup fuels under hypothetical accident conditions, we should always consider the ultimate goal: to prevent and/or to mitigate radiological consequences to the public.

The second level of defense against potential radiation release is, of course, the reactor coolant system. We have been working on the prevention and mitigation of loss of coolant accidents (LOCAs) and other reactor transients for more years than we care to remember. Experimental research has yielded valuable data in boiling, dryout, blowdown effects, and other phenomena that have been used in empirical correlations for the thermal hydraulic codes and for reactor design. Thermal hydraulic analysis of hypothetical accidents has also been evolving. Advances in computer technology and telecommunications have created opportunities and challenges for

us to advance state-of-the-art thermal hydraulics and neutronics. How many of you remember bringing punched cards to be read by the mainframe computer while praying that you do not drop those thousands of cards? Today we can sit by our PCs, build our analytical models through the Graphic User Interaction (GUI) input, verify the intended configuration through a 3-D Model Viewer and press "Go". Using new know-how in thermal hydraulics and developments in computer technology, we have made significant advances in conventional thermal hydraulics and computational fluid dynamics.

At the NRC, our initiatives include the thermal hydraulic codes consolidation program and the development of alternatives to 10 CFR 50.46 (acceptance criteria for emergency core cooling systems). As the recent Commission paper (SECY-00-0198) on the status of risk-informed changes to 10 CFR Part 50 stated, the staff is working with various owners' groups on the feasibility of applying risk insights and more realistic models to assess potential alternatives to large break LOCA acceptance criteria and their implications for other plant design and performance requirements. Other research activities include the materials reliability program sponsored by Electric Power Research Institute (EPRI) and the study of steam generator tube integrity under severe accident conditions. The NRC is also conducting research, in cooperation with EPRI and other countries, in areas related to steam generator tube flaw development and propagation, tube leak and burst behavior, as well as improved non-destructive examination (NDE) techniques. I believe that steam generator issues are presently of the highest priority and they require prompt resolution. As I see it, these research activities are directed toward the goal of preventing and/or mitigating undue radiological consequences to public health and safety.

The other regulatory principle, very much applicable to our research, forms the basis for risk-informed regulation: regulatory requirements are to be commensurate with safety and risk significance. Under a risk-informed regulatory structure, attention and resources are devoted primarily to structure, systems, and components (SSCs) and issues of higher safety and risk significance. Requirements would be reduced accordingly for lower safety and risk significant SSCs and issues. For those with little or no safety significance, there would be minimal to no requirements. Of course, the devil is in the details: how to define these risk levels to make sound, state-of-the-art decisions.

The radiological protection and the risk-informed principles are not antagonistic, rather, they are complementary and synergistic. Today, like the "Old Blue Eyes" song said about love and marriage: "You can't have one without the other."

Based on these two fundamental principles, the U.S. NRC is developing a risk-informed regulatory framework. They have been stated in multiple ways. For example, in 1995, the probabilistic risk assessment (PRA) Policy Statement focused on the use of PRA, or should I say, probabilistic safety assessment (PSA)?, with statements like:

- the use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data as well as in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

- PRA should be used to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices.

- PRA evaluations in support of regulatory decisions should be as realistic as practicable. Appropriate supporting data should be publicly available for review.

Later, the Commission issued the White Paper on Risk-Informed Regulation (SECY-98-144), broadening the risk-informed framework to include the concurrent use of operational safety experience and conventional engineering and risk analysis, both coupled with defense-in-depth and probabilistic safety analysis. SECY-98-144 provides the first Commission definition of risk-informed regulation and it states:

"Risk-informed Approach to Regulation"
A risk-informed approach to regulatory decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety.

"A risk-informed approach enhances the deterministic approach" by (a) allowing explicit consideration of a broader set of potential challenges to safety, (b) providing a logical means for prioritizing these challenges based on risk significance, operating experience, and/or engineering judgment, (c) facilitating consideration of a broader set of resources to defend against these challenges, (d) explicitly identifying and quantifying sources of uncertainty in the analysis (although such analyses do not necessarily reflect all important sources of uncertainty), and (e) leading to better decision-making by providing a means to test the sensitivity of the results to key assumptions.

A risk-informed regulatory approach can also be used to reduce unnecessary conservatism in purely deterministic approaches, or can be used to identify areas with insufficient conservatism in deterministic analyses and provide the bases for additional requirements or regulatory actions.

One example of how these two principles work together relates to the third level of protection against radiological release, the containment. Experimental and analytical work has been done to assess various containment designs and containment cooling and depressurization systems. Recent research has focused more on the severe accident and its effects on containment. Among other things, this work addresses direct containment heating phenomena and the combustible gas control system. Research results and risk insights associated with combustible gas generation and combustion have led to our belief that combustible gases are not a significant challenge to containment integrity in the near term (about 24 hours) following a hypothetical severe accident. For the longer term, combustible gases may be a concern for containment integrity and should be addressed in the plant's severe accident management program. Because of this improved understanding, NRC staff has initiated work to develop risk-informed alternatives to 10 CFR 50.44 (standards for combustible gas control system). In SECY-00-0198, the staff recommended that the NRC eliminate the requirement to measure hydrogen concentration in containment and the requirement to control combustible gas concentration resulting from a postulated LOCA. However, the staff recommended that the requirement for licensees to provide an inerted atmosphere for Mark I and II containments be retained.

Risk-informed regulation is not a panacea; it will not solve all problems or settle all issues. However, I believe that the increased use of a balanced, risk-informed approach to regulation

will enhance radiological protection with these measurable outcomes: better safety decision-making, more effective and efficient use of agency and licensee resources, and elimination of unnecessary burden. Achieving all of the above should result in an increase in public confidence in the regulation and use of nuclear materials.

There are multiple current challenges to the development and implementation of a risk-informed regulatory regime, including cultural, technical and institutional issues. For example, important challenges stem from the voluntary nature of key risk-informed rulemakings, as well as the complexity of dealing simultaneously with multiple regulatory approaches. Today, the challenge of PRA quality is still hampering the development of risk-informed regulation. PRA quality means the effective combination of traditional PRA methodology with the underlying engineering analytical methods as well as the supporting operational database. There is an urgent need to settle the issue of PRA quality in a manner that serves the regulation and utilization of nuclear power with better definition of assurance of public health and safety.

Our greatest developmental need then is the skillful molding of the present scientific and engineering know-how into research tasks that will result in the advances needed to use risk-informed methods, as appropriate, and to characterize and prevent undue radiological risks to public health and safety.

I know these tasks are in good hands.

United States
Nuclear Regulatory Commission

CHALLENGES IN THE FUTURE FOR RISK-INFORMED REGULATION

MATERIALS AND WASTE

2000 WATER REACTOR SAFETY MEETING
October 23, 2000

Margaret V. Federline
Deputy Director
Office of Research

BACKGROUND

- 1995 PRA Policy Statement
  - PRA technology should be increased in all regulatory matters
    - -To the extent supported by the state-of-the-art in methods and data
    - -To complement deterministic approach and support defense-in-depth
  - Single approach for incorporating risk analyses may not be appropriate given dissimilaritites in nature and consequences of uses
  - PRA evaluations supporting decisions should be as realistic as practicable and supporting data should be publically available

- Objective is to highlight challenges to risk-informed regulation in materials and waste

RANGE OF MATERIALS/WASTE ACTIVITIES

- Thousands of Agreement State and NRC licensees

- About 40 activities
  - medical diagnosis and treatment
  - waste storage and disposal - high and low levels
  - well logging
  - gauging
  - process control
  - radiography
  - irradiators
  - decommissioning
  - transportation
  - fuel cycle

CHARACTERISTICS CHALLENGE USE OF RISK INFORMATION

- Diversity in characteristics of activities, devices, systems, and licensees

- Complexity varies from simple sealed-sources to large fuel cycle facilities

- Hazard, accident potential, and failure modes vary
  - Human error is frequent cause of system failure

- Data available for risk analyses varies from large sets of event data from similar systems to few, if any, event data

- Licensee communities vary in terms of technical sophistication

- Time scales vary significantly
  - Millennia for waste disposal - hours for medical uses
  - Intergenerational risk

QUESTIONS FOR RESEARCH CONSIDERATION

- Are diverse types of activities amenable to risk-informed approaches?
- What questions are appropriate for each activity or for each approach to decide on feasibility?
- What tools are available and how do we decide which one to use?
  - Integrated Safety Assessment
  - Hazards Analysis
  - Performance Assessment
- How will we decide if sufficient information exists for risk-informed approaches for a particular activity?
- What new/improved tools and databases are needed?
  - Predictive or actuarial
- What are the considerations for risk management strategies?

ACCEPTANCE ISSUES

- Public Acceptance
  - Will they trust evaluations covering millennia?
  - Public perception of risk is often a driver in risk management

- Licensee Acceptance
  - Will they be willing to pay for development costs?
  - Will the increased cost of analysis be mitigated by reduced cost of compliance and increased flexibility?

- Agreement State Acceptance
  - Will they agree with the new approach?

QUESTIONS FOR IMPLEMENTATION

- What specific use is the staff expected to make of risk insights?
  - Developing rules and guidance
  - Licensing
  - Inspection
  - Enforcement

- What specific use is the licensee expected to make of risk insights?
  - Design
  - Planning
  - Operations

- Answers will be different for different technologies
  - Hazard and complexity of operations
  - Degree of human involvement
  - Technical sophistication of the licensee community
  - Agreement State issues
  - Public perception

PATH FORWARD

- RES will continue to provide tools and scientific bases
- Overarching strategy and Risk Informed Regulation Implementation Plan
- NMSS Framework in SECY-99-100 - high level parallel of the reactor framework
- Performance Assessment methods for high level waste
  - EPA standard for doses from radioactive material is probabilistic
- PRA of transportation and storage systems
- Integrated Safety Assessments required for fuel cycle facilities
- Hazards analysis being developed for medical and industrial applications
- Case studies to investigate draft screening criteria and feasibility of safety goals

"Perspectives on Research's Role in Regulation"

The Honorable Greta Joy Dicus
Commissioner, U.S. Nuclear Regulatory Commission

at the

28th Water Reactor Safety Information Meeting
Bethesda, Maryland
October 24, 2000

Good morning ladies and gentlemen. I am very pleased to have the opportunity to speak to you at this conference. Today I intend to provide my perspectives on some of the activities within Research which I believe are a very important part of the NRC mission. In particular, my remarks will be focused on the following: (1) how important I perceive the office of Research's role to be; (2) current initiatives which benefit from Research's support; and (3) challenges which provide opportunities to shape Research's future, But first I would like to recall what Congress had in mind when it formed the office of Research.

The Energy Reorganization Act of 1974 stipulated that the Director of Nuclear Regulatory Research shall perform such functions as the Commission shall delegate including: (1) developing recommendations for research deemed necessary for performance by the Commission of its licensing and related regulatory functions, and (2) engaging in or contracting for research which the Commission deems necessary for the performance of its licensing and related regulatory functions.

Of note as stipulated in the Act, was that the head of every other Federal Agency shall cooperate with respect to the establishment of priorities for the furnishing of such research services as requested by the Commission for the conduct of its functions. This is a mandate that we should continue to exploit to the maximum benefit for our research activities. As I'm sure many of you have heard from those within the NRC the research budget has decreased from a high of over 200 million in the past to about 42 million in the last fiscal year. This is due in part to the fact that the nuclear industry has matured. This has provided challenges for the NRC to get the most from each research dollar to support both short term and longer term activities that support the Agency's mission.

As I hope that most of you know by now, in meeting this challenge the NRC has adopted a strategic plan that articulates four primary objectives: (1) to maintain safety; (2) to improve public confidence; (3) to make our regulatory processes more effective, efficient, and (4) to reduce unnecessary regulatory burden. In the process of meeting these objectives I believe we are benefitting in that we are focusing our research efforts to gain the maximum benefit for the stakeholders we serve.

Over our recent history the NRC has been challenged to redefine or at least re-examine Research's role and future direction. It pretty much began with an issue paper, Direction Setting Issue 22 written in 1996, which posed fundamental questions about what role research should play in meeting the Agency's mission and it also provided several recommendations. Since then there have been several status reports to the Commission and one of the outcomes of NRC's efforts to increase its efficiency and effectiveness has been to fold many of the responsibilities

previously charged to the NRC Office of Analysis and Evaluation of Operational Data into the Office of Research.

I'm sure some of you may have heard that recently a panel was convened to review what role research should have in our current and future regulatory environment in an effort to gain input from stakeholders. And I will do a little advertising and mention that tomorrow, my fellow Commissioners Merrifield and McGaffigan will be part of a discussion on this subject. I'm pleased by the diversity that has been brought to the panel which is chaired by former Commissioner Kenneth Rogers and includes membership from academia, public interest, industry, other federal agencies, former NRC executive managers, as well as, congressional and senate staff representation. I have studied some of their preliminary recommendations and I understand that they are only about half way through their study; but I am intrigued by the scope of their individual recommendations. And while the focus of the panel so far has not specifically identified the role of research with respect to materials issues, I am sure this panel will give appropriate consideration to those research activities because there are many materials challenges that go hand in hand with the future of nuclear power in the U.S. Also, I noted a question posed by several members of the panel was whether the Offices of Nuclear Reactor Regulation and Nuclear Materials Safety and Safeguards should also be solicited to provide input. However, even if these offices do not participate as part of this panel, I am confident that any future changes to the direction of our research programs would surely be weighted in on by all NRC stakeholders at the appropriate juncture.

One particular aspect I would hope to see as an outcome of this effort would be recommendations regarding what minimum staffing level or minimum core areas of research might be necessary to maintain research's ability to respond to future challenges. Recently, I read where the technology boom in the Silicon Valley and other similar technology centers is taking the best and brightest from government research laboratories. It can only stand to reason that the same might hold true for our University expertise base.

Because the chance to become an Internet millionaire is very alluring, I think we might need to start looking at ways to ensure our current base of technical expertise which we frequently draw upon, the national laboratories, does not become too watered down. One thing I am very mindful of every time I review the NRC's budget is, what level of funding will ensure that RES can efficiently and effectively function to support the NRC mission while maintaining highly qualified respected technical staff who produce high quality products.

CHALLENGES THAT TRANSLATE TO OPPORTUNITIES

Regulatory Initiatives

One of NRC's management challenges is to develop and implement a risk-informed, performance based regulatory oversight program. We are answering this challenge by working with industry on risk-informing 10 CFR Part 50 through several initiatives focusing on what has been referred to as "special treatment" requirement and piloting risk-informing regulations such as 10 CFR Part 50.44. Years ago when research for much of today's regulatory framework was conducted using experience, testing programs, defense-in-depth philosophy and engineering margins incorporated to account for areas of uncertainty, we didn't have the benefit of quantitative estimates of risk. This framework has served our nation quite well for many years, and we don't expect to throw it out and start over. Rather, given that the margin of

safety is a recurring issue in the implementation of risk-informed regulation we must not lose sight of the benefits of research to identify which margins do -- and which do not -- contribute to safety. As we move into the 21st century, continued research directed at quantifying margins should NOT be confused with the perception that while reducing regulatory burden, to support risk-informed regulation we are also improving safety. Remember we now have much commercial operating experience and research to consider as a result of the ensuing years of inquiry and challenges the nuclear industry has brought us all -- and we should try and benefit from this knowledge in every way possible.

We must also be mindful of the impact of industry deregulation and license transfers on those we regulate. While we will always conduct our activities so as to be true to our mission to regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, to promote the common defense and security, and to protect the environment -- that does not mean that we cannot support industry initiatives such as the development of technical basis to support license renewal, or risk inform our current regulatory requirements and appropriately reconcile these requirements to allow licensee's to more efficiently and effectively focus their resources in those areas where their impact on improving safety will have the greatest result.

I believe the NRC has been responding to the changing environment well, but I'll be the first to agree we can continue to do more. And I believe that the staff is up to this challenge. For example, earlier this year we launched implementation of the new power reactor oversight program for all plants. If you will recall, last year we piloted the new program with a few plants, made adjustments and subsequently initiated the program for all reactor licensees in April of this year. A key part of this initiative is that risk insights were used and we are making every aspect of it transparent as possible -- one just needs to visit the revised reactor oversight program webpage accessible through the NRC's homepage to see what I mean. And while there is agreement that lessons learned since its recent wide scale implementation suggest that more changes to the program will probably be necessary, I think we all can agree that overall the effort has been a success to a large part because of stakeholder input. And I think experience gained through research has contributed to this effort and we are currently looking to Research in conducting studies aimed at developing data and methods to risk-inform the various performance measures.

Decommissioning is another area where we have been working with stakeholders to remedy inefficiencies in our current regulatory framework which was largely established from the perspective of operating reactors. As a result, in the power reactor area, the NRC is taking a formal look at our whole approach to decommissioning to see if we need to create a new regulatory framework, and to see if we can focus on the areas of greatest risk. This year the staff proposed an integrated rulemaking plan and has been discussing its recommendations with stakeholders. Research is contributing by examining various analytical tools and studying the viability of possible approaches to decommissioning, such as entombment.

Participation & Communication

Closer involvement and improved dialog with the industry and all stakeholders is required in order to better define and focus NRC research efforts. Only through such interactions will it be possible to obtain broader support for research programs. And meetings like this one are just

one of the many ways we can actively achieve education of and input from all of our stakeholders. Looking at the various topics that will be discussed I see there are papers from both the staff and industry experts which give me the impression that we are making progress toward working together on challenging technical issues. Another way to raise consciousness for the value of research is to ensure that our research products provide relevant recommendations toward improving our regulatory structure.

I think if we are going to be successful in making the case for maintaining the current funding levels or perhaps even increasing funding we will have to get better at communicating and demonstrating how research dollars have benefitted safety and are providing products to support concerns such as license renewal, power up-rates, increased fuel burnup, and mixed oxide fuels. To quote Mr. Thadani "we would have had a difficult time moving as rapidly as we did on license renewal without anticipatory research." Much of which contributed significantly to the beginnings of the first Generic Aging Lessons Learned report. Obviously, explaining to stakeholders the costs of such efforts in terms of anticipatory research dollars should increase confidence in what we consider to be forward thinking research activities.

Timeliness of Our Activities

However, there is one aspect with respect to our research activities that I am very sensitive to, which is timeliness of outcomes. Frequently, we find real world uses for our anticipatory research, but we end up taking many years to see the results to fruition. Our research programs must be timely and responsive to both internal and external stakeholders. I suppose resources could be part of this mix, but I would also argue that management oversight might also be a contributing factor. I believe one way to ensure we can improve performance in this area is to get input early on from all stakeholders. I can assure you that while I am on the Commission I will be very critical of research activities that lend themselves to improving our regulatory infrastructure but do not have an aggressive schedule for seeing their contribution through to improving our regulatory framework.

Cooperation with Independence

As resources for research become more subject to challenge, I think we can really benefit by maintaining our existing relationships and looking to develop new relationships and cooperative agreements with our Federal colleagues, private sector stakeholders, and international colleagues. For example, I noted that with respect to one of the topics that will be discussed, digital instrumentation and control, the research staff have identified that digital failure assessment methods are currently used by defense and aerospace industries to determine types of failures and their impact on overall safety. Also, the railroad industry has experience with systems which we foresee as being potentially viable for the nuclear industry. Obviously the practical experience and research results from these parties could serve as a minimum -- as a starting point as the NRC begins to determine and gather information on digital instrumentation and control failure rates to better assess the risk from the increased use of this type of equipment. Another example that has already yielded significant results is the successful collaboration between the NRC and industry in the 1980's on research projects under the auspices of the Nuclear Plant Aging Research Program which lead to development of much of the basis for our conclusions that license renewal was viable. And just recently at the conference of the International Atomic Energy Agency the U.S. and France signed an agreement on scientific and technological cooperation for developing an advanced

type of nuclear reactor. Under the agreement, the two countries will cooperate in developing an advance type of nuclear reactor, establishing research programs in materials and combustibles for future reactors and in developing medical and industrial uses for radio-isotopes. Another very good example of working to achieve unique solutions as the nuclear industry moves to a deregulated environment is the Research-Energy Power Research Institute memorandum of understanding which advocates sharing available data and sharing costs of generating new data, when required. I would hope this would go a long way towards ending disagreements over data which has traditionally been one area where contentions arise between the staff and industry when facing new challenges. This is especially useful as those facilities which the NRC has traditionally relied upon are scaling down or closing down as the need for research in new areas has dwindled as the industry has matured and also in the face of declining budgets. In the area of cooperation aimed at risk-informing regulations, I noted that last month the NRC's PRA Steering Committee and the NEI Risk-Informed Regulation Working Group held their second meeting to discuss the various initiatives which could be used to support the framework for risk-informing 10 CFR Part 50.

While working with the industry is becoming more of a reality in our current environment we must also remain vigilant to insure that the public's confidence in the NRC's independence is not eroded by blindly accepting results from others. Confirmatory research or anticipatory research for industry initiatives has been, is, and will always be necessary to insure we maintain our charge as an independent regulator. I think upon reflection of the lessons we have learned from Millstone and those we are still learning from Indian Point Unit 2, I am convinced that communicating what we do and how we do it in a way that is open to all stakeholders is very important to maintaining public confidence.

Research's overall budget has decreased. However, as I just stated the NRC has a management challenge to redefine the role of research in a mature industry I think we can't be too short sighted as we implement this challenge. If you look at the current challenges facing Ford and Firestone I think you will agree that the consequence of not aggressively investigating suspicious safety problems has resulted in a significant loss of credibility for both these companies. We cannot allow that to happen to the NRC. There are many past and recent examples which demonstrated the benefits of being a forward thinking organization and I will use remarks made by the Chairman which I whole heartedly agree with, to illustrate my point. " . . . Virtually every major new initiative that the agency has undertaken over the past few years, license renewal, risk-informed regulation, design certification of advanced reactor designs, assessment of digital instrumentation and control systems, steam generator tube integrity programs, and the new source term, have required technical guidance derived from our research programs. I do not believe that the NRC would have either the reputation that it enjoys as a world leader in nuclear regulation, or the credibility and the technical wherewithal to proceed with the implementation of a risk-informed regulatory structure, were it not for the contributions of the Office of Research."

We are hearing rumblings today that utilities are beginning to explore the possibility of building a new reactor in the United States. I can't see how the NRC can wait until we see an application at the door to begin exploring what new regulatory requirements might be necessary if an application was received. At some point, as soon as the picture focuses a little more on this issue, we might need to embark on what some might perceive to be anticipatory research. Performing the research now to better understand where the uncertainties lie with possible new

technologies will not only provide short term benefits but long term benefits if and when we see future power plant applications.

CONCLUSIONS

In closing, I would just like to add that my vision of the NRC Office of Research would be a center of excellence and source of expertise. This center would maintain a cadre of reactor and materials safety specialists in various key areas, with independent and unbiased expertise across a broad spectrum of advanced nuclear technology, to provide the technical basis for robust and transparent regulatory decisions. Experimental facilities and resources would be maintained to ensure our ability to respond in a timely manner to new or emerging issues. The office would complement the front-line regulatory activities of the agency and independently examine evolving technology and anticipated issues. While I am pleased to see that we are soliciting stakeholders more in what we do, I would expect we do more and focus on making what we produce more timely and more useful.

One final thought that I would like to leave with you regards the issue of funding. The current funding process of NRC research through users fees has the unintended impact of discouraging user support in the face of economic pressures. As a result, some are starting to pose the question as to where the NRC's research activities, if not the anticipatory activities, should be funded from the general fund rather than from those we regulate, since the public at large benefits from activities such as establishment of new regulatory requirements to support new reactor designs for example. I find this proposition very interesting and must study it more before I reach my final conclusion, but nevertheless I appreciate new ideas from our stakeholders as we continue to explore the future role of research and what mix of anticipatory and confirmatory research is optimum.

Thank you for your attention, I would be pleased to answer any questions you might have at this time.

Remarks of

Jeffrey S. Merrifield
Commissioner, U.S. Nuclear Regulatory Commission

at the

28th Water Reactor Safety Information Meeting
Bethesda, Maryland
October 25, 2000

Good Morning. Thank you very much for the opportunity to speak to you today. It is a pleasure to be here.

I would like to begin by reflecting on the speech I gave a year ago, and share with you my current views on the state of the NRC's research program. I also want to spend some time looking at the future and the role research will have in shaping our regulatory landscape. Frankly, my view of this landscape is remarkably different today than it was just one year ago. Let me begin by reflecting on what I said last year and by giving you my current impressions of the NRC's research program. For the sake of those who are not familiar with my comments last year, I'll briefly summarize them. I challenged our Office of Research in five critical areas:

First, I stated that the growing economic pressures facing the NRC and our licensees would result in even greater scrutiny of each and every research dollar we spend. Given the fact that these economic pressures are undoubtably here to stay, I challenged our research staff to adapt to a higher standard of fiscal accountability and more effectively demonstrate to their stakeholders that the NRC's research activities represent a valuable and prudent use of agency resources.

Second, I challenged our staff to reinvent the way in which they defend their research activities. Contrary to popular belief, good research does not speak for itself. I stated that if we have a defendable research program, our staff must learn to market it, sell it, and clearly make the case for why it should be funded. If research activities are not important to the NRC's mission or closely linked to the agency's strategic and performance goals, then the NRC should sunset these activities and move on to higher agency priorities.

Third, I told our staff that while it is important to have a research program that is visionary in its approach and capable of providing an independent view on important agency matters, that independence must be carefully managed so that it does not lead to isolation. I challenged the research staff to work closely with our program offices - the primary end users of the research - to ensure that these parties share similar priorities and a consistent, or at least a compatible, vision of the future.

Fourth, I challenged our research staff and our stakeholders to stop their fixation with the bottom line of the research budget. From my perspective, the fact that the NRC's reactor research budget declined from over $100M in the early 1990s to around $40M in FY 2000 is not relevant to the decisions we are tasked with today. Budget realities dictate that we approach our research budget, line item by line item. I challenged those who argue that our research budget is too big, or too small, to move beyond the bottom line and instead make the case for

either adding research initiatives that we **should** be doing but <u>aren't</u>, or for eliminating research initiatives that we **are** doing but <u>shouldn't</u>.

Fifth, I challenged our staff to seek ways to expand their efforts to capitalize on research work being conducted by the international nuclear community. As economic pressures drive greater fiscal restraint, we must leverage our international research efforts and not foolishly aspire to be the premier nuclear research agency in every discipline.

I believe the challenges I laid out last year were clear and meant to be constructive. However, some who attended the conference viewed my speech as an attack on research - somehow reflecting a lack of appreciation on my part for the contribution our research program makes to the effective fulfillment of our safety mission. With all due respect, I would argue that anyone who left last year's conference with that impression either did not listen carefully, felt threatened by the challenges, or did not recognize the realities we face. Let me make one thing perfectly clear - I believe our research program is absolutely essential to the long-term viability and success of our agency. However, if the program can't be managed properly, if its value can't be adequately conveyed to internal and external stakeholders, or if its links to the agency's strategic goals can't be clearly demonstrated, I assure you the agency will lose its ability to control the program's destiny. Others will decide that destiny for us. Like it or not, this **<u>is</u>** our reality.

With that said, let me now shift my focus to where I think our research program currently stands.

As I assess our research program today, I am pleased to say that it is healthier than it was just a year ago. Ashok and his management team deserve credit for what they have been able to accomplish in such a short period. While it is far too early to declare victory, the program has become more responsive to stakeholders, more fiscally disciplined, and frankly, more defendable. Given the importance of this matter, I believe it is essential that I articulate my thoughts more thoroughly.

First, let me focus on our external environment. The financial challenges facing our agency are greater today than they were last year, and I anticipate that these challenges will continue to intensify as our licensees - those that pay our fees - face greater competitive challenges associated with a deregulated electric market. This situation will only be compounded by the trend toward fewer reactor owners. It would be naive to think that distributing the fees associated with our research program among far fewer licensees will not bring with it an escalation of external scrutiny.

In regard to the research program itself, the Commission recently completed its review of the agency's research budget for FY 2002. As I promised at last year's conference, I vigorously challenged the merits of every line item in that budget. I am pleased to say that my expectations were exceeded. There were clear links between proposed research activities and the NRC's strategic and performance goals. There was a clear and defensible articulation of why each research project was necessary. There was less focus on the bottom line and greater focus on the merits of each project. In fact, without divulging too much about the agency's internal matters, the Commission, with my full support, approved a research budget virtually unchanged from that requested by our staff. Nobody in this room should underestimate the significance of that action.

As you know, I am a lawyer, not an engineer. Nonetheless, I understand the hazards associated with trying to identify a trend from a single data point, and I recognize that the recent budget cycle was but one data point. For me, another significant data point came during a recent visit I made to the Argonne National Laboratory, a lab that performs about $5.5M of research annually for the NRC. As you might expect, I was briefed on the status of the research initiatives they are conducting for the NRC. To my surprise, however, I was also briefed on how these initiatives are linked to the strategic and performance goals of the agency, and how the Argonne staff is exercising the fiscal discipline necessary to obtain the greatest return from every dollar the NRC spends. To me, this was especially gratifying because it demonstrated that the expectation of greater fiscal accountability that I and the other members of the Commission have been preaching has been embraced not only by our staff but also by our contractors.

A third data point came during a recent trip I made to Norway where I had the opportunity to visit the Halden Reactor Project. Over 100 nuclear organizations from around the world participate in research activities at Halden on such important matters as high burn-up fuel, MOX fuel, material properties, and human performance. While we spend less than one million dollars annually on research at Halden, our participation provides us with access to tens of millions of dollars of international research activities. My experience at Halden left me with little doubt that our staff <u>is</u> placing greater emphasis on leveraging our research dollars by looking for opportunities to capitalize on the research carried out by our international counterparts.
Data point #4 is not so encouraging because it represents a challenge that remains unanswered - a challenge requiring greater management attention. I voice this as constructive criticism in the hope that significant progress can be made this coming year. Despite efforts by our research staff, our attempts to reach out to stakeholders have resulted in limited success. Frankly, some of our internal and external stakeholders still do not have an appreciation of the value provided by our research initiatives. When the research management team attempts to articulate the value of the agency's research program, they are met with significant skepticism among our stakeholder communities - skepticism that is centered around the critical question, "Valuable to whom?" The accuracy of the perception is irrelevant. When you are dealing with stakeholders, perception **is** reality and thus it cannot be ignored.

Let me give you an example that illustrates my point.

In the May 8th edition of **<u>Inside NRC</u>**, Oliver Kingsley, Unicom's President of Nuclear Generation, provided his views of the NRC's research program. Mr. Kingsley stated that he does not support more money for the NRC's research program. More importantly, Mr. Kingsley added, "What would [the] NRC need research for? We've been operating plants for decades. Unless there's some type of advanced reactor program, I don't see a great deal of need [to fund NRC research]." Now, I have not talked to Mr. Kingsley about the article or the context in which his comments were made, but, assuming the article is accurate, the NRC cannot afford to underestimate the significance of his comments. As most of you know, Mr. Kingsley is responsible for the largest commercial nuclear program in the U.S.; a stakeholder that is well-respected throughout the industry for his emphasis on operational safety and technical excellence. The fact that such a well-informed and respected stakeholder does not see a need to fund NRC research should serve as a wake-up call to our agency. The fact that he made those comments in the same article that he discussed license renewal, the new reactor oversight process, and risk-informed regulation - all matters in which NRC research initiatives

were instrumental - only serves to highlight just how high a hurdle our research program must overcome.

The message I want to leave today is that the NRC's research team has been successful in meeting many of the challenges I put before them last year. Nevertheless, challenges remain. Maintaining fiscal discipline and accountability requires continuous vigilance. Cultural changes of this magnitude typically take years before sustainable benefits are recognized. Our research staff must redouble their efforts to ensure that our stakeholders understand the value the agency hopes to derive from each and every research initiative. Frankly, if we are not successful in clearly defining the value of our research program, our critics will undoubtably define it for us. I am not willing to accept such a scenario.

The Future Landscape

I'm now going to change course and share my views on the future research needs of the agency. From my perspective, the future landscape of the nuclear industry, and the research associated with it, look much different today than just a few years ago. There are challenges looming on the horizon that could serve to reshape the commercial nuclear industry in the United States - challenges that will tax the NRC's technical capabilities. While some of these challenges may never come to fruition, I believe it is essential that the Commission assess our staff's readiness for them, and take the steps necessary to develop our capabilities at a rate commensurate with the pace of change we face. I'll take a few minutes to discuss some of these challenges.

If you have been reading the trade press, I am sure you are aware that several utilities are exploring the option of building new nuclear plants in the United States. Joe Colvin, the President of the Nuclear Energy Institute, recently told a gathering in London that a new plant may be ordered in the United States within five years, but that conditions for doing so may be ready in as little as two years. I am not prepared to address the likelihood of such an initiative, and I certainly do not want to give the impression that I am promoting it - as I am not. As a Commissioner of the NRC, to do so would be irresponsible. However, it would be just as irresponsible for us not to take the initial steps necessary to ensure that the staff is prepared to carry out its responsibilities should new plant orders emerge. We must critically assess our staff's technical and licensing capabilities to ensure that we can effectively and efficiently carry out our responsibilities. Given that we have not overseen the construction of a new plant in many years, we must assess our inspection assets to determine where there are gaps in knowledge and expertise. We must also critically assess the quality and stability of the regulatory infrastructure supporting Part 52. These tasks simply cannot be accomplished overnight. Thus, the NRC cannot wait until a licensee knocks on our door with an application. I believe the Commission must act soon to reallocate the funds necessary to at least assess whether the agency is up to the challenges associated with new plant orders. Clearly, the Office of Research will play a critical role in this effort.

We must also be prepared to address advanced reactor designs. It is not inconceivable that one day it may be more appropriate to call this conference the Water and Pebble Bed Reactor Safety Meeting. Again, I am not prepared to address the likelihood of such an eventuality, nor am I promoting the ongoing Pebble Bed initiatives; however, it would be irresponsible for us to stick our head in the sand and ignore reality. The reality associated with this issue is that one of our licensees, PECo Energy (PECo), is actively involved in Pebble Bed reactor initiatives in

South Africa. According to recent comments attributed to Corbin McNeill, PECo's President and CEO, PECo could apply for a design certification in as few as 15 months. Such a development would be a real challenge for the NRC. The fact is, expertise associated with such a new reactor technology cannot be developed overnight. We must take steps now to develop this expertise so that we do not one day find ourselves incapable of carrying out our responsibilities associated with Part 52. I believe that our Offices of Research and NRR must, at a minimum, follow the activities in South Africa so that we can gradually build a prudent regulatory foundation and an appropriate level of expertise commensurate with the rate of progress made on the Pebble Bed initiative. One should not underestimate the safety and public confidence ramifications of falling short in our preparations. Clearly, our responsibilities in the area of new plant designs will not be limited to the Pebble Bed reactor. As you know, the NRC has already been approached by Westinghouse on an AP-1000 design. With escalating global warming concerns and the growing emphasis being placed around the world on energy independence, there is little doubt in my mind that domestic and international initiatives related to advanced reactor designs will intensify and that the NRC will be called upon to play a significant role in the safety reviews associated with these designs.

Another area that undoubtably will dot our landscape is the issue of extended power uprates. As many of you know, Alliant Energy is pursuing a 15% power uprate for their Duane Arnold facility. In addition, it appears that the Dresden and Quad Cities plants may submit similar licensing amendment requests in late 2000 and that the Brunswick plant may do the same in 2001. I am confident that the NRC is prepared to meet the technical challenges associated with 15% uprates. However, we should not kid ourselves that this represents the limit of future uprate requests. In a deregulated environment, our licensee's will look to squeeze as many megawatts as prudently possible out of their existing nuclear plants. How this incentive will manifest itself in the power uprate arena, I simply do not know. However, I do not believe it is unrealistic to expect that licensees could seek power uprates that extend beyond 15%. Should we face uprate requests of this magnitude, we have an obligation to all of our stakeholders to maintain safety and carry out our regulatory responsibilities in an effective, efficient, and realistic manner. In order to do that, we must ensure that our engineering analyses, our thermal-hydraulic code expertise, and our understanding of plant systems and safety margins, are sound. It is clear to me that our research program must be at the forefront of the NRC's efforts to address the realities we likely will face in the power uprate arena.

Steam generator research must also be a significant component of the NRC's research program in the future. It is essential that both we and our licensees develop better tube inspection methods, improve the accuracy of our data evaluation processes, and make further progress in our understanding of flaw growth predictions. Our goal must be to prevent, with greater certainty, tube failure events like the one that recently occurred at Indian Point 2. Now, some may argue that the Indian Point event was <u>not</u> of particularly high risk significance and thus preventing such events should <u>not</u> receive higher priority by the agency. I could not disagree more, and here's why. While we can argue risk numbers until we are blue in the face, I believe it would be irresponsible to assess the significance of such events so narrowly. This event certainly was significant to the public. It certainly was significant to the media. It certainly was significant to the New York Congressional delegation. It certainly was significant to our staff who faced the wrath of stakeholders and who ultimately will spend thousands of hours conducting event follow-up activities. It certainly was significant to ConEd, which is not only bearing the financial implications of an extended plant shutdown, but also the heavy burdens associated with facing a public that has lost confidence in their ability to operate the plant

safely. So, as the NRC and our licensees go about assessing risk in the traditional safety sense, we must not ignore the enormous business, social, and political risks associated with a steam generator tube failure. Events like the one at Indian Point 2 could damage our credibility as a regulator and serve to erode public, Congressional, and to some extent, regulatory confidence in each of the 103 reactors operating throughout the U.S. Therefore, I believe we owe it to our staff and our stakeholders to continue the valuable steam generator research we are sponsoring at Argonne and to provide the resources necessary to further enhance our knowledge and capabilities in this very important area. Our research program will also face challenges associated with the growing use of risk insights to support operational and maintenance decisions, licensing actions, and regulatory reforms. While we have started down the road toward risk-informing Part 50, I believe we are just now scratching the surface. At some point, licensees will undoubtably attempt to use risk-insights in applications that we cannot even imagine today, and the NRC will be called upon to effectively and efficiently carry out its regulatory responsibilities related to those applications. The NRC's research program must ensure that the agency's risk capabilities are sound and evolve in a manner commensurate with the applications they are being called upon to support. Our research program must proactively identify vulnerabilities and knowledge gaps, and ensure that our program offices recognize them, respect them, and compensate for them in their regulatory decisions. Let's face it, the use of risk insights is here to stay. The NRC can either manage them, or be managed by them. From my perspective, I believe our research program must be especially robust in this area so that our capabilities and expertise stay one step ahead of the applications we are being called upon to address. One should not underestimate the safety implications or the difficulty of this task.

Last but not least, I believe that the time has come for our research program to reassess whether the NRC's quality assurance (QA) requirements are continuing to produce outcomes that are consistent with the agency's performance goals. As most of you know, Appendix B to Part 50 lays out the quality assurance criteria for nuclear power plants. It is a regulation that has served an important role in our regulatory framework for many years. However, during my visits to 60 nuclear units over the last two years, it has been common to see maintenance activities involving the replacement of plant components and equally common to hear licensee concerns over the difficulty they face finding suppliers that maintain an Appendix B QA program. During a recent briefing I received from our staff, I learned that the number of suppliers with Appendix B QA programs has declined. I also learned that this type of problem is not new to the nuclear industry. In our discussions on related matters like the ASME Code and the N-stamp process, I learned that during the 1989 time-frame, a number of utilities experienced difficulties obtaining replacements for components that were originally constructed in accordance with Section III of the ASME Code. In that case, the NRC was compelled to issue Generic Letter 89-09 to provide appropriate regulatory relief. Here's my concern. Are the agency's quality assurance requirements inappropriately discouraging high-quality component suppliers from participating in the U.S. nuclear market, and if so, do we fully understand the consequences? Are these requirements unwittingly inhibiting potential safety enhancements? More broadly, are the agency's QA requirements consistent with our performance goals of maintaining safety, reducing unnecessary regulatory burden, increasing public confidence, and carrying out our responsibilities more effectively, efficiently, and realistically? I understand the commercial-grade dedication process and I am familiar with our ongoing efforts in the risk-informed arena. While these are important initiatives, I believe the time has come to take a more fundamental look at our quality assurance requirements to determine whether they are effectively and efficiently achieving their intended outcomes. I believe our staff should take a

fresh look at Appendix B and our regulatory framework surrounding quality assurance. The staff should also assess whether there are insights that can be drawn from more widely utilized national and international quality standards. For example, the ISO 9000 family of standards has become one of the most widely utilized quality standards in the world, already adopted by thousands of organizations, many of which have outstanding quality records. While I understand the staff has conducted some limited comparisons between Appendix B and ISO 9001, quite frankly, that's simply not enough. I want to know why ISO banners are rapidly going up as Appendix B banners are coming down. I want a better understanding of what is driving suppliers away from Appendix B quality assurance programs. We owe it to our stakeholders to critically assess Appendix B, compare it to more widely accepted quality standards like ISO 9001, identify where there are differences, and assess whether these differences are meaningful in our efforts to protect public health and safety. If particular Appendix B requirements cannot be linked to safety or to the NRC's performance goals, we should consider eliminating them. To the extent feasible and prudent, we must seize opportunities to bring Appendix B in line with widely accepted quality standards. Simply put, I believe the Commission must provide the resources necessary to ensure the agency's quality assurance requirements are not inappropriately driving high-quality component suppliers from the U.S. nuclear market, are aligned with our performance goals, and are in the best interests of the American people.

In closing, these are very dynamic times for the NRC and the U.S. nuclear industry, and the future promises to be even more dynamic. As I have outlined, there are many challenges on the horizon - challenges that bring with them opportunities. For us to seize these opportunities, the NRC must have the vision and leadership to not only recognize them, but to be prepared for them. Our research program must play an instrumental role in this process. It must be visionary in its approach and must provide the technical foundation necessary to support the bold decisions our agency will be called upon to make. I believe the next 10 years will prove to be some of the most challenging and rewarding our research program has ever faced. Winston Churchill once said, "A pessimist sees the difficulty in every opportunity; an optimist sees the opportunity in every difficulty." I am an optimist and I truly see tremendous opportunities embedded in the difficulties facing our research program. As a Commissioner, I believe I have an obligation to ensure that our research program and our staff are well-positioned to seize these opportunities. I assure you, I take that obligation very seriously.

The Future Role of Nuclear Power and the Need for Nuclear Regulatory Research

Robert J. Budnitz, President
Future Resources Associates, Inc.

My name is Bob Budnitz, and I was the Director of the Office of Research in the year 1979-80. I'm going to provide a perspective that is going to sound to some of you like we're washing dirty laundry, but it's intended. I'm going to talk almost exclusively about the problems that the Office of Research faces inside NRC. Not exclusively, there are a couple of other things that it faces inside the Nuclear Regulatory Commission itself in carrying out its designated mission.
The first item I want to talk about is the need for agency-wide support for research within NRC itself, and I'm going to try to describe the reason.

You see, I see it as almost inevitable that many staffers in the regulatory offices, NRR and NMSS, will not understand the need for, or the benefits of, the Office of Research or the work they do. The reason is that they're often faced with a technical need for a regulatory decision this year, next year, or sometime six months from now. So they call up someone in the Office of Research who is an expert in their field - it might be in instrumentation and controls or metallurgy or PRA or thermal hydraulics - and find out, to their surprise, that the programs that are going on have a four to seven to ten year horizon. And they're wondering why the agency is wasting that money when they can't answer the question that I have now. That's almost inevitable. It's just the way the world is if you're on the pressure line in NRR and NMSS. I don't see any way that that can be overcome. It's just the way it is. It's regrettable, but it's sure to exist as long as there is a technical need in those offices for things -- for technical answers to questions that come up. Now, I understand the need, and I hope everybody in this room understands the need, but the way to overcome that has got to be for strong, continuing, overt, open support for the mission, the need, and so on from the top.

When I was the Director of Research, we had five Commissioners and a couple of them, frankly, didn't support the Office of Research, and it was terrible. It was terrible. I just can't tell you how terrible it was, being the Director and talking to these people, because people down the ranks get the message. The Commissioners, the EDO, the office directors, the division directors, people at that level, there are some dozen or two, need to speak out often and forcefully, proactively, not in reaction to events, just proactively all the time to point out that there is a long range, as well as a short range need here, and that the Office of Research is designated by the Congress and agreed to by everybody. That's why it's there. They need to speak out in ways that are recognized within the agency. By the way, this is going on now and it's terrific. It needs to continue. You just can't wait five years and do it again. It's very, very important.

Now, regrettably, during the last 25 years, there have been periods when this hasn't been strong, and there has really been some suffering because of that. Now, when people speak out, the message gets -- I know, because I've been there and people here that are there know -- the message gets there. It really does get there. People listen. They understand the direction from the top, and people understand the situation, and it really helps them in the Office of Research, as well as helps their effectiveness.

Now, there has been a very serious problem in all of this separate from that support, and that's the long-term decline in the budget that everybody understands. When I was the Director, we had $205 million, and that was real money. It was 1980 and '81, fiscal '81, 205 million. Today, Thadani can buy research that's more or less 20 percent as much as I could buy. Just think. I'm not arguing for a budget that would be 300 million. I don't think that that's called for. But it really does make a difference to the culture, and the reason it makes a difference to the culture is that it's very hard to attract the very best people, although, thank God, a lot of them are still coming. It's really hard to in that situation. That's my second bullet. Because this long-term, what I would call in physics, a secular decline, is a problem.

Now, there's another problem I want to be sure I outline in here. Although in my 60 years of life, I've never read anything, I'm going to read you something, because I crafted the words carefully. This long decline helps to explain an aspect of NRC's culture -- I wrote these words very carefully and you'll read them in the report, if you get it tomorrow. It helps to explain an aspect of NRC's culture wherein technical staffers in the Office of Research often get a clear, if unspoken message that their career advancement within the agency will be expedited if they transfer to NRR and NMSS, or retarded if they remain in Research. That's a terrible thing to say, but that's an interjection. But let me go on. Not everybody in Research hears this message, thank God. And not all respond to it. But all too many of them do. Just go and ask.

In my view -- I'm reading -- in my view, enough I've heard and responded to make a difference over the years and over the two decades. This ugly little fact of life, I'm reading, has been there from the start, but a crucial counter-force occurs when senior NRC managers outside of Research create an environment in which Research's contributions are given due recognition and value, and which this comes often and loudly from the top and it needs to continue. It's a high priority for everybody, because it's going to be there, in my view, as long as there is an NRC that has an Office of Research. That's a really important point and it sounds ugly, and it's something that I at least need to say for me.

Number two, the technical strength of the staff, and Ray just outlined something that I'll just try to reiterate. When I was there, and this was not my doing, Herb Kouts did this in the early '70s. Every technical discipline necessary for reactor safety had experts on the staff who are recognized peers in the world. If you ask who are the 15 materials people in the world, four of them were on the staff. If you asked who are the 12 thermal hydraulics people, three of them were on the staff. Experts, peers in the research community. Now, there are still a few, thank God, but it's not the way it was. Why? Part of it is this long trend. Another thing, it's mature, it's not as exciting and so on. So even though there are some, this has diminished and there's nothing we can do about that. I mean, it just doesn't seem like it's feasible to do anything about it. But recognizing that, there is something that this same effect I was talking about can do something about it. That is, to the extent that the Commissioners and the people in the senior staff offices make this point clear, to that extent, really first class people could -- more of them, because there are still some -- more of them could say yeah, yeah, yeah, that's where I want to work, because it's on her majesty's service. But if you're going to be on her majesty's service, you have to have the feeling that there is value recognized all around and the more the better, and the less the worse. I'm not saying it isn't going on now. Thank God, the Commissioners have been supportive and so on, but it needs to continue. It just can't stop. There used to be a time when -- there used to be a time when -- of course, it was the AEC, so there was a lot of mixing between regulation and the others, which is why they were, in some cases, able to attract some of those people.

Now, let me go on to the next point, because I only have a few minutes. The need for flexibility. It is, of course, crucial that every NRC program respond to the regulatory needs. It's not research. It's research in support of regulation. That is absolutely crucial. But if you read the legislation, the Energy Reorganization Act, that said the Director of Research shall, that's Thadani, I was there once, shall recommend the research necessary to do X, Y and Z. Doesn't have to carry it out. That's important. But it's shall recommend. There have been times in which important programs were recommended and they didn't get carried out for a few years, because they didn't have the support because people were more focused on the short-term than the long-term. You need both and, crucially, you need flexibility. The Director needs flexibility. The office needs flexibility, so that some of the stuff that not everybody thinks they want gets done anyway, because the Director or the office thinks it's important. Sometimes it's developing a capability for analysis, sometimes it's an area that's emerging that others don't think is quite so important, and so on. And without -- it doesn't have to be a large percent. A few percent is enough. But it can't be zero percent.

There was a time right after I left I the early `80s when the Commissioners did something that was nuts that has since been rescinded. They passed some rule or whatever that every new research program had to have a sign-off of a user need letter from those guys over there, and a whole lot of stuff that the agency wanted -- that Research wanted to do, they said we don't need that, for 11 years. It didn't get done and there was a period, long since gone, when that policy was a debilitating problem for the Office of Research. Thank God, it's not there now. Now, the way, again, to correct that is the same way. Inside the agency, there needs to be this strong support and it has to be verbal, it has to be continual, and it has to be proactive, and it doesn't have to deal with details. It has to deal with philosophy.

Now, finally, as support for the regulated industry, we all know perfectly well that a good deal of the budget problem is because the utilities themselves say, gee, why should I pay for this research. I know that. Go and read what happens on the Hill. You know that. Everybody that doesn't know that isn't looking. The fact is that there are two things to do to fix that, and the first is people and -- the Commissioner just spoke about Kingsley, for example -- people need to go out and talk to the leaders and explain what it is that's going on and why it's important and what we did four years ago, what regulatory actions were taken four years ago, because Research 12 years got started that made that possible, and how much money it saved and involved or averted, because it's money that's speaking, and it's not wrong that it's money, consistent with health and safety. It's really important to do that. Nobody can do that but the senior people. But that's not enough. And the other stuff that's necessary is underway.

I was thrilled to hear Chairman Meserve say that the Congress is considering, although it's not in law yet, putting a very small percentage of the money into the agency that comes from general revenues -- that is, taxpayer dollars -- instead of from the fees. It doesn't take very much to take that heat off. If you have four percent, you can undertake a million dollar project and they won't complain that it's my money. It doesn't take very much. It's really an important thing.

So I guess my ten minutes are up, and I'm just going to leave you with a perspective that although I've been gone for -- it was 1980s, I've been gone 20 years, I have remained close to most of the old friends in the office, whom I see a lot, a third or half of them are still there, and I've talked to them. And what surprised me was that after I wrote this thing, which has been circulated, it's my bit of the report, it was circulated around informally to many people.

I was approached just yesterday afternoon by a senior officer in NRR, who read that and said, "Gee, you're out of touch. It ain't that way anymore." And all I could do is look him in the eye and say, "No, no, you're out of touch. It is that way." The reason I know is that when I wrote it and it was circulated around, the members of the Office of Research, many of them in this room, said, "Boy, you're right on the mark, Budnitz." And that made me feel as if I'm not out of touch, that, in fact, this isn't something that was true in 1980 only. It's an ongoing thing that's going to remain true because it's something in the culture of the way we deal with each other and the only way to overcome that is leadership from the top.

Thank you.

The Future Role of Nuclear Power and the Need for Nuclear Regulatory Research

Raymond Durante, President
Durante Associates, Inc.

Thank you very much. My name is Ray Durante, and I was privileged to work with the expert panel. I'm going to tell you a little bit about how this all came about. You've heard many times in this conference about the nuclear industry and the important and far-reaching changes that are going on and how these changes will create new issues and new challenges for the NRC.

As a result, there's an agency-wide internal evaluation to determine how they can meet these challenges; at the same time, continue their objectives and improve safety, regulatory efficiency and public confidence. An essential part of this effort is a thorough review of the activities of the Office of Nuclear Regulatory Research, and, by the way, I'll refer to that simply as Research from here on in.

Since it was established by Congress in 1975, Research has provided a significant part of NRC's independent capability for developing and analyzing technical information related to reactor safety, safeguards, environmental protection, in support of the licensing and regulatory issues. Now, as a means of supplementing their internal planning, input from stakeholders was sought on the role and future direction of Research in this rapidly and changing environment. A 16-member panel of experts, chaired by former Commissioner Ken Rogers, representing industry, academia, government and public interest, was assembled and asked to present their views and comments on the vision, the mission, and the role and general direction of regulatory research and provide their insight and guidance for future activities.

I put this slide up to give you an idea of the makeup of this committee. I'm sure you recognize all the names. I would point out, however, that Aloysius Hogan was on the committee as a representative of Senator Joe Knollenberg, Kristine Svinicki was on as a representative of Senator Larry Craig, and Andrew Wheeler (who is on the Senate Environmental Committee) represented Senator Inhofe's office. So we did have good representation from the Hill.

The panel convened for two meetings. The first two-day meeting was opened by NRC Chair Richard Meserve and followed by presentations and open discussions with the Research senior staff. The panel then met the next day for internal discussions and then each of the members prepared individual written statements which identifyed what they believed were the key issues and recommendations.

The second meeting, held about six weeks later, involved only the panel and focused on more detailed discussions of individual statements, which were then finalized by the authors and submitted as part of this report. It should be strongly emphasized, as Commissioner McGaffigan said, that this was a non-FACA committee and no attempt was made to reach a consensus.

The views of each of the panel members, including the Chairman, are their own with no editing or modification. The summary (written by a non-member of the panel) included in this report, points out those issues which were commonly held by a majority of the panel members. No order of priority or rank is intended, but only those issues considered important by a number of the panelists have been identified, and this is done for the convenience of the reader.

The material contained in the report represents phase one of a two phase program. The second phase will utilize the same panel members and follow the same format, but will focus on methods of dealing with the issues identified in phase one and how you implement the recommendations and conclusions. Phase two will be concluded in March 2001 and the results of both phases will be chronicled in a NUREG report and issued to the public.

Now, let me very briefly go over some of the important points that came out. The 17statements represented in this report are the views of the individual panel members, including the Chairman. No attempt has been made to reach a consensus or establish a uniform set of recommendations. It is clear, however, that there were many issues and conclusions that were independently considered by more than one panel member and, in some cases, the majority of the panel members.

I'm going to try and briefly summarize some of the more important issues. However, I'm sure you will hear more about them from the individual panel members as they come up to talk. Number one, it was unanimous among the panel members that a strong and viable Research must be retained in order to assure a sound technical base for all regulatory activities and to maintain the credibility and leadership role of the NRC, both domestically and internationally. To do this, Research must expand in-house expertise by adding experienced professionals, qualified in areas directly related to current and anticipated regulatory activities. While there was no specific criticism of current personnel, it was felt that through attrition and budget reduction, technical expertise has been steadily eroded and it was suggested a full-time cadre of scholarly technical experts be available to Research to keep abreast of worldwide technical developments that might impact on regulatory activities.

Number two, almost all the panel members recommended that Research increase its cooperative research efforts with DOE, industry, EPRI, and international organizations. It was felt that with declining budgets, greater returns could be realized by pooling research efforts with others. While it is neither possible or necessary for Research to actually participate in all the work, it was important to be in on the planning and initiation of research programs. It was recommended that the current working agreements with DOE and EPRI be reexamined and strengthened, wherever possible. This led to extensive discussions regarding the question of whether NRC can still maintain independence in their decision-making, while utilizing data and test results obtained by others. It was generally agreed this was not an insurmountable problem and a solution can be achieved. I'm sure you'll hear more about that from the panel.

An underlying concern with several members of the panel was whether Research was operating in accordance with the intent of the congressional mandate. The question was raised as to whether all research should be conducted in a single organization. There was considerable discussion regarding the proper balance between anticipatory and confirmatory research and technical support.

Several panel members suggested the definition of research, as it is conducted by NRC, needs to be more specifically defined and better methods are needed to decide what research needs to be done and when to start and stop the research programs. Almost all the panel members agreed the facilities available to Research are aging rapidly and becoming obsolete and expensive to operate, particularly those in the national laboratories. This prompted further discussion by several members of the panel on the need for more collaborative efforts, using the resources and facilities of industry and international sources. The matter of maintaining independence was considered and the panel felt it could be done and successful collaborative efforts in the past with foreign-owned facilities were cited as examples.

The majority of the panel agreed that Research must improve its communication efforts with the stakeholders, other government agencies, and internally with the Commission at all organization levels. Concern was expressed that in many instances, the public and even industry are not aware of research programs underway, what objectives were being pursued, and what the final results were.

The cross-cutting issue that impacted all other issues discussed was the matter of funding research efforts. It was generally agreed that funding was at a dangerously low level and any further cuts would make the viability of research questionable. The need for full cost recovery places too much of a burden on stakeholders and opinions ranged from funding research completely from general funds to at least providing a significant percentage from that source.

Several panel members felt stakeholders should not be required to fund any anticipatory research, even though they admitted that research has value and may be needed in future regulatory actions. It was suggested the NRC, at the highest levels, increase contact and dialogue with the Congress to obtain budget relief and reconsideration of the requirement for full cost recovery. Several panel members urged more active and direct leadership by the Commissioners in support of research.

Finally, it should be noted that at the first meeting of the panel, in his opening remarks, Chairman Meserve posed three questions to the panel -- Are we funding research at the right level? Are we doing the right kinds of research? And are we using the right R&D performance? Most of the members of the panel tried to respond directly to those questions; however, they felt they needed more information to properly respond. We plan to address that aspect of the questions more completely in phase two of the program.

Commissioner Rogers regrets not being here, but has wanted to emphasize three things. One, research must increase its core capability by maintaining a cadre of top-notch technical experts in each of the disciplines NRC will be making regulatory decisions. Two, research must have access to physical facilities capable of testing the validity of results of their computer programs. With present and contemplated budgets, NRC will never be able to support such facilities on their own, but they must find a way to gain access to them. Number three, research must maintain both confirmatory and anticipatory research programs. We must be ready for the future. We must get the funding situation worked out to allow more advanced work, whether its with industry, academia, or foreign entities.

Thank you very much.

The Future Role of Nuclear Power and the Need for Nuclear Regulatory Research

Michel Livolant, Director
Institute de Protection et de Surete Nucleaire (IPSN)

So I am the only non-U.S. member of this panel, so I consider this position as suitable to give a sort of international view on the topics, which look like U.S. questions, but, in fact, more or less, international questions, too, similar in other countries. Some words were given concerning the future role of nuclear power and I am glad to see that in the United States the view is really very optimistic.

Seen from Europe, I have to say that it is not so optimistic and in Europe now, there is only one project, which is a common French-German project of advanced reactor, which continues, but which is not decided. And with the political situation, the decision was not so easy to take. I am sure that what is happening in this country will have a great influence in Europe, but this will take some years. So for us, at the present time, the main issue is to have a safe operation of reactors, without accidents and wait till the situation changes for whatever reason, as Mr. Colvin really well explained. In this safe operation, the main responsibilities for operators, that is clear, and they have to do the activity and research which is necessary, but there is still a role for public safety to control safety and that the safety and health of the public are maintained. And as nuclear operations are complex, that requires a high level of competence, which generally recognized is better given by some research. And so it is generally agreed that this is the position of international panels, too. That regulatory agency have to do some research to do by themselves or to subcontract some research.

And that was one of the questions asked of the panel in the U.S. situation, what should be the regulatory research, in quantity and in quality. For that, I have made also one transparency only, which looks like the usual. As you see, some of the points which are here consider key points for regulatory research, were all presented by Bob Budnitz, but this is not abnormal, because as was said before, members of the panels agree on many points. So some points I find very important. There is a need in the regulatory research for some independence in the choice of subjects. In a given situation, the research program may be considered as more or less straightforward or going on normally, but in some cases, some issues are really not really clear, is this a real issue, not a real issue. And in that case, industry is not very willing to do research and so it's good to have a public agency who push some more or less prototypic research to very far, if it is a serious issue or not.

We know also that in some domains, there are potential improvements which can be made, and it's interesting that a public agency can make some prototypic research. We have done that once in the question of in-service inspection, for example, to be able to say to industry you can do better and we can prove it. If you arrive before industry people saying you can do better and you have no way to prove that, it's difficult to push them to do it. And in such cases, it's important that regulation agency can decide by itself to do the research work and this come, again, to the question of the who gave the finance to industry and the story of the fee or public money. But this is a pure U.S. question and I cannot interfere with it.

Another topic which is very important is the collaboration with other research financing organization. It's clear that to obtain access to objective experimental results, it's really acceptable and efficient to share research sponsoring between regulation organization, public research organization, and industry. But this is reality, sometimes some question of is it good that the industry and public agency make the same research. Well, I think this is good, but there are some conditions and the conditions are to keep independence for the public agency and for industry in the interpretation and use of results. Also, that the public agency has sufficient understanding and control by its staff to ensure that the research program take care of their needs. If you have a program share with other people, you have discussion necessarily. Research is objective, but the parameters used in the research may change more or less what you do. For example, depend if you want just to verify that the criteria fulfill or to find the margin in the rupture of a given component, for example. And in that sense, the idea to go to risk-informed regulation, in my opinion, should give rise to some results, because we often know how to make a structure which did not break, but where it works and the level of it works is difficult. If you want to estimate the risk, you have to know that.

Another point, and I think that Bob Budnitz spoke of that, the agency needs real competence to decide appropriate research and conduct it and with the type of work which is the agency, especially in the United States, don't make the results by themselves. They subcontract, and in that case, it's difficult when you are a young engineer to become a real expert. And if you have competence, if you are an expert, it's not so easy to keep it long. So for me, there are some things to do, but I think that Bob Budnitz has really said the best way on that and I don't insist on this point.

Another point which is of importance is the dialogue with people who do the licensing work. Clearly, in the United States, the dialogue with NRR and the RES. But this is not special to the United States, I consider, because we have two populations of people and one of my great concerns is how do they dialogue, how do they express their research needs, their needs. They don't express research needs. How do you express the needs and how to transform their needs in real programming research. That's a difficult situation. And after that, I want to associate them to this research, because necessarily everybody has the kind of frame of time. The time schedules are not the same and this is a very difficult question.

We tried to solve it by mentioning generic research in a discipline, like thermal hydraulics and mechanics, just to have a group of people able to answer very quickly to questions. But in other cases, you need to have some research programs and in that case, you have to try to better organize relation between these two. This is a problem for everybody. The same type of relation, you have to transfer research results in regulatory terms. So this is not straightforward, generally. And for that, you cannot rely on people external to the agency. You have the in-house capability to do that, to be able to understand research on regulatory needs, and this you cannot subcontract it. This is very important. But I don't think this is not made in your organization. I think you are very good in that direction.

My last point concerns the facilities and capabilities. It's not possible to do research if you have not the capabilities for that and the facilities which are needed, and at the present time, the situation is not very good. You need to keep a good scientist in the nuclear field to be able to treat whatever problem will happen and it's not so much a question of money. It's a question of stimulating research for them, stimulating where to do, and certainly the regulatory agency has to be promotive for that. But, again, in collaboration with industry, because the problem is national, it's not a problem of one organization or another. In some sense, the capabilities, that

is a more responsive question. You need to have a research reactor in operation. You need ten to 20 million dollars a year. And you have to continue to give -- even if you are not a given program at a given time, you have to maintain operation. So this is certainly not a pure RES or NRC problem. It's a national problem to keep the capabilities. I understand that you can rely, more or less, on international capabilities, but the burden is also for the foreign countries. I was very interested to hear Commissioner Merrifield said that for one dollar invested, you obtain ten dollars value. So I would expect that you could -- you will be beneficial with two dollars invested. That will be better for the other people on the other side.

So I think I have taken my time. But I think I have expressed what I wanted to do. So I insist on more freedom in the choice of research subjects, collaboration with industry, but with good fixed condition, and competence as a measure point. And so I think that this is really a responsibility of nuclear business to keep an active research community, and I say community, not to say one side or another. It's a community of research. And this is necessary to prevent for routine, even if you have no new reactor, your reactor, to operate, you have changes, you have fuel buildup and so on, and so you have to keep people who know all these questions and especially certainly you have to increase research if you want to have future new reactors.

Thank you.

The Future Role of Nuclear Power and the Need for Nuclear Regulatory Research

David Lochbaum, Nuclear Safety Engineer
Union of Concerned Scientists

Good morning: I would like to start with an informal survey by a show of hands. How many of you have hands? If you do, please raise one of them? Thank you.

Second, I'd like to ask how many of you work primarily in the research community, developing reports, conducting research and developing reports, can I have a show of hands? I'm not going to count, but quite a few.

Now, how many of you make regulatory decisions or are basically the end users of the research from the people who just raised their hands? A smattering. Not quite as many.

I think that leads to the contention or the issue that we had the greatest concern with on the expert panel, and that contention was, in our view, the NRC Office of Research is issuing reports that are not being adopted and used by the NRC staff in its regulatory decision-making. If that contention is valid, then whether the budget is increased ten-fold or reduced to zero makes no difference whatsoever, the regulatory decision process is unaffected. I think that may be true and I want to explain some of the reasons we believe that, since it's not probably going to be universally adopted.

One example was just two years ago, two years ago tomorrow, we were down at Brown's Ferry meeting with the NRC staff to talk about a 2.206 petition we had submitted. We had cited NUREG-CR-6451, that was prepared by the Brookhaven National Laboratory and issued by the NRC staff in August of 1997, a year before our meeting. Here is how the NRC staff reacted when we cited this NUREG report.

John Zwolinski of the NRC staff was the staff member saying it, and he said "You referenced the Brookhaven report, which the staff has not adopted. The staff does rely on NUREG-1353 generated in 1989. The numbers are significantly different as far as off-site exposures and we'll certainly continue to look at the Brookhaven work, but I will say this is not fully mature."

Now, the questions that that posed, in our minds, or in my mind, was how does the NRC staff adopt a report and, equally importantly, how does the public know when a NUREG report is an orphan or not, how do we know when a report has been adopted by the NRC staff. Is there a seal of approval, is there an adoption letter? What is the process that these things go through. And it also begs the issue of why does the NRC Office of Research issue reports that are immature. And lastly, why should the public have confidence in NRC research, when the NRC staff doesn't have confidence in that research?

The second example I'd like to use or cite is more recent. It was from May of this year. The NRC issued for peer review a draft study of reactor core isolation cooling system performance from 1987 to 1998. This is Figure 13 from that draft NUREG, that shows the results for 13 boiling water reactors in the United States -- 30 -- I'm sorry -- 30.

The solid circles are the actual performance from those plants over that 11-year period. The open circles are the results that were used by these plants in their individual plant examinations. The average performance is -- the bottom axis is unreliability. The average performance for these plants is, for the systems at these plants is 75 percent reliable, 25 percent unreliable. That's the actual reality. The IPEs use the reliability figure of over 92 percent reliable. Makes a big difference.

We're concerned that reality is not being a factor in the regulatory retreat, because this is not the only example where Research has done a report and it just doesn't seem to be factored into the regulatory decision-making process. The questions that this and other reports like this raise, in our minds, is how does the NRC staff use a NUREG report? What objective evidence is there that results like these are actually being used by the people making regulatory decisions? Related to that question is how does the public know when the NRC staff is using a NUREG report like this in its decision-making process? We look at safety evaluations issued by the NRR staff and we can't see references to NUREG reports and other products of the Office of Research.

So it seems to us as if these decisions are being made independently of the Office of Research. That also begs the question why does the NRC Office of Research issue reports that aren't being used by the NRC staff and, again, repeats the question, why should the public have confidence in the Office of Research when the NRC staff does not have confidence.

So I think the question that we were asked on the expert panel by the Chairman was, is the level of funding at the right level, and I can't determine that until I see what the need is and why that research has to be done at all. I think on a gut level and on an intellectual basis, academic basis, you can understand why the Office of Research needs to do its work. We need to see how that work is being used by the NRC staff. We just don't see it. We didn't see it during the presentations to the panel. I didn't see the link. There wasn't anybody from NRR that came in and addressed the panel on why we need this work done or why we've used this work in the past.

We heard from the Office of Research about why they need to do the work, but that's kind of like us asking our funders, asking the UCS funders why they need to give us more money. There seems to be a somewhat conflict of interest in that message.

Thank you.

The Future Role of Nuclear Power and the Need for Nuclear Regulatory Research

Theodore U. Marston, Vice President and Chief Nuclear Officer
Electric Power Research Institute

Good morning. I'm Ted Marston, from EPRI, and, as you might imagine, research is near and dear to my heart. What I would like to do this morning in the brief time I have is cover four items, very briefly, some general remarks on the value of research in the regulatory process, some specific remarks about the relationship between NRC Research and EPRI over the years, and then try to very briefly answer Chairman Meserve's three questions, and then conclude with some conclusions and recommendations.

First of all, research is a necessary and critical part of the licensing process. I think that's a given. And the credibility of the overall regulatory process is, to some extent, dependent upon the reputation and credibility of the supporting research organization. And that will depend on issue by issue. Oftentimes, the toughest questions are answered by research. You go back historically and look at that, and I'll only review the recent past, we've got risk-informed regulations, license renewal, digital I&C, inspection technology, new reactor design, et cetera.

Now, let me shift to the EPRI/NRC Research relationship. As I say, it's been a long and very constructive relationship, in my opinion, and I guess you'd have to ask Ashok about his. But during the 1985 to '95 time frame, because of legal considerations, this issue of independence came up and then all of a sudden the ability to cooperate really became impeded. And there are a number of reasons it turned that around. The Commission went through a strategic assessment and baselining process. There were reduced resources, which really forced us together. We had strong input from our utility direction to cooperate more with research and the risk-based, risk-informed regulation is driving us that way, with the drive towards more realistic information. So we executed a memorandum of understanding several years ago to really force us into this area, and I think that's worked out very well.

I would like to make a comment on independence. I think independence is a major concern, but I think the generation, co-generation of data and information is important, very cost - effective. However, the independent interpretation of that information is imperative on both the industry, as well as the regulatory agencies.

Now, why am I not concerned about us generating the same data? Well, simply because we can't afford to do redundant tests. Second of all, if you look at research, you look at the owner's group, and you look at ourselves, we're basically interested in the same safety issues. And most importantly is that the nuclear power plant owner-operators are equally concerned about the public health and safety of the public, and it's really because of their large investment in their business, as Joe mentioned. And you take that large investment and you put it at risk if you challenge safety. So there's a fundamental alignment in that area. The bottom line is the cooperation and collaboration has significantly improved, but there are areas that we should improve it further.

First question from the Chairman was, is the funding at the right level. I think the first and foremost thing, and it's been said several times, but I will say it again, because it's really a must do, is the NRC Research must attract and retain highly qualified and respected technical staff members, especially in critical core areas. That is not an optional – that's a mandatory step. And it's going to require, in decreasing budgets, I can tell you from our own experience, in an era of decreasing budgets, it's going to require extraordinary insight and planning and management to keep the R&D priorities focused on the changing efforts. This is a major task. And the real question, only the NRC can determine if this is the right amount of information. We heard David had mentioned about NRR and RES misalignment. We don't see as much of that. There are certain examples, and I think David cited a few of those, but that dialogue is improving. It's very important that NRR establish that.

Now, I am concerned, personally, because of what Joe Colvin mentioned, that I think we may be, and I stress the term may be, on the verge of a build program in this country and we really need to have the licensing process focused on that as well. And as I understand it, there's essentially no budget allocated to that right now. The right kind of research; again, it would be very presumptive of me to say what kind of research should NRC Research do. That's really an NRC question. But I can tell you, from my own experience, it's very, very tough to stop research, to change direction. It's difficult to align priorities with needs. We need to have stability in the planning and budgeting process and yet we got instability in the issues. So it's a difficult job. So my sympathies go to Ashok and his people and I think they've done a very good job. Again, coordination with the input between NRR and RES is absolutely essential. We had that same issue with our members and without that dialogue, it fails. However, I do believe that there are some areas that we should increase the amount of research. This would be digital I&C, advanced sensors, monitoring diagnostics, digital information management, chemistry possibly, and, also, more openness to risk-informed regulations without unnecessary conservatism. I think those are some areas.

Now, the next question was, do we have the correct R&D performance, and I don't think that's really the right question. I don't think we should ask who should be doing the work. We should say what attributes does the performance have. And this has been mentioned several times, but I'd like to give you my spin on it. Clearly, we need a requisite level of expertise to do the high quality job. The contractors have to be able to have that expertise. They have to have the correct tools and facilities to do that. And now in today's times of greater cost competition, they must be cost-effective and timely. And most importantly, they must be objective. There really can't be any personal or hidden agendas in the work they do and they must stand behind their work. So whether it's a university, it's a private contractor, it's a national lab or an international lab is really not the question. The question is do they have these attributes and can they perform the research. That does not, however, remove the responsibility of interpreting those results and managing the research away from NRC Research. So, again, they must have the quality people on the staff. They must have the focus. They must have the will to drive the research to its objectives.

So I think that's my perspective on those questions. I do have some recommendations. Fees have come up a number of times. They came up in our discussions, and I think this is a central issue. I think the NRC and Congress really have to revisit the full fee recovery basis. Apparently, there is some movement in that direction. We had a modest proposal, that possibly a way we could do that is to take the total cost of NRC Research, that would be not only the

contracted, but the direct and indirect costs associated with that, and split it between user fees and appropriations. That's just a proposal on our part.

NMSS research: should it, in fact, be folded back into research? Our feeling was that at this critical juncture, we should not tamper with it. Possibly in the future that would be a good move, but right now, at this point in time, we ought to leave things alive. Because any time you make changes, that possibly could compromise the situation.

Advanced reactor: I was very pleased to hear a balanced approach. We hear a lot of discussion about Gen IV, but if we have an early order, it's not going to be Gen IV. It's going to be advanced light water reactor or modification of an advanced light water reactor.

And, finally, in terms of cooperation, I think we, including everyone in this room, we really need to re-double our effort to cooperate, because we don't have enough resources, either domestically or internationally, either in government or in industry, to do this all alone. So I think we're going to see our cooperation paradigm change dramatically in the near future, particularly if we start thinking about going into a build program.

Thank you.

The Need for Nuclear Regulatory Research

Kenneth L. Mossman
Professor Health Physics and Director, Office of Radiation Safety
Arizona State University

Good morning. My name is Ken Mossman. I am Professor of Health Physics and the Director of Radiation Safety at Arizona State University. My professional interests are in radiation protection and, in this regard, I bring a different perspective to the Panel. I appreciate the opportunity to come to this meeting, and I want to take this opportunity to thank Ashok Thadani and Margaret Federline for extending an invitation to me to serve on this panel.

I thought the panel was superb. The membership represented a very wide spectrum of perspectives on research. I must admit that I probably got more out of working with the panel members and listening to the discussions than the NRC got out of my contributions.

Let me address a number of issues. I will begin by reviewing what I consider to be the overarching principles that ought to govern how we look at research. These overarching principles are not only applicable to the U.S. NRC and RES, but any other Agency or any other institution that supports or carries out research. I will then summarize my responses to Chairman Meserve's questions posed to the Panel.

OVERARCHING PRINCIPLES

The first principle is how do we go about supporting research in a declining budget environment, and that's already been mentioned by several of the panelists here. In a declining budget environment, some really important projects either don't get funded at all or get funded at a level that is inadequate, and we need to look very hard at that situation. Strategies should be developed to maximize extramural collaborations. Support for research may be leveraged by coordinating research activities among agencies with common interests. Although RES deals with research problems unique to U.S. NRC (e.g., reactor safety), there are many other regulatory research problems (e.g., worker health and safety) that cross agency boundaries for which a coordinated research effort may be useful.

Second, we should consider all research programs. Interestingly, the panel was given an excellent presentation by RES staff, but that's all we heard about in terms of research programs. We did not hear anything about what is going on in the Office of Nuclear Material Safety and Safeguards (NMSS) and, clearly, there are research programs there that need to be considered.

Third, we need to maintain public confidence in Agency activities. Research initiatives should be subject to stakeholder input. The planning of research programs and the implementation of research findings in licensing and regulatory activities should include affected and interested stakeholders. To enhance quality, research findings should be subject to peer review. Support of independent research enhances public confidence in regulation decision-making by minimizing perceived or real conflicts of interest.

Fourth, we need to coordinate regulatory programs. By my count, there are over a dozen Federal agencies in the United States that have regulatory responsibility for radiological health and safety, including the U.S. NRC. Because of differences in philosophical approaches to standard-setting and statutory authorities, some regulations among Federal agencies are conflicting or overlapping. A 1994 GAO report ("Nuclear Health and Safety: Consensus on Acceptable Radiation Risk to the Public is Lacking" GAO/RCED-94-190, September 19, 1994) addressed this very serious problem. Federal agencies should work closely together to minimize regulatory conflicts. For instance, the EPA and the U.S. NRC continue to have difficulties on how restrictive U.S. protective standards should be.

Let me now address the three questions that Chairman Meserve put to the panel.

IS RESEARCH BEING FUNDED AT THE RIGHT LEVEL?

Whether the Agency is conducting research at the right level is obviously a difficult question to answer and requires a thorough understanding of short-term and long-term research problems. The Agency must be able to evaluate critically what it can and what it cannot do. The Agency should carefully look at opportunities for collaboration with other Federal agencies and with industry, where joint efforts may be considered mutually beneficial. As discussed by other panelists, research budgets for some industries have been estimated to be about ten percent of the total budget. If we use that as a guideline, the U.S. NRC research budget should be about $50 million for FY-2001, based on a total budget of $488 million. U.S. NRC sponsored research activities should be funded by congressional appropriations rather than from licensing fees. Placing the burden for research solely on licensees is inappropriate, since they are but one of a number of constituencies that benefit from the research.

How do we define research that is done by the U.S. NRC? If we go to Congress and we talk to them about what it is that we mean by research, are they able to readily distinguish U.S. NRC research from research that is done by the National Science Foundation or by the National Institutes of Health? The Agency needs to explain to the general public and to Congress why research is supported by the Agency and how research findings benefit licensing and regulatory activities. Anticipatory and confirmatory research need to be clearly distinguished. How should research be prioritized in a declining economic environment? Allocation of funds should be based on a rational system of prioritization of research projects. In a climate of declining research, not every project can be funded at the desired level and, unfortunately, many excellent projects may go unfunded. The Agency currently uses a prioritization system that emphasizes safety significance, scope of license impact, realistic decision-making, industrial participation leverage, and economic impacts. Are these appropriate priority determinants or do other priority determinants need to be considered?

Agency research staff expertise is also a serious question that has been more than adequately addressed by other panelists. My views are essentially congruent with what has already been said.

ARE THE RIGHT QUESTIONS BEING ASKED?

The Commission's commitment to research is critical. The importance of research in support of regulatory and licensing activities must come from the top, and this was a point on which many

panelists concurred. The Commissioners must clearly articulate research goals and the significance of the research as part of the Agency's mandate. The goals, however, should be broad enough so as not to constrain needed flexibility within research programs. Anticipatory research, in particular, requires flexibility. With respect to anticipatory research, asking the right scientific questions and designing experiments to answer such questions are characteristics of quality research. For confirmatory research, problems are likely to be self-evident and it is clear what direction the research should take. However, in the case of anticipatory research, the investigative direction may not be clear. Interagency collaborations and a U.S. NRC office of interagency research should be established to deal with broad research questions of health and safety and coordinate research programs with other Federal agencies with radiological health and safety mandates.

The Agency should consider broadening its research scope to include research in the communication sciences, statistical modeling as it relates to risk assessment, and issues pertaining to bridging policy and science questions. Examples of research questions in the science policy arena include: what radiation dose is safe, and should the Agency attempt to target a particular dose level (e.g. 1 mSv/y) as safe? How should a "safe" dose be distinguished from an exposure limit? What are the advantages and disadvantages of returning to a dose-based system of radiation protection? Is the linear no threshold theory (LNT) an appropriate basis for setting standards? What is the cost of retaining LNT-based risk assessment as a basis for standard-setting?

ARE THE RIGHT PERFORMERS BEING USED?

The Agency should review its practices in selecting and monitoring research done by contracting organizations. Are the best research organizations doing the job? Is the Agency getting the most from its research dollars? Collaborative opportunities should be pursued wherever possible. For example, research reactors are located at universities. These facilities conduct important research projects that cannot be performed elsewhere. Without funding from the U.S. NRC, it is likely that universities would have to shut down these facilities. Support of university-based research is important in education and training of future nuclear engineers and scientists. As an academician I cannot emphasize enough the importance of maintaining academic nuclear engineering and health physics programs. We need more trained engineers and scientists in the nuclear technologies. Unless current trends are reversed, the U.S. is in real danger of having an inadequate manpower supply.

The Agency should establish science/engineering advisory committees to advise the Agency on matters of research. I realize that the Agency already has advisory committees but I am proposing something different. Each advisory committee would deal with a single broad issue, such as reactor safety or nuclear waste. Members (e.g., nationally recognized experts from universities, other government agencies, and industry) would be appointed by the Commissioners, with input from appropriate U.S. NRC offices. The committees would report directly to the Commissioners and have two main functions: (1) provide independent advice to the Commission on research matters, and (2) coordinate peer review of U.S. NRC funded research proposals.

This concludes my remarks. Thank you for your attention.

Risk-Informing Technical Requirements

Mary Drouin, Alan Kuritzky, Nathan Siu, Mark Cunningham, Thomas King
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission

Abstract

The NRC staff is now working to modify its basic nuclear reactor safety regulations, contained in 10CFR50, to make these regulations impose regulatory burdens on licensees that are commensurate with their safety importance. One part of this work involves making changes to specific requirements in Part 50. Current staff assessments, particularly with respect to potential changes to hydrogen control requirements, are discussed. As the staff undertakes this work, it is understood that there remain technical impediments which, until overcome, constrain the potential uses of risk information in regulatory activities. Two of these impediments - the lack of formal PRA standards and gaps in PRA technology - are discussed.

Introduction

The United States Nuclear Regulatory Commission (NRC) has made use of probabilistic risk analysis (PRA) information for many years. A key milestone in this use was the issuance of the Commission's 1995 PRA Policy Statement (Reference 1), which indicated that: "the use of PRA technology should be increased in all regulatory matters ...in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." More recently, the Commission's Strategic Plan (Reference 2) has been published; this plan includes a number of strategies to accomplish the intent of the policy statement.

Using this guidance, the NRC staff is now working to modify its basic nuclear reactor safety regulations, contained in 10CFR50 (Reference 3), to make these regulations impose regulatory burdens on licensees that are commensurate with their safety importance. One part of this work involves making changes to specific requirements in the body of regulations. As discussed below, the staff is now studying the Part 50 technical requirements to identify areas of unnecessary conservatism and needed additional safety requirements.

The staff's work to risk-inform the Part 50 technical requirements is, of course, dependent on the quality of PRA information being used by NRC and its licensees. While the PRA library of information being used today is reasonably mature, there remain technical impediments that, until overcome, constrain the potential uses of this information. Two of these impediments - the lack of PRA standards and the gaps in PRA technology - have been the subject of considerable work in the past several years, and are discussed further below.

Risk-Informed 10CFR50 Technical Requirements

In one part of its program to risk-inform 10CFR50, the staff is studying the Part 50 technical requirements to identify areas of unnecessary conservatism and potential additional safety requirements. The staff has developed, and is now using, a general framework for identifying and prioritizing potential changes. This framework is described in Reference 4.

An early result of this work was the identification and review of potentially valuable changes to requirements contained in 10CFR50.44 (Reference 5). Based upon current risk information and research results, the staff believes that little to no risk significance or benefit is associated with some of the combustible gas control requirements of this regulation, potentially resulting in unnecessary burden. In addition, the staff also believes that the current requirements do not address some risk-significant concerns from accident scenarios.

The staff considers the work described in Reference 4 sufficient to establish the feasibility for risk-informed changes to the technical requirements of 10 CFR 50.44. In Reference 4, it has recommended the following characteristics for a risk-informed alternative to 10 CFR 50.44[1]:

1. Specify in the regulation a specific combustible gas source term using best available calculational methods for a severe accident that includes in-vessel (and ex-vessel) hydrogen and carbon monoxide generation in such a way that the alternative regulation addresses the likely sources of combustible gases. These sources would only address challenges to the containment that could potentially result in a large radionuclide release within 24 hours after the onset of core damage.

2. Eliminate the requirement to measure hydrogen concentration in containment. Hydrogen monitoring is not needed to initiate or activate the combustible gas control systems for each type of containment, hence hydrogen monitors have a limited significance in mitigating the threat to containment in the early stages of a core-melt accident. Hydrogen monitoring for emergency response purposes is addressed separately from 10 CFR 50.44.

3. Retain the requirement to insure a mixed atmosphere. The intent of this requirement is to maintain those plant design features (e.g., open compartments) that promote atmospheric mixing and is considered an important defense-in-depth element (i.e., meeting the intent of Part 50's General Design Criterion 50 (Reference 6)).

4. Eliminate the requirement to control combustible gas concentration resulting from a postulated loss of coolant accident. This type of accident is not risk significant and the means to control combustible gas concentration (e.g., recombiners) does not provide any benefit for the risk-significant accidents or, if a vent-purge method is used, can result in unnecessary releases of radioactive material to the atmosphere. Long-term combustible gas control is addressed in Item 9 below.

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[1]The Commission is presently reviewing the staff's recommendations.

5. Retain the requirement to inert Mark I and Mark II containments. Removal of this requirement would result in the integrity of these containments being highly vulnerable to gas combustion.

6. Retain the requirement for high point vents in the reactor coolant system (RCS). Combustible gases in the RCS can inhibit flow of coolant to the core, therefore, the capability to vent the RCS provides a safety benefit in its ability to terminate core damage.

7. Modify the requirement for the hydrogen control system for Mark III and ice condenser containments to control combustible gas during risk-significant core-melt accidents (e.g., station blackout accidents). Since the control system uses igniters that are alternating current (ac) dependent, under station blackout conditions, these containments may remain vulnerable to gas combustion. Alternately, if station blackout could be shown by the licensee to be of low enough frequency, with due consideration of uncertainties and defense-in-depth, then the sequence would not be risk significant and the licensee would have complied with the requirement via the current igniter system. Such an approach represents a performance-based aspect of this recommendation.

8. Include a performance-based second alternative within this regulation that would allow a licensee to use risk information and plant-specific analysis on the generation and control of combustible gases to demonstrate that the plant would meet specified performance criteria (e.g., maintain containment integrity for at least 24 hours for all risk-significant events). This may be especially attractive to future plants.

9. Recommend that long-term (more than 24 hours) control of combustible gas be included as part of a licensee's severe accident management guidelines. This is recommended since combustible gases still pose a challenge to containment integrity in the long term with the possibility of a large, late radionuclide release.

Since these recommendations are based upon a feasibility study, additional work would be required to support an actual rule change. In addition to the calculation of the combustible gas source term discussed earlier, such work would include:

- detailed regulatory analysis on safety enhancements that have the potential to pass the provisions of NRC's Backfit Rule (Reference 7),

- assessing the relation to and need for conforming changes in emergency operating procedures and severe accident management guidelines,

- assessing the implications of fire and seismic events on the combustible gas control system requirements in Mark III and ice condenser plants, and

- developing regulatory guides to implement the performance-based aspects of the recommended risk-informed alternative rule.

The staff is now evaluating the potential value of changing requirements contained other Part 50 regulations. Specifically, the staff is assessing possible changes to 10CFR50.46 (Reference 8). The staff is also considering ways for modifying and consolidating the 10CFR50 requirements for

"special treatment" of important systems, structures, and components. This work represents a longer-term assessment of these requirements, building on the proposed changes to the scope of special treatment requirements now also under consideration (Reference 9). That is, if implemented, this work could potentially result in more fundamental changes to the requirements beyond those now being considered to change the scope of plant structures, systems, and components subject to these requirements. The staff's work to assess the feasibility of making such changes, and develop recommendations for transmittal to the Commission, is expected to be completed in 2001.

Role of PRA Standards

In any regulatory decision, the goal is to make a sound safety decision based on technically defensible information. Therefore, for a regulatory decision relying upon risk perspectives as one source of information, there needs to be confidence in the PRA results from which the perspectives are derived. Consequently, the PRA needs to have the proper scope and technical attributes to give an appropriate level of confidence in the results used in the regulatory decision-making.

Consensus PRA standards can be used to define the needed scope and technical attributes, and an industry peer review program can provide an assessment of the weaknesses of a PRA. Such standards have been under development for several years by the American Society of Mechanical Engineers (ASME)(Reference 10), the American Nuclear Society (ANS), and the National Fire Protection Association (NFPA)(Reference 11). Industry peer review programs (Reference 12) have also been under development during this time. The staff is now reviewing, or expects to soon review, industry peer review programs, and the ASME and ANS PRA standards, as well as the PRA portion of the NFPA fire protection standard, in this light. To support this review, the staff is developing acceptance criteria for the technical requirements and peer review process (Reference 13).

Reference 13 describes the minimal functional attributes necessary to ensure that a PRA is capable of providing the information needed in a regulatory decision. Although these are the minimal requirements at a functional level defining a PRA, they do not by themselves ensure confidence in the PRA results. This confidence can be gained by defining supporting technical requirements. For example, in the technical element of systems analysis, one functional attribute is that "the model is developed in sufficient detail to capture the impact of dependencies." To ensure that the intent of this attribute is met, it is necessary to understand the dependencies that could impact the availability and operability of the system and components under consideration. However, what the dependencies are and how they support a specific system or component are not always evident. Dependencies such as the need for DC power for the Reactor Core Isolation Cooling (RCIC) system (in a BWR) are evident. However, for continued operation of RCIC, there is also a need for suppression pool cooling. The steam from the RCIC turbine exhausts to the suppression pool, and loss of cooling to the pool can cause the RCIC turbine to trip on high exhaust pressure. This type of dependency is not as evident. Consequently, to ensure that a PRA being used to support regulatory decisions has properly accounted for the impact of dependencies, supporting technical requirements interpreting this functional requirement (and the others) are needed. In this example, the supporting requirements may specify the types of dependencies (e.g., motive and control power, design and operational conditions) that need to be considered in looking at the availability and operability of a particular type of component (e.g., turbine-driven pump).

As noted above, the staff is developing acceptance criteria for the technical requirements and peer review process. These reviews are summarized below.

- Review of the ASME, ANS and NFPA standards The staff intends to use Reference 13 to provide the general basis for its review of the ASME and ANS standards and the PRA portion of the NFPA fire protection standard. The staff will review the ASME and ANS final versions as well as the final version of the appendix on fire PRA included in the NFPA fire protection standard. If appropriate, the staff will endorse them in an update of Regulatory Guide 1.174 (Reference 14) or elsewhere to support other risk-informed activities. The staff endorsement may take exception to or include additional specific criteria to address any identified weaknesses in the standards to ensure that PRAs used in regulatory decisionmaking will have an adequate technical basis.

- Review of the Peer Review Program The staff review will encompass both the process and technical aspects of the peer review program discussed in Reference 12. The staff process review will involve comparing the Reference 12 review methodology against the criteria described in Reference 13. These criteria provide a set of "peer review attributes" on what constitutes an acceptable peer review process. The staff technical review will involve comparing the Reference 12 technical elements against the set of "functional technical attributes" in Reference 13 and sub-tier criteria against the acceptance criteria being developed to supplement the functional attributes.

Until the review of the standards and peer review program is completed and the staff endorsement final, the staff will continue to use the guidance of Regulatory Guide 1.174 and SRP Chapter 19 (Reference 15) to ensure that PRA information used in regulatory decisions is of the appropriate scope and quality for the decision being made. To strengthen this guidance and thus improve the efficiency and consistency of the staff review process, the staff intends to include the information in Attachments 1 and 2 of Reference 13 in the next update of the guide and SRP chapter. The staff is now developing an updated version of these documents with the intent of publishing them in early 2001 for public review and comment.

Filling Gaps in PRA Technology

As noted above, one important impediment to the greater use of risk information in regulatory decision making is that gaps in PRA methods still remain. One function of NRC's research program in PRA is to identify such gaps and perform research to fill them. More specifically, the staff now has work underway to improve:

- Human reliability analysis (HRA) methods It has been accepted for some time that failures in human performance are one of the principal sources of risk. Although techniques have been used in the past to quantify both pre-accident and post-accident human error, one of the remaining questions is how to treat "errors of commission." This question has been the subject of recent NRC-sponsored work (Reference 16) as well as an international study (Reference 17). Based on this and other work, the NRC staff is now developing its plans for future HRA research.

- Fire risk analysis methods Fire-initiated accident scenarios continue to be a significant contributor to the calculated risk of most plants. Recognizing this, the NRC has initiated a

program to improve fire risk analysis methods and data. The overall program is described in Reference 18. Recent products of this program include an evaluation of the effect of exemptions to NRC fire protection requirements on fire risk (Reference 19), and a study of electrical cable failure modes, their effect on associated circuits, and a rough probabilistic quantification (Reference 20).

- Treatment of aging effects in PRAs The staff has recently completed an initial study to incorporate physical models of aging effects into a PRA (Reference 21). This initial work focused on one aging mechanism of piping - flow accelerated corrosion - but the general approach may be applicable to other aging mechanisms. Follow up research on this topic is now being planned.

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Use Of PRA Results in Regulatory Decision Making

G. Holahan
U.S. Nuclear Regulatory Commission

presented at the

28th Water Reactor Safety Information Meeting

October, 2000
Bethesda, Maryland

DECISION MAKING AND PSA QUALITY (SECY-00- 162)

- The Goal is to Make Good Safety Decisions. Therefore, the Quality Required of a PSA for a Specific Application is a Function of the Role the Results Play in That Decision.

- Some Applications Depend on Results from a Limited Number of PSA Elements, Others Require a Broader Scope.

- The Decision Making Process Has to Determine What Is Required to Generate the Required Insights.

- The Degree of Confidence in the PSA Insights Used to Support a Decision is a Function Both of the Quality of the Underlying Analysis and of the Treatment of Uncertainty.

DECISION MAKING AND PSA QUALITY (Continued)

- ASME and ANS Are Developing Standards for PSA Which Should Define the Requirements for an Analysis to Be Considered a PSA.
- The Standards Will Not Be Prescriptive: They Will Define What Issues Should Be Addressed, but Not How They Are to Be Addressed.
- Because of this Flexibility, Some Level of Review Will Always Be Required. A Peer Review of the PSA Can Help to Focus Regulatory Review by Identifying Key Assumptions and Approximations and Assessing Their Potential Impact on the PSA Results and Insights.

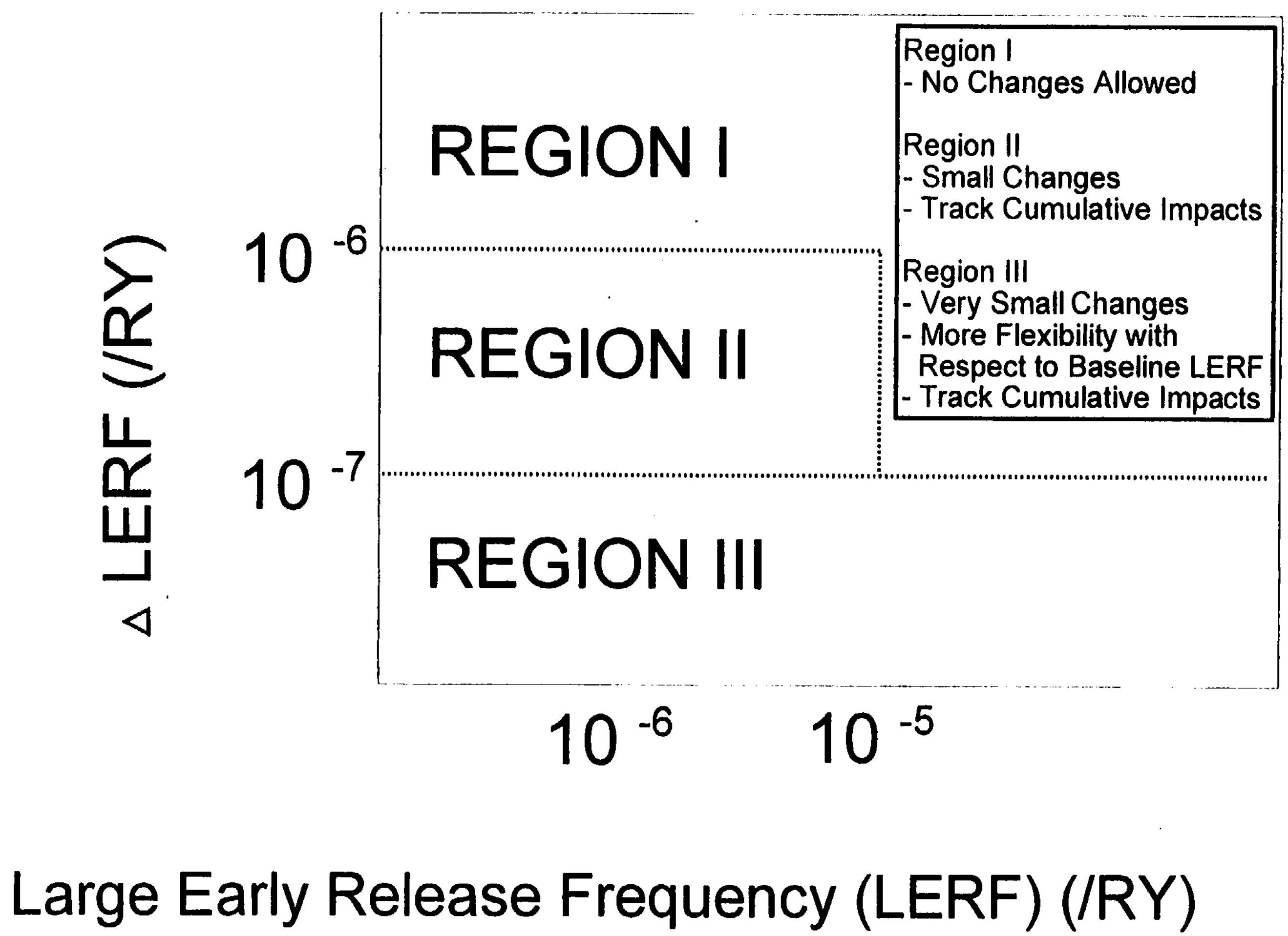
DECISION MAKING AND PSA QUALITY (Continued)

- Not Meeting the Standard in All Aspects Does Not Preclude Using the PSA for a Specific Decision if Those Elements That Do Not Meet the Standard Have No Impact on the Necessary Insights.

- The Degree of Confidence in the Results Can Vary Even if the Standards Are Met.

- Alternate Approaches to Improve Confidence That a Good Safety Decision Has Been Made:
  - Refine the Analysis
  - Rely on a Performance Monitoring Program to Confirm That the Assumptions Made Are Not Invalidated
  - Restrict the Scope of Implementation by Defaulting to More Conservative Decisions.

LERF ACCEPTANCE GUIDELINES



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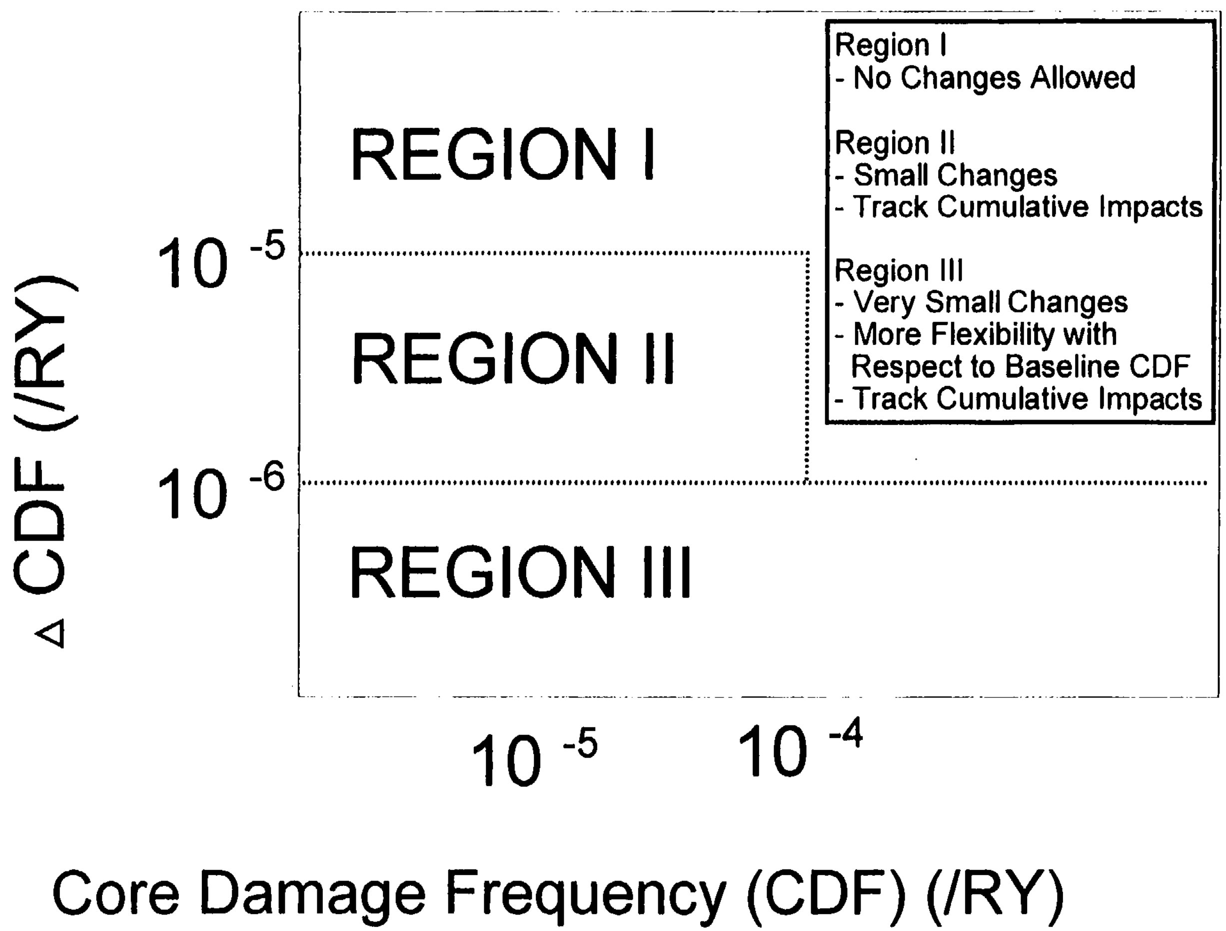

RISK-INFORMED INITIATIVES OVERVIEW

- Activities Are Underway in a Number of Areas to Take a Risk-Informed Approach to Regulation in Both Plant-Specific and Generic Activities:
  - Regulations
  - Licensing
  - Plant Oversight
  - Events Assessment

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- Risk-Informed Activities Build Upon Current Infrastructure, Policies, and Practices:
  - PRA Policy
  - Risk-Informed Regulatory Guides/SRPs

RISK-INFORMED REGULATORY GUIDANCE

Principles of Risk-Informed Decisionmaking

The proposed change meets the current regulations unless it is explicitly related to a request exemption or rule change

The proposed change is consistent with the defense-in-depth philosophy

The proposed change maintains sufficient safety margins

Integrated Decisionmaking

The impact of the proposed change should be monitored using performance measurement strategies

When proposed changes result in an increase in core damage frequency and/or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement

OBJECTIVES FOR RISK-INFORMED REGULATION

- Enhance Safety Decisions (e.g., Configutation Control, Accident Management)
- Efficient Use of NRC Resources (e.g., IPE Insights, Risk-Informed Inspections)
- Reduce Industry Burden (e.g., Grade QA, Risk-Informed IST)

Transition to Risk - Informed Regulation

By

Robert A. Bari
Brookhaven National Laboratory
P.O. Box 5000
Upton, NY 11973-5000

ABSTRACT

Laboratory initiatives and contributions in support of the transition to risk-informed regulation are presented and discussed. Key accomplishments are noted and their impacts on current approaches to regulatory activities are summarized. Particular attention is given to how regulatory decision-making is impacted by uncertainty in PRA results and to the key technical impediments to moving to risk-informed regulation. Some challenges are presented for the research community in facilitating the practical applications of probabilistic risk assessment in the regulatory area.

1. Laboratory Initiatives

Several of the U.S. Department of Energy National Laboratories have contributed to the advancement of probabilistic risk assessment as it relates to the safe operation of commercial nuclear power. The U. S. Nuclear Regulatory Commission (NRC), in particular, has benefited from the availability of this vital national resource as it carries out its mission to assure the safety of the public with regard to nuclear power plants. Indeed, the Energy Reorganization Act of 1974 enabled the NRC to have the availability of the scientific and technological knowledge and resources of the national laboratories in order to fulfill its mission.

Notably, the national laboratories have participated in every major PRA-related initiative of the NRC. This includes: the ground breaking Reactor Safety Study (WASH-1400) in the 1970s, the plethora of studies that were performed in the aftermath of the accident at Three Mile Island in 1979, the severe accident analysis and management programs of the 1980s, the major risk updating study (known as NUREG-1150), offsite consequence modeling improvements, external events methodologies, and human performance assessments.

The laboratories have also been actively involved in the formulation and execution of the Individual Plant Examination programs (beginning in the late 1980s and continuing through much of the 1990s) in which probabilistic risk assessments were performed by the plant owners. This led to the identification of specific vulnerabilities at each of the plants and to greater insight, by the owners, to improved operation of their facilities. Over the same time frame, studies were undertaken of the risk posed by power reactors while in shutdown or low power conditions. These studies resulted in and enhanced understanding of the overall risk envelope for reactor operation and to improvements in activities associated with these other modes of operation.

More recently, two of the national laboratories, Brookhaven and Sandia, have been working in close collaboration with the NRC research staff on the development of approaches to risk-informing 10 CFR 50. This includes a framework document and specific applications of an approach to risk informing 10 CFR 50.44 and 10 CFR 50.46. The former refers to combustible gas control in containment and the latter refers to the requirements on the emergency core cooling function. A summary of progress on these activities is contained in a paper by the research staff in this session.

Finally, the laboratories have played key roles in the development of standards for the performance of probabilistic risk assessments. The need for standards has long been recognized. A fundamental dilemma has been how to develop a standard in an area that is still emerging and for which improved methods may yet be in the offing. Early attempts at standardization can be found in NUREG-2300 and NUREG/CR-2815. The advantage of a standard to the industry is that if they submit a request for a regulatory review to the NRC that is based on PRA methods and if it is done within an accepted standard, then the review would not be encumbered by a case-specific review of the methodology itself.

2. Key Accomplishments and Regulatory Impact

The foregoing activities have led to many accomplishments, which have in turn had lasting regulatory impact. For example, the focus on the impact of human performance on nuclear power plant risk has been underscored in many PRAs. These have led the way to the formulation and execution of research programs by NRC, the industry, and by organizations in many countries on human performance. This has led to valuable insights that have affected emergency plans, accident management programs, and the day-to-day efficient operation of each unit.

The "level two" portion of the PRAs has been invaluable in providing great insight into the performance of containment under severe accident loads. This includes the timing of pressure and temperature loads as well as the behavior of combustible gases and core debris within containment. The relative benefits of each containment type are now far better understood because the phenomenology was evaluated and prioritized within the context of a level two PRA.

Risk perspectives from performing PRAs for various modes of power operation also were of great benefit to the agency and the industry. These studies led to utilities having enhanced flexibility in managing outages and in extending allowed outage times. Further, the studies led to better decision making by the NRC staff and to more effective implementation of Regulatory Guide 1.174.

The Individual Plant Examination Program, which was essentially a PRA enterprise, led to many improvements, directly by the owners in the procedures and operations of their plants. The program also led to improvements in systems, structures, and components and to insights for improved decisions making by the regulator and the industry.

More timely review and approval of requests for specific regulatory actions, that are based on PRA methods, will come to fruition as the standards program reaches an acceptable level of maturity.

3. Decision Making and Uncertainty

Much has been said and written over the past two decades on this topic. Hopefully we are coming to closure on new insights on this subject. It has been a strength of PRA and the risk informed approach that

it lends itself well to the expression of uncertainties. After all, uncertainty is at the very heart of risk. On the other hand, the elucidation of uncertainties by PRA has been taken by some to be a limitation of the PRA methodology (i.e. it deals in vagaries and cannot be trusted). It is also unfortunate that PRA practitioners and advocates have so often felt the need to dwell on the limitations of PRA. However, it should be recognized that deterministic methodologies are also fraught with uncertainties, but they are not as glibly expressed as they are by the PRA methodologists. PRA allows decision-makers to be informed about *what they know about what they do not know.* And this is valuable information to have.

4. Deterministic vs. Probabilistic vs. Prescriptive vs. Performance Based

The title of this subsection refers to four notions that are related, sometimes used conjunctively, and sometimes interchanged. Figure 1 provides a two dimensional view of how they should be correctly related. Prescriptive and deterministic are sometimes confused and used interchangeably. They tend to be within the comfort zone of some participants in regulatory matters and perhaps that is the source of the interchangeability. Deterministic is really an approach to analysis and its opposite is probabilistic. Prescriptive is an approach to decision making and its opposite is performance based. Most analyses are not purely probabilistic or deterministic but an admixture of the two extremes. A classic "Chapter 15" safety analysis really starts by determining, however informally, what is likely and unlikely. Similarly, a probabilistic analysis as embodied in a PRA, usually has some form of deterministic analysis--e.g. a heat transfer calculation. The important point is that this is a two dimensional space with various regulatory activities falling in different parts of the plane. For example, if one were interested in compliance with a numerical safety goal related to the likelihood of an event, it would fall in the lower right quadrant.

5. Risk Currency

In the early days of PRA, both regulators and the industry for the most part were not very comfortable with PRA and the use of risk management concepts to guide decisions with regard to the safe operation of plants. This has changed, in an evolutionary way, over the past quarter century. Now it is quite the custom to see, hear, and read of exchanges between the industry and the NRC (and among the organizations themselves) in which concepts like risk assessment and risk management form the currency for their exchange of "safety thought". There has been an ever-increasing attempt by the NRC and the industry to bring other stakeholders (states, public interest groups, and concerned citizens) into the communication loop on safety matters. However, the process, while moving forward, involves the exchange of different currency with regard to safety. This can be termed risk perception or more broadly, risk communication (see Figure 2). These are very important aspects of the safety enterprise and require much attention and development. It should be noted that the simple figure does not capture the fact that the indicated modes of currency are not exclusive to any of the interchanges. Each interchange is really an admixture of the two indicated types. The figure just indicates the predominate exchange mode.

6. Challenges for the Research Community

Safety research, being an inductive discipline, will always have its frontiers. The following are three areas that might benefit from the now vast repertoire of knowledge and methodology that is contained in the PRA field.

Much of the work in PRA has focussed on the severe accident. This type of accident (core melt, large radiological release to the environment) tends to be unacceptable to all parties concerned. Fortunately they also are the very unlikely events--not expected to occur in the lifetime of a facility. There are events,

however, that are much more likely, that do occur and, fortunately, have little or no health or environmental impact. Yet they do attract public attention and require much attention by all parties involved. A recent example of such an event is the steam generator tube rupture at the Indian Point 2 plant. The offsite radiological release was insignificant but the event drew much attention from the press and the political representatives (and therefore the NRC and the industry). From the PRA perspective, this was a very low risk event that would typically be disregarded in the analysis. While the PRA methodology can easily express the risk of such a small consequence event, can it be used to flag such events? Can these risks be managed? How should they be managed and by whom?

A related topic is risk perception. It is well known that an analytical quantification of health and environmental risks, an objective expression of reality, does not necessarily represent the risk that an individual or a group perceives. Further, it is sometimes said that perception is reality. The question is: can psychosocial measures of risk be defined and calculated in a way that is comparable to physical risks? If so, can risk management programs be developed that recognize these measures?

Finally, it should be recognized that the NRC and the civilian nuclear power industry have been at the vanguard of PRA methods development and applications. This has, no doubt, contributed to the safety and reliability of this technology. Other organizations and industries would benefit from the advances made in the uses of PRA and there should be a wider scale adoption (and adaptation) of these methods and applications.

Acknowledgements

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Figure 1

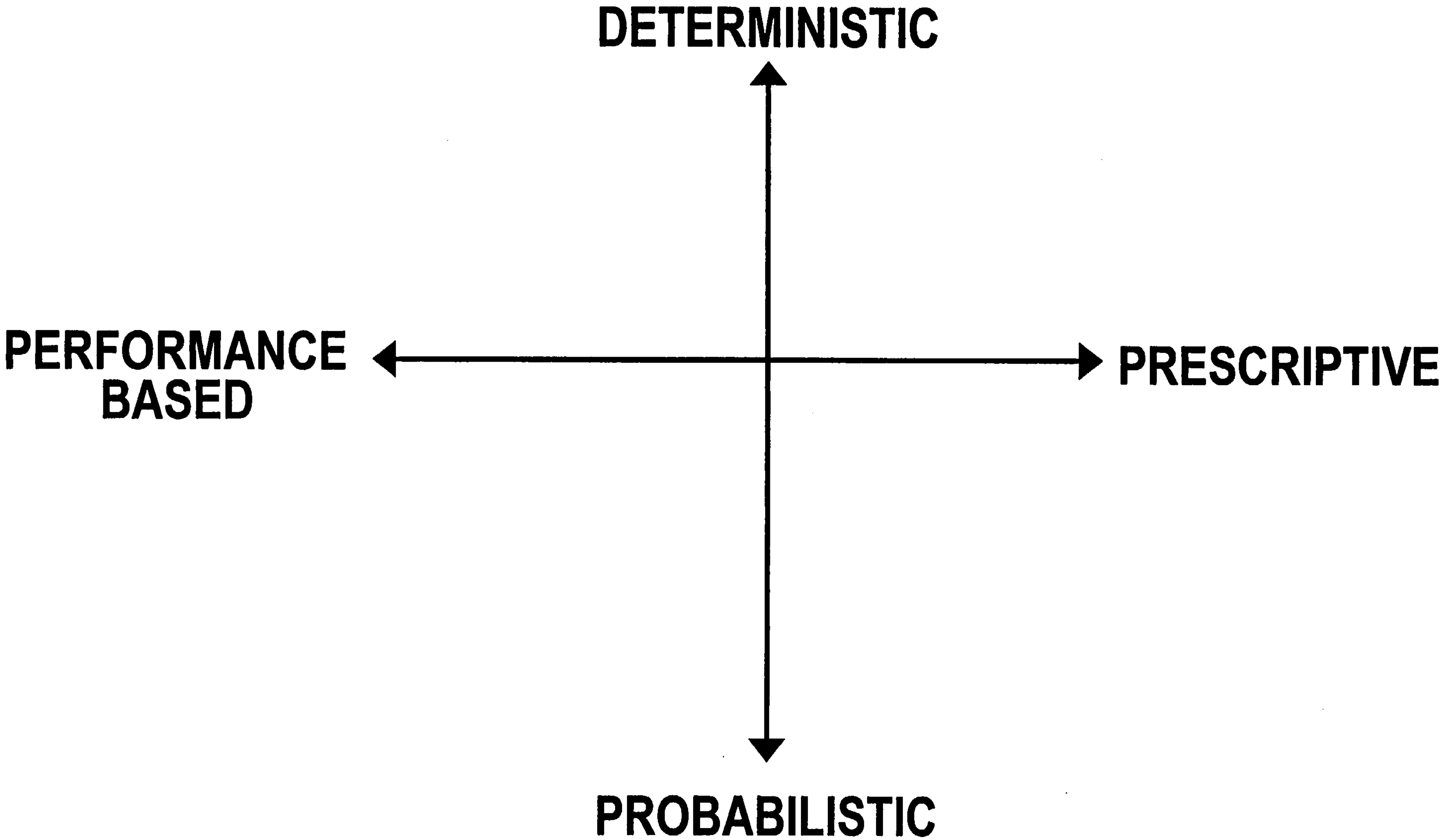
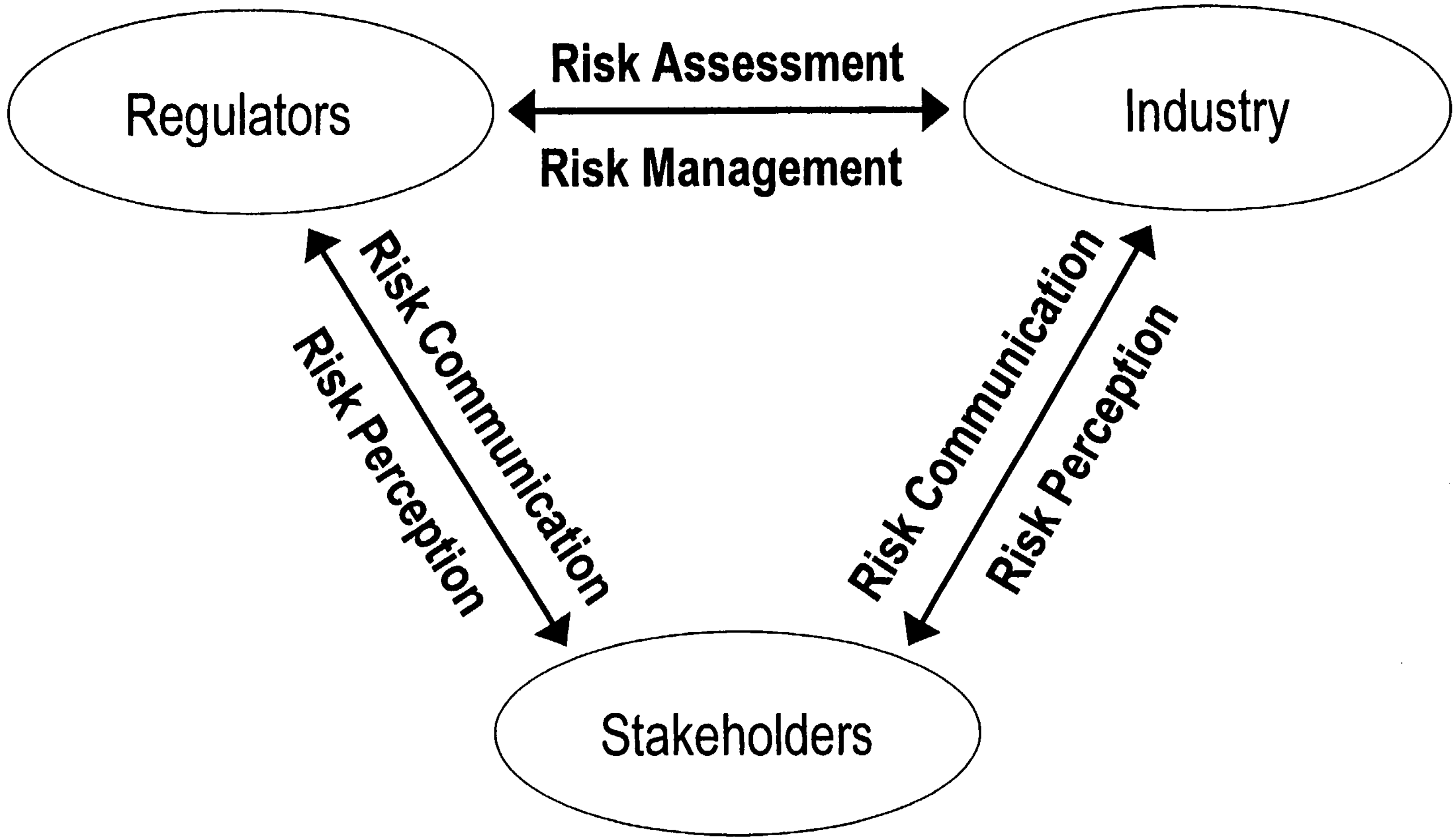

DETERMINISTIC
PERFORMANCE BASED
PRESCRIPTIVE
PROBABILISTIC

Figure 2



Regulators
Risk Assessment
Risk Management
Industry
Risk Communication
Risk Perception
Risk Communication
Risk Perception
Stakeholders

Risk-Informed Regulation

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Biff Bradley
NEI
October 23, 2000

Objectives

Discuss:

- Impact of uncertain PRA results on regulatory decisionmaking
- Key technical impediments to risk-informed regulation

NEI

NEI Role in Risk-Informed Regulation

- Goal: Widespread implementation of risk-informed regulation
  - Represent industry in achieving framework
  - Facilitate change within industry
  - Provide input into regulatory development

NEI

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NEI Role

- Primary role is in policy, direction
- Technical support provided by industry:
  - Owners Groups
  - EPRI
  - Industry technical experts

Observations

- PRA = state of knowledge
- Variability exists in industry
  - Usage of risk insights
  - State of development of models
  - Enthusiasm for change

Observations

- State of knowledge continues to improve
- Major Incentives
  - Oversight process
  - Plant configuration control
  - Applications (ISI, Tech Specs)
  - Deregulation of market

NEI

Impact of Uncertain PRA Results on Regulatory Decisionmaking

- All regulatory methods contain uncertainty
  - Obvious impact is increased conservatism in decision
  - Uncertainty results in need for balance of deterministic and risk methods

NEI

Uncertainty

- Can be minimized through structure of application
  - Delta versus absolute risk
  - Reliance on overall risk insights rather than numbers
  - Selection of applications

Uncertainty

- Parameter (data)
- Modeling (eg., RCP seals)
- Incompleteness (scope)

Examples

- Risk-informed ISI
  - Known versus unknown degradation mechanisms
- Spent fuel pool study
  - Regulation based on one very low probability initiating event

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Technical Impediments

- Scope and detail of risk evaluations
  - Internal events
  - Containment performance
  - Fire
  - Seismic
  - Shutdown

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Technical Impediments

- Degree of detail in specificity of PRA requirements
- Degree of reliance on risk metrics in decisionmaking
- Low probability high consequence events
- Blending of deterministic and risk insights

Cultural Impediments

- Acceptance of change
- "Belief" in PRA results
- Uncertainty in outcomes

NEI

INSPECTION OF THE CASTOR-V/21 CASK AND CONTENTS

Roger M. Kenneally
U.S. Nuclear Regulatory Commission

John H. Kessler
Electric Power Research Institute

A Castor-V/21 cask containing 21 spent PWR rods (burnups in the 30-35 GWd/MTU range) has been in storage at the Idaho National Environmental and Engineering Laboratory (INEEL) since 1985. This cask represents one of the longest storage periods in the current fleet of licensed dry storage containers in the United States. Given that current dry storage cask licenses are only for 20 years, and several cask systems are approaching the end of the initial license period, it is necessary to establish a technical basis for extended storage. Consequently, the Nuclear Regulatory Commission, the Electric Power Research Institute, and the Department of Energy have embarked upon a cooperative research program to assess the integrity of this cask in order to establish a partial basis for extended dry storage in existing licensed casks. The Castor cask has been reopened and the cask internals, fuel assemblies, and selected rods from one fuel assembly have been visually inspected at the INEEL. The cask, and the stored fuel rods appeared to be unchanged by the long storage duration.

INTRODUCTION

Most nuclear power plants in the United States were not originally designed with a storage capacity for the spent fuel generated over the operating life by their reactors. Utilities originally planned for spent fuel to remain in the spent fuel pool for a few years after discharge, and then to be sent to a reprocessing facility. Since reprocessing has been eliminated, and no other option for spent fuel disposition currently exists, utilities expanded the storage capacity of their spent fuel pools by using high-density storage racks. This has been a generally short-term solution with many utilities having reached, or soon will reach, their spent fuel pool storage capacity (Fisher and Howe, 1998). Utilities have developed independent spent fuel storage installations as a means of expanding their spent fuel storage capacity on an interim basis until a geologic repository is available to accept spent fuel for permanent storage.

The U.S. Nuclear Regulatory Commission (NRC) promulgated 10 CFR Part 72 (Title 10, 2000) for the independent storage of spent nuclear fuel and high-level radioactive waste outside reactor spent fuel pools. Part 72 currently limits the license term for an independent spent fuel storage installation to 20 years from the date of issuance. Licenses may be renewed by the Commission at or before the expiration of the license term. Applications for renewal of a license should be filed at least two years prior to the expiration of the existing license.

In preparation for possible license renewal, the NRC Office of Nuclear Material and Safeguards, Spent Fuel Project Office, is developing the technical basis for renewals of licenses and Certificates of Compliance for dry storage systems for spent nuclear fuel and high-level radioactive waste at independent spent fuel storage installation sites. These renewals would

cover periods from 20 to 100 years, and would require development of a technical basis for ensuring continued safe performance under the extended service conditions. An analysis of past performance of selected components of these systems is required as part of that technical basis.

In the 1980s through the early 1990s, the Department of Energy (DOE) procured four prototype dry storage casks for testing at the Idaho National Engineering and Environmental Laboratory (INEEL): Castor-V/21, MC-10, TN-24P, and VSC-17. The primary purpose of the testing was to benchmark thermal and radiological codes and to determine the thermal and radiological characteristics of the casks.

The Castor-V/21 cask was loaded in 1985 with irradiated assemblies from the Surry Nuclear Station and then tested in a series of configurations using a variety of fill gases. Since the tests were not intended to be fundamental fuel behavior tests, the fuel prior to the tests had undergone only minimal characterization consisting of visual examination of the outside of the assemblies and ultrasonic examination to ensure no breached rods would be included. During the tests, the temperature at various locations was monitored and the cover gas was periodically analyzed to determine if any leaking rods had developed. No leaking rods were found. The details of these tests have been reported in a number of documents. Since the conclusion of the testing in 1985, the Castor-V/21 cask containing the Surry fuel assemblies remained on the storage pad at INEEL.

The NRC, the Electric Power Research Institute (EPRI), and the DOE Offices of Civilian Radioactive Waste Management (DOE-RW) and Environmental Management (DOE-EM) are participating in a cooperative research program (Dry Cask Storage Characterization Program) to determine the long-term integrity of dry cask storage systems and spent nuclear fuel under dry storage conditions. The program objectives are (1) determine the long-term integrity of dry storage cask systems and spent nuclear fuel under dry storage conditions, and (2) provide data to augment the technical bases and criteria for evaluating the safety of spent-fuel storage and transportation systems, and for extending dry cask storage licenses. The Castor-V/21 cask at INEEL was selected for study under this Program. This cask represents the "lead" storage cask in the United States - the cask with fuel assemblies stored inside for the longest amount of time. A summary of the scope of work performed on the Castor-V/21 cask and fuel can be found in Kenneally and Kessler (2000).

CASTOR-V/21 CASK DESCRIPTION

Cask Body

The cask body is a one piece cylindrical structure composed of ductile cast iron in modular graphite form. This material exhibits good strength and ductility, as well as providing effective gamma shielding. The external dimensions of the cask body are 4886 mm (16 ft.) high and 2385 mm (7.8 ft.) in diameter. The external surface has 73 heat transfer fins that run circumferentially around the cask, and is coated with epoxy paint for corrosion protection and ease of decontamination (Figure 1). The cask body wall, excluding fins, is 380 mm (15 in.) thick. Incorporated within the wall of the body are polyethylene moderator rods to provide

neutron shielding. Two concentric rows of these 60 mm (2.4 in.) nominal diameter rods are distributed around the cask perimeter. Two lifting trunnions are bolted on each end of the cask body.

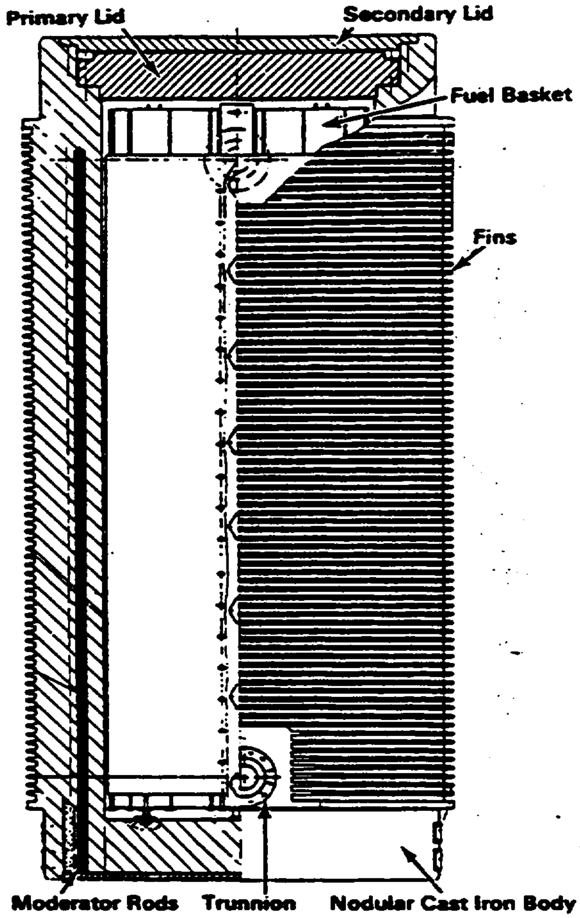

Figure 1. Castor-V/21 Cask Vertical Cross Section

(From Virginia Power, et al., 1986. Permission to use this copyrighted material is granted by the EPRI)

Spent Fuel Basket

The spent fuel basket is a cylindrical structure of welded stainless steel plate, and borated stainless steel plate, having a boron content of approximately 1% for criticality control (Figure 2). The basket comprises an array of 21 square fuel tubes/channels that provide structural support and positive positioning of the fuel assemblies. The basket overall height is 4110 mm (13.5 ft.) including the four 130 mm diameter (5 in.) pedestals that support the basket and fuel weight on the bottom of the cask cavity. The basket outside diameter of 1524 mm (5 ft.) fits tightly in the cask cavity inner diameter of 1527 mm (5 ft.).

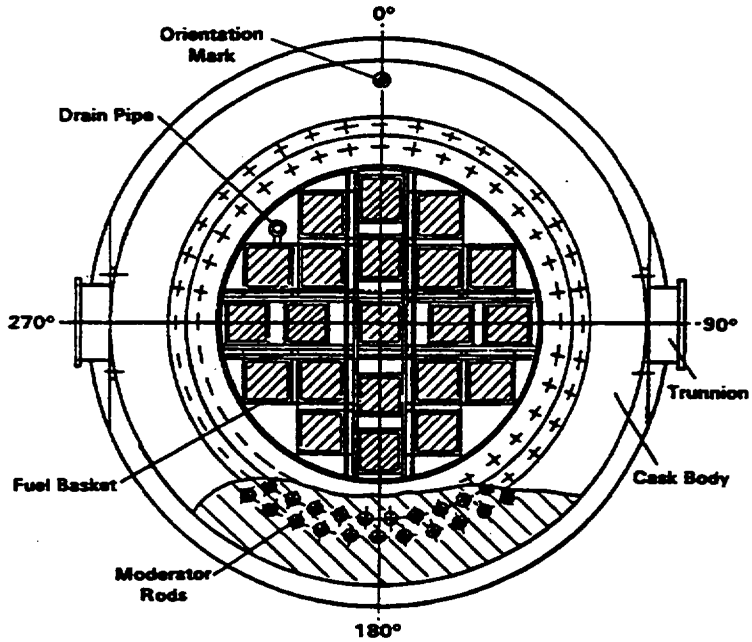

Figure 2. Castor-V/21 Cask Cross Section
(From Virginia Power, et al., 1986. Permission to use this copyrighted material is granted by the EPRI)

Primary Lid

The stainless steel primary lid is 1785 mm (5.8 ft.) in diameter and 290 mm (12 in.) thick. Forty-four bolt holes are machined near the lid perimeter to secure the lid to the cask body. Two grooves machined around the lid underside, inside the bolt circle, are provided for O-ring gaskets (Figure 3). The inner groove accepts a metal O-ring, which serves as the first barrier between stored fuel and the environment. The outer grove accepts an elastomer O-ring. A 10 mm diameter (0.4 in.) penetration through the lid provides access to the annulus between the two seals to perform post-assembly leak testing. This penetration is plugged when not in use.

Secondary Lid

The stainless steel secondary lid is 2007 mm (6.6 ft.) in diameter and 90 mm (3.5 in.) thick (Figure 3). Forty-eight bolt holes are machined near the lid perimeter to secure the lid to the cask body. Two concentric grooves located inside the bolt circle on the underside are provided for a metal O-ring/elastomer O-ring sealing system of the same design as that used on the primary lid. Three normally sealed penetrations are provided for various cask operations. A 10 mm diameter (0.4 in.) penetration through the lid provides access to the annulus between the two seals for post-assembly seal testing. A gasketed seal plug is used to close this penetration.

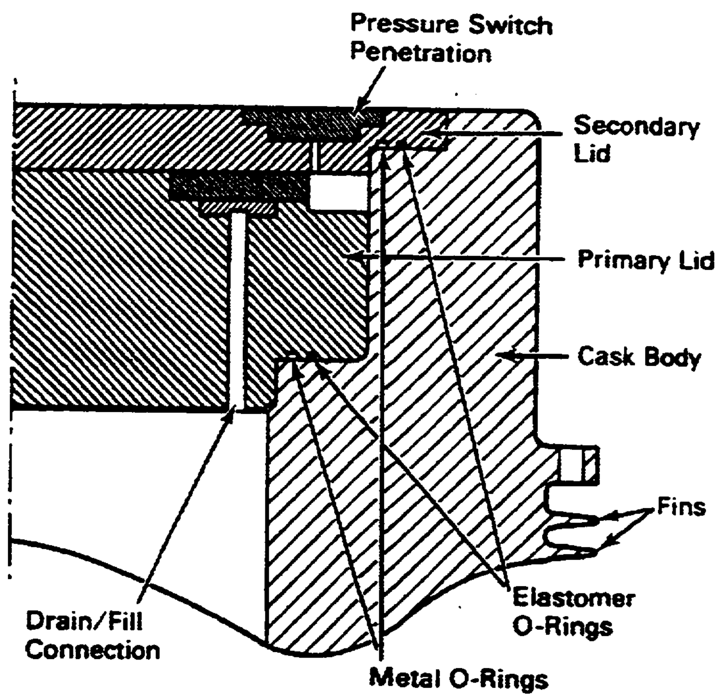

Figure 3. Castor-V/21 Seal System
(From Virginia Power, et al., 1986. Permission to use this copyrighted material is granted by the EPRI)

The secondary lid was not used in this particular cask because of interference with fuel assembly instrumentation leads that were installed during the Castor-V/21 cask performance test conducted in 1985.

CASK SUPPORT PAD DESCRIPTION

The concrete pad was designed to hold six spent fuel storage casks. The size of the pad is approximately 29 m (95 ft.) long by 12 m (40 ft.) wide. The pad consists of 600 mm (2 ft.) thick concrete on top of a minimum of 300 mm (12 in.) of compacted subbase of pit run gravel. The concrete was reinforced with two mats of No. 6 steel reinforcement bar spaced 180 mm (7 in.) on center (each way); the mats were each embedded 100 mm (4 in.) below and above the top and bottom surfaces of the pad, respectively. The concrete was covered and kept wet during the first few weeks of the curing period to ensure maximum strength and durability. The design strength of the concrete was 28MPa (4000 psi); the 28-day post-cure compression strength averaged 30MPa (4400 psi).

Although it was designed to hold six storage casks, the pad held four dry storage casks, including the Castor V/21. The Castor V/21 cask was located approximately 13 m (43 ft) from the west edge and 4.3 m (14 ft) from the north edge of the pad.

SUPPORT PAD, CASK COMPONENT, AND FUEL ASSEMBLY INSPECTIONS

Inspections were performed on the concrete support pad, cask exterior, primary lid bolts, primary cask lid seals, cask interior, and fuel assemblies to assess if degradation occurred, and if applicable, the mechanisms of the degradation such as, corrosion, wear, or accumulation of crud.

Support Pad

Approach

The evaluation of the integrity of the 15-year old support pad consisted of the testing of the structural soundness of the surface of the concrete and a visual assessment of the physical condition of the concrete surface, particularly at and immediately around the cask location to identify evidence of degradation or structural failure.

The structural soundness of the concrete was determined by ASTM test standard C805-94 (Test Method for Rebound Number of Hardened Concrete, also known as the Swiss hammer test). The Swiss hammer test was performed on the concrete surface in 9 places in a 37.2 m^2 (400 ft^2) area centered on the placement of the Castor V/21 cask, of which 5 places were selected in the area under the cask.

Results and Observations

The Swiss hammer test results, which ranged from 28MPa (4050 psi) to 41 MPa (5900 psi), averaged at 33 MPa (4800 psi) and demonstrated that the structural integrity of the concrete pad still meets or exceeds the 28 MPa (4000 psi) design strength of the concrete.

The surface of the whole pad did not exhibit any evidence of structural failure of the concrete, such as open cracks or cracks with displacements in elevation of the surface. The surface of the concrete did not exhibit any evidence of spallation of the surface, exposed aggregate, or aggregate pop-out from the surface. The surface was solid and exhibited only minor wear and environmental weathering, well within the extent of weathering expected for the cold and windy climate of Idaho. The broom-finished unpainted surface exhibited only a network of faint, fine surface shrinkage cracks, less than 0.8 mm (1/32 inch) wide and of superficial depth, and a few rust stains under the cask from lightly rusted bolts on the cask. Similar cracks were prevalent across the entire surface of the pad, and were not associated with the cask locations. Tests with a straight taught line across the 6.1 m x 6.1 m (20 ft x 20 ft) grid indicated that there was no sag or vertical displacement in the concrete associated with the crack network; measurements with a straight edge and the taught line indicated only localized variations in the elevation of the concrete that were less than 3.2 mm (1/8 in.). The localized variations in elevation were not associated with the location of the cask, and were most likely an artifact of the screeding and finishing when the concrete was originally poured.

Cask Exterior

It appeared that the cask had not undergone any real damage, although some small superficially corroded areas were noticed where the epoxy paint had peeled. The epoxy may contain UV inhibitors, which would not have been uniformly mixed, and the densification of

which may have cause peeling of the exterior layer. Alternatively, the paint loss could be due to abrasion by chocks during the handling of the container when it was previously moved.

Primary Lid Bolts

The 44 bolts of the primary lid were individually inspected visually for their physical condition, specifically for evidence of cracks, pitting corrosion, general corrosion, thread damage, and any discoloration.

All bolts were in satisfactory condition. None had any indications of pitting or general corrosion, cracks, thread damage, discoloration, or any defects or indications of potential failure.

Primary Lid Seals

Approach

The accessible surfaces of the primary lid O-rings were inspected immediately after the cask was opened. In early 2000, at the end of the cask and fuel assembly inspections, the original O-rings were replaced and the entire surface of the original O-rings was subjected to a direct visual examination.

The objectives of the inspection were to evaluate the condition of the seals for potential degradation due to (1) oxidation of the elastomer and metal seals; (2) thermal degradation of the elastomer seal; (3) embrittlement or hardening, including cracking, crazing and evidence of loss of elasticity or ductility; and (4) physical damage to the seals, such as scratches across the seal surfaces, dents, and seal deformation.

The remote inspection used three video cameras mounted on a work stand (work platform) at 120° intervals around the top perimeter of the cask. The resolution and color rendition of the cameras were checked daily with a resolution chart. The magnification and resolution of the remote cameras were sufficient to discern fine defects. For example, in the initial inspection immediately upon opening the cask, it was possible to clearly identify a long fine hair (presumably human hair) that was looped across the two O-rings of the primary lid.

Observations

The O-rings in the primary lid were in excellent condition. The remote visual inspection immediately upon opening the cask and removal of the primary lid indicated that the compression area of the elastomer and the metal O-rings were free of breaks, cracks, crazing, delamination, pull-outs, oxidation or other evidence of degradation of the O-rings.

Elastomer O-Ring Seal.

The only observed defect in the elastomer O-ring was an imperfect splice joint that was slightly misaligned and partially open. The glue did not completely fill the gaps in the joint; however, the joint still had good strength, and could not be pulled apart manually. Furthermore, this imperfection was not significant enough to cause air ingress during the 15 years of storage.

The elastomer O-ring was still firmly resilient in consistency, flexible, and limber, with no evidence of embrittlement, stiffness, or depolymerization. Bending, pulling, twisting, and coiling the elastomer into a 300 mm (12 inches) diameter coil did not cause fracture or stress failure.

The elastomer O-ring did exhibit random, crisply-delineated patches of light surface discoloration, appearing gray against the black color of the elastomer. These patches were not associated with surface relief or differences in flexibility, resiliency, or firmness of the polymer and had a 'graphitic' sheen, suggesting that they are probably caused by excess anti-seize lubricant that was used on the lid bolts. These 'graphitic' patches were much more numerous on the back side of the elastomer, which contacted the seat of the O-ring groove in the lid, suggesting that the anti-seize lubricant was used as an aid to hold the seal in place during lid assembly.

Metal O-Ring Seal

The metal O-ring compression surface did not show any evidence of breaks, scratches, dents, distortion, or corrosion. The compression area of the metal O-ring was textured due to the impression of the machining marks from the mating metal seal surface of the cask body. The metal O-ring was still ductile, as indicated by a few slight kinks, bends, and fresh surface scratches imparted by handling during removal from the lid.

The compression sealing area of the metal O-ring was, in general, quite reflective, glinting in the natural illumination in the hot shop, indicating that no significant corrosion or oxidation had occurred. The discolorations on the compression surface of the metal O-ring were usually associated with similar gray discoloration on the elastomer seal and with deposits/films of material on the metal flange of the bolt circle. It seems possible that excess fluid anti-seize compound may have run off the lid bolts onto the solid sealing surface of the cask body, and wicked onto the elastomer and the metal O-rings before the bolts and the lid were torqued down.

Cask Interior

Approach

The objectives of the cask interior examinations were to inspect the exposed, accessible, internal surfaces of the cask structure for evidence of cask and/or basket degradation caused by long-term storage. For the cask cavity, the visual inspection focused on evidence of corrosion and crack formation in the sidewalls and the bottom of the cask, particularly in the bottom corner, as well as the failure of the nickel coating by blistering, delamination, corrosion, or discoloration. For the fuel basket, the inspection focused on evidence of new cracks in welds or in walls of fuel tubes (cracks had been identified during the initial thermal testing conducted in 1985), propagation of existing cracks in welds, corrosion and discoloration of fuel tube walls, and accumulations of oxide particles on bottom support brackets and at the bottom of cask in each fuel tube.

The inspection of the inner wall at the top of the cask was performed by remotely using three video cameras mounted on a work stand (work platform) at 120° intervals around the top perimeter of the cask. The floor and the bottom corner of the cask were examined with a radiation-tolerant miniature (pencil) camera and light mounted at the end of a 4.5 m (15 ft.)

pole. As with the video cameras, the pencil camera resolution was checked daily with the resolution and color charts.

Most of the cask inner wall and bottom was not accessible to visual inspection due to the size and tightly fitting characteristics of the basket. At the top of the cask, only approximately 80 mm (3 in.) of sidewall was exposed above the top of the basket and the rebate below the seal area of the cask body (i.e., the sidewall area between the top of the basket and the bottom of the lid), the 50 mm (2 in.) step of the rebate, and approximately 250 mm (10 in.) of sidewall between the primary and secondary seal seats. The floor of the cask was accessible only through the 21 fuel tubes. The bottom corner and 20 - 50 mm (1 - 2 in.) height of cask sidewall was partially accessible through a few of the larger flux traps (approx 90 mm [3.5 in.] at the widest) at the periphery of the basket.

Because of the tight clearances for access to the bottom corner and sidewall of the cask, the examination was attempted initially with a borescope, but failed because of the narrow field of view, short working distance, short depth of field, and poor dynamic response of the borescope camera.

Interior Cask Sidewall Observations

The upper exposed area of the inner sidewall of the cask was in very good condition. The galvanically-applied nickel coating was still intact and did not show any evidence of blistering, peeling, cracking, delamination, or corrosion.

The nickel-plated sidewall was free of significant defects. However, a few isolated minor, superficial features or imperfections were visible in the visual inspection; these appeared to be light scuff and faint scrape marks that were most likely created during the initial installation of the basket in the cask. Adjacent to fuel tube D3 (Figure 4, at the 270° position), the sidewall had an imperfection that initial inspection identified as a blister or dimple. However, close examination of the illumination shadows indicated that the feature was a shallow depression (dimple) about 2 cm in diameter and probably only approximately a millimeter deep, with the nickel coating still intact. The visible surface of the sidewall also had several isolated, randomly-oriented superficial lines that could be surface deposits (from abrasion by a softer material) or superficial scrapes. These features are quite faint, with no discernible vertical dimensions, burrs, ridged edges or plow marks that usually are associated with scratches that penetrate coatings or gall a surface.

Considering the tight fit of the basket within the cask body, the nickel coating on the upper sidewall shows little evidence of damage due to insertion of the basket. The only discernible feature that might constitute significant coating damage was a black mark on the sidewall at the level of and coincident with the corner of fuel tube D3. However, the black surface mark appeared to be superficial, and did not have any burrs or dimensional relief indicative of substantial abrasion damage, corrosion product formation, or cracks. While the feature is coincident in location with the corner of fuel tube D3, it cannot be the result of abrasion by the corner of the fuel tube (by vibration from cask handling), for the fuel tube is separated from the wall by the thickness of the steel barrel plate comprising the outer rim of the basket. Instead, this feature may be the result of abrasion during insertion of the tightly-fitting basket into the cask or from vertical thermal expansion of the tightly-fitting basket barrel wall during the 1985 thermal tests. The upper sidewall has several similar, less distinct blemishes that could be

construed as light scuffing or abrasion of the nickel coating from contact during the insertion of the basket into the cask body. These features consist of black 'scuffs' and spots on the nickel surface, as if the nickel plating was lightly abraded from the high points of the rough as-machined surface of the cask body. These features have no discernible relief, implying negligible superficial damage at worst, and have no evidence of more than possibly superficial surface corrosion, as might have occurred prior to sealing the cask in 1985. Furthermore, there was no evidence of delamination and peeling of the nickel layer around these features, or of subsurface corrosion or blistering in the areas surrounding the features, which could be the expected effect from a corrosive, oxidizing environment.

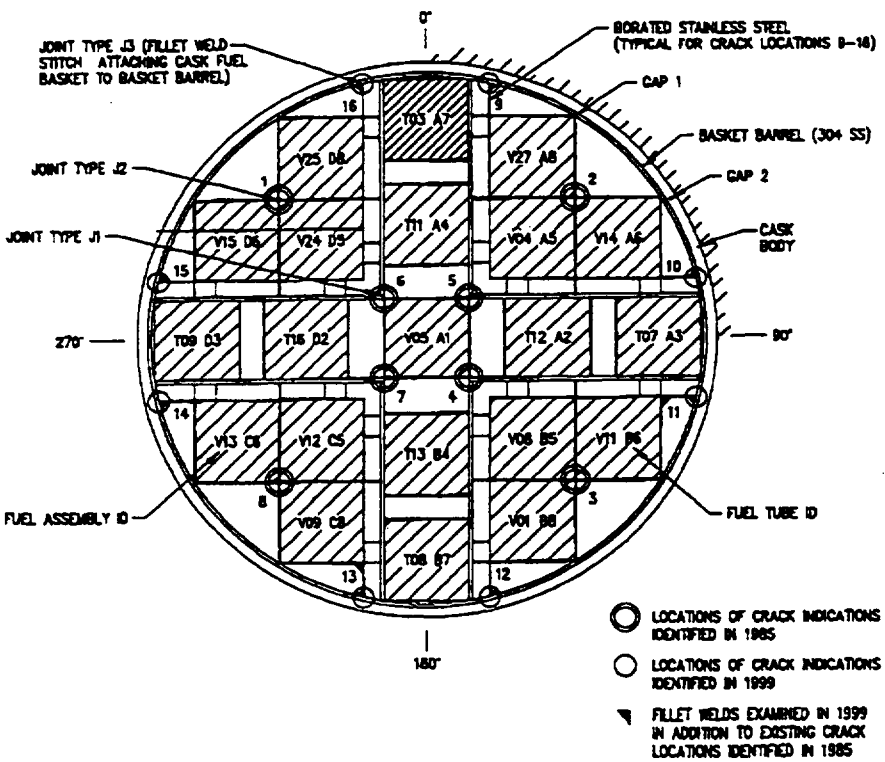

Figure 4. Castor-V/21 Cask Basket Crack Indication Locations

Interior Cask Bottom and Bottom Sidewall Observations

The cask bottom and bottom sidewall could be inspected only to a limited extent by access through the 21 fuel tubes and the eight channels at the perimeter of the basket. Access via the remaining flux traps was prevented by the tight dimensions of the traps and the structural gussets and spacers in the cavities of the traps. In addition, the inspection of the whole area of the cask floor was hampered by the small clearance (approximately 3.8 cm [1.5 inches]) between the bottom of the basket and the cask floor.

The floor of the cask was of roughly-grained as-cast texture, overcoated with the nickel plating. The floor of the cask turned smoothly up into the sidewall, so that the first centimeter or two of sidewall was also generally of rough as-cast texture. The sidewall above the bottom corner radius was smoother than the floor, as if it had been machined to remove the as-cast texture prior to nickel plating.

The nickel plating on the floor and bottom sidewall was generally clean and quite reflective despite the as-cast texture. There was no evidence of any corrosion, cracks, or flaws in the nickel plating, such as blistering or delamination, in the floor, corner, or sidewall of the cask. In general, the bottom sidewall was quite clean and reflective, particularly those areas that were machined prior to nickel plating. There were, however, isolated areas that appeared to be covered with light-colored spots of material that were not reflective and had no relief. These patches appeared to be mineral spots, as if deposited from residual water in the cask (as from evaporation of residual plating solution or rinse water). These flat, light-colored spots did not appear to contain much material as they had no relief (depth); neighboring areas were free of these deposits. They did not appear to be caused by corrosion or oxidation, nor was the integrity or adherence of the nickel plating affected by them.

Small grains of debris were thinly scattered over most of the cask floor. The debris ranged from sandlike particles of submillimeter to several millimeter size, to long slivers of material several millimeters in length. These generally appeared to have been deposited after the nickel plating of the surface, since the larger particles were dark in color and not reflective. Similar material had accumulated on the horizontal bars at the bottom of each fuel tube, on which the fuel assemblies rested. Much of the sand-like debris probably consists welding slag or grinding swarf from the basket. However, some of the debris appeared to consist of slivers of metal, and may be slivers of stainless steel gouged from the fuel tube walls by insertion and extraction of the fuel assemblies, since the fuel tube walls exhibited much evidence of scraping by the fuel assemblies.

Fuel Assembly Basket

Approach

The fuel basket was examined for evidence of further corrosion of the plate surface, the welds and associated heat affected zone, the junction between stainless steel and borated stainless, and contact points between the stainless steel structure and the zircaloy fuel assembly structure, such as on the steel brackets at the bottom of each fuel tube that support the weight of the fuel assemblies. In addition, the welds in the basket structure were inspected for failure, both for propagation of the cracks in the known broken welds and for initiation of new cracks in other welds.

The accessible portions of the fuel assembly basket inside the cask were inspected visually, using the three remote video cameras positioned around the top rim of the cask, and the pencil camera used to inspect the floor.

Only the surfaces of the basket directly accessible to the video and pencil cameras were inspected. The basket was inspected while in place within the cask. The extremely tight diametral clearance between the basket and cask wall (approx. 3 mm [0.1 in.]) prevented the unloading and extraction of the basket from the cask. The top surfaces of the basket were

inspected by the three video cameras mounted around the top of the cask. With the fuel assemblies removed, the interior surfaces of the 21 fuel tubes and eight ungussetted air channels were inspected with the pencil camera system. The interior surfaces of the flux traps and the triangular air spaces at the perimeter of the basket could not be inspected with the pencil camera, for these spaces were obstructed by welded spacers and gussets, or were too narrow to permit insertion of the pencil camera.

<u>Observations</u>

General

The fuel basket was in good condition, comparable to the surface condition observed in 1985 (Virginia Power, et al., 1986). In fact, some of the images of the tops of the basket in 1985 looked worse (more oxide scale) that in the 1999 inspections, an effect of the difference in lighting conditions. The basket structure showed no evidence of corrosion beyond the mill surface finish and the heat tarnish in the heat affected zones of the welds. The fabricator of the basket had left the mill surface finish on the steel plate components of the basket; no attempt had been made to remove the native oxide, stencils, construction layout marks, or environmental stains on the as-supplied steel stock.

Therefore, most of the surfaces of the basket structure had a light-gray non-reflective surface, as well as superficial oxide tarnish in the region of many of the welds. However, some of the interior surfaces of the fuel tubes bore the marks of spot (rotary) surface grinding that 'skinned' the flat surfaces; these ground surfaces were still brightly reflective under the camera illumination, indicating that neither significant air oxidation nor corrosion had occurred since the fabrication of the basket. There was no evidence of corrosion due to incompatibility between the stainless steel and the borated steel, nor was there evidence of corrosion or degradation at the contact between the zircaloy bottom nozzle of the fuel assemblies and the bottom support plates in the fuel tubes. No cracks or similar degradation was seen in the steel plate components, except for some of the welds as noted below.

Basket Welds

The 1985 inspection of the basket after the completion of the heat transfer performance tests identified eight broken welds (Figure 4, welds 1 - 8) in the top of the basket (Virginia Power, et al., 1986). The welds cracked as a consequence of the stresses created by the differential thermal expansion of the tightly-fitting basket within the cask during the tests.[1] An objective of the 1999 inspections was to reexamine the affected welds for any changes in configuration, and to examine other accessible welds in the basket structure for cracks or corrosion. Unfortunately, the stitch welds of the structure are located in the flux trap and spacer channels, not inside the 21 fuel tubes. Therefore, the only accessible welds were the welds visible at the top of the basket and a few others.

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[1]The basket used in these tests was specifically designed to be more tight-fitting than those used in all other Castor casks. Thus, the weld cracking observed in the 1985 INEEL tests is not expected to be indicative of the performance of other Castor casks.

Welds numbered 1 - 8 in Figure 4 appeared to be the same as in the 1985 inspection. The four welds in the corners of the central fuel tube A1 (welds numbered 4 - 7) involved welds of stainless steel to borated stainless steel, whereas the four welds joining the fuel tubes in other locations (welds numbered 2, 3, 8, and 1) involved only stainless steel. The 1999 inspection confirmed that five welds were broken clear through (as initially observed after the 1985 thermal testing), and three had substantial cracks that propagated partially through the top stitch weld. The narrowness of the flux traps and the supports within blocked the views of the stitch welds below the top welds from the top-side video cameras, and prevented the insertion of the pencil camera assembly. Therefore, the condition of those welds could not be determined.

The top stitch welds throughout the top of the basket were inspected, as well as the welds of the top-most struts within the flux traps. Except for the eight known cracked welds, the remaining welds were in good condition.

The stitch welds in the triangular air channels at the perimeter of the basket (fillet weld stitch attaching the cask fuel basket to the cask barrel) were examined with the pencil camera system. The gusset- and strut-free channels were just large enough to permit insertion of the pencil camera for viewing the stitch welds attaching the fuel basket partition plates to the basket barrel.

The inspections found that all eight of the top stitch welds were cracked, and seven of the eight bottom welds. The intermediate welds did not appear to be cracked.

Additional cracks in basket welds were associated with stainless/borated steel junctions welds in fuel tubes.

It is not possible to unequivocally state these stitch welds cracked during the 1985 thermal testing since no visual inspection of these welds was performed at that time. However, stress analyses undertaken to assess the weld cracks that were observed immediately after the 1985 thermal testing suggest that the yield strength of these stitch welds was also exceeded during the thermal testing (Virginia Power et al., 1986). Thus, it is probable that these weld cracks also occurred during the initial thermal testing and not during the subsequent years of dry storage.

Fuel Assemblies

Approach

The Surry PWR fuel assemblies in the Castor-V/21 have burnups in the range of 30-36 GWd/MTU. The assemblies were out of the reactor for a total of 26 or 46 months at the time of initial loading and testing in 1985. Thus, the initial decay heat at the time of initial loading was estimated to be 28kW. While these burnup levels are now more indicative of 'medium' burnup fuel, the relatively short decay time at the time of loading means that the initial thermal output for this cask is also roughly representative of higher burnup fuels with longer initial decay times prior to dry storage. Thus, the peak temperatures experienced by this cask are broadly representative of what would be experienced by a cask containing higher burnup fuel assemblies with longer out-of-reactor times prior to dry storage.

The inspection searched for evidence of change in the structure and integrity of the fuel assemblies: changes in corrosion and crud deposits on nozzles, grid spacers, rod cladding; additional corrosion, loose or lightly adherent corrosion product or crud; evidence of spallation or flaking; physical degradation or damage to nozzles, spacers, fuel rods; cracks, bowing of rods, or distortion.

The visual inspection required the removal of the fuel assembly from the basket. Once the assembly was lifted out of cask basket, the inspectors identified the assembly serial number and its orientation with respect to basket. They checked the relative uniformity of fuel rod lengths by clearance between tops of rods and top nozzle, then scanned the four sides using the three remotely operated cameras with zoom capabilities. After the visual inspection, the fuel assembly was returned to its original channel in the cask, maintaining the original orientation.

Observations

Table 1 provides the fuel assembly weight and the force required to start lifting each assembly out of the V-21 cask. The lifting force measurements indicate little 'sticking' of the assemblies during removal suggesting that no significant bowing of the assemblies or development of corrosion products causing adherence to the cask floor occurred.

Table 1. Fuel Assembly Examination Sequence and Lift Force Measurements

Fuel Assembly ID	Inspection Sequence	Grapple + fuel weight (lb)	Force to start lifting (lb)	Fuel Assembly ID	Inspection Sequence	Grapple + fuel weight (lb)	Force to start lifting (lb)
V05	1*	1414.5	1436	V09	12	1412	1440
TO3	2	1421	1472	V12	13*	1403	1420
V27	3	1419	1457	V13	14	1417	1442
V04	4	1419	1439	T09	15	1410	1428
V14	5	1415	1446	T16	16	1413	1441
TO7	6	1419	1434	V24	17*	1410	1430
T12	7	1415	1442	V15	18*	1408	1433
V08	8	1412	1431	V25	19	1413	1426
V11	9	1407	1421	T13	20	1412	1427
V01	10	1415	1433	T11	21*	1410	1420
T08	11	1410	1431				

* Indicates a fuel assembly that was selected for closer examination

The assemblies were in a generally good condition, which had not changed since the 1985 inspection. The general visual survey revealed a dark gray oxide layer under ambient cell lights, and light tan by video. The inspection found no increase in the oxide layer thickness. There was no formation of a loose oxide scale or particles between the fuel rods of the grid spacers or on the bottom nozzles.

CASK TEMPERATURES

Approach

The spent fuel cladding must be protected against thermally activated processes by keeping the storage temperature within the limits postulated in the cask design. Spent fuel storage or handling systems must be designed with a heat-removal capacity without active cooling systems. However, the conditions in the second storage period (e.g., dry cask) will be less severe than in the first storage period (e.g., fuel pool) since the decay heat decreases with time. Therefore, the decreasing decay heat requires less heat removal capacity during the extended licensing period.

Internal temperature measurements were taken during the 1999 testing, but only to provide a general indication of temperatures inside the cask. This is because the thermocouple lance system used in 1985 was no longer available for the 1999 tests. Thus, the temperatures inside the cask had to be measured with the lid off.

The primary lid bolts were removed on September 7, 1999, and the primary lid was first removed on September 8, 1999. The cask lid was removed every workday morning at approximately 7:30 - 8:00 am, and by procedure was replaced nightly at the end of the day's activities (generally between 7:00 and 9:30 pm). Between September 8 and 29, 1999, when the cask internal temperatures were recorded, the lid had been removed generally 5 days per week, for 10 hours per day.

Internal temperatures were measured with a Type J thermocouple inserted into the control rod guide tube between approximately 4:00 and 5:00 pm. This means that there was at least 8 hours of convective cooling on the day that the temperatures were recorded, in addition to the gradual convective cooling achieved during the working days prior to that measurement. Therefore, the contents would have cooled considerably.

Observations

It was expected that the upper portions of the fuel assembly would exhibit the highest temperatures due to convection. To approximate the best position, the temperature in fuel assembly V05 (Figure 4, fuel tube A1) were quickly measured at three positions:

after 12 minutes,	0.6 m (2 ft.) below the top nozzle,	152.1° C (305.8° F);
after 5 minutes,	1.5 m (5 ft.) below the top nozzle,	<140° C (284° F); and
after 10 minutes,	0.3 m (1 ft.) below the top nozzle,	146.5° C (295.7° F)

Within 10 minutes or so, the temperatures equilibrated to within 0.1° C/min (0.2° F/min) rise; the temperature readings were recorded for at least the last five minutes of equilibration. The final readings at 10 minutes (12 mins for V05), measured approximately 0.6 m (2 ft.) beneath the top nozzle, are presented in Table 2

The hottest zone, in the hottest of the three measured assemblies, was approximately 155° C (311° F) at the end of 10 minutes equilibration, when the rate of rise was still 0.1° C/min (0.2° F/min). A plot of the data indicated that the temperature would eventually equilibrate between 155 and 160° C (311 and 320° F).

Table 2. Internal Temperatures

Assembly ID	Fuel Tube ID (location)	Temperature
V05	A1 (center of basket)	152.1° C (305.8° F)
T11	A4 (between center and outer tubes)	154.6° C (310.3° F)
T03	A7 (outer tube, at 0° mark on cask)	122.8° C (253.0° F)

It must be emphasized that these results pertain only to the conditions at the time of measurement and represent an estimation of the maximal temperature of the assembly. It was considered as satisfactory that the air temperatures were well below 200° C (392° F). However, with the cask lid in place, the final equilibrium temperature will be higher.

GAS SURVEY

One of the primary concerns of the study of the Castor V-21 cask was whether degradation of the spent fuel cladding due to the initial thermal testing or long term storage would lead to the release of gaseous fission products. In addition, it is important to maintain a low oxygen environment inside the cask to minimize oxidation of the cladding and spent fuel. Thus, the cask internal cavity was backfilled with helium after completion of the thermal testing in 1985.

In 1985, the cask cover gas was sampled several times during performance testing, to evaluate the integrity of the spent fuel rods and the cask lid seals.

In August 1999, a mass spectrometric analyses of the Castor-V/21 cask gas samples were performed. Radiochemical analyses were performed on approximately 10 std-cc of gas from each bomb. In addition, the analytical procedure followed also checked for the presence of Ne, Kr and Xe; measurable quantities were not detected. A separate scan for organic species was also run on each sample, none were detected.

It appears that no major leakage of air into the cask occurred between 1985 and 1999. It also appears that none of the fuel rods in the stored assemblies have leaked over the same time period.

CONCLUSION

A series of examinations in 1999 and early 2000 to investigate the integrity of the Castor V/21 cask were undertaken. There is no evidence of significant degradation of the Castor V/21 cask systems important to safety from the time of initial loading of the cask in 1985 up to the time of testing in 1999. Supporting evidence for this lack of significant degradation are (1) gas analyses show neither signs of air ingress into the container nor signs of cladding failure leading to fission product release, (2) visual examination of the cask lid O-rings suggest they were in adequate condition to maintain a seal; (3) there was no evidence of major crud

spallation from the fuel rod surfaces; and (4) all materials inside the cask, including the assemblies, appeared the same as they did in 1985.

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DISCLAIMER

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Research Supporting Implementation of Burnup Credit in the Criticality Safety Assessment of Transport and Storage Casks

C. V. Parks, I. C. Gauld, J. C. Wagner, B. L. Broadhead, M. D. DeHart
Oak Ridge National Laboratory

D. D. Ebert
U. S. Nuclear Regulatory Commission

Abstract

The Nuclear Regulatory Commission (NRC) Office of Regulatory Research (RES) has initiated a program to support effective implementation of burnup credit in the criticality safety assessment of transport and dry storage casks. The goal is to develop technical bases that can be used to provide criteria and guidance for use in licensing activities. The program is being conducted in a phased approach, with the initial focus on unresolved issues related to the use of actinide-only burnup credit in transport and dry storage casks designed for spent fuel from pressurized-water reactors (PWRs). The work will gradually expand to investigate credit for fission products in PWR casks and application to burnup credit for boiling-water-reactor (BWR) fuel. This summary will review the status of progress to date and will identify planned activities and priorities.

Introduction

In the past, criticality safety analyses for commercial light-water-reactor (LWR) spent fuel storage and transport canisters assumed the spent fuel to be fresh (unirradiated) fuel with uniform isotopic compositions corresponding to the maximum allowable enrichment. This "*fresh-fuel assumption*" provides a well-defined, bounding approach to the criticality safety analysis that eliminates all concerns related to the fuel operating history, and thus considerably simplifies the safety analysis. However, because this assumption ignores the decrease in reactivity as a result of irradiation, it is very conservative and can result in a significant reduction in spent nuclear fuel (SNF) capacity for a given package volume. *The concept of taking credit for the reduction in reactivity due to fuel burnup is commonly referred to as burnup credit.* Numerous publications have demonstrated that increases in SNF cask capacities from the use of burnup credit can enable a reduction in the number of casks and shipments, and thus have notable economic benefits.

The use of burnup credit in criticality safety analyses for away-from-reactor applications (transport and storage) necessitates that the reactor operating history and conditions experienced by the fuel be considered. In contrast to the fresh fuel assumption, the use of burnup credit requires additional validation of calculational methods used to predict the SNF nuclide compositions applied in the safety analyses. Studies performed in the United States (sponsored largely by the Department of Energy and the Electric Power Research Institute) and abroad (primarily France, the United Kingdom, and Japan) have provided a significantly advanced understanding of the issues and developing approaches for a safety evaluation. However, a consensus has not been reached on how to answer such questions as: What constitutes adequate validation per the guidance of the ANSI/ANS-8.1 standard for nuclear criticality safety outside reactors? How does one select the appropriate axial-burnup profile for the licensing analysis? How should

the variation/uncertainty in operating histories, fuel design, and SNF composition be quantified and incorporated in the safety analysis? The NRC is seeking to develop and document technical bases for criteria and guidance that will facilitate the review of licensing applications that use burnup credit. Such technical bases will allow the identification of areas where additional understanding or experimental information can enhance the safe and effective use of burnup credit.

The goal of the NRC/RES project directed at burnup credit is to develop technical bases and to provide recommendations on criteria and guidance for consideration by the licensing office (Office of Nuclear Material Safety and Safeguards, NMSS). The purpose of this paper is to review the progress of the NRC/RES efforts since its inception in the summer of 1999 and discuss ongoing and planned activities.

Current NRC Guidance on Burnup Credit

One of the initial activities of the NRC/RES project on burnup credit was to provide the NRC/NMSS Spent Fuel Project Office (SFPO) with confirmatory analyses and technical assistance to support the issuance of Revision 1 of the Interim Staff Guidance - 8 (ISG8),[1] which provides recommendations for the use of burnup credit with PWR spent fuel in transport and dry storage casks. A discussion of the technical considerations that helped form the development of ISG8 can be found in Ref. 2.

The recommendations within ISG8 limit the burnup credit to that available from actinide-only nuclides for SNF with an assembly-average burnup of 40 GWd/t or less and a cooling time of 5 years. The ISG8 recommendations allow spent fuel with burnup values greater than 40 GWd/t to be loaded in a cask, but burnup to only 40 GWd/t can be credited in the safety analysis. Initial enrichments up to 5.0 wt % ^{235}U are allowed but, for each 0.1 wt % increase above 4.0 wt %, the assigned burnup loading value must be 1 GWd/t higher than the credited burnup used in the safety analysis. This loading offset accounts for the lack of assay data for fuels with an initial enrichment greater than 4 wt %. In addition, assemblies with burnable absorbers are not allowed. The ISG8 recommends that the analysis methods used to predict the SNF isotopics and the neutron multiplication factor (k_{eff}) for the cask be validated against measured data. Potential uncertainties caused by a variation in reactor operating histories, a lack of measured data for validation, and a spatial variation of the burnup within the assembly (axial and horizontal) need to be quantified and/or bounded in the safety analysis. Further, ISG8 recommends the use of a measurement prior to or during the loading procedure to ensure that each assembly is within the loading specifications for the approved contents (e.g., a burnup measurement). The recommendations for a bounding approach and preshipment measurements are consistent with the international regulations for the transport of fissile material.

Although ISG8 does not recommend that credit be sought for the presence of fission products, it is recommended that applicants provide an estimate of the cask-specific reactivity margin provided by the fission products and actinides for which the computational methods cannot be adequately validated. It is recommended that the methods used for such estimations be verified against any available experimental data and/or computational benchmarks to demonstrate the performance of the applicant's methods in comparison with independent methods and analyses.

A key element of ISG8 is the recognition that the "staff will issue additional guidance and/or recommendations as information is obtained from its research program on burnup credit and as experience is gained through future licensing activities." No commercial LWR cask has been licensed for burnup credit in the United States. ISG8 represents an initial step towards regulatory guidance that enables

industry to effectively proceed with design and licensing of a burnup-credit cask. The goal of the research program is to provide information that can serve as a basis for decisions on potential future modifications to the ISG8 recommendations. Such future modifications should lead to enhanced usage of burnup-credit casks while maintaining an adequate margin of safety.

Overview of Current Research Efforts

With the criteria and guidance of ISG8 established by the licensing staff, the effort of the research project shifted to identifying work needed to develop expanded guidance relative to selected elements of ISG8, to implement software enhancements that can facilitate the use of computational methods in safety analyses, and to develop the technical basis for the NRC/SFPO to use in considering future revisions of ISG8. A baseline report[3] was developed to review the status of burnup credit and to provide a strawman prioritization for areas where additional guidance, information, and/or improved understanding were judged to be beneficial to the effective implementation of burnup credit in transport and dry storage casks. The prioritization considered input obtained at public workshops sponsored by the NRC and Nuclear Energy Institute (NEI).

As a result of the initial review and input from industry and licensing staff, the current focus areas for the NRC research program were established:

1. Development of a comprehensive reference report that uses current cask designs (rail and truck) to provide a consistent basis for demonstrating the magnitude of the various negative reactivity components as a function of burnup, initial enrichment, and cooling time.

2. Development of an automated process for coupling the depletion/decay analysis to the criticality analysis to support initial license reviews. Eventually the analysis tool will be released as a module of the SCALE code system.[4]

3. Development of a computational benchmark for a generic rail cask design to support the calibration of an applicant's estimation of fission product margin per ISG8 recommendations.

4. Development of criteria and guidance for the selection of an appropriate axial profile for use in the safety assessment.

5. Development of an initial recommendation and associated technical basis for potential near-term modifications to the ISG8 relative to the use of cooling times other than 5 years.

6. Development of an initial recommendation and associated technical basis for potential near-term modifications to the ISG8 relative to the use of burnup credit with PWR fuel containing burnable poison rods and/or integral burnable absorbers.

7. Investigation of the potential for modifying or removing the loading offset (the added burnup margin required for fuel with initial enrichments above 4.0 wt %) based on existing and potential experimental data.

8. Review and evaluation of existing and proposed experimental data to (a) demonstrate and rank the relevance of experiments for methods validation using quantitative criteria, (b) identify experimental

needs, and (c) assess technical bases for "certifying" a minimum reactivity margin accountable to fission products.

The first two of these eight areas are aimed at assisting the licensing staff in preparation for review of applications which use burnup credit in the safety analysis. Areas 3 and 4 are being developed to provide additional guidance to assist in effective implementation of the ISG8 recommendations. The remaining areas are directed at expanding the inventory of fuel that will be allowed in a burnup-credit cask. Progress in each of these eight areas will be discussed in the following sections.

Reference Report on Components of Negative Reactivity

A significant number of domestic and international studies have been performed to help understand the components that contribute to the negative reactivity available with burnup credit. However, most of these studies were not comprehensive in nature and a comparison between studies is often difficult due to different assumptions used in the analysis (e.g., nuclide sets, cask models, etc.) A study is being performed to provide the NRC with a comprehensive reference report that uses a consistent set of assumptions to demonstrate the components of negative reactivity as a function of initial enrichment, burnup, and cooling time. Two cask configurations have been utilized: a generic burnup-credit rail cask model with 32 PWR spent fuel assemblies (GBC-32), as shown in Fig. 1, and a truck cask model (not shown). The negative reactivity provided by various nuclide sets are considered. The goal of the report is to quantify the various contributors to the negative reactivity available in SNF and to provide explanations that will assist readers in understanding the physics associated with the various effects. As an example of the type of information to be presented and discussed in the report, consider the nuclide sets of Table 1 and the results shown in Figs. 2–4 for the GBC-32 cask configuration. These figures demonstrate the reactivity decrease from various nuclide data sets as a function of burnup and cooling time.

Burnup-Credit Analysis Sequence

The ISG8 highlights the need for applicants employing burnup credit in criticality safety assessments to account for the axial and horizontal variation of the burnup within a spent fuel assembly. In practice, the axial-burnup variation (i.e., the axial-burnup profile) is commonly modeled in a criticality calculation using a finite number of axial segments or zones (10 to 20 is typical) to represent the burnup profile, each zone having a uniform average burnup for that segment. Consequently, implementation of burnup credit using this approach requires separate fuel depletion calculations for each axial zone, and the subsequent application of these spent fuel compositions in the criticality safety analysis. Implementation of this approach therefore requires that numerous spent fuel depletion calculations must be performed, and potentially large amounts of data must be managed, converted, and transferred between the various codes.

To simplify this analysis process and assist the NRC staff in their review of criticality safety assessments of transport and storage casks that apply burnup credit, a new SCALE control sequence, STARBUCS (**S**tandardized **A**nalysis of **R**eactivity for **Bu**rnup **C**redit using **S**CALE) has been created. STARBUCS automates the generation of axially varying isotopic compositions in a spent fuel assembly, and applies the assembly compositions in a three-dimensional (3-D) Monte Carlo analysis of the assemblies in a cask environment.

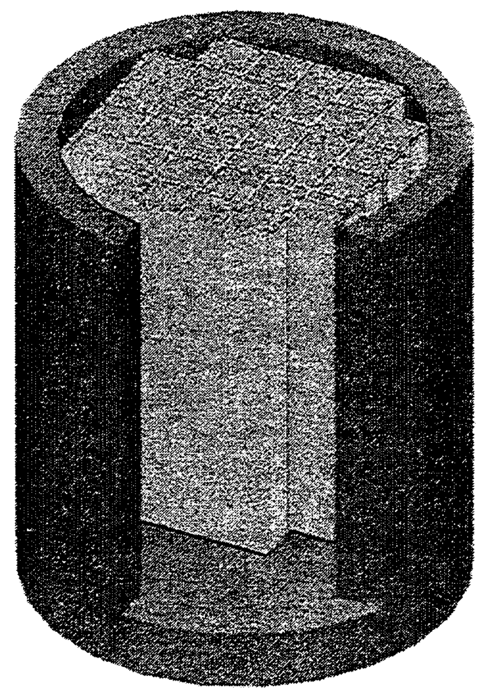

Figure 1. Cutaway view of GBC-32 burnup-credit cask model (one-half full height).

Table 1. Nuclide sets used for analysis

SET 1: Major actinides* (10 total)									
U-234	U-235	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Am-241	O[†]
SET 2: Minor actinides and major fission products (19 total)									
U-236	Am-243	Np-237	Mo-95[‡]	Tc-99	Ru-101[‡]	Rh-103[‡]	Ag-109[‡]	Cs-133	Sm-147
Sm-149	Sm-150	Sm-151	Sm-152	Nd-143	Nd-145	Eu-151[‡]	Eu-153	Gd-155	

*Actinides are consistent with those specified in the DOE Topical Report (Ref. 5).

†Oxygen is neither an actinide nor a fission product, but is included in this list because it is included in the calculations.

‡Nuclides for which measured chemical assay data are not currently available in the United States.

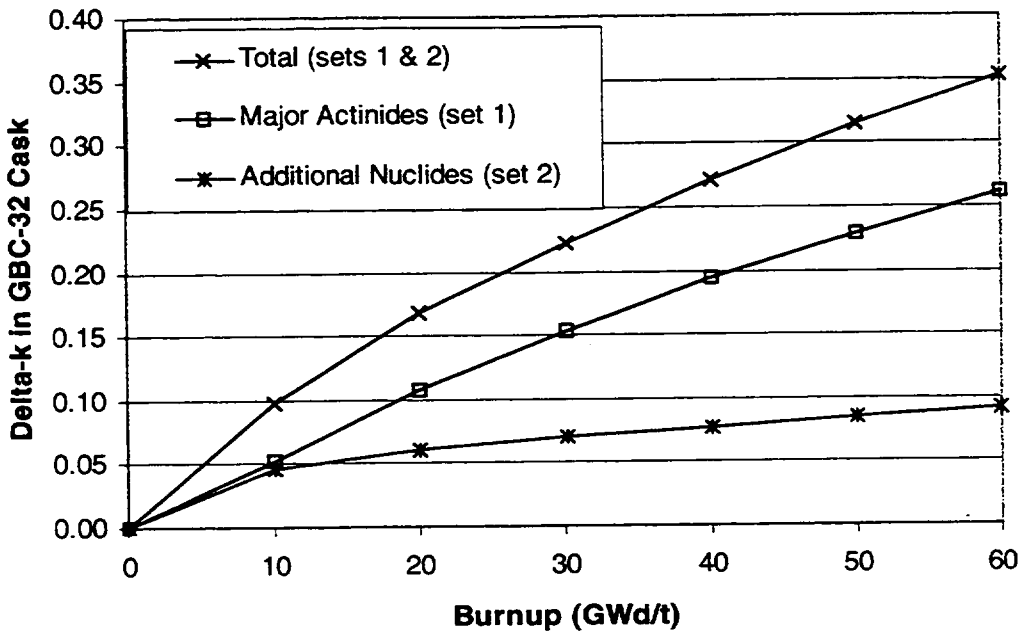

Figure 2. Δk values (relative to fresh fuel) in the GBC-32 cask as a function of burnup using the different nuclide sets and 5-year cooling time for fuel of 4.0-wt % ^{235}U initial enrichment.

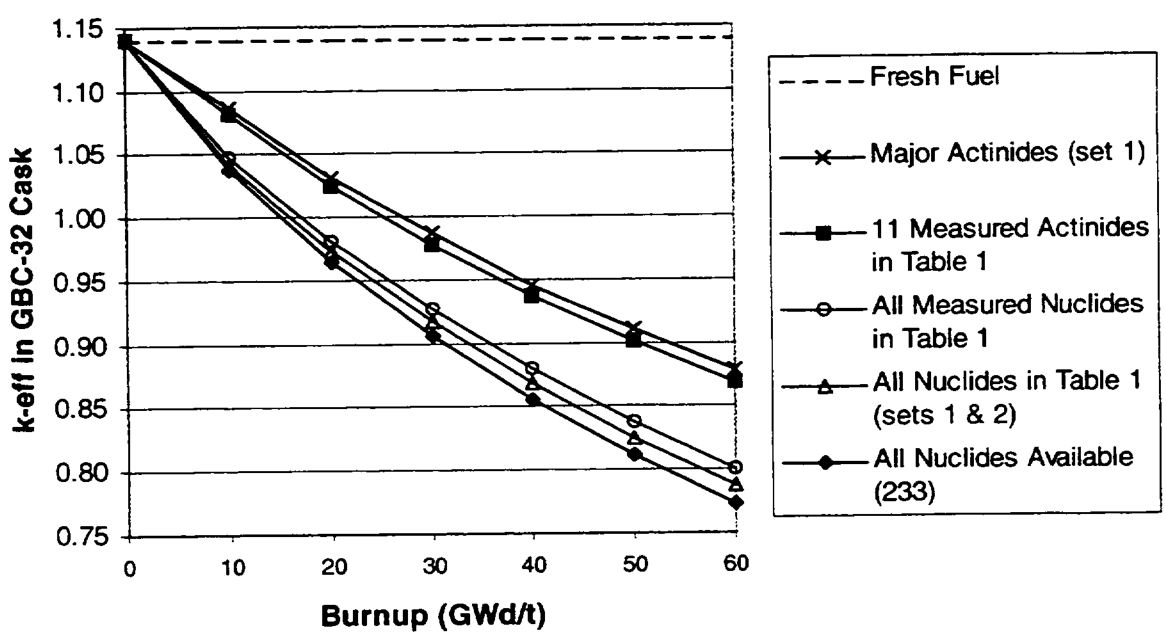

Figure 3. Values of k_{eff} in the GBC-32 cask as a function of burnup using various nuclide sets and 5-year cooling time for fuel of 4.0-wt % ^{235}U initial enrichment.

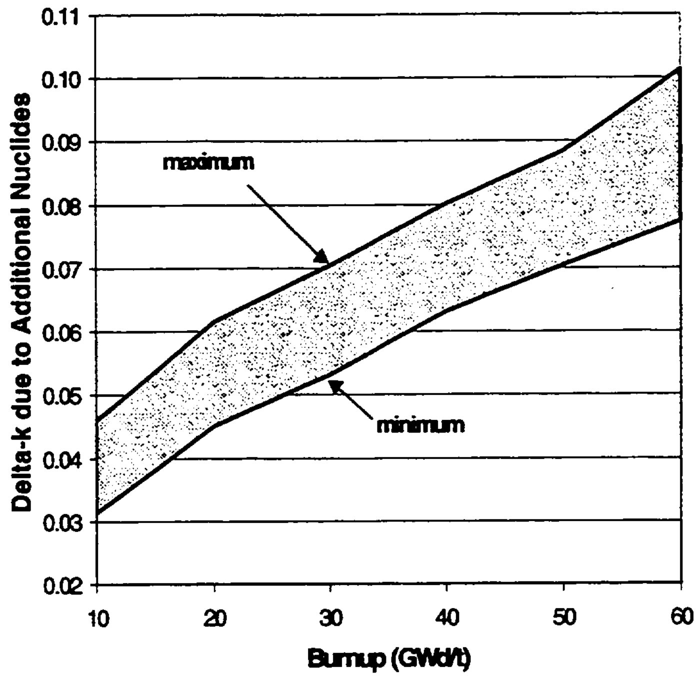

Figure 4. Range of Δk values in the GBC-32 cask due to the additional nuclides (set 2, defined in Table 1) as a function of burnup for all cooling times and initial enrichments considered (cooling times from 0 to 40 years; initial enrichments from 2.0 to 5.0 wt %).

The STARBUCS control sequence uses the new ORIGEN-ARP methodology of SCALE to perform automated and rapid-depletion calculations to generate spent fuel isotopic inventories in each axially varying burnup zone of a fuel assembly. The user need only specify the average assembly irradiation history, the axially varying burnup profile, the actinides and, optionally, the fission products that are to be credited in the criticality analysis. An arbitrary number of axial zones may be employed. The user may input a profile or select from several predefined profiles. This series of calculations is used to generate a comprehensive set of spent fuel compositions for each axial zone of the assembly. The STARBUCS sequence uses the SNF inventories provided for each zone to automatically prepare cross sections for the criticality analysis. A 3-D KENO V.a criticality calculation is performed using cask geometry specifications provided by the user.

Isotopic correction factors may also be applied to correct for known bias and/or uncertainty in the prediction of the isotopic concentrations. Development of STARBUCS is ongoing and will include the ability to model horizontal burnup effects and may include an automated source-starting routine for the Monte Carlo criticality safety calculation to help ensure proper source convergence within a reasonable number of neutron histories.

Computational Benchmark for Estimation of Additional Reactivity Margin

A recommendation of ISG8 suggests that an applicant estimate the design-specific reactivity margin provided by fission products and actinides that are excluded from the safety analysis. The applicant is encouraged to assess this estimated margin against estimated uncertainties and/or potential nonconservative approximations that are not readily quantified. The ISG8 points to the small amount of experimental data available and computational benchmarks as a means to verify the estimate of the additional reactivity margin. The NRC research program has worked to develop and document a computational benchmark problem based on the GBC-32 model of Figure 1. While preserving design features common to current storage and transport cask designs and deemed important to the neutronic analysis, the benchmark problem approximates (or eliminates) nonessential detail and is not constrained by proprietary information. Thus, the benchmark provides a reference configuration that applicants can readily use to evaluate their estimate of the reactivity margin from nuclides not included in their explicit validation. Version 4.4a of the SCALE code system has been used to provide reference estimations of additional reactivity margin as a function of initial enrichment, burnup, and cooling time. Although the reference solutions are not directly or indirectly based on experimental results, the SCALE 4.4a system is being used to analyze available assay data and proprietary reactivity worth experiments to obtain partial validation of this particular methodology.

Guidance for Selection of Axial Profile

The ISG8 recommends the use of analyses that provide an "adequate representation of the physics" and notes particular concern with the axial and horizontal variation of the burnup. The horizontal variation of burnup is a relatively minor effect which has been investigated only within the context of the development of Ref. 5. Future work of the research program will seek to further investigate the horizontal variation of the burnup. However, the axial burnup profile is an extremely important component of the safety analysis. The profile is dependent on the fuel assembly design, burnup, and the operating conditions of the reactor. Work sponsored by the DOE has provided a database[6] of more than 3000 PWR axial-burnup profiles, and studies[7] have identified the axial profiles that provide bounding k_{eff} values over selected burnup ranges and developed artificial bounding profiles over select burnup ranges. The database provides a large, but not exhaustive, set of profiles that hopefully represents a statistical sampling of typical and atypical profiles resulting from irradiation in U.S. PWR reactors. Figure 5 shows the spread of k_{inf} values that result from the set of profiles available from a selected burnup range, together with the bounding "real" (i.e., actual profile from the database) and "artificial" profiles. Although applicants will always have the flexibility to extend the existing database and/or create and use alternative databases, the current research program is seeking to initially develop a technical basis which demonstrates that bounding profiles developed from the existing publicly available database are adequate for use in burnup-credit safety analyses with actinide-only assumptions. The technical basis for this recommended position is currently being documented for review by NRC staff.

As evidenced from Fig. 5, the use of a bounding profile provides a considerable increase in reactivity over the predominant "typical" or average profiles. Future work will seek to use risk-informed insights to enable criteria for the development and use of an "average" profile. For example, if axial-profile measurements for each assembly were performed prior to loading, a profile deemed bounding of the "typical" profiles could be used in the safety analysis and the profile for the as-loaded assembly would be

checked for adherence. However, alternative approaches to allow the use of an average profile without such axial measurements are being investigated.

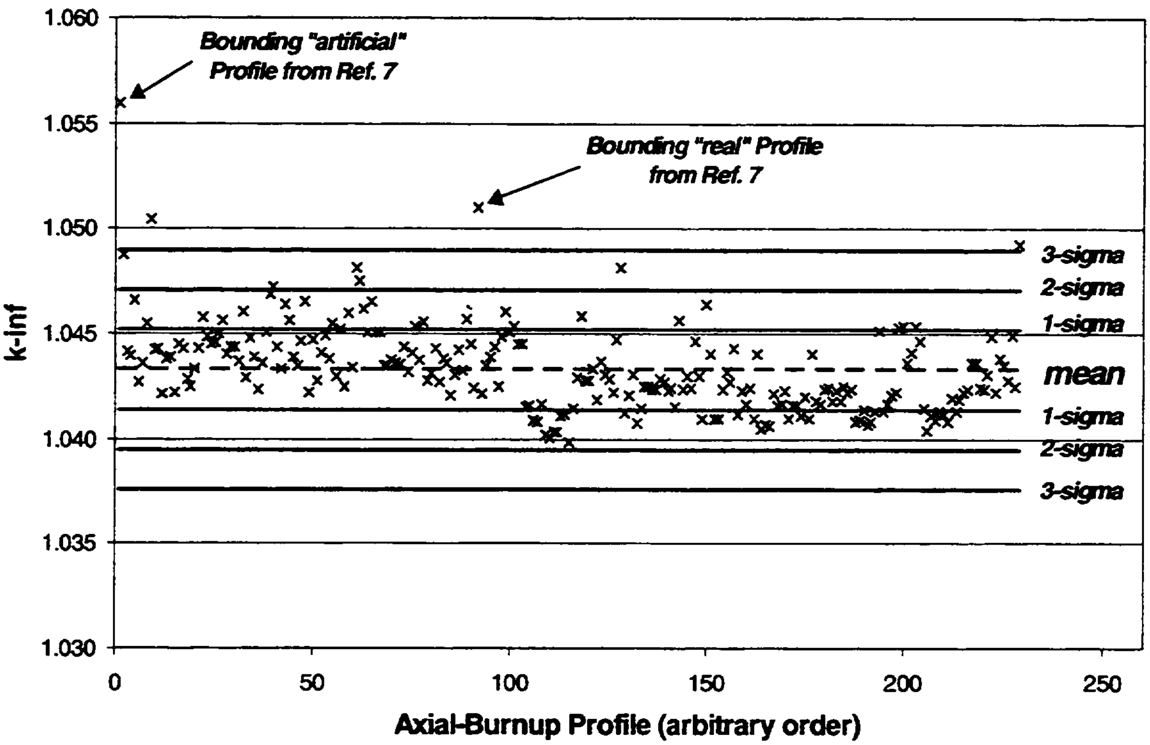

Figure 5. k_{inf} values based on database axial profiles for 38–42 GWd/t.

Modification of Cooling-Time Recommendation

The ISG8 recommends that a fixed cooling time of 5 years be used in the criticality safety analysis and that fuel with less cooling not be loaded in a burnup credit cask. In response to industry comments requesting more flexibility, work has been performed to demonstrate the effect of cooling time from discharge to 100,000 years and make recommendations relative to allowing additional cooling times. Figure 6 illustrates the general variation in k_{eff} as a function of cooling time for the GBC-32 cask loaded with SNF at 4.0-wt % initial enrichment and 40-GWd/t burnup. The "dip" at around 100 years is due to the decay of ^{241}Pu and the buildup of ^{241}Am and becomes less pronounced as the burnup decreases for a constant initial enrichment (i.e., under-burned fuel). For burnup-credit criticality safety analyses performed at 5 years, increasing cooling time results in an increasing conservative safety margin out to approximately 100 years. The magnitude of the conservatism depends on the initial enrichment and burnup of the fuel. Uncertainty associated with reactivity changes due to cooling time in the 1-to-100-year time period should be small because decay data important to the changes in this time period are known with very good accuracy.

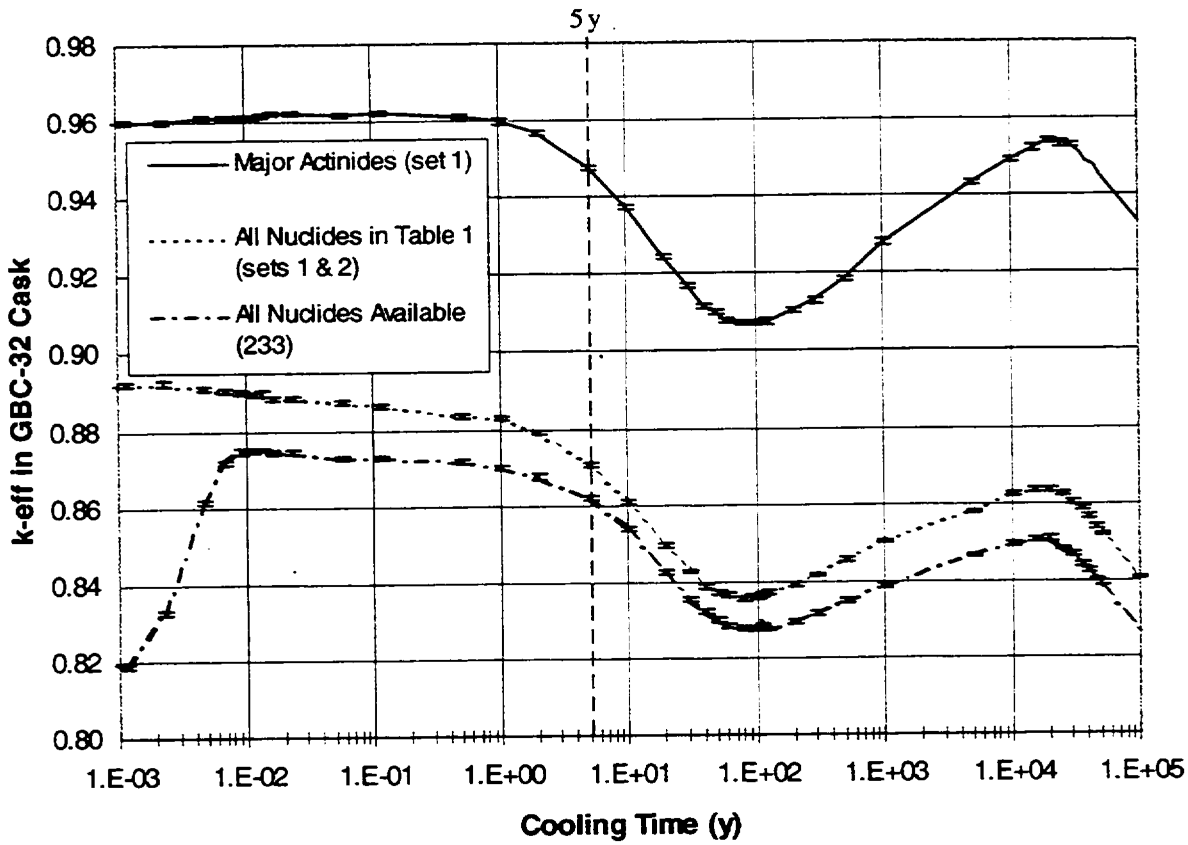

Figure 6. Values of k_{eff} in the GBC-32 cask as a function of cooling time for the three classifications of burnup credit (burnup-credit classifications are defined in Table 1). The results correspond to fuel with 4.0-wt % ^{235}U initial enrichment that has accumulated a 40-GWd/t burnup.

As evidenced by Fig. 7, there is an insignificant benefit in performing a safety analysis with cooling times greater than 50 years. A cooling time of 40 years provides a k_{eff} value that approximately equates to the k_{eff} value at 200-year cooling, which might be considered a practical lifetime for dry storage and transport casks. Thus, this rationale leads to a conclusion that cooling times up to 40 years can be assumed in developing the safety basis. However, if SNF loaded with an assumed cooling time of 40 years remains in the cask beyond the 200-year time frame, then the potential may exist for a reactivity increase beyond that allowed in the safety assessment. A study of the reactivity margin provided by the actinide-only assumption could be used to dispense with this concern. To address this concern and lay a consistent foundation that enables future extension beyond the actinide-only assumption, it has been suggested that a value of 10 years be assumed as the cooling time limit for safety analysis. The rationale is that, except for SNF that is highly under-burned (e.g., 5.0 wt %, 20 GWd/t), the best-estimate results for k_{eff} at a 10-year cooling time are always greater than the maximum k_{eff} in the secondary peak (10,000-to-30,000-year time frame). Finally, a lower limit on cooling time will continue to be dictated by thermal and shielding requirements.

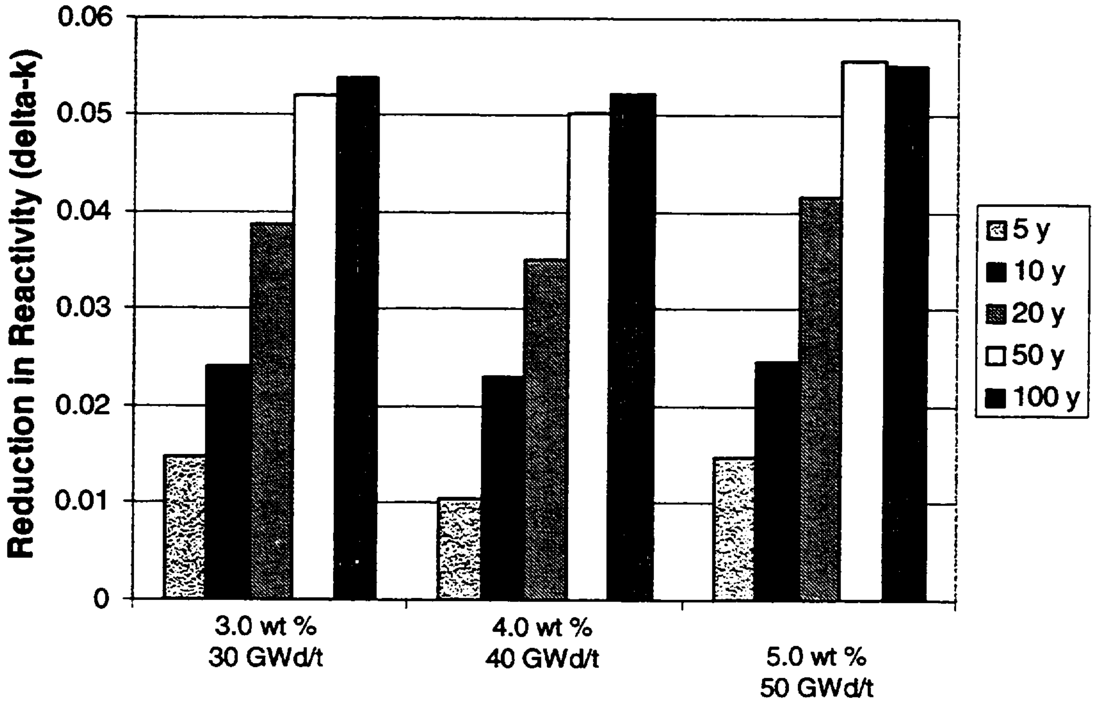

Figure 7. Reactivity reduction as a function of cooling time for some typical initial enrichment and discharge burnup combinations with actinide-only burnup credit.

Modifications to Allow SNF Exposed to Burnable Absorbers

The ISG8 restricts the use of burnup credit to assemblies that have not contained burnable absorbers during any part of their exposure. This restriction eliminates a large portion of the currently discharged spent fuel assemblies from cask loading, and thus severely limits the practical usefulness of burnup credit. Burnable absorbers may be classified into two distinct categories: (1) burnable poison rods (BPRs) and (2) integral burnable absorbers. BPRs are rods containing neutron-absorbing material that are inserted into the guide tubes of a PWR assembly during normal operation and are commonly used for reactivity control and enhanced fuel utilization. Due to the depletion of the neutron-absorbing material, BPRs are often (but not always) withdrawn after one-cycle residence in the core. In contrast to BPRs, integral burnable absorbers refer to burnable poisons that are a nonremovable or integral part of the fuel assembly. An example of an integral burnable absorber is the Westinghouse Integral Fuel Burnable Absorber (IFBA) rod, which has a coating of zirconium diboride (ZrB_2) on the fuel pellets.

The presence of BPRs during depletion hardens the neutron spectrum because of the removal of thermal neutrons by capture in ^{10}B and by displacement of moderator, resulting in lower ^{235}U depletion and higher production of fissile plutonium isotopes. Enhanced plutonium production and the concurrent diminished fission of ^{235}U due to increased plutonium fission have the effect of increasing the reactivity of the fuel at discharge and beyond. Consequently, an SNF assembly exposed to BPRs will have a higher reactivity for a given burnup than an assembly that has not been exposed to BPRs.

SCALE/SAS2H depletion calculations were performed assuming the BPRs were present during (1) the first cycle of irradiation, (2) the first two cycles of irradiation, and (3) the entire irradiation period (i.e., three cycles). For comparison purposes, isotopics were also calculated assuming no BPRs present. These four sets of isotopics were then used to determine the reactivity effect of each BPR design as a function of burnup for out-of-reactor conditions at burnup steps of 1 GWd/t and zero cooling time. The criticality calculations were based on an infinite array of spent fuel pin cells using isotopics from the various BPR depletion cases, and thus the effect of the BPRs is determined based on their effect on the depletion isotopics alone (i.e., the BPRs are not included in the criticality models).

Figure 8 plots the reactivity differences (Δk values relative to no-BPR depletion calculations) as a function of burnup using the actinides from Table 1. The isotopics used in the criticality calculations correspond to spent fuel with 4.0-wt % ^{235}U initial enrichment that has been exposed to Westinghouse Wet Annular Burnable Absorber (WABA) rods during depletion. For the purpose of the depletion calculations, three cycles of 15-GWd/t burnup per cycle were assumed. The results shown in Fig. 8 demonstrate that the reactivity effect increases with BPR exposure (burnup and number of BPRs present) and that calculations based on continuous exposure during the entire depletion yield higher (more conservative) reactivity than analyses based on actual/typical one-cycle exposures. For the same conditions plotted in Fig. 8, but with the inclusion of the major fission products, the reactivity behavior is very similar to that of the actinide-only condition.

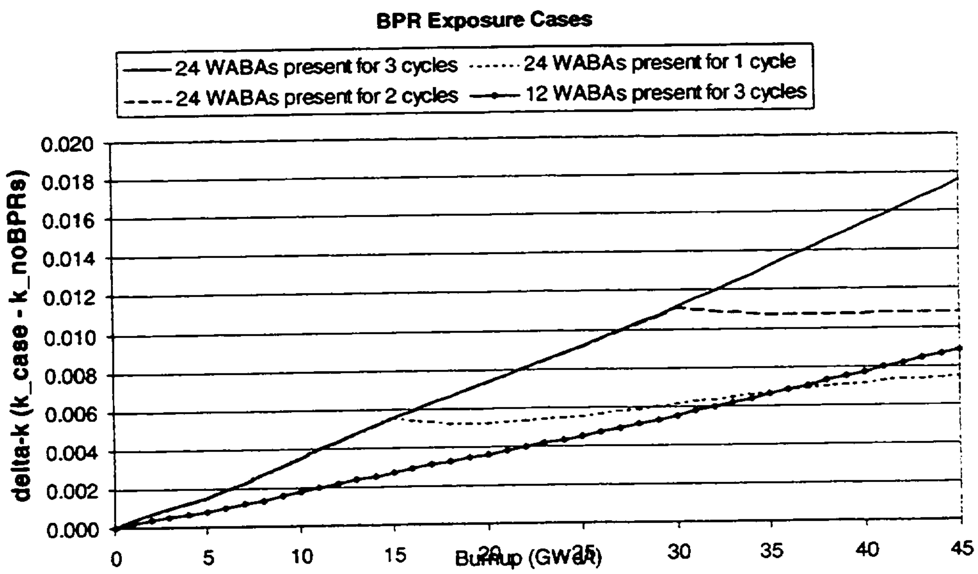

Figure 8. Reactivity differences (Δk values relative to the no-BPR condition) as a function of burnup for various BPR exposures using actinide-only assumption. Results correspond to fuel with 4.0-wt % ^{235}U initial enrichment that has been exposed to Westinghouse WABA rods (three cycles of 15-GWd/t burnup per cycle were assumed).

Analysis of the GBC-32 cask loaded with Westinghouse 17 × 17 OFA assemblies provides k_{eff} values for actinide-only and actinide + fission-product-burnup credit that demonstrate a BPR effect very similar to that exhibited for an infinite array of fuel pins. To determine the impact of incorporating the axial-burnup distribution, k_{eff} values were also calculated for the GBC-32 cask for various BPR exposures with the axial-burnup distribution included. The results reveal that the inclusion of the axial-burnup distribution reduces the reactivity increase associated with the BPRs. This reduction is due to the fact that the lower-burnup regions near the ends (that control the reactivity of the fuel when the axial-burnup distribution is included) have less burnup, and thus less-than-average burnup exposure to the BPRs.

A SAS2H fuel assembly model is limited to a one-dimensional radial model with a single smeared fuel region. Geometric modeling approximations are made in an effort to achieve a reasonable assembly-average neutron energy spectrum during the depletion process. However, the presence of BPRs challenges SAS2H modeling capabilities. Therefore, HELIOS, a two-dimensional, generalized-geometry transport theory code was utilized for selected cases to compare the reactivity differences (Δk values relative to the no BPR condition) as a function of burnup against those established using SAS2H. The results were comparable within a few tenths of a percent, with SAS2H isotopics predicting slightly larger reactivity effects. Further, very good agreement was achieved between k_{inf} values based on isotopics from the two methods.

The reactivity effect of BPRs increases nearly linearly with burnup and is dependent upon the number and poison loading of the rods and the initial fuel enrichment. Although variations are observed for the various BPR designs, maximum reactivity increases have been found to be ~1 to 3% when maximum BPR loading and exposure time are assumed for typical initial enrichment and discharge burnup combinations. Based on the analysis summarized here, guidance for an appropriate approach for calculating bounding spent nuclear fuel isotopic data for assemblies exposed to BPRs may be developed. For example, assuming maximum BPR exposure during depletion would be a simple, conservative approach to bound the reactivity effect of BPRs, where maximum BPR exposure may be defined as the maximum possible number of BPRs with the most bounding BPR design (i.e., most bounding geometric design and maximum possible poison loading) for the entire exposure. Other, less-conservative approaches that incorporate information regarding the percentage of assemblies exposed to mutiple cycles will be explored during the coming year.

A study has recently been completed that investigated the impact of integral burnable absorbers on the k_{eff} values in cask environments. Depending on the design and loading of neutron poison, the presence of integral burnable absorbers can slightly lower or raise the k_{eff} values of SNF assemblies, in comparison to assemblies without the integral burnable absorber. Integral burnable absorber analyses for multiple designs have been studied, and the maximum increase in k_{eff} is less than that identified for assemblies exposed to BPRs. The technical basis for including assemblies with integral burnable absorbers is currently being documented for review by NRC staff.

Modification or Removal of Loading Offset

The present experimental database of public domain actinide assay data consists largely of samples from older fuel assembly designs with enrichments below 3.5 wt %, and contains only one measurement for fuel above 3.4 wt % (a 3.89-wt % sample with a low burnup of 12 GWd/t). Only seven of the approximately

50 samples had BPRs present during irradiation. The enrichments and burnup ranges of the spent fuel samples used in recent validation studies performed for burnup-credit studies are shown in Fig. 9. The figure illustrates the paucity of experimental data in both the high-enrichment and high-burnup regimes. The loading offset of ISG8 provides a means of extending the usefulness of ISG8 to include spent fuel with initial enrichments above 4 wt %, using an engineering approach to compensate for potentially larger uncertainties. The loading offset, expressed in terms of the reactivity penalty Δk, is illustrated in Fig. 10 for the GBC-32 cask design employing actinide-only burnup credit for a 5-year cooling time. The reactivity margin for 5-wt % fuel, the maximum enrichment considered by ISG8, ranges from typically 0.035 to 0.045 Δk, depending on the fuel burnup.

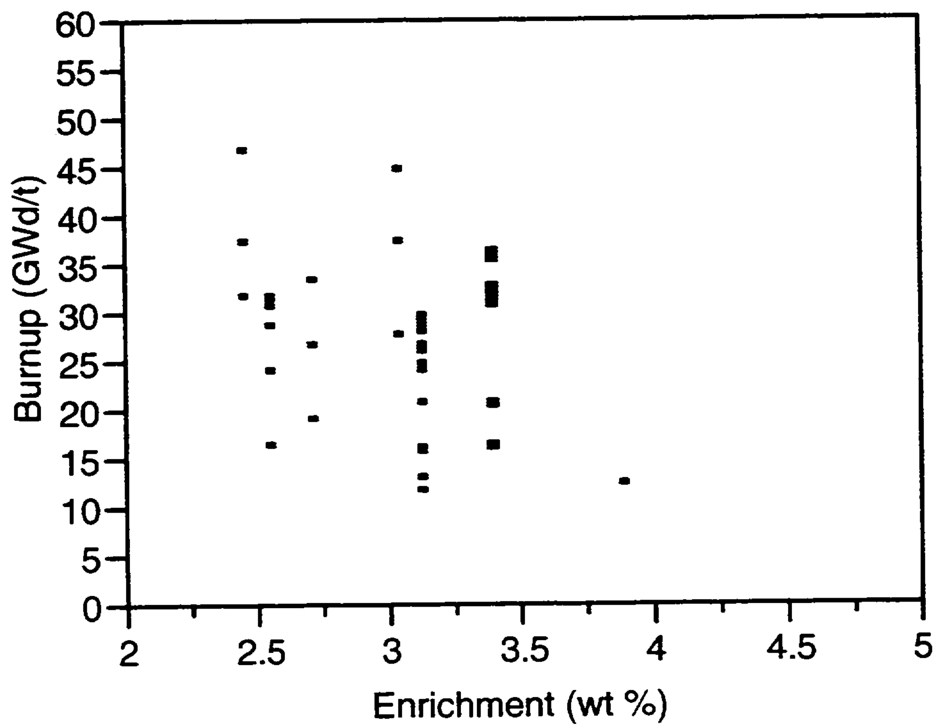

Figure 9. Enrichment and burnup of 46 PWR assay samples used in recent burnup-credit isotopic validation studies.

This added margin, shown in Fig. 10, can be compared with the actinide isotopic uncertainties for which it is intended to compensate as a means of estimating the conservatism in ISG8 with respect to existing isotopic assay data and spent fuel characterization methods. The influence of actinide uncertainties on the predicted k_{eff} of a spent fuel cask was estimated using isotopic correction factors derived from the publicly available experimental assay data obtained with the depletion analysis methods in SCALE and ENDF/B-5 cross-section data. The correction factors represent the amount by which the isotopic compositions must be adjusted to account for known calculational bias and uncertainty. This uncertainty is typically accounted for at a 95% confidence level and allows for the variance of the predicted bias and the number of assay measurements available. The influence of the uncertainties on the calculated k_{eff} was estimated

using sensitivity coefficients that have been generated for each of the important actinides over a wide range of enrichments and burnup values. The sensitivity coefficients represent the relative change in k_{eff} with respect to a 1% change in a nuclide concentration. Thus, the coefficients provide a method of predicting the change in k_{eff} with respect to a change in the isotopic concentrations required to account for the uncertainties in the concentrations.

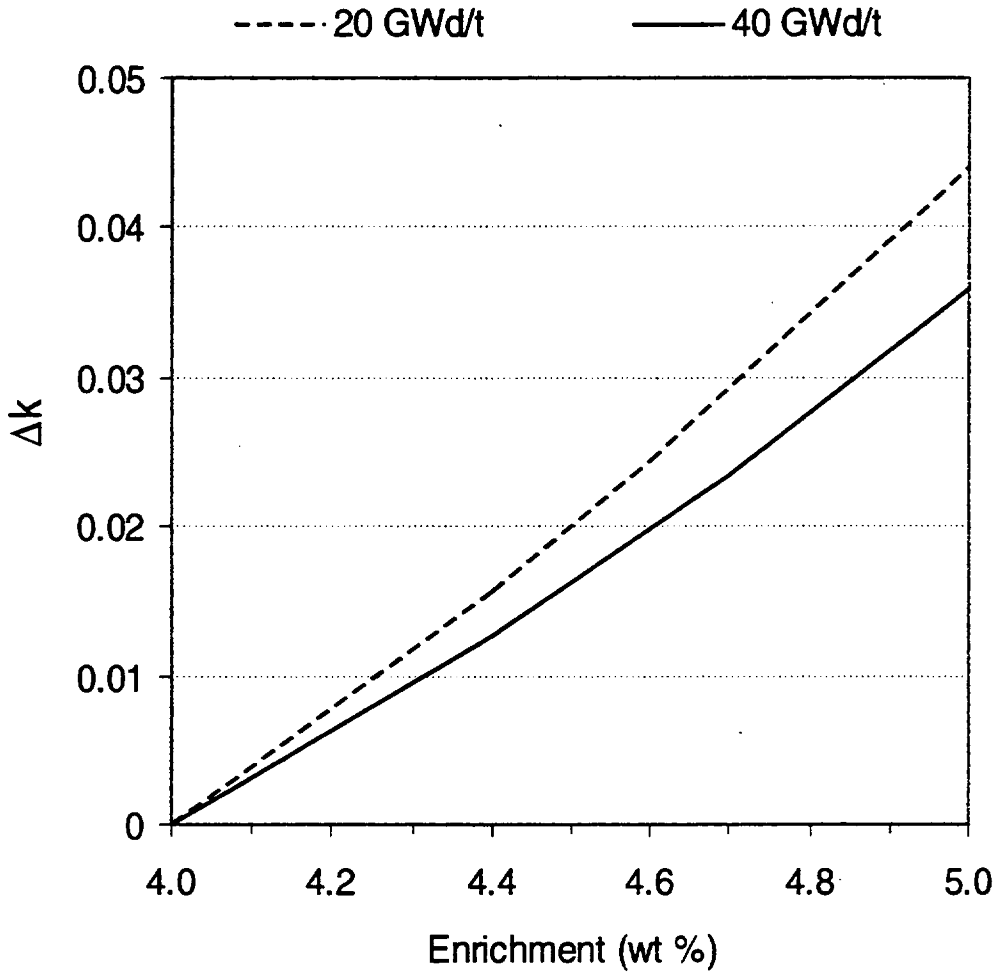

Figure 10. ISG8 loading offset reactivity for the GBC-32 cask design.

An important consideration is how to properly combine the uncertainties of the individual isotopes. The most conservative approach adjusts the concentration of every nuclide in such a way as to always create a more reactive system. Perhaps a more realistic strategy is to assume each uncertainty is independent (i.e., random) and combine the uncertainties using a root-mean-square (RMS) approach. However, the RMS method does not consider potentially correlated uncertainties in transmutation or decay chains. The actual effect is somewhere between these two approaches. Assuming the more conservative strategy, the net reactivity margin associated with the actinide uncertainties is illustrated in Fig. 11 for a range of

enrichments and burnup. The figure shows the increase in the reactivity margin associated with uncertainties in the concentration of the dominant burnup-credit actinides with increasing burnup. The changes in the margin reflect the changing actinide compositions with burnup and enrichment, the bias and uncertainty associated with each actinide, and the changing relative importance of each actinide to the system reactivity. As enrichment increases, the overall uncertainty exhibits a marginal decrease. For high-burnup fuel the combined reactivity change associated with all actinide isotopic uncertainties is about 4 to 5% $\Delta k/k$. If the actinide uncertainties are combined using a less-conservative RMS approach, the margin is reduced considerably to about 2% $\Delta k/k$. The reactivity margin due to the isotopic uncertainties is considerably larger than that due to the average bias.

Figure 11 inherently assumes that the isotopic uncertainties do not change with increasing enrichment. That is, the isotopic correction factors derived using the existing database of lower-enrichment and moderate-burnup fuel are assumed to be applicable in the extended regimes. The ISG8 loading offset above 4 wt % (see Fig. 10) amounts to about an added 4% $\Delta k/k$ penalty (assuming a neutron multiplication factor near unity) for an enrichment of 5 wt %, a reactivity margin similar to that associated with current actinide uncertainties. Therefore, the ISG8 loading offset penalty (corresponding to 5.0 wt %) is approximately equivalent to doubling the isotopic correction factors derived using existing isotopic assay data below roughly 3.5 wt % and 40 GWd/t.

A number of new sources of experimental assay data have been identified that could potentially be used to assess isotopic bias for the higher-enrichment and higher-burnup regimes in the near term. Some assay data may become available from an experiment performed on 3.8-wt % high-burnup fuel in Japan. Although this fuel does not extend beyond 4 wt %, it would significantly improve the coverage by existing data and provide improved confidence in code predictions above 3.5 wt %, thus providing a potential basis to extend the range of applicability above 4 wt % using bias trends. Additional information on the reactor operating conditions is currently being pursued to enable their accurate analysis.

The most attractive sources of existing higher-enrichment data that have been identified are the proprietary French programs, primarily the Gravelines-3 program involving 4.5-wt % fuel with a wide range of burnup. Acquisition of these data is currently viewed as a high priority within the NRC research project, particularly with the exclusion of some data sets from future consideration due to the use of nonstandard (reconstituted) assemblies. Published differences between French calculations and experiments[8] indicate no significant trends with burnup for the major burnup-credit actinides and, notably, the magnitude of the calculated isotopic biases for the 4.5-wt % fuel are comparable to the biases observed in benchmarks in the U.S. studies involving lower-enrichment fuels. However, the French results were obtained using cross-section data from the Joint European Files (JEF) of evaluated data and two-dimensional depletion analysis methods. Consequently, the reported biases may not be indicative of different code systems and data. Nevertheless, the results suggest that with up-to-date nuclear data and appropriately rigorous computational methods the burnup-credit actinides can be predicted in high-enrichment and high-burnup PWR fuel to a level of accuracy that is not significantly different than that for conventional enrichment and burnup fuel.

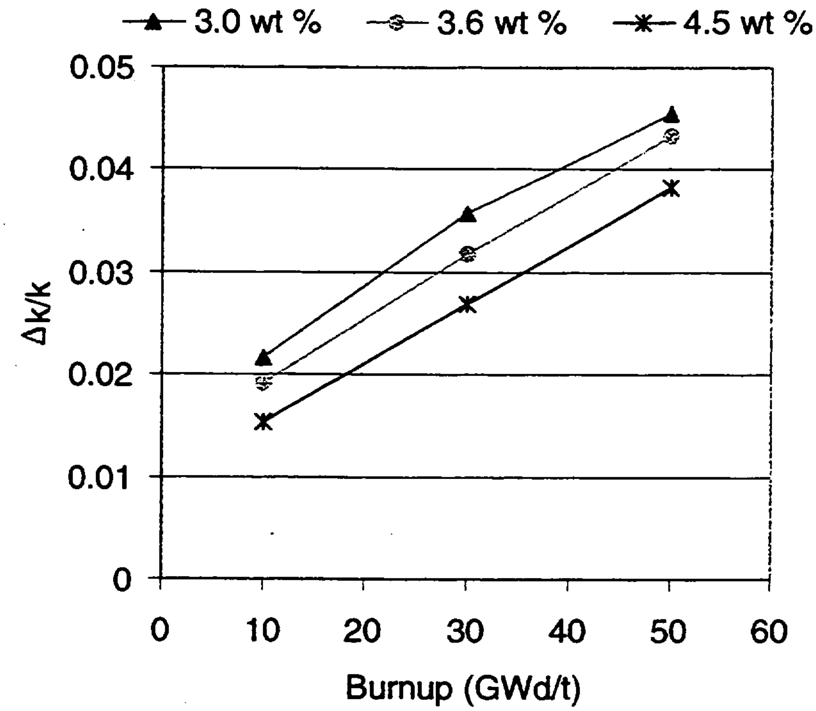

Figure 11. Reactivity penalty associated with actinide uncertainties as a function of burnup.

Isotopic analyses of spent PWR fuel will be performed as part of the REBUS program[9] and Phase II of the LWR-PROTEUS program.[10] These programs are proprietary. NRC is currently a participant in REBUS and will have access to the isotopic assay data. The projected schedule for reporting the commercial UO_2 radiochemical analysis results from REBUS is late 2002 to early 2003. The LWR-PROTEUS Phase II measurements are scheduled for completion in July 2001. The only new data likely to become available in the United States in the near term are the spent fuel samples from TMI-1 (4 and 4.65 wt %) currently being analyzed for the DOE repository program. However, release of the final results from this program are not expected before June 2001. The PWR enrichment and burnup regimes covered by these programs, the Japanese assay data, and the Gravelines-3 proprietary data are shown in Fig. 12, together with the existing publicly available assay data. A report providing a detailed review of available isotopic assay information and discussing the sensitivity methods being applied to investigate similarity and expected trends has been drafted and provided to NRC for review and comment.

Extending the area of applicability by making use of trends in the bias and uncertainty has proven to be challenging due to a relatively large variability in the existing data and the many factors that may influence the bias, including fuel enrichment, burnup, assembly design complexity, calculational methods, nuclear data, and uncertainties in reactor operating conditions, irradiation history, and sample burnup. A reliable trending assessment is challenged by the limited amount of experimental data and the large number of different parameters that can affect the bias.

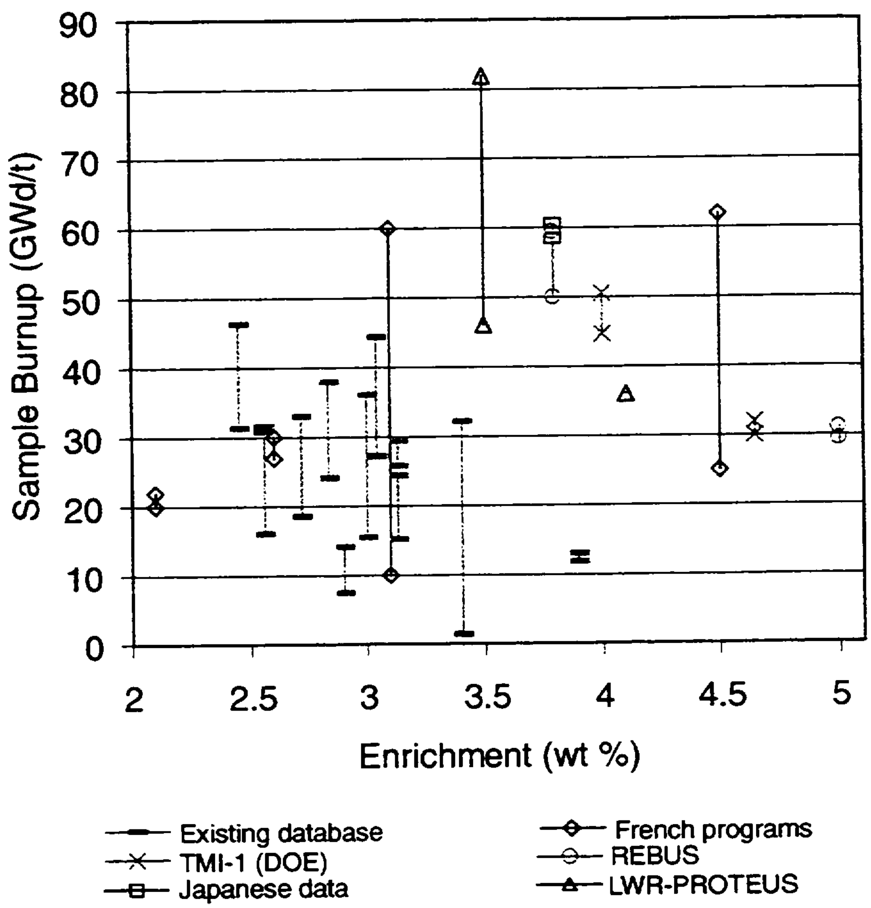

Figure 12. Available and potential future PWR isotopic assay data.

Several studies do suggest, however, that the effect of enrichment on the isotopic uncertainties should be minimal. The published French results for Gravelines spent fuel using French computational methods and JEF cross-section data indicate a level of agreement that is comparable to that of lower-enrichment fuel. In addition, sensitivity-based methods have been applied to assess the influence of nuclear data bias and uncertainties on the isotopic compositions and the k_{eff} of a spent fuel storage cask. These studies indicate that there is a strong correlation between spent fuel systems with a constant enrichment-to-burnup ratio. The results suggest that existing isotopic assay data may be highly applicable to regimes well beyond that of the data. However, there is currently insufficient experimental data to validate these findings. It is anticipated that as new assay data become available it will be possible to combine the limited amount of experimental data with the sensitivity-based methods to provide additional evidence to support predictions beyond the range where the majority of experimental data exist.

Evaluation and Use of Experimental Data

The nature of experimental data appropriate for use in the validation of burnup-credit analysis methodologies and the value and applicability of such data have been debated topics for over a decade. Available (albeit some are proprietary) experimental data include chemical assays of SNF inventories, critical experiments performed with fresh fuel in cask-like geometries, reactivity-worth measurements, subcritical experiments, and critical configurations in operating reactors. The potential value and limitations of each of these types of experiments were reviewed in Ref. 3.

A summary of chemical assay data was discussed earlier in the section on the loading offset. Reactor-critical configurations and some planned reactivity worth experiments with spent fuel are integral experiments that require the prediction of the nuclide composition and k_{eff} analysis. Thus, these experiments are also potential sources of experimental data that may be used to supplement, or potentially replace, the use of assay data. These options will continue to be considered as the research project proceeds.

Currently the NRC/RES is actively participating in the REBUS experimental program[9] which will provide chemical assay data and reactivity worths from insertion of SNF rods into a matrix of fresh fuel pins. The NRC is also discussing with the French the various avenues available for potential use of portions of their critical experiment and chemical assay data. To assist in this assessment, sensitivity/uncertainty (S/U) methods discussed in Ref. 11 are being used to provide information on the strengths and potential limitations of various types of experiments relative to validation needs for burnup credit. The S/U methodology utilizes two different parameters as measures of applicability: one is a global measure for system-to-system applicability (c_k value); the other is a nuclide-specific measure of applicability (T value). Existing fresh fuel (UO_2-fuel and mixed-oxide) critical experiments, reactor- critical configurations, reactivity worth experiments, and measured chemical assay data are being studied with the prototypic S/U methods.

The experimental programs evaluated using S/U methods include the French critical experiments with HTC pins,[12] which simulate the actinide concentrations of burned spent fuel (37 GWd/t) without the presence of fission products, and the Valduc[13] fission-product-solution experiments in which fission products are individually placed into solutions at the center of an experimental core. Other experiments that are to be analyzed include the worth experiments from the CERES/MINERVE[14] program, the planned ANL(NRAD) program,[15] the REBUS program[9] in Belgium, the PROTEUS program[10] in Switzerland, and the planned DOE-sponsored program[16] at Sandia National Laboratories.

The c_k values for the HTC experiments indicate a high degree of applicability to a series of infinite pin-cell calculations for burnups ranging from 10 to 60 GWd/t. The T values also indicate a high degree of applicability for the primary plutonium isotopes for burnups less than 60 GWd/t. Thus, these experiments are believed to be beneficial to actinide-only burnup-credit validation efforts.

The Valduc fission-product experiments are evaluated using only the nuclide-specific T parameters. This method is used because the system-to-system parameters are not currently appropriate for fission products due to the lack of uncertainty data on the fission-product cross sections. Also, an examination of the

T values is performed only for ^{149}Sm, because this is the only experiment available in the open literature. The T values obtained for the single Valduc experiment indicated that it is highly applicable to ^{149}Sm capture in the series of pin-cell applications for 10 to 60 GWd/t. This indicates that the fission-product-solution experiments should be good experiments for the validation of the fission products in a cask environment. These fission-product-solution experiments are valuable in that they allow for the effect of individual fission-product cross-section uncertainties on the system k_{eff} to be evaluated separately. This information is useful in combination with the additional data derived from the CERES-type measurements with doped-fission products, where the contribution of fission-product cross-section uncertainties to the worth of the fission product itself is obtained. The CERES-type measurements are very sensitive to the fission-product cross-section uncertainties; however, the Valduc-type experiments have fission- product cross-section sensitivities that are nearly the same as those in an actual cask environment. The details on the configurations for the REBUS, PROTEUS, and ANL(NRAD) are not currently available, and only approximate models have been considered. Further analyses are in progress.

Three PWR commercial reactor-critical state points have also been analyzed using the S/U methodology, and comparisons made with SNF cask environments. The results indicate that the reactor-critical state points have adequate similarity to cask environments. Reactor-critical configurations are the only measured information where significant quantities of SNF are used and, from an integral perspective, provide a viable source of validation information for both actinides and fission products.

A prudent approach to burnup-credit validation should involve assay-data validation, followed by cross-section validation for the actinides and fission products. The existing mixed-oxide fuel criticals, combined with French HTC experiments, are believed to be sufficient for actinide-only cross-section validation purposes. Additionally, applications that take credit for fission products need to consider experiments which validate individual fission-product cross sections. Validation is best accomplished by a combined approach of large-sample, individual fission-product worth measurements, like the Valduc or DOE/SNL experiments and the small-sample, individual doped-fission-product worth measurements like CERES/MINERVE. The remaining REBUS, PROTEUS, and ANL spent-fuel-sample worth experiments may be useful as an overall check on the reactivity effects of spent fuel. Although more complex to model, commercial-reactor-critical data provide a valuable source of experimental information for integral validation of the SNF compositions and cross sections and the effect of neutronic interaction between assemblies.

The estimation of the benefits of inclusion of fission products in the reactivity effects of burnup is a complicated process. An approach that has been offered is to quantify two independent factors to account for the effects of isotopic prediction inaccuracies and isotope cross-section inaccuracies. The product of these two factors and the predicted worth values in the cask configuration gives an estimate of the "guaranteed" fission-product worth in the cask application of interest. Efforts are underway to quantify these effects for an example application.

Expert Input and Review

The NRC research program is working to obtain input from domestic and international experts and organizations with experience in burnup-credit research, experiments, criticality safety practice, and operations of transport and dry storage casks. One primary tool for this input is the expert panel convened

to participate in a process of developing Phenomena Identification and Ranking Tables (PIRT). The main goal of the PIRT panel is to identify phenomena, parameters, procedures, etc., that influence the determination of k_{eff} for spent fuel in a cask environment, provide a graded (e.g., high-importance, moderate-importance, low-importance) ranking of the phenomena and, as appropriate, judge the uncertainty associated with each phenomena. Besides its primary objective, the PIRT process can also facilitate a beneficial exchange of information and ideas that will hopefully lead to improved understanding of the issues and practical approaches for effective implementation of burnup credit within the licensing process. Issues related to axial profile, cooling time, presence of BPRs, and loading offset have all been presented and discussed with the panel, and valuable insight and feedback has been obtained. The progress of the PIRT panel can be followed by reviewing the following web site: http://www.nrc.gov/RES/pirt/BUC.

Planned Activities

Several planned activities that relate to the work in progress have been noted in the previous sections. Another planned activity scheduled to begin soon is a parametric study of the impact of control rod insertions on the SNF isotopic inventory and subsequent k_{eff} values in a cask environment. Also, work to review preshipment measurement approaches has been initiated by the NRC staff.

A major focus over the next year is to investigate various approaches for increasing the allowed inventory of SNF that can be inserted in a burnup-credit cask design. Using the current recommendations of ISG8, a significant portion of the current and anticipated SNF would not be allowed in a cask designed for burnup credit. If the restriction on burnable absorbers is removed, the potential inventory that can be considered for loading in a burnup credit cask will be expanded. However, the loading curves (burnup vs initial enrichment) developed with the current recommendations would be such that a large portion of the SNF inventory would be eliminated because the burnup value would be too low for the specified initial enrichment. Efforts in the coming year will seek to study various risk-informed approaches that may reduce the conservatism associated with development of the loading curve (i.e., lower the required burnup value needed for a specific initial enrichment). For example, the use of typical or average axial profiles may be acceptable if it can be demonstrated that the impact of using bounding profiles for a portion (some realistic upper limit based on the probability for multiple assemblies with atypical profiles) of the loading does not present an unacceptable risk to safety. Another example would be the assumption that BPRs are only used for one cycle even though the bounding case would provide for their use for three cycles. To investigate such approaches extensively will require additional information from industry regarding the range of parameter values (e.g., soluble boron concentration, moderator temperature) seen in typical and atypical reactor operations. Such information could allow the use of statistical analyses to help determine appropriate "typical" conditions and help assess the probability of "outlier" conditions that would normally be the basis for bounding values. *The goal is to develop criteria and/or recommendations that are technically credible, practical, and cost effective while maintaining needed safety margins.*

Of course a major component that will lower the loading-curve profile is the inclusion of fission products. The current work to estimate a "certified" fission product margin using the CERES experiments and the SCALE code system needs to be expanded to investigate more general approaches that might provide acceptable means for taking fission-product credit.

All of the work discussed in this paper has focused on PWR spent fuel. Subsequent to completion of that work, it is anticipated that similar efforts will be pursued to develop technical bases and guidance for BWR spent fuel.

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