OCT 2 5 1989

TO: ALL LICENSEES OF OPERATING NUCLEAR POWER PLANTS AND HOLDERS OF CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS, AND INDIVIDUALS ON THE ATTACHED DISTRIBUTION LIST

SUBJECT: MINUTES OF THE PUBLIC MEETINGS ON GENERIC LETTER 89-04

In June 1989, the NRC staff held four public meetings to discuss Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs." This generic letter, issued on April 3, 1989, provides guidance aimed at improving inservice testing (IST) programs at nuclear power plants.

Attached for the use of licensees and construction permit holders in implementing the generic letter are the minutes of the public meetings. These minutes contain a summary of the opening remarks by NRC representatives at the meetings and responses to all of the questions raised at the four public meetings. Licensees and permit holders should review the entire package because specific staff guidance must be considered in the context of all questions and responses. These minutes are not intended to convey any new requirements and are not considered a backfit.

Please direct questions or comments regarding the meeting minutes to the appropriate NRC Project Manager.

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosure: As stated

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FMiraglia
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9001040128 891025
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Document Name: ALL LICENSEES

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NAME	:TScarbrough:rsc	:TSullivan	:LMarsh	:JERichardson	:FMiraglia	:JGPartlow
DATE	:10/18/89	:10/18/89	:10/19/89	:10/20/89	:10/24/89	:10/ /89

IDH:R-5
GENERIC
LTR

PMs for plants which are not listed on either Table of the GL should ensure that all plant specific TACs for review of IST programs issued prior to the issuance of the GL are closed out as noted in J. Hayes' May 17, 1989 memorandum. In addition, there were some inadvertent omissions in Tables 1 and 2 of the GL. ANO-2 and Dresden 2 & 3 should have been included as Table 1 plants and Catawba Units 1 & 2 should have been included as a Table 2 plant. TACS for these plants, therefore, do not appear on the attached list except for Dresden 2 & 3 which should be deleted.

Original Signed By:

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/enclosures:
J. Taylor
H. Thompson
J. H. Sniezek
Division Directors, NRR
Associate Directors, NRR
Project Directors, NRR
Regional Administrators
J. Conrad, CRGR
C. Berlinger, DOEA
S. Treby, OGC
NRR Technical Assistants

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NAME	:JHayes:sw	:EAdensam	:GLainas	:SVarga	:GHolahan	:JPartlow	:
DATE	:11/03/89	:11/03/89	:11/03/89	:11/06/89	:11/09/89	:11/13/89	

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LIST OF TAC NUMBERS FOR PLANTS NOT IN TABLES 1 OR 2

OF GENERIC LETTER 89-04.

0

1	2 DOCKET NO.	3 POWER PLANT	4 LEAD PROJECT MANAGER	5 RITS INIT
M 74756	50-313	Arkansas 1	Harbuck	NCH
	~~50-368~~	~~Arkansas 2~~	~~Poslusny~~	~~CPP~~
	~~50-334~~	~~Beaver Valley 1~~	~~Tam~~	~~PST~~
74757	50-412	Beaver Valley 2	Tam	PST
74758	50-438	Bellefonte 1	Auluck	RCA
	~~50-439~~	~~Bellefonte 2~~	~~Auluck~~	~~RCA~~
74759	50-155	Big Rock Point	Pulsifer	RPV
	~~50-456~~	~~Braidwood 1~~	~~Sands~~	~~SPS~~
	~~50-457~~	~~Braidwood 2~~	~~Sands~~	~~SPS~~
	~~50-259~~	~~Browns Ferry 1~~	~~Gears~~	~~GEG~~
	~~50-260~~	~~Browns Ferry 2~~	~~Gears~~	~~GEG~~
	~~50-296~~	~~Browns Ferry 3~~	~~Gears~~	~~GEG~~
	~~50-325~~	~~Brunswick 1~~	~~Tourigny/Le~~	~~EGT/NAL~~
	~~50-324~~	~~Brunswick 2~~	~~Tourigny/Le~~	~~EGT/NAL~~
	~~50-454~~	~~Byron 1~~	~~Olshan~~	~~LNO~~
	~~50-455~~	~~Byron 2~~	~~Olshan~~	~~LNO~~
	~~50-483~~	~~Callaway 1~~	~~Alexion~~	~~TWA~~
	~~50-317~~	~~Calvert Cliffs 1~~	~~McNeil~~	~~SWM~~
	~~50-318~~	~~Calvert Cliffs 2~~	~~McNeil~~	~~SWM~~
	~~50-413~~	~~Catawba 1~~	~~Jabbour~~	~~KNJ~~
	~~50-414~~	~~Catawba 2~~	~~Jabbour~~	~~KNJ~~
	~~50-470~~	~~CE-CESSAR-F~~	~~Kenyon~~	~~TBK~~
	~~00-675~~	~~CE-CESSAR-DC~~	~~Kenyon~~	~~TBK~~
	~~50-461~~	~~Clinton~~	~~Hickman~~	~~ZZY~~
	~~50-445~~	~~Comanche Peak 1~~	~~Malloy~~	~~VMM~~
	~~50-446~~	~~Comanche Peak 2~~	~~Malloy~~	~~VMM~~
	~~50-315~~	~~Cook 1~~	~~Gitter~~	~~IJG~~
	~~50-316~~	~~Cook 2~~	~~Gitter~~	~~IJG~~
M 74760	50-298	Cooper	O'Connor	PWO
M 74761	50-302	Crystal River 3	Silver	HAS
	~~50-346~~	~~Davis Besse~~	~~Wambach~~	~~TVW~~
	~~50-275~~	~~Diablo Canyon 1~~	~~Rood~~	~~HAR~~
	~~50-323~~	~~Diablo Canyon 2~~	~~Rood~~	~~HAR~~
~~74762~~	~~50-010~~	~~Dresden 1~~	~~Erickson~~	~~PBE~~
~~74763~~	~~50-237~~	~~Dresden 2~~	~~Siegel~~	~~XBS~~
~~74764~~	~~50-249~~	~~Dresden 3~~	~~Siegel~~	~~XBS~~
74765	50-331	Duane Arnold	Hall	JRH
	~~00-669~~	~~EPRI~~	~~Long~~	~~WAL~~
	~~50-348~~	~~Farley 1~~	~~Reeves~~	~~EAR~~
	~~50-364~~	~~Farley 2~~	~~Reeves~~	~~EAR~~
	~~50-016~~	~~Fermi 1~~	~~Erickson~~	~~PBE~~
	~~50-341~~	~~Fermi 2~~	~~Stang~~	~~SFJ~~
74766	50-333	Fitzpatrick	LaBarge	DWL
	~~50-285~~	~~Fort Calhoun 1~~	~~Milano~~	~~PZM~~
	~~50-267~~	~~Fort St. Vrain~~	~~Heitner~~	~~KLH~~
	~~50-605~~	~~GE-ABWR~~	~~Scaletti~~	~~DCS~~
74767	50-244	Ginna	Johnson	AGJ
74768	50-416	Grand Gulf 1	Kintner	LLK

2

1	2 DOCKET NO.	3 POWER PLANT	4 LEAD PROJECT MANAGER	5 RITS INIT
	~~50-417~~	~~Grand Gulf 2~~	~~Kintner~~	~~LLK~~
74769	50-213	Haddam Neck	Wang	ADW
	~~50-400~~	~~Harris 1~~	~~Becker~~	~~RBI~~
	~~50-321~~	~~Hatch 1~~	~~Crocker~~	~~LXC~~
	~~50-366~~	~~Hatch 2~~	~~Crocker~~	~~LXC~~
	~~50-354~~	~~Hope Creek~~	~~Shiraki~~	~~CSE~~
	~~50-133~~	~~Humboldt Bay~~	~~Erickson~~	~~PBE~~
	~~50-003~~	~~Indian Point 1~~	~~Erickson~~	~~PBE~~
74770	50-247	Indian Point 2	Brinkman	SBU
74771	50-286	Indian Point 3	Neighbors	JDN
	~~50-305~~	~~Kewaunee~~	~~Gody~~	~~AOG~~
	~~50-409~~	~~La Crosse~~	~~Erickson~~	~~PBE~~
74772	50-373	LaSalle 1	Shemanski	PCS
74773	50-374	LaSalle 2	Shemanski	PCS
	~~50-352~~	~~Limerick 1~~	~~Clark~~	~~RJC~~
	~~50-353~~	~~Limerick 2~~	~~Clark~~	~~RJC~~
74774	50-309	Maine Yankee	Sears	PMS
	~~50-369~~	~~McGuire 1~~	~~Hood~~	~~DSH~~
	~~50-370~~	~~McGuire 2~~	~~Hood~~	~~DSH~~
74775	50-245	Millstone 1	Boyle	MLB
	~~50-336~~	~~Millstone 2~~	~~Vissing~~	~~GSV~~
	~~50-423~~	~~Millstone 3~~	~~Jaffe~~	~~DHJ~~
74776	50-263	Monticello	Long	WAL
	~~50-220~~	~~Nine Mile Point 1~~	~~Slosson~~	~~MMS~~
	~~50-410~~	~~Nine Mile Point 2~~	~~Slosson~~	~~MMS~~
74777	50-338	North Anna 1	Engle	LBE
8	50-339	North Anna 2	Engle	LBE
9	50-269	Oconee 1	Wiens	LHW
74780	50-270	Oconee 2	Wiens	LHW
1	50-287	Oconee 3	Wiens	LHW
2	50-219	Oyster Creek	Dromerick	AID
3	50-255	Palisades	DeAgazio	ABD
	~~50-528~~	~~Palo Verde 1~~	~~Chan~~	~~TCC~~
	~~50-529~~	~~Palo Verde 2~~	~~Chan~~	~~TCC~~
	~~50-530~~	~~Palo Verde 3~~	~~Davis~~	~~MND~~
	~~50-171~~	~~Peach Bottom 1~~	~~Erickson~~	~~PBE~~
	~~50-277~~	~~Peach Bottom 2~~	~~Martin~~	~~REM~~
	~~50-278~~	~~Peach Bottom 3~~	~~Martin~~	~~REM~~
74784	50-440	Perry 1	Colburn	TGC
5	50-293	Pilgrim 1	McDonald	DGM
6	50-266	Point Beach 1	Swenson	WOS
74787	50-301	Point Beach 2	Swenson	WOS
	~~50-282~~	~~Prairie Island 1~~	~~DiIanni~~	~~DCD~~
	~~50-306~~	~~Prairie Island 2~~	~~DiIanni~~	~~DCD~~
74788	50-254	Quad Cities 1	Ross	TER
74789	50-265	Quad Cities 2	Ross	TER
	~~50-312~~	~~Rancho Seco~~	~~Kalman~~	~~GCK~~
	~~50-458~~	~~River Bend 1~~	~~Paulson~~	~~WAP~~
	~~50-261~~	~~Robinson 2~~	~~Lo~~	~~RHL~~
74790	50-272	Salem 1	Stone	JTF

1	2 DOCKET NO.	3 POWER PLANT	4 LEAD PROJECT MANAGER	5 RITS INIT
74791	50-311	Salem 2	Stone	JTF
74792	50-206	San Onofre 1	Trammell	CMT
	~~50-361~~	~~San Onofre 2~~	~~Hickman~~	~~DHZ~~
	~~50-362~~	~~San Onofre 3~~	~~Hickman~~	~~DHZ~~
	~~50-443~~	~~Seabrook 1~~	~~Nerses~~	~~VXN~~
	~~50-444~~	~~Seabrook 2~~	~~Nerses~~	~~VXN~~
	~~50-327~~	~~Sequoyah 1~~	~~Donohew~~	~~JND~~
	~~50-328~~	~~Sequoyah 2~~	~~Donohew~~	~~JND~~
74793	50-322	Shoreham	Brown	SWB
	~~50-498~~	~~South Texas 1~~	~~Dick~~	~~GFD~~
	~~50-499~~	~~South Texas 2~~	~~Dick~~	~~GFD~~
74794	50-335	St. Lucie 1	Norris	JAN
	~~50-389~~	~~St. Lucie 2~~	~~Norris~~	~~JAN~~
	~~50-395~~	~~Summer 1~~	~~Hayes~~	~~JGH~~
	~~50-280~~	~~Surry 1~~	~~Buckley~~	~~BCB~~
	~~50-281~~	~~Surry 2~~	~~Buckley~~	~~BCB~~
74795	50-387	Susquehanna 1	Thadani	MBT
74796	50-388	Susquehanna 2	Thadani	MBT
	~~50-289~~	~~Three Mile Island 1~~	~~Hernan~~	~~RHH~~
	~~50-320~~	~~Three Mile Island 2~~	~~Masnik~~	~~MTM~~
74797	50-344	Trojan	Bevan	RBB
74798	50-250	Turkey Point 3	Edison	GEE
74799	50-251	Turkey Point 4	Edison	GEE
74800	50-271	Vermont Yankee	Fairtile	MBF
	~~50-424~~	~~Vogtle 1~~	~~Hopkins~~	~~JSH~~
	~~50-425~~	~~Vogtle 2~~	~~Hopkins~~	~~JSH~~
	~~50-382~~	~~Waterford~~	~~Wigginton~~	~~DXW~~
74801	50-390	Watts Bar 1	Auluck	RCA
	~~50-391~~	~~Watts Bar 2~~	~~Auluck~~	~~RCA~~
	~~50-601~~	~~RESAR SP/90~~	~~Kenyon~~	~~TBK~~
	~~50-460~~	~~WNP 1~~	~~Kenyon~~	~~TBK~~
	~~50-397~~	~~WNP 2~~	~~Samworth~~	~~RBS~~
	~~50-508~~	~~WNP 3~~	~~Kenyon~~	~~TBK~~
	~~50-482~~	~~Wolf Creek~~	~~Pickett~~	~~DLP~~
74802	50-029	Yankee Rowe	Fairtile	MBF
	~~50-295~~	~~Zion 1~~	~~Patel~~	~~CBP~~
	~~50-304~~	~~Zion 2~~	~~Patel~~	~~CBP~~
	~~50-018~~	~~VBWR~~	~~Erickson~~	~~PBE~~

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

OCT 25 1989

TO: ALL LICENSEES OF OPERATING NUCLEAR POWER PLANTS AND HOLDERS OF CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS, AND INDIVIDUALS ON THE ATTACHED DISTRIBUTION LIST

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Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosure: As stated

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C. E. Rossi	11 E 4	
O. O. Rothberg	NLS 302	
H. K. Shaw	9 H 8	Region V:
J. Sniezek	12 G 18	C. A. Clark
E. J. Sullivan	9 H 2	R. J. Pate

Region I
6/5/89

NAME	ORGANIZATION
Paul Cervenka	GPUN
Jim Connolly	PSNH/NHY
Bill Kittle	PSE&G-Salem
Cornelius Coddington	PP&L
Jeffrey Lomm	NYPA
Deborah K. Schultz	PSE&G
John Rigert	LILCO
Bob Knight	GPUN
Shafi Rokerya	NYPA
Noah Fetherston	Yankee Atomic
Safio Toth	NYPA
Patrick Sheldon	Yankee Atomic
Francis Kamiski	PSE & G
Joann West	Beaver Valley
Clive Callaway	NUMARC
John T. Lindberg	PP&L
Douglas B. Ritter	PP&L
Eugene Perry	Con Edison
Jeff Neyhard	Niagara Mohawk
Joan F. Etzweiler	Con Edison
J.R. Bashista	TMI-1
Albert A. Koehl	NES
V. C. Ruppert	PECC
R. Binz IV	PSE&G
D. Wallace	NYPA Fitzpatrick
K. Woodard	NYPA Fitzpatrick
R. Haladyna	BECO
J. L. Sabina	BECO
W. G. Carroll	BECO

Region II
6/8/89

NAME	ORGANIZATION
W. E. Galbreath	Duke Power
J. A. Witherspoon	Duke Power
John Zeiler	USNRC
Paul Burnett	USNRC/DRS/TPS
J. J. Lenahan	NRC
M. Belford	Southern Company
Steve A. Saunders	SERI Grand Gulf
J. S. Jackson	TVA
C. L. Dunkerly	Baltimore Gas & Electric
J. M. Duke	TSE
Eben Burns	BCP Techical Services, Inc.
Philip J. North	Duke Power Company
Robin Dyle	Southern Company Services, Inc.
W. E. Campbell, Jr.	USNRC/RES
Gary Smith	System Energy
Wavel Justice	System Energy
Stephen E. Mohn	Florida Power & Light
A. L. Koon	South Carolina Electric & Gas
Gene G. Sowlt	SCE&G V. C. Summer Station
Ken Kmetz	Enercon Services
John Zudans	Florida Power & Light
A. Ronald Jacobstein	
Art Caudill	Georgia Power Company
Herbert P. Walker	Georgia Power Company
Sid Burns	Alabama Power Company
Bud Syx	Georgia Power Company
Kris Miller	Florida Power & Light
Jim Holton	Florida Power Corporation
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Stan Pruitt	Carolina Power & Light
Al Schneider	Enercon Services
John Kin	Virginia Power
Peter Taylor	NRC
Arthur Szczepaniec	NRC
Karl Jacobs	N.Y.P.A.
John B. Lee, Jr.	Virginia Power
S. L. Nader	Duke Power Company
John J. Hayes, Jr.	NRC

Region III
6/13/89

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Bruce J. Sheffel	Detroit Edison Company
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Paul Shemanski	NRC
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Laurence Attochman	NUTECH Engineers
Dale L. Jones	NUTECH Engineers
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Timothy P. Jaeger	Combustion Engineering
John Ozol	Commonwealth Edison
Steve Sovich	Duquesne Light Company
Dave Jones	Duquesne Light Company
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Norm Peterson	Iowa Electric Light & Power
Mark Harris	IMPELL
Patrick M. Finnemore	Wisconsin Public Service Corp.
Gordon Svendsen	Commonwealth Edison - Zion Station
Jeff Grzeszczak	Commonwealth Edison - Braidwood Station
Gary Bal	Commonwealth Edison - Braidwood Station
Pat Tobin	Northern States Power
Doug Kerr	NUTECH Engineers
Mark Horbaczewski	Dresden Station
Vince Treagne	Point Beach Nuclear Plant
Brent Metrow	Illinois Dept. of Nuclear Safety
Lawrence Sage	Illinois Dept. of Nuclear Safety
Steven M. Hutton	Energy Testing Services
Bill Carroll	Pilgrim Station
Joseph L. Sabina	Pilgrim Station
John C. Rivers	Cleveland Electric Illuminating Company
Vince Concel	Perry Power Plant
Stephen Forsha	Impell Corporation
Stephen P. Brown	NUTECH Engineers
Dennis Carlson	Northern States Power
Frank Dunder	Commonwealth Edison - Dresden Station
Russ Tamminga	Commonwealth Edison
A. John Birkle	Consumer Power Company - Big Rock Point
Jeff Cook	Omaha Public Power District
George Schrader	Consumers Power Company
David Kanuch	Impell Corporation - Dresden Station
Steve Bell	Illinois Power

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6/15/89

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Ken Trippel	Houston Lighting & Power - South Texas Project
Steve Wideman	Wolf Creek Nuclear Operating Corporation
Alan Harris	Waterford 3
Bruce Wadley	TU Electric - Commanche Peak
Don Ringle	TU Electric
Clifford Clark	NRC
John Arhar	Pacific Gas & Electric - Diablo Canyon
Terry Pellisero	Pacific Gas & Electric Co. - Diablo Canyon
Steve L. Scammon	Supply System
Ali Abbasi	Southern California Edison
John DeBonis	Stone & Webster (c/o TU Electric)
Paul Croy	So. Cal Ed.
Don Hickman	NRC
Rocky Schultz	Cooper Nuclear Station
Wayne Walling	Gulf States Utility - River Bend
Steve Asztalos	Cygna Energy Services

PROJECT MANAGERS

Plant	Project Manager	Ext. No.	Backup Manager	Ext. No.
Arkansas 1/2	Craig Harbuck	21341	Chet Poslusny	21336
Beaver Valley 1/2	Peter Tam	21307	Frank Orr	21321
Bellefonte 1/2 TVA	Raj Auluck	20759		
Big Rock Point	Robert Pulsifer	21330	Lynn Kelly	21305
Braidwood 1/2	Stephen Sands	21396	Leonard Olshan	23018
Browns Ferry (TWA)	Gerry Gears	20767		
Brunswich 1/2	Edmond Tourigny	21474	Tommy Le/Les Kintner	21455
Byron	Leonard Olshan	23018	Stephen Sands	21396
Callaway	Tom Alexion	21387	Randy Hall	21391
Calvert Cliffs 1/2	Scott McNeil	21438	David LaBarge	21421
Catawba 1/2	Kahtan Jabbour	21496	Darl Hood	21442
CECessar			Tom Kenyon	21120
Clinton	John Hickman	23101	Paul Shemanski	23017
Comanche Peak (SP)	Melinda Malloy	20439		
Cook 1/2	John Stang	21328	Tony Gody	21305
Cooper	Paul O'Connor	23026	Doug Pickett	21336
Crystal River 3	Harley Silver	21470	George Wunder	21480
DavisBesse	Tome Wambach	21323	Joseph Giiter	21379
Diablo Canyon 1/2	Harry Rood	21352	Roby Bevan	21361
Cresden 2/3	Byron Siegel	23019	Thierry Ross	23016
Duane Arnold	Randy Hall	21391	Tom Wambach	21323
EPRI	Paul Leech	21103	Dino Scaletti	21104
Farley 1/2	Edward Reeves	21457	Jack Hayes	21456
Fermi 2	John Stang	21328	Lynn Kelly	21305
Fitzpatrick	David LaBarge	21421	Scott McNeil	21438
Fort Calhourn	Patrick Milano	21347		
Ft. St. Vrain	Keneth Heitner	21333	Edward Tomlinson	23024
GEABWR	Dino Scaletti	21104	Paul Leech	21103
Ginna	Carl Stahle	21435	Patrick Sears	21429
Grand Gulf 1	Les Kintner	21458	Edmond Tourigny	21474
Haddam Neck	Alan Wang	21313	Michael Boyle	21308
Harris 1	Dick Becker	21465	Edward Reeves	21457
Hatch 1/2	Lawrence Crocker	23049	Jon Hopkins	21494
Hope Creek	Clyde Shiraki	21445	Stu Brown	21444
Indian Point 2	Don Brinkman	21420	Don Neighbors	21409
Indian Point 3	Don Neighbors	21409	Don Brinkman	21420
Kewaunee	Joseph Giitter	21390	Tom Alexion	21389

Plant	Project Manager	Ext.No.	Backup Manager	Ext. No.
LaSalle	Paul Shemanski	23017	John Hickman	23101
Limerick 1/2	Richard Clark	23041	Robert Martin	21426
Maine Yankee	Patrick Sears	21437	Carl Stahle	21435
McGuire 1/2	Dearl Hood	21442	Kahtan Jabbour	21496
Millstone 1	Michael Boyle	21308	Alan Wang	21313
Millstone 2	Guy Vissing	21314	David Jaffee	21312
Millstone 3	David Jaffe	21312	Guy Vissing	21314
Monticello	John Stefano	21309	Dom DiIanni	21324
Nine Mile Point 1/2	Marylee Slossom	21412	Robert Benedict	21402
North Anna 1/2	Leon Engle	21484	Bart Buckley	21452
Oconee 1/2/3	Leon Engle	21404	Kahtan Jabbour	21496
Oyster Creek	Alexander Dromerick	21301	Ronald Herman	21320
Palisades	Al DeAgazio	23063		
Palo Verde 1/2	Terence Chan	21366	Micheal Davis	21368
Palo Verde 3	Micheal Davis	21368	Terence Chan	21366
Peach Bottom 2/3	Robert Martin	21426	Richard Clark	23041
Perry	Timothy Colburn	21389	David Lynch	23023
Pilgrim 1	Daniel McDonald	21436	Vernon Rooney	21440
Point Beach 1/2	Warren Swenson	21386	Timothy Colburn	21369
Prairie Island 1/2	Albert DeAgaqio	21323	Theodore Quay	21315
Quad Cites 1/2	Thierry Ross	23016	Byron Siegel	23019
Rancho Seco	George Halman	21367	Steve Reynolds	21366
River Bend 1/2	Walter Paulson	23028	David Wigginton	23027
Robinson 2	Ronnie Lo	21463	Les Kintner	21458
Salem 1/2	Jim Stone	21422	Mohan Thadani	21427
San Onofre 1	Charles Trammell	21363	Donald Hickman	21380
San Onofre 2/3	Donald Hickman	21380	Harry Rood	23062
Seabrook 1/2	Victor Nerses	21441	Norton Fairtile	21443
Sequoyah (TVA)	Thomas Rotella	20760		
Shoreham	Stewart Brown	21444	Clyde Shiraki	21445
South Texas	George Dick	21326	Anthony Bournia	21345
St. Lucie 1/2	Jan Norris	21483	Gordon Edison	21471
Summer 1	Jack Hayes	21456	Ronnie Lo	21463
Surry 1/2	Bart Buckley	21452	Leon Engle	21484
Susequehanna 1/2	Mohan Thadani	21427	Jim Stone	21422
Three Mile Island 1	Ronald Herman	21320	Alexander Dromerick	21301
Three Mile Island 2	Michael Masnike	21373	Lee Thomas	(717) 9481151
Trojan	Roby Bevan	21361	Charles Trammell	21363
Turkey Point 3/4	Gordon Edison	21471	Jan Norris	21483

Plant	Project ManagerI	Ext. No.	Backup Manager	Ext. No.
Vermont Jankee	Vernon Rooney	21440	Daniel McDonald	21436
Vogtle 1/2	Jon Hopkins	21494	Lawrence Crocker	23049
Waterford 3	David Wigginton	23027	George Dick	21326
Watts Bar (TVA)	Rejender Auluck	20759		
Wapwr	Thomas Kenyon	21120	Guy Vissing	21101
WNP2	Robert Samworth	21304	John Bradfute	21381
Wolf Creek	Doug Pickett	21336	Paul O'Connor	23026
Yankee Rowe	Morton Fairtile	21443	Victor Nerses	21441
Zion 1/2	Chandu Patel	21395	Lloyd Zerr	23100

MINUTES OF THE PUBLIC MEETINGS TO DISCUSS

GENERIC LETTER 89-04

"GUIDANCE ON DEVELOPING ACCEPTABLE INSERVICE TESTING PROGRAMS"

On April 3, 1989, the NRC issued Generic Letter 89-04 which provides guidance aimed at correcting several weaknesses found by the NRC staff in inservice testing (IST) programs at nuclear power plants. The issuance of the generic letter is part of an overall NRC staff effort to improve IST programs. The staff also has a long-term goal of making IST programs essentially self-implementing such that establishment of a proper IST program would be determined through audits and inspections at the plant site rather than by staff review before a program is implemented.

The NRC staff held four public meetings to discuss Generic Letter 89-04 with holders of nuclear power plant operating licenses and construction permits. These meetings were noticed in the Federal Register (54 FR 23305) on May 31, 1989. In addition, the NRC Project Managers were requested to inform the individual licensees of the meeting dates and locations. The meetings took place in King of Prussia, Pennsylvania on June 5 for NRC Region I plants; Atlanta, Georgia, on June 8 for Region II plants; Chicago, Illinois, on June 13 for Region III plants; and South San Francisco, California, on June 15 for Regions IV and V plants.

Transcripts of the Chicago and San Francisco meetings were taken to provide assistance in the preparation of meeting minutes. The minutes will be distributed to meeting attendees who provided their address and holders of nuclear power plant operating licenses and construction permits. In addition, the meeting minutes together with the transcripts will be placed in the NRC Public Document Room.

At each meeting, a management representative of the NRC Region where the meeting was held provided opening remarks. Following those remarks, Tad Marsh, Chief of the Mechanical Engineering Branch (EMEB) of the Office of Nuclear Reactor Regulation (NRR) discussed the objective of inservice testing, its regulatory foundation, and problems found in IST programs. He also provided a brief overview of the NRC effort to improve inservice testing at nuclear power plants. Ted Sullivan, Section Chief of the Inservice Testing section in the MEB, then presented a detailed explanation of Generic Letter 89-04 and its applicability. Summaries of these three presentations are provided below. Copies of the slides used during the presentations by Tad Marsh and Ted Sullivan are attached to these meeting minutes.

SUMMARY OF OPENING REMARKS BY REGION MANAGEMENT

(Bill Johnston, Region I; Al Gibson, Region II; Carl Paperiello, Region III; and Dennis Kirsch, Region V)

Inservice testing of pumps and valves is explicitly required by the NRC regulations. This testing however is not performed merely to satisfy the Commission. Inservice testing is highly important to the operational safety of a nuclear power plant.

It is well understood that components important to the operational safety of the plant must function when needed. Two activities that provide assurance of the operability of these components are maintenance and inservice testing. In this regard, inservice testing is an equal partner with good maintenance practices.

The NRC headquarters and regional staffs are increasing their attention to inservice testing. As evidence of this increased attention, Generic Letter 89-04 was issued to provide the first NRC generic guidance on inservice testing. This guidance was developed to address frequently encountered issues involving IST programs, relief requests, procedural implementation, and technical specification provisions for operability. This generic letter will be followed by additional guidance that the NRC staff is preparing on inservice testing.

As indicated in the generic letter, less emphasis will be placed on program review by the NRC staff for determining the acceptability of IST programs before this implementation. Rather, the focus will be on audits and inspections of the IST program and procedures at the plant site by NRC personnel. In light of their importance in ensuring the acceptability of the program and procedures, these IST audits and inspections will be more detailed than in the past.

SUMMARY OF PRESENTATION BY TAD MARSH

The objective of inservice testing is to assess the operational readiness of safety-related pumps and valves. The scope of Section XI of the ASME Code, however, is limited to Code Class 1, 2, and 3 components. Thus, there is a disparity between the objective of inservice testing and the scope of Section XI.

The Commission regulations in 10 CFR 50.55a require compliance with the inservice testing provisions of Section XI. In order to account for improvements to the Code, the regulations were developed to require that IST programs be updated to the current Code edition and addenda every ten years. As has been seen, however, the IST provisions of Section XI have changed little in the last ten years and, in fact, have become quite stagnant. The regulations also allow licensees to submit for NRC review requests for relief from Code requirements where those requirements are impractical.

The establishment of effective IST programs is plagued by a variety of problems. Many of these problems are the result of inadequate testing requirements in the ASME Code. For example, the provisions of Section XI for performance testing of motor-operated valves extend only to stroke time. Further, the Code requirements for pump vibration testing are weak. Code Class 2 and 3 safety valves are not explicitly required to be tested. The Code incorrectly implies that check valves have a safety function in only one direction. The trending requirements in the Code are insufficient. As is apparent, the inservice testing provisions of the Code are lacking in many respects.

In addition to the Code guidance, there are several other sources of IST problems. For example, there has been an absence of NRC guidance on inservice testing. Previously, a regulatory guide was begun by the NRC staff, but was never completed. Further, the large number of revisions to IST programs and relief requests that require NRC review has caused a backlog in the approval process. Contrary to Standard Technical Specification 4.0.5, licensees have been implementing relief requests prior to NRC approval. Lastly, inspection efforts by NRC personnel have been made more difficult by the unavailability in some instances of a Safety Evaluation Report (SER).

The quality of inservice testing programs varies significantly from one nuclear plant to another. While some licensees have good IST programs, other licensees lack coordination among the groups (including corporate personnel) involved with inservice testing. At certain plants, inservice testing has been combined with inservice inspection despite the fact that these are distinct activities requiring personnel with different expertise. This combination of inservice testing and inservice inspection might be the result of their being addressed together in Section XI. Another problem is that inservice testing is often used as a training ground for junior personnel with those individuals moving to other areas as they progress. Further, many plant organizations do not have a single individual or organizational unit responsible for inservice testing. Unfortunately, inservice testing is viewed, on occasion, only as an activity to satisfy the NRC. The IST program, however, can be a true benefit to a licensee by initiating corrective action before a component must be declared inoperable. In this manner, inservice testing can also provide important information to be used in the maintenance program.

The issuance of Generic Letter 89-04 is a first step toward resolving the large number of IST problems. It provides generic guidance on eleven significant IST issues involving alternatives to Code requirements, and interpretation of the Code and technical specifications. Guidance is also provided to assist licensees in the development of acceptable IST programs. The generic letter clarifies the approval status of current IST program and relief request submittals that are under staff review. Finally, the generic letter presents a method for preparing revisions to IST programs in an acceptable manner.

In addition to Generic Letter 89-04, several NRC activities are intended to improve inservice testing at nuclear power plants. In particular, efforts are underway to revise Section 50.55a of the Commission regulations to endorse ASME standards OM-6 on pumps and OM-10 on valves. These standards provide improved guidance for inservice testing. Consideration is also being given to revision of the regulations in other respects. One proposed change would simply separate inservice testing from inservice inspection in paragraph (g) of 10 CFR 50.55a for administrative purposes. Another change under consideration would involve the long range plan for inservice testing, such as emphasizing the need for inservice testing to provide assurance that a component will perform all of its safety functions as necessary. In addition to Generic Letter 89-04, other generic guidance may be prepared. Finally, it was noted that an ASME/NRC symposium on inservice testing was scheduled for August 1-3 of this year.

SUMMARY OF PRESENTATION BY TED SULLIVAN

In the past, the process to obtain NRC staff review of an IST program and approval of the relief requests could consume a considerable amount of time. First, the NRC staff would review the IST program submitted by an applicant for or holder of a nuclear power plant operating license. From this review, a list of questions would be sent to the utility through the NRC Project Manager. An IST review meeting would then be held at the plant site. At the conclusion of the meeting, the staff would request that the program be revised to respond to issues raised at the meeting. Following receipt of that response and its review, the staff would issue an SER. Generic Letter 89-04 is intended to improve IST programs and also to simplify the process for obtaining NRC approval of IST program relief requests.

To help specify the method of response by the individual licensees, the operating nuclear power plants are categorized in Generic Letter 89-04 according to the status of the SER for their IST program. In this regard, the generic letter provides two tables listing particular nuclear power plants. For those plants in Table 1, the staff is nearing completion of an SER. That is, the IST review meeting has taken place fairly recently with a subsequent resubmittal by the licensee. With respect to Table 2, these listed plants have received an SER on their currently submitted IST program. If a plant received an SER several years ago but significantly revised the program in the meantime, that plant was not listed in Table 1 or 2. Similarly, if a plant had not received an SER on a prior IST program but had submitted one or more significant program updates, that plant was also excluded from Tables 1 and 2. About half of the operating plants are not listed in either Table 1 or 2. The staff is aware of minor errors in the tables but these have been resolved through the NRC Project Managers for those plants.

Plants listed in Table 1 or 2 do not need to submit a confirmation letter in response to Generic Letter 89-04. Nevertheless, it is essential that these licensees review the plant procedures to ensure their consistency with the provisions of the generic letter. For plants listed in Table 1 or 2, the SER for the particular plant will constitute approval of the IST program relief requests, including any deviations from the ASME Code.

In the case of plants not listed in Table 1 or 2, the generic letter provides the means for approval of IST program relief requests from the ASME Code. Certain steps should be completed, however, for the approval to be valid. First, the licensee is expected to review the IST program and procedures against the positions in Generic Letter 89-04 and then revise as necessary to conform to those positions. A confirmation letter is to be submitted by the licensee within six months of the issuance of the generic letter to indicate conformance with its provisions. For any necessary equipment modification, the licensee should provide in its confirmation letter a schedule for completing those modifications that is consistent with the time period specified in the generic letter. The NRC staff does not intend to perform detailed reviews of the confirmation letter and any alternatives discussed in those letters. Thus, an SER will not be issued. Nevertheless, NRC personnel could review this documentation during plant inspections.

Where a generic letter position is impractical for a particular licensee, a mechanism for approval of an alternative to that position is provided in Paragraph B of the generic letter. This mechanism requires evaluation of the maintenance and degradation history of the component. In this regard, all four criteria listed on page 3 of Generic Letter 89-04 must be addressed and documented in the IST program. If each criterion cannot be addressed, then Paragraph B is not the proper means to obtain approval of an alternative to a generic letter position. Further, the use of the Paragraph B mechanism for obtaining approval of an alternative to a position in the generic letter is limited to areas within the scope of those positions. Deviations to the ASME Code outside the scope of the generic letter positions will require submission of a relief request for review by the NRC staff.

It is recognized that the staff approach simplifies the review process for previously submitted relief requests that are not covered by the generic letter positions. When the NRC staff prepared the generic letter, it was determined that technical guidance would be provided on eleven issues. This determination was based on the total number of relief requests and their particular safety significance. Therefore, if a plant not listed in Table 1 or 2 had a program submitted and docketed before April 3, 1989, any relief requests outside the scope of the generic letter positions are approved provided that they are not subsequently changed.

At present, some plants might have aspects of their IST program that have not been approved by the NRC staff. For those plants, licensees should specify in their confirmation letter the relief requiring NRC staff review and approval, and the time frame in which that relief is needed. The staff will make a concerted effort to complete those reviews within the specified time frame. Overall, the goal is to have each licensee implementing a fully approve IST program.

A copy of the current IST program for each plant should be provided to the NRC staff. In addition, each licensee should provide an updated copy of its IST program to the staff when substantive changes are made to the program. The submittal should clearly identify those deviations from the ASME Code that are approved through the mechanism of the generic letter. Other deviations from the ASME Code that have received staff approval or must undergo staff review should be so indicated.

Licensees should evaluate deviations from the ASME Code included in the current IST program to determine if plant conditions continue to require relief from the Code. If the situation has changed, then approval of that relief through the generic letter may not be appropriate. Of course, where a licensee has received an SER on a particular relief request, that SER may be followed even if it appears to conflict with the generic letter. Where the staff believes that the relief is inappropriate, discussions may be held with the licensee to request a program revision. In significant cases, the staff may institute backfit procedures.

Generic Letter 89-04 is intended as a vehicle for the future as well as the present. For revisions to the IST program covered by the generic letter positions, the generic letter should be used as guidance for approval of the revisions. If a program revision is outside the scope of the generic letter positions and the licensee intends not to follow the ASME Code, a request for relief must be submitted for review by the NRC staff, which will then prepare an SER.

Upon implementation of the generic letter, some NRC staff resources will be shifted from IST program reviews to providing assistance in the inspection of IST programs. An inspection instruction will be prepared with a focus on the generic letter positions. The NRC staff has a goal of conducting an inspection of the IST program at each plant on a five-year schedule.

QUESTIONS

Following the presentations at each meeting, the NRC staff responded to the extent possible to questions submitted before the meeting, as well as to written and verbal questions and comments from the audience. These questions have been grouped according to their subject and then answered by the staff. In some instances, the staff responses at the meetings have been modified or expanded to answer the question in a more complete manner. The applicable regional meeting (together with the question number for that meeting) and, where known, the name of the individual asking the question, are noted in parentheses after each question.

QUESTIONS ON ATTACHMENT 1

POTENTIAL GENERIC DEFICIENCIES RELATED TO IST PROGRAMS AND PROCEDURES TO GENERIC LETTER 89-04

Position 1: Full Flow Testing of Check Valves

Question 1

Item 1 of Attachment 1 to the generic letter request that flow through a check valve be known for a valid full-stroke exercise test. Does this mean a direct flow indication and a recorded flow rate is the only acceptable method for the test? For example, BWR minimum flow lines are not instrumented with flow indicators. (Region 1 meeting, Question #9 at the meeting, questioner: Dave Wallace of Fitzpatrick)

Is direct flow rate instrumentation required for verification of full-stroke capability for all check valves? For example, the diesel cooling water check valves? (I #46)

Verifying full flow through small check valves in auxiliary systems or gas systems is typically impractical. As an alternate, will the NRC accept a qualitative evaluation of system response or performance in the place of flow measurements? (II #1c, John Zudans, Florida Power & Light)

For check valves where design accident flow is not specified, what guidance can you give for full-flow testing? (III #28, Don Zebrauskas, Commonwealth Edison Co.)

Response

Any quantitative measure that has acceptance criteria that demonstrate the required flow through the check valve may be used to satisfy the full-stroke requirement. An indirect measure of flow may be acceptable. For example, a change in tank level over a specified period could be used. In another case, the acceptance criterion could be based on a change in flow rate of an instrumented line when flow is admitted from a non-instrumented line containing the check valve being tested. In any event, some form of quantitative criteria should be established to demonstrate full-stroke capability.

Question 2

Why isn't knowledge of total flow through multiple parallel lines acceptale, when the total flow through each path was known when it was established? (I #13, J. W. Connolly, PSNH-Seabrook Station)

Regarding full flow testing of check valves, why is knowledge of total flow through parallel flow lines unacceptable? This seems to challenge conservative Technical Specification requirements for flow balancing. (III #34, Gary J. Roesner, Callaway Nuclear)

Response

The objective of inservice testing is to evaluate and investigate the possibility of degradation of components and to take corrective action before the components fail. Verification of total header flow rate might not identify a problem, developing or occurring, with an individual check valve in one of the parallel flow paths. With respect to the balancing of flow, the Technical Specification requirement is based on the flow from one loop being lost through a break. Consequently, that flow path is restricted or throttled to minimize significant diversion of flow. The Technical Specification requirement was not intended to verify individual check valve operability. The licensee is expected to justify the use of a test method that does not verify full stroke of individual check valves.

Question 3

Can check valves with external operators and position indicators be tested only with these devices and never exercised with flow or disassembled (I #47)

Is it the intent of the NRC to require full-stroke flow testing of all check valves or is it acceptable to perform manual exercising and partial stroke testing of check valves as permitted by IWV-3522(b)? (II #1a, John Zudans, Florida Power & Light)

Position 1 implies that the only method acceptable to the NRC for full stroke exercising is a full flow test. No mention is made of check valves with external features which can be used for full stroke exercising. Do the 6 criteria presented have to be addressed in the IST program to justify using an external operator? (III #43, Pat Tobin, Northern States Power, Monticello)

Response

The ASME Code in IWV-3522(b) allows full stroke testing of check valves either with flow or with a mechanical exerciser. Full flow testing is preferable where practical, but Position 1 of Generic Letter 89-04 was not intended to imply that the ASME Code provisions for mechanical exercising were not acceptable. Such mechanical exercising is clearly acceptable and is certainly preferable to valve disassembly as a means of ensuring valve operability. If an external operator is used to exercise a check valve, the provisions of IWV-3522(b) must be met, but the six criteria in Position 1 of the generic letter need not be addressed.

Question 4

What is considered the maximum required accident condition flow? (I #14, J. W. Connolly, PSNH-Seabrook Station)

In reference to Items 1 and 2 of Attachment 1, please clarify the term "maximum required accident condition flow." (IV & V #22, John DeBonis, Stone & Webster/Comanche Peak)

Response

The phrase "maximum required accident condition flow" is intended to mean at least the largest flow rate for which credit is taken for this component in a safety analysis in any flow configuration. The safety analyses are those contained in the plant's Final Safety Analysis Report (FSAR), or equivalent, but are not limited to the accident and transient analyses.

Question 5

Is it the intent of the stated position of Attachment 1 that a satisfactory test of a valve in the open direction requires only measurement of full accident flow through the valve and not the measurement of differential pressure (with associated acceptance criteria) as per IWV-3522(b)? (II #1f, John Zudans, Florida Power & Light)

Response

The ASME Code does not require the measurement of valve differential pressure when exercising check valves with flow. It should be recognized, however, that such a measurement might provide useful information for evaluating the condition of the valve.

Question 6

For check valves which are never required to open fully (i.e., thermal expansion or siphon breakers), verification of design (safety) function is the testing required for forward flow. Is this correct? (III #42)

Response

In addition to verifying its safety function performance, quantifiable acceptance criteria should be developed for the testing of these components. For example, a pressure decay test with specified acceptance criteria would be considered a reasonable test.

Question 7

In reference to Item 1 of Attachment 1, for non-parallel full flow test, does the flow obtained need to be documented quantitatively, or can it be qualitative (i.e., greater than ____ gallons per minute)? (IV & V #23, John DeBonis, Stone & Webster/Comanche Peak)

What is an acceptable flow condition when, for example, the safety analysis requires 250 gallons per minute (gpm) flow but 600 gpm can be delivered? Would passing greater than, or equal to, 250 gpm be a valid full flow test, or would 600 gpm need to be delivered? (IV & V #24, D. G. Dobson, Texas Utilities/Comanche Peak)

Response

The full flow test is intended to demonstrate that the necessary flow rate can be achieved and to detect any degradation of the check valve. Therefore, acceptance criteria for the test should involve more than the achievement of flow above a minimum rate. The acceptance criteria should also include the allowable variation of test results. To enable the test results to be compared, the initial parameters for the test should be standardized to the maximum extent feasible. The acceptance criteria for the full flow test and the bases for those criteria should be documented and available for review by NRC inspectors.

Question 8

In reference to Item 1.3 of Attachment 1, please clarify what the NRC would expect a "qualification program" to include (i.e., how extensive). (IV & V #25, John DeBonis, Stone & Weber/Comanche Peak)

Response

Position 1 of Generic Letter 89-04 indicates that, where full flow testing is impractical, it might be possible to qualify other techniques to confirm that the check valve is exercised to the position required to perform its safety function. One of the stated conditions for this approach is that the licensee should describe the test method and results of the program to qualify the alternate technique for meeting the ASME Code. The language of Position 1 in this regard was chosen to allow the licensees flexibility in qualifying alternatives to full flow testing. In general, the licensee should demonstrate that the alternate test is quantifiable and repeatable. The alternate test should also meet the intent of the ASME Code. This qualification of the alternate test should be documented by the licensee and available for review by NRC inspectors. The Nuclear Industry Check Valve Group (NIC) is said to be investigating the qualification of various testing techniques, such as ultrasonics and radiography for check valves. The results of those and other industry efforts might be of value to the individual licensee in providing for the use of alternatives to full flow testing.

Position 2: Alternative to Full Flow Testing of Check Valves

Question 9

Does the Generic Letter Attachment 1, item 2c use of "orientation" refer to physical orientation (e.g., horizontal or vertical) or plant orientation? (I #15, J. W. Connolly, PSNH-Seabrook Station)

Response

Orientation, as used in Generic letter 89-04, refers to the physical orientation (horizontal or vertical) as well as the physical relationship to major components. For example, a check valve at the discharge of a pump has a different orientation than one at the pump suction.

Question 10

When manually exercising per position 2c, is this done per Code or just a physical stroke checking for binding? (I #16, J. W. Connolly, PSNH-Seabrook Station)

When valves are disassembled and manually exercised in lieu of full-flow testing, is adherence to the quantitative aspects and acceptance criteria of IWV-3522(b) required? (II #1e, John Zudans, Florida Power & Light)

Response

The staff believes the requirement in IWV-3522 (b) of the ASME Code to measure the force or torque while manually exercising check valves only applies to manual exercising from outside the valve where the observation of the valve internals cannot be made. This measurement permits a quantitative evaluation of the performance of the valve in that changes in the measured force or torque may be indicative of degradation of the valve internals. While the valve is in a partially disassembled condition the valve internals should be inspected and the condition of the moving parts evaluated. This inspection and evaluation should include verification by hand that the valve disk is free to move, but measurement of force or torque is not required. Following reassembly, a partial flow test is expected to be performed.

Question 11

Does the utility have the option of either inspection through disassembly or performing functional testing to satisfy IST requirements? Can either be used regardless of the previous testing mode? (I #31, John Wiedemann, PSE&G)

Response

Disassembly, together with inspection, to verify full stroke capability of check valves is an option only where full stroke exercising cannot practically be performed by flow or by the other positive means allowed by IWV-3522. Additionally, partial stroke exercise testing with flow is expected to be performed after the disassembly and inspection is completed but before returning the valve to service. If the previous test was performed using flow, the licensee is expected to document the justification for any change from that test method. Also, for the case where plant conditions prevent full stroke testing with flow, the licensee should periodically evaluate whether plant conditions have been altered in such a way that full stroke testing using flow is possible. If so, the licensee should revise the test procedures to provide for such testing.

Question 12

In light of the stated position of requiring check valve internal inspection at least once every six years, is it permissible to schedule the inspections for the total group of valves on a six year frequency vs. each refuel outage? This is especially important where plant preparations for inspection of multiple valves are essentially equal to those for a single valve and they represent a considerable cost in terms of monetary outlay as well as schedule and availability impacts. (II #1d, John Zudans, Florida Power & Light)

Response

Position 2 of Generic Letter 89-04 takes advantage of the benefits that can be obtained through sampling techniques. The NRC staff, however, recognizes that the position may have a significant impact on outage time. For example, some plants have combined injection header check valves that are physically located in a position relative to the reactor coolant system (RCS) loops such that their disassembly would require draining the RCS to a level that would necessitate core offload. In order to alter the inspection frequency as suggested by this question, licensees should use the criteria in Position 2 to justify and to document the proposed disassembly schedule. The justification should address the significance of the loss of benefits of sampling in light of the condition, service history, and application of the valves. For additional discussion of this issue, see the response to Question 19.

Question 13

Does disassembly/inspection require certified visual testing personnel, or can detailed inspection procedures be performed by maintenance personnel without certified inspectors? (II #25, Jim Holton, Florida Power Corp.

Do personnel performing the visual inspections addressed on Position 2 have to be VT-3 certified, ANSI 45.2.6 (i.e., Mech Inspector) certified, or may engineering personnel competent in check valve technical requirements perform this visual inspection (III #2, Larry Campbell, Toledo Edison)

Response

The personnel performing the disassembly/inspection must be qualified to evaluate the condition of the valve and to assess its continued operability. The licensee is responsible for the development and implementation of a program to ensure that IST personnel are appropriately trained and qualified for performing the valve disassembly/inspections. Generic Letter 89-04 alone does not impose any requirements for visual testing certifications (such as VT-3) beyond those currently in the ASME Code. Nevertheless, licensees must implement the provisions of ANSI/ASME N45.2.6, "Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants," according to their commitments based on the implementation section of Regulatory Guide 1.58. The NRC staff encourages those licensees that have not formally committed to following Regulatory Guide 1.58 to review the ANSI standard and regulatory guide for guidance in developing a program for the qualification of inservice testing personnel.

Question 14

If a check valve within a sample group is disassembled/inspected in a non-refueling outage, does the next valve need to be inspected at the next refueling outage, or can it still be scheduled for its original refueling outage? (II #26, Jim Holton, Florida Power Corp.)

Response

This question is difficult to answer without more detailed information. In general, in order to alter the disassembly/inspection schedule as suggested by the question, the licensee should justify and document the proposed change. The justification should address the effect of the proposed disassembly/inspection schedule on the sampling program. The justification should rely on the maintenance history and known valve condition from previous inspections rather than subjective qualitative judgement. Position 2 in Generic Letter 89-04 indicates the criteria that need to be addressed.

Question 15

Is it the intent of Position 2 of the Generic Letter 89-04 that during valve testing by disassembly, that the valve be completely disassembled and each internal valve part removed, if possible, and 100% of the part visually inspected, or may only the valve bonnet be removed and the valve internals inspected in place without the removal of the internal valve parts unless evidence of discrepant conditions are found which then would require further inspection and probable removal of the part? Note: Inspection of the valve internal parts without removal of the part would be by direct visual inspection, use of mirrors, or by remote inspection equipment such as boroscope fiberoptics. (III #1, Larry Campbell, Toledo Edison)

Response

When performing check valve disassembly and inspection to satisfy the requirements of the ASME Code for inservice testing, disassembly is required only as far as necessary to assess the condition of the valve and to allow manual exercising of the disk. (It must be recognized, however, that the Code requirements for inservice inspection are different from those associated with inservice testing.) If a partial stroke exercise with flow can be performed, this testing is expected to be performed after the disassembly and inspection are completed but before returning the valve to service.

Disassembly and inspection of a check valve is not considered a "test" as implied by the question. Disassembly is not a true substitute for an operability test using flow, but is allowed as an alternative to a flow test where that test is not practical. Disassembly and inspection does, however, provide a valuable means of determining the internal condition of the valve. A recent example of the value of disassembly and inspection involved the identification of broken bolting material in Anchor Darling check valves at two nuclear power plants. This occurrence is discussed in NRC Information Notice 88-85, dated October 14, 1988.

The NRC staff is encouraging the development and use of alternate techniques to evaluate the position of check valve disks. The Electric Power Research Institute (EPRI) and the Institute of Nuclear Power Operations (INPO) are recommending an inspection periodically for check valves that are subjected to potentially harsh service conditions. The NRC staff encourages these activities as well. The industry group NIC is also investigating methods to demonstrate the operability of check valves.

Question 16

Even though the check valve flow testing can be performed as required by ASME Section XI, may the valve test be performed by disassembly as permitted by Position 2 in Generic Letter 89-04 when it is considered by the utility that testing by disassembly will provide the same or greater assurance that the valve will function properly? (Note: If possible, partial valve stroking quarterly, or at cold shutdown, or after re-assembly would be performed.) If the answer is yes, (a) can the test frequency, sample, etc., as described in Generic Letter 89-04 Postion 2 be used in lieu of ASME Section XI requirement-even if the Section XI test could be performed, i.e., at cold shutdown; (b) must a relief request be processed or may this "test by disassembly" be noted in the valve IST program submittal to the NRC; and (c) must a relief request be processed or may the frequency sample, etc., be noted in the valve IST program submittal to the NRC? (III #3, 4, 5, 6, Larry Campbell, Toledo Edison)

Response

The various methods aimed at evaluating the operability of check valves are not equally acceptable to the NRC staff. At the outset, the ASME Code requires a full stroke exercise using flow (or a mechanical exerciser) to be performed quarterly. Where full stroke exercising cannot be performed quarterly, the Code allows the performance of this test during cold shutdowns. Full stroke exercising during refueling outages may be an acceptable alternative if the test cannot be performed at cold shutdown, but this approach would require submission of a relief request. For those cases where full stroke exercising cannot be performed quarterly, during cold shutdown, or during refueling outages, disassembly and inspection in conformance with Position 2 of Generic letter 89-04 is allowed as an alternative. If the provisions of Position 2 are followed, a relief request need not be submitted for NRC review but this deviation from the ASME Code should be documented. (See also the response to Question 15)

Question 17

May the valve testing by disassembly/visual inspection identified in Position 2 of Generic Letter 89-04 be applied to reverse flow testing of check valves? (III #7, Larry Campbell, Toledo Edison)

Response

Position 2 of Generic Letter 89-04 addresses the use of disassembly and inspection as an alternative to forward flow testing of check valves. The use of disassembly and inspection to verify closure capability (i.e., back flow) may be found to be acceptable depending on whether verification by flow or pressure measurements is practical. As the generic letter does not address this use, however, the submission and approval of a relief request before implementation is required. Disassembly and inspection is not acceptable for demonstration of leak-tight integrity.

Question 18

We are only able to perform a partial flow test of the accumulator discharge check valves due to limitations based on system configuration. Do we have to supplement this test with disassembly of the check valves? (III #20, Wisconsin Public Service Corp.)

Response

The safety injection accumulator discharge check valves are typically very difficult to exercise with flow to the position required to perform their safety function. If a partial flow exercise is all that can be performed, then some other technique, as discussed in Position 1 of Generic Letter 89-04, might be developed to periodically verify the capability of these valves to move to their safety function position. If this is not feasible, the licensee is expected to follow the provisions for the disassembly alternative contained in Position 2 of the generic letter.

Question 19

Regarding disassembly of check valves, please define "extreme hardship" when speaking with regard to extension of disassembly interval. (III #36, Gary J. Roesner, Callaway Nuclear)

Response

The existence of "extreme hardship" that would allow extension of the disassembly schedule in Position 2 of Generic Letter 89-04 is dependent on the particular circumstances at the plant. To determine whether extreme hardship exists, the licensee should conduct a detailed evaluation of the various competing factors. First, the licensee should determine the effect on plant safety that would result from the proposed schedule extension. The maintenance history of the component and other information relevant to its reliability should be reviewed to determine whether the decrease in assurance of plant safety resulting from the schedule extension is justified. A need to offload the reactor core, such as when testing the combined injection header check valves at some plants, or to operate at mid-level of the reactor coolant loops may be considered. The radiation exposure that would result from the disassembly and inspection is a factor to be considered under the ALARA (As Low As Reasonably Achievable) principle, but it should be judged in combination with all of the other factors.

Question 20

Position 2 goes into the scheduling of disassembly/inspection in a very detailed manner. Are other scheduling schemes acceptable as long as they have each valve disassembled/inspected within 6 years? Would approval of an alternate schedule have to be in the form of an SER or acceptance of details provided in a confirmation letter (existing schedule for disassembly/inspection agreed upon in IST program review with NRC, but SER never issued)? (III #44, Pat Tobin, Northern States Power, Monticello)

Response

As stated in Position 2 of Generic Letter 89-04, the burden is on the licensee to demonstrate the extreme hardship necessary to comply with the identified sample disassembly/inspection schedule. The staff considers the sampling aspect of the position to provide assurance of the continued operability of the valves that are not inspected during any given outage. Therefore, the licensee should justify through the provisions listed in Position 2, any deviation from the stated schedule. That justification should be provided in the IST program submitted to the NRC staff, but need not be included in the confirmation letter. Where the provisions of Position 2 for an alternate disassembly schedule are followed, it is acceptable to implement the alternative and an SER will not be issued. The NRC staff, however, may review the alternative and its justification during plant inspections.

Position 3: Back Flow Testing of Check Valves

Question 21

With reference to generic letter item 3, if a leak test is performed to verify Category C check valve seat position, would any leak rate be acceptable so long as the system meets its minimum requirements to perform its safety function? (I #18, Al Koehl, NES)

Response

When performing a test to verify closure capability of a check valve that does not have a defined seat leakage limit, the achievement of the necessary system flow rate through the intended flow path might be an adequate demonstration of the closure capability of a check valve. For example, when verifying the closure capability of the check valves on the discharge of parallel pumps, achievement of the required safety flow rate from one running pump with the idle pump's discharge check valve providing the barrier for recirculation flow would be considered an acceptable test configuration. In addition, the licensee should evaluate the consequences of the back flow through the check valve. This evaluation should consider the loss of water from that system and connecting systems, the effect that the leakage might have on components and piping downstream of the valve, and any increase in radiological exposure resulting from the leakage.

Question 22

Are the items listed in Attachment 1, number 3a, d, e, f, specific to PWR's? The nomenclature is not familiar to BWRs. (I #24, John Lindburg, PP&L)

Section 3 of Generic Letter 89-04 deals with back flow testing of check valves. It has a list of several valves that NRC states provide a safety function. Some of these valves do not appear to provide a safety function and we would like to hear the NRC's reason for classifying these valves as safety related. (III #19, Wisconsin Public Service Corp.)

Response

All of the listed systems do not necessarily apply to each plant. A licensee should evaluate at least the listed systems to determine if they apply to its facility and should make any necessary modifications to its IST program. In regard to a particular question, items 3d, e, and f are specific to pressurized water reactors (PWRs) while 3a (feedwater header check valves) may be applicable to both boiling water reactors (BWRs) and PWRs. One example provided in Position 3 to the generic letter is the volume control tank outlet check valve in the chemical and volume control system. This check valve may serve an important safety function at some PWR plants to separate the non-safety grade water source from the safety grade source.

Question 23

In regard to Attachment, Position 3, how is individual seat leakage determined for 10 CFR 50, Appendix J, Type C, tested valves? Tech Specs specify only penetration totals. (I #35, J. W. Connolly, Seabrook Station)

Response

IWV-3426 of Section XI of the ASME Code requires that a permissible leak rate be specified by the plant owner (licensee) for a specific valve. If leak rates are not specified by the licensee, permissible leak rates are provided in IWV-3426. It should be noted that Section XI provides no criteria or guidance for licensees on the method to establish or to specify the permissible leak rate of a particular valve. Apparently, the Code recognizes that leak behavior of a valve varies according to the type of valve, the vendor, the valve size, the service conditions, the safety-related functions, and other factors, and that there is no simple leak rate rule that may be applicable to all valves.

In general, the leak rate limits should be set within certain bounds. If the leak limits are too low, unnecessary repairs or adjustments to the valve can result. If too high, failure of the tests required by Appendix J to 10 CFR Part 50 could occur, leading to concerns for leak-tight integrity of the containment. Appropriate permissible leak rates can only be developed and refined by analyzing and trending the leak rate data of specific valves or leak rate data from similar valves at other plants. Therefore, the NRC staff is not in a position to specify leak rates. The licensee should document its methods for establishing the initial permissible leak rates and procedures for improving the leak rate limits.

Question 24

In regard to Attachment 1, Position 3, does this backseat check require a full-stroke exercise and is it performed at the Code specified frequency regardless of normal plant positions? (I #36, J. W. Connolly, Seabrook Station)

In reference to Item 3 of Attachment 1, does a valid back-flow test on a check valve first require the valve to be exercised to the open position then back tested, or is it valid to merely perform the back flow test? (IV & V #29, D. G. Dobson, Texas Utilities/Comanche Peak)

Response

If a particular valve performs a safety function only in the closed position, demonstration of a full-stroke open before verification of closure capability is not required by the ASME Code. This closure verification is required to be performed at the frequency specified by the Code. If (1) the valve performs a safety function in the closed position, (2) the normal position for the valve is closed, and (3) this position can be verified during normal plant operation, then quarterly documentation of this verification satisfies the Code requirements. If a valve performs a safety function in both the open and closed positions, however, the Code requires that the valve be exercised to the open position and then be verified to close.

Question 25

Previous to this, it was permissible to verify closure of stop-check valves simply by operation of the stem (shaft). Is this acceptable instead of reverse flow testing? (II #1b, John Zudans, Florida Power & Light)

Response

Verification of closure capability of stop check valves by using the handwheel meets the ASME Code requirements. This, however, is not the preferred method of test. The NRC staff considers reverse flow testing to be a more reliable indication of valve operability.

Question 26

Regarding back flow testing of check valves, what is the position of the generic letter in the phrase "verify by other means?"
(III #39, Mort Khazrai, Toledo Edison)

Response

The majority of the wording in the sentence in which this particular phrase appears was taken directly from IWV-3522 of Section XI of the ASME Code. The NRC staff included the phrase "by other positive means" to be consistent with the wording of the Code. When Generic Letter 89-04 was written, the staff did not have in mind any particular techniques that it would consider acceptable.

Position 4, Pressure Isolation Valves

Question 27

Is it the intent of Generic Letter 89-04 that the only Reactor Coolant System Pressure Isolation Valves (PIVs) to be included in the IST program are those listed in the Technical Specifications and those which are Event V PIVs? (III #8, Larry Campbell, Toledo Edison)

For plants licensed prior to 1979 which do not list all RCS Pressure Isolation Valves in their Technical Specifications, is it the intent of Position 4 of Generic Letter 89-04 that only PIVs listed in the Technical Specifications and PIVs which are "Event V" be included in the IST Program? (III #9, Larry Campbell, Toledo Edison)

Does the NRC anticipate requiring (in the future) that all RCS PIVs be included in the IST program? (III #10, Larry Campbell, Toledo Edison)

Response

The position in Generic Letter 89-04 represents only a limited area of the staff's concerns regarding PIVs. The generic letter position only applies to those PIVs listed in individual plant Technical Specifications. However, the staff recognizes that the PIVs in the Technical Specifications for many plants, particularly older plants, are a subset of the PIVs in the plant. In view of this fact and other concerns regarding PIVs, the staff has recently undertaken a program to reevaluate various aspects of PIVs, including testing. Sample inspections are underway as part of this NRC program.

Question 28

What, if anything, is being done with the licensee responses to Generic Letter 87-06? The generic letter references PIVs in Section 4; however, it appears that there are no changes required due to Generic Letter 87-06. Is this true? (III #18, Wisconsin Public Service Corp.)

Response

The responses to Generic Letter 87-06 are being used as input for the resolution of Generic Issue 105, "Interfacing Systems LOCAs at Light Water Reactors," under investigation by the NRC Office of Nuclear Regulatory Research. No further licensee action is required at this time with respect to Generic Letter 87-06.

Position 5, Limiting Values of Full-Stroke Times for Power-Operated Valves

Question 29

Attachment 1, Position 5 in part states: "The deviation should not be so restrictive that it results in a valve being declared inoperable due to reasonable stroke time variations. However, the deviation used to establish the limit should be such that corrective action would be taken for a valve that may not perform its intended function." Given that MOVs operated by AC induction motors fail if slowed by more than approximately 10%, a valve normally stroking in 15 seconds will fail to operate by a change of 1.5 seconds. By comparison, a reasonable deviation from normal stroke time of 15 seconds caused by error in measurement might be 2 seconds. The fact that the reasonable deviation for this 15 second valve is larger than the possible actual deviation before failure makes the two quoted goals of Attachment 1, Position 5, mutually exclusive. Request resolution.
(I #32, D. B. Ritter, PP&L)

Response

The staff agrees that stroke times for AC motor-operated valves probably will not change appreciably before failure, especially for MOVs that have relatively short stroke times. If the ASME Code-identified testing does not provide useful information for evaluating the continued operability of these valves, then the licensee should propose an alternative to the Code requirements that does provide such information. The Code requires the licensee to establish limiting values of full stroke time for all power-operated valves and also requires measurement of stroke time to an accuracy of within 10 percent for this particular case. The Code does not prohibit the measurement of stroke time more accurately or the setting of the limiting value at less than 25 percent above the normal stroke time. The NRC and industry recognize that the Code-specific criteria are not sufficient for assuring operability of AC motor-operated valves. In light of this recognition, the staff issued Bulletin 85-03 to require that licensees establish programs to ensure that operator switches for MOVs in certain important plant systems are selected, set, and maintained properly. As a result, in part, of the responses to that bulletin, the scope of the effort has been expanded in Generic Letter 89-10 to include many other MOVs important to plant safety. NRC staff actions such as these will be need to compensate for weaknesses in the IST provisions of the ASME Code until an adequate IST standard is available.

Question 30

In regard to Attachment 1, Position 5, what is considered a reasonable deviation from the reference stroke time? (I #37, J. W. Connolly, Seabrook Station)

In regard to Attachment 1, Position 5, can the deviation be different for valves with different functions and/or actuators? (I #38, J. W. Connolly, Seabrook Station)

What is meant by "reasonably limiting value of full-stroke time?" (I #48)

What methods are considered acceptable for establishing the limiting value for full stroke times for power operated valves as given in Position 5 of Generic Letter 89-04? (III #50)

In reference to Item 5 of Attachment 1, is there any generic guidance on what is acceptable to the NRC on this item? (IV & V #11, T. F. Hoyle, Washington Nuclear 2)

What is "reasonable" value for deviating from the reference stroke time established for valve testing? (IV & V #16, Arkansas Nuclear 1 and 2)

Response

The NRC staff has attempted to provide the general philosophy for establishing the limiting stroke time. The establishment of specific values for the limiting stroke time is dependent on a variety of parameters relevant to the particular valve and the conditions at the plant. The parameters include operating characteristics, operating environments, actuator types, and valve stroke times. In that the test should confirm the operability of the component and not the system, the limiting value is not to be considered a function of the valves's safety significance. As the limiting value is specific to the valve, the staff is not in a position to provide values for limiting stroke times. The licensee needs to use its best judgement in assigning these values. The justification for the assigned values is expected to be documented and available to the plant site for review by NRC personnel. One aspect of the staff review will be a comparison of the limiting stroke time to the technical specification value.

Question 31

In regard to Attachment 1, Position 5 (paragraphs 2, 3 and 5), why are Tech Specs or Safety Analysis limiting criteria not acceptable for valve operability if maintenance is triggered by component evaluation? (I #41, Eugene Perry, Consolidated Edison)

With respect to the application of stricter acceptance criteria for valve stroke times, apparently the NRC has some idea as to the philosophy and limits that would be acceptable. This information should be shared with licensees. (II #17)

Define the "limiting value of full-stroke time." Is this number the operability number for the valve even if the Tech Spec stroke is much higher? (II #14, Mark Cardile, Georgia Power)

Response

The Technical Specifications provide assurance that important plant systems are capable of performing their safety functions in a timely manner during selected plant accidents. The provisions of Section XI of the ASME Code are intended to ensure the continued operability of particular plant components. The distinct bases for these two documents lead to criteria that may differ significantly. Nevertheless, the Technical Specifications and ASME Code are both needed to provide confidence that the nuclear power plant can be operated safely. Therefore, the more restrictive criteria of the two documents must be followed even though this might result in a component or system being declared inoperable. The response to questions on position 8 of Generic Letter 89-04 also address the relationship of the ASME Code to the Technical Specifications.

Question 32

Is it required to measure stroke times of valves that are not provided with remote position indication? (II #11, John Zudans, Florida Power and Light)

Response

The ASME Code requires the measurement of stroke time for all power-operated valves regardless of whether they have remote position indication. The staff has endorsed this requirement. Without specifics, the staff is not in a position to comment on alternate techniques that may be found acceptable.

Question 33

When considering comparison of power-operated (stroke time) valves according to valve type, valve actuators, valve size, etc., we find there is no consistency when using this comparison. However each valve consistently tests well. We are currently looking at a quantitative method of establishing maximum allowable stroke times. Is this an acceptable method?
(II #28, Jim Holton, Florida Power Corp.)

Response

If we understand the intent of the opening sentence of the question, we agree that criteria for setting the limiting value of full-stroke time may vary for each valve type, stroke time, size, etc. The use of a quantitative multiplier on a reference time may be an acceptable method for setting these values. However, as discussed in some of the responses above, the licensee should document the justification for its quantitative methods of establishing maximum allowable stroke times. This justification should be available at the plant site for review by NRC personnel.

Question 34

When the stroke time of a power operated valve exceeds its [limiting value for] stroke time, as established in accordance with Position 5 of the Generic Letter 89-04, but is still within its plant Technical Specification or FSAR stroke time limit, can performing an evaluation which determines if the valve may remain operable be used to satisfy Position 5 in lieu of making it mandatory that the valve be declared inoperable? (III #12, M. J. Richter, Commonwealth Edison)

Response

The limiting value of full stroke time is required to be established for all power-operated valves. The limiting value should be that point at which the licensee seriously questions the continued operability of the valve. It is expected to be a value determined to be reasonable for the individual valve based on that valve's characteristics and past performance, but not to exceed any safety analysis requirements. The value should not be based solely on the system requirements or values specified in safety analyses for system performance. When the identified limiting value is exceeded, the licensee shall declare the component inoperable and shall enter any applicable Technical Specification limiting condition for operation (LCO). Following the declaration that the valve is inoperable, the licensee may perform an analysis to identify the root cause of the problem with the valve. If this analysis clearly demonstrates that the valve remains capable of performing its safety function, the analysis might constitute the corrective action required by the Code. The analysis should be documented.

Question 35

If the limiting value of full stroke time is less than the "alert limit" identified in the Code, does the trending still have to be done? (III #51)

Response

If the limiting value of full stroke time is exceeded, then the licensee shall declare the valve inoperable and shall perform corrective action. Where the limiting value is less than the 25 percent or 50 percent "alert limits" for trending as specified in the ASME Code, trending as envisioned by the Code becomes a moot point. The licensee could identify a reduced percentage alert limit for this valve to provide early warning of problems with this valve, but this is not required either by the Code or by Generic Letter 89-04.

Question 36

In reference to Item 5 of Attachment 1, is Item 5 in fact a rewrite of the stroke time criteria that are to be applied in accordance with OM-10? (IV & V #31, D. G. Dobson, Texas Utilities/Comanche Peak)

Response

The information in Position 5 of Generic Letter 89-04 was not intended simply to be a rewrite of the information in ASME Standard OM-10. This position has evolved over the years and is considered reasonable by the staff for establishing limiting values of full stroke time for power-operated valves. As such, the position represents a clarification of existing ASME Code requirements. For its part, ASME Standard OM-10 does not provide guidance for the establishment of the limiting value of full stroke time. This standard, however, does require that a valve be declared inoperable immediately upon discovering that it fails to exhibit the required change of obturator position or exceeds the limiting value of full stroke time.

Question 37

Since establishing maximum stroke time limits may in some cases at first prove too restrictive, is it acceptable for corrective action to be an engineering evaluation which increases the time limit (based on more detailed analysis)? (IV & V #33, Alan Harris, Waterford 3)

Response

The Commission regulations in 10 CFR 50.59 allow licensees to perform engineering evaluations of plant structures, systems, and components. If the stroke time limit is exceeded, the valve must be declared inoperable and any applicable Technical Specification limiting condition for operation entered. At that point, an engineering analysis may be performed to verify that the valve is capable of performing its safety function. This analysis should include more than a determination that the new value is less than the FSAR or Technical Specification limit. For example, a root cause investigation should be performed to determine the reasons for the stroke time increase.

Question 38

We have been informed that we could omit the valve stroke time limits from our IST Submittal. Where can we find guidance on what is really required in a submittal (minimum scope)? (IV & V #37, Paul Croy, Southern California Edison, San Onofre)

Do specific valve stroke time requirements (or limits) need to be specified in the IST plan, or is specification in implementing procedures sufficient? If procedures are sufficient, can existing limits referenced in the plan be removed in a future revision? If plan specification is required, is this limited to Technical Specification and safety analysis stroke time limits, or must owner specified stroke time limits that are required also be in the plan? (IV & V #38, Terry Pellisero, Pacific Gas & Electric, Diablo Canyon)

Response

The specific limiting values of full stroke time for each power operated valve as determined according to Position 5 of Generic Letter 89-04 are not required to be identified in the IST program. These limiting values, however, should be provided in a document such as the individual test procedure or a general procedure that identifies the criteria for establishing these values. The concern for the specification of limiting values is the result of weaknesses that the NRC staff has found while reviewing IST procedures. As a general rule, IST programs should contain sufficient information to indicate what parameters are being measured, how tests are being performed, and the bases for the acceptability of any departures from the ASME Code. For example, the program should indicate forward flow testing or back flow testing, or both, for check valves.

Postion 6, Stroke Time Measurements for Rapid Acting Valves

Question 39

With reference to the Generic Letter item 6, paragraph 4, where does the two-seconds come from and what is the bases for the two-second only criteria, could this be a minimum of 3 or 4 seconds? (I #19, Al Koehl, NES)

Response

The two-second criterion is based on the staff's consideration of the response time of personnel and equipment and the difficulties involved in applying the ASME Code requirements in this situation. Any alternative to Position 6 of Generic Letter 89-04 or the ASME Code requirements may be submitted, along with a sound basis, for staff review through a relief request. As relief requests containing alternatives to the Code requirements are expected to address the fundamental purpose of inservice testing, see the summaries of the opening presentations for a discussion of this subject.

Question 40

Generic Letter 89-04 states that previous analysis (IWV-3417(a)) can be replaced with a conservative "reference value" comparison. Generic Letter 89-04 states this should be documented in the IST program. Should this change be made by relief request or by a text change to the program body? (I #23, Jeff Nyhard, Nine Mile Station)

Generic letter position on power operated valve stroke times of greater than ten seconds is to place the valve in increased frequency if stroke time is greater than 25% of the base line stroke time.
(III #38, Mort Khazrai, Toledo Edison)

Response

When the staff prepared the discussion in Position 6 of Generic Letter 89-04, the objective of the first paragraph was to set the stage for the discussion on "rapid acting" valves, and it was not intended to address all aspects of stroke time for power-operated valves. Nevertheless, the staff believes that the use of a reference value stroke time as a base line for comparison of routine test values is a better method of evaluating change in valve performance than that specified ASME Code IWV-3400. Therefore, if a licensee wishes to use reference values rather than previous test values for comparing stroke times for valves with normal stroke times equal to or less than ten seconds, the generic letter provides the vehicle for this deviation from the Code and a relief request need not be submitted. As the generic letter does not address valves with normal stroke times greater than ten seconds, a licensee must submit a relief request for staff review and approval before using reference values as a base line for stroke times for these valves.

Question 41

Can an MOV or power-operated valve have a dual classification under "rapid acting" and "less than 10 seconds?" For example, we have valves that stroke closed in less than 2 seconds and open in less than ten seconds. Therefore, is the classification and the previous test (or reference test) percentage based on opening time or closing time? (I #34, Jeff Neyhard, Nine Mile Station)

Response

If the valve performs a safety function in both positions, and the stroke time in one direction is less than two seconds, then for that stroke direction, the licensee may use either the acceptance criteria of ASME Code or the staff's position for rapid acting valves. Where the stroke time for the valve in the other direction is greater than two seconds, the acceptance criteria for that stroke time range, as identified in the Code, should be followed when testing the valve in the greater-than-two-second direction. Similarly, the alternative concerning measurements of changes in stroke time allowed by Generic Letter 89-04 may be used for the stroke direction that has a stroke time of less than ten seconds. (NOTE: Although both MOVs and power-operated valves are mentioned, the question is more applicable to air-operated valves. Normally, MOVs do not have widely different stroke times for the open and close directions.)

Position 7, Testing Individual Control Rod Scram Valves in BWRs

No questions.

Position 8, Starting Point for Time Period in TS ACTION Statements

Question 42

10 CFR 50.55a(g) states that IST programs comply with Section XI. Section XI states for valves that "If the condition is not, or cannot be, corrected within 24 hours, the valve shall be declared inoperative." This is in direct disagreement with the Generic Letter which states that the LCO must be declared immediately. How do you justify this disagreement with the Code? (I #5, Dave Wallace, Fitzpatrick)

Generic Letter 89-04 implies that the 24 hour time period for declaring valves operable versus inoperable does not apply. Can the utility continue to use the 24 hours before declaring a valve inoperable? (I #27, Jeff Neyhard, Nine Mile Station)

Position 8 specifically states that licensees cannot use the 24-hour grace period for declaring a valve inoperable (IWV-3417(b)) and must make such declaration immediately upon recognition of exceeding a stroke time limit. Position 5 states that the intent of developing more restrictive stroke time limits is to identify a valve problem "before the valve reaches the point where there is a high probability of failure to perform if its safety function is called upon. Per Position 5, exceeding the more restrictive limit does not imply that the valve is inoperable but that the probability of failure is increased. With this philosophy, the 24-hour grace period is even more reasonable. (II #8, John Zudans, Florida Power and Light)

This question is in reference to Item 8 of Attachment 1: "Starting point for time period in Technical Specifications ACTION statement." This item eliminates the 24-hour clock for valves which exceed Section XI limits. In most cases, the Technical Specifications limits are higher than the Section XI limit. This item needs discussion. (II #15, John Kin, Virginia Power)

Response

The Standard Technical Specifications in Section 4.0.5 specifically state that the more restrictive requirements of the Technical Specifications take precedence over the ASME Code. For example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable. That definition takes precedence over the ASME Code, which allows up to 24 hours before declaring inoperable a valve that (1) is incapable of exhibiting the required change of disk position or (2) has exceeded its limiting value of full stroke time. Therefore, if a valve is tested and the data indicate that it is inoperable as defined by the required action range, then that valve must be declared inoperable at that time and not 24 hours later. This elimination of the 24-hour grace period before declaring a valve inoperable is consistent with the requirements of ASME Standard OM-10.

Question 43

When a piece of equipment enters the required action range, why must the Tech Specs action statement be entered without some time to reflect on why it has entered the required action range? A reasonable approach would be to establish a limited reflection time, for example the existing shift, to review how the test was conducted and review previous tests to see what the problem is. In declaring equipment inoperable when it really may not be upon review of how the test was conducted, generates needless paperwork and impacts INPO avail-ability statistics (i.e., HPCI, RCIC, RHR). (I #28, Bob Binz, PSE&G Hope Creek)

Response

For some time, the NRC staff has been concerned with the unrestricted grace period for declaring a component inoperable allowed by the ASME Code. One example of this grace period is the 24-hour delay allowed by IWV-3417 of Section XI following a failure of a valve to exhibit the required change of disk position. The staff's concern in this area has been expressed to individual licensees on many occasions. In order to provide guidance that is consistent with the Standard Technical Specifications and that can be applied generically, the staff developed Position 8 of Generic Letter 89-04 which states that the unrestricted grace period in the ASME Code is unacceptable. Once a component is declared inoperable, the action statement in the Technical Specifications would provide time for evaluation of the situation, including performing the test, before change is required in plant operating mode. A licensee may propose alternatives to the NRC staff's position. For example, a valve stroke time that is less than the limiting stroke time could be established as an alert time. If the alert time is exceeded and the limiting time is not, the licensee would initiate a 24-hour period for evaluating the condition of the valve before declaring it inoperable.

Question 44

Address the conflicts between the background of the generic letter which states "The intent of testing is to detect degradation affecting operation and assess whether adequate margins are maintained" and Position 8 regarding the starting point for Technical Specification ACTION statements. This will require declaring components inoperable which are capable of fulfilling their safety function (i.e., operable). (II #33, Philip J. North, Duke Power)

Response

The staff does not see a conflict between the statement in the background and Position 8 of Generic Letter 89-04. Testing is intended to detect degradation of a component and to provide assurance that adequate margins are maintained. Where testing indicates that a component has undergone such degradation that its operability is in question (e.g., the limiting value of full stroke time for a valve has been exceeded), Position 8 of the generic letter requires that the component be declared inoperable.

Question 45

Referring to paragraph 8, after testing a pump and declaring it inoperable, is it acceptable to replace the process instruments with test instruments which are more accurate then retest, rather than recalibrating process instruments? (IV & V #14, Arkansas Nuclear 1 and 2)

Response

Accuracy of the instrumentation is an important consideration in the performance of a test. In addition, the test must be performed in a manner that allows the test results to be compared for trends. This consistent performance of a test is sometimes referred to as "repeatability." Where instruments with different characteristics (such as with respect to range and accuracy) are used for each test, the ability to monitor the results for trends may be lost. Therefore, the staff prefers that the same set of instruments be used in performing tests on a particular component. This can be accomplished most readily by use of properly calibrated process instruments installed in the system. The installation of test instrumentation that are more accurate than the process instruments is allowed by the ASME Code. For the example cited by the question, after declaring the pump inoperable because of the test results from process instruments, the operability of the pump may be verified by more accurate test equipment. Because the same instruments should be used for tests to monitor the results for trends, the licensee should recalibrate the process instruments for their continued use or should establish a procedure to use the more accurate test instruments from that point forward.

Question 46

In reference to Item 8 of Attachment 1, it states that the provisions to recalibrate in IWP-3230(d) can only be done after the component is declared inoperable. What if, during a pump test, before test data is taken, it is clearly observed that a gauge is malfunctioning. Do I need to declare the pump inoperable, or can I stop testing and recalibrate?
(IV & V #36, Ken Trippel, Houston Lighting & Power/South Texas Project)

If it is obvious that a test has been run incorrectly (i.e., a recorded parameter is out of the range of the device being tested), do we still enter the action statement before re-running the test? (I #26, Bill Kittle, PSE&G - Salem)

Response

If a test is under way (regardless of whether test data have been taken) and it is obvious that a gauge is malfunctioning, the test may be halted and the instruments should be promptly recalibrated. One example might be a wildly fluctuating gauge. It should be noted, however, that, in many situations where anomalous data are indicated, it may not be clear that the problem lies with the gauge. In these cases, the licensee should attribute the problem to pump performance. The licensee would then declare the pump inoperable and evaluate the condition of the pump during the time allotted by the applicable Technical Specification.

Position 9, Pump Testing Using Minimum-Flow Return Line or Without Flow Measuring Devices

Question 47

With reference to the Generic Letter item 9, in cases where only the minimum flow return line is the available path, would the generic letter be revised to consider reducing the 5 minute time required for stabilizing the pump as required by IWP-3500(a) to a lesser time such as 2 or 3 minutes in order to minimize the possibility of pump damage occurring during the pump's operational test? (I #20, Al Koehl, NES)

Response

The staff does not intend to revise Generic Letter 89-04 to change any current positions or to address additional issues. If there is a problem concerning compliance with the ASME Code, requests for relief from the Code may be submitted.

Question 48

If mini-flow recirculation lines are instrumented for flow, are quarterly tests alone, which measure flow, differential pressure, and vibration, acceptable? (IV & V #18, Waterford 3)

Response

Mini-flow recirculation line tests are not prohibited by Section XI of the ASME Code. The staff, however, believes that a mini-flow test can be detrimental to a pump and is not a desirable test configuration. These tests produce data of marginal value and provide little confidence in the continued operability of the pump. The staff would prefer a more comprehensive test performed at some reduced frequency rather than relying only on the mini-flow test that is performed quarterly. This particular issue may be a topic of another generic letter addressing inservice testing in the future.

Question 49

Many mini-recirculation lines have no means to adjust flow to a reference value prior to taking data. Thus, this recirculation flow is relatively fixed. Since Table IWP 3100-2 limits are placed in differential pressure, what criteria should be used to place limits on flow? Even with a fixed-flow system, measured flow will demonstrate some variation test-to-test due to instrument repeatability, operator interpolation of needle position on meter face, etc. Table IWP 3100-2 limits do not seem appropriate for flow in this case. To allow both flow and differential pressure to vary within 13% ranges does not appear to meet the intent of Section IWP. (IV & V #19, Waterford 3)

Response

In most cases, mini-flow recirculation lines do not have flow adjustment capability. The ASME Code recognizes this in IWP-3110, which permits the use of one or more fixed sets of reference values for pump testing. The Code identifies acceptance criteria for both differential pressure and flow rate in Table IWP-3100-2. It is not permissible for both parameters to vary during a test. With one parameter set at a reference value, the other parameter is compared to the acceptance criteria.

Question 50

It is more desirable to test pumps at substantial flow conditions than on mini-recirculation lines. Should entire trains of safety systems be declared inoperable and 72 hour action statements entered solely to realign these systems for inservice testing? Does the obtaining of "better" pump data justify the increased risk to the public during the time the system cannot perform its safety function? (IV & V #20, Waterford 3)

Response

As stated in the question, it is more desirable to test pumps with substantial flow than in mini-flow recirculation configurations. The NRC staff, however, does not agree with the questioner that the performance of inservice testing results in increased risk to the public. Inservice testing is intended to provide assurance of the continued operability of pumps and valves. To provide this assurance, it is considered acceptable for a Technical Specification action statement to be entered on infrequent occasions in order to test a component. Where a system must be taken out of service to perform a test, it is likely that, in the event of a plant emergency, the system could be realigned for operation in short order. Where one train of a safety system will be disabled for an extended period or both trains of the system must be made inoperable to perform a test, the licensee should propose a testing schedule that provides for verification of component operability with testing performed during period (e.g., refueling outages) when availability of the system is not essential to plant safety.

Position 10, Containment Isolation Valve Testing

Question 51

In regard to Attachment 1, Position 10, why can't valves other than containment isolation valves (CIVs) that are 6 inches or larger be exempt from the needless requirement of IWV-3427(b)? (I #40, J. W. Connolly, Seabrook Station)

Does the exemption from IWV-3427(b) pertain to pressure isolation valves (PIVs) as well as Appendix J valves? (II #4, John Zudans, Florida Power and Light)

Do PIVs have relief from IWV-3427(b)? Item 10 on Attachment 1 only discusses CIVs (III #46)

Response

The relief from IWV-3427(b) of the ASME Code granted through Generic Letter 89-04 only applies to CIVs under containment leak rate testing. This position was written in response to numerous relief requests concerning CIVs from licensees that cited difficulties in trending leak rate data. We were not aware of similar difficulties with PIVs during reactor coolant system leak testing. The relief from the explicit requirements of IWV-3427(b) should not be taken as an indication that the NRC staff is disregarding the value of trending CIV leak testing data. Until more information is available on appropriate leak rate limits and on reasonable scatter of data, however, Position 10 will remain in effect for CIVs. The NRC staff anticipates developing a more comprehensive position of the subject in a future generic communication to licensees.

Position 11, IST Program Scope

Question 52

IWV-1200 specifically exempts control valves from testing. Why are these valves included in the list of examples in IST program scope as part of Attachment 1? (I #6, Dave Wallace, Fitzpatrick)

Response

IWV-1200 of the ASME Code does not exempt valves that have a required safety function from the provisions of Section XI. Code interpretation XI-1-83-59 states that it is a requirement of Section XI that flow control valves that have one or more defined safety-related functional requirements be classifieed Category A or B, as applicable, and tested in accordance with the requirements of Subsection IWV. This philosophy applies to all control valves that have one or more defined safety-related functional requirements.

Question 53

Please clarify the last three lines of Generic Letter item 11 of Attachment 1. (I #10, Shafi Rokerya, New York Power Authority)

The scope statement of Position 11 is much too vague. The position with respect to program scope must be clarified and explained to provide further guidance and should also address the backfit issue. In addition, in the past, it has been the practice of adding additional components to the scope of IST Programs via the authority of 10 CFR 50.55a(g)(ii). How will this be addressed in the future? (II #5, John Zudans, Florida Power & Light)

Do safety-related components outside of Class 1, 2, and 3 need to be tested in accordance with the Code and be included in the IST program, or is it the intent to have some form of testing to demonstrate operability. (III #29, Vince Treague, Point Beach)

In reference to Item 11 of Attachment 1, please clarify the intent of the last sentence of this item: "Therefore, while 10 CFR 50.55a delineates the testing requirements for ASME Code Class 1, 2, and 3 pumps and valves, the testing of pumps and valves is not to be limited to only those covered by 10 CFR 50.55a." (IV & V #10, T. f. Hoyle, Washington Nuclear 2)

How will the NRC review pump and valve testing not included in the scope of the IST program? Will the ASME Code requirements be applied to these components? (IV & V #15, Arkansas Nuclear 1 and 2)

Response

Criterion 1 in Appendix A to 10 CFR Part 50 requires, among other things, that components important to safety be tested to quality standards commensurate with the importance of the safety functions to be performed. Appendix B to Part 50 describes the quality assurance program, which includes testing, for safety-related components. Paragraph (g) of 10 CFR 50.55a requires the use of Section XI of the ASME Code for inservice testing of components covered by the Code. For other components important to safety, the licensee also has the burden of demonstrating their continued operability. The list provided in Position 11 contains examples of components that have been shown by our experience to be frequently omitted from a routine testing program. The licensee should review the safety significance of these identified components to ensure that the inservice testing is adequate to demonstrate their continued operability. NRC inspectors will evaluate the adequacy of such testing. The Code-required IST program is a reasonable vehicle to provide a periodic demonstration of the operability of pumps and valves not covered by the Code.
If non-Code components are included in the ASME Code IST program (or some other licensee-developed inservice testing program) and certain Code provisions cannot be met, the Commission regulations (10 CFR 50.55a) do not require a "request for relief" to be submitted to the staff. Nevertheless, documentation that provides assurance of the continued operability of the non-Code components through the performed tests should be available at the plant site.

Question 54

The Diesel Generator air start system direction that was in the initial draft of Generic Letter 89-04 has now been dropped. Can we remove the testing from our program? (Not that we would, I feel it is a good practice) (I #22, Jeff Neyhard, Nine Mile Station)

In Position 11, why were the emergency diesel generator support system components deleted from the list in the final version of the letter? (II #3, John Zudans, Florida Power & Light)

Response

Typically, the Emergency Diesel Generator air start system is not Code Class 1, 2, or 3 and, therefore 10 CFR 50.55a does not require the testing of these components to be performed under the provisions of the ASME Code. Emergency Diesel Generator air start, cooling water, and fuel oil transfer systems, however, are considered safety related. As such, Appendices A and B to Part 50 require that they undergo component testing.

Question 55

Are the items listed in Attachment 1 number 11c, d, and e specific to PWRs? (I #24, John Lindburg, PP&l)

Response

The listed items were not intended to apply to every plant. Each licensee should review the list and determine those items applicable to its facility. In response to the specific question, items 11c, d, and e do not apply to BWRs.

OTHER QUESTIONS DURING GENERIC LETTER 89-04 MEETINGS

Schedule for Implementing the Generic Letter

Question 56

The scope of the Generic Letter is broad and requires more than the allotted 6 months for response. What guidance can be given for extension of the response date? (I #8, Dave Wallace, Fitzpatrick)

How much is expected to be done at the end of 6 months? (I#50)

What is the schedule requirement for implementing additional or revised testing arising from the activities related to the generic letter? Keep in mind that the results of reviews and evaluations must be available prior to revising and implementing the related procedures. (II #9, John Zudans, Florida Power & Light)

Do the requirements to conform to the stated positions of the generic letter within 6 months of the date of the letter mean that all procedures have to be revised and approved within this 6 month period, or is it acceptable to have procedures in the process of being revised within the 6 month period? (III #15, M. H. Richter, Commonwealth Edison)

Due to outage schedules and constraints, are there any provisions for not completing all equipment modifications within 18 months of the date of confirmatory letter, or the first scheduled refueling outage following the confirmation letter? (III #16, M. H. Richter, Commonwealth Edison)

How are extensions of the October 3, 1989 deadline viewed; what factors are considered on such requests? (III#21, Point Beach Nuclear Plant)

Do utilities have to contact their Project Managers to schedule immediately a meeting to resolve any requested relief requests outside the generic letter (prior to required test frequency) to obtain approval and avoid violation after submittal, or will there be a grace period? (III#41)

Response

With regard to plants not listed in Table 1 or 2 of Generic Letter 89-04, the intent has been that, by the end of six months, (1) the IST program would be revised to incorporate all the requirements of the generic letter, (2) the procedures would be written and implemented, (3) the confirmation letter and any necessary additional relief requests would be submitted to the NRC, and (4) a schedule would be provided for any plant modifications necessary to comply with the requirements. It has been additionally intended that any necessary equipment modifications be completed within 18 months of the date of the confirmation letter or the first scheduled refueling outage following the confirmation, whichever occurs later.

We have received several comments stating that this schedule may not be achievable. For example, one licensee noted that acceptance criteria need to be developed before procedures can be prepared and implemented. Following preparation of the procedures, several weeks were said to be needed to provide the necessary training to plant personnel on various shifts. Another licensee indicated that the resources necessary to implement the generic letter had to be determined to justify to management the need for contractor assistance. Even where licensee management accepts the justification for contractor assistance, it was said that few highly qualified contractors in the area of inservice testing are available. With respect to equipment modifications, one licensee hypothesized a situation where a refueling outage began soon after the confirmation letter and the next refueling outage would be a month or two beyond the 18-month limit.

Several reasons that the NRC staff does not consider sufficient to justify not meeting the schedule in the generic letter were also given by meeting attendees. These insufficient reasons include (1) the lack of activity relative to Generic Letter 89-04 until the NRC meetings took place and (2) the lack of a designated individual responsible for IST at the plant when the generic letter was issued. If any particular plant anticipates a problem in meeting the schedule, this should have been discussed with the NRC Project Manager. In determining the necessary schedule extensions, licensees should have limited the request for schedule relief to the smallest set of revisions to the IST program and procedures, and modifications to equipment. The information submitted to the NRC by the licensee to justify a delay in meeting the schedule established in Generic Letter 89-04 should have contained at least (1) a description of the actions to be completed by October 3, 1989, including an interim schedule of accomplishments by system and component, (2) a description of the action for which an extension in the schedule is being requested with the specific proposed schedules for the program, procedures, and any necessary equipment modifications, and (3) a discussion of the specific reasons for the need to extend the schedule, including the hierarchy of the proposed schedule extensions as established by their importance and dependence on the completion of other actions.

Question 57

Does the NRC expect the licensee to take any specific action prior to receipt of the SER? (IV & V #1, T. F. Hoyle, Washington Nuclear 2)

Is it the intent to have all implementing procedures of changes required by Attachment 1 be completed within 6 months? Does this apply to Table 1 and Table 2 plants? (IV & V #6, T. F. Hoyle, Washington Nuclear 2)

Response

The positions in Generic Letter 89-04 address both program and procedural issues. Positions 4, 5, and 8 are related to procedures and would not be covered by a review of the IST program. The remainder of these positions are related to both the IST program and the procedures. For Table 1 plants, we believe that it would be reasonable for the generic letter provisions to be implemented within six months of issuance of the SER. The precise schedule, however, will be specified in the SER. The schedule for Table 1 plants is keyed to the SER because the licensee needs an opportunity to review the SER before having to commit to an implementation schedule. Nevertheless, the staff encourages Table 1 plants to begin verifying that plant procedures are consistent with the generic letter before receipt of their SER. Table 2 licensees should verify that plant procedures are consistent with the generic letter positions within six months of issuance of Generic Letter 89-04.

Confirmation Letter

Question 58

With our confirmation letter will be a couple of relief requests. How will they be handled? Can we assume relief is granted? Do we have to wait for your SER? (I #30, Joann West, Beaver Valley)

What is the level of information expected in the response to the generic letter? How detailed must it be? (II #22, Garry Galbreath, Duke Power)

Is "relief" required for items per Generic Letter 89-04 which differ from the ASME Code? (III #22, Point Beach Nuclear Plant)

Response

A confirmation letter from a particular licensee may contain several forms of information, depending on the IST program. The confirmation letter should address the extent to which the licensee's program and procedures meet the positions attached to Generic Letter 89-04. It is anticipated that most licensees will have to modify their IST programs as a result of the generic letter. The revised program should accompany the confirmation letter. In cases where a generic letter position that approves an alternative to the ASME Code is being followed, a relief request is not required, but the deviation from the Code should be documented in the IST program along with its method of approval (i.e., through the relevant generic letter position). As a suggestion, licensees may reserve the use of the term "relief request" for those cases where specific staff review and approval are needed before implementation.

If a licensee cannot meet one of the generic letter positions, an alternate test method may be performed, providing the provisions of Paragraph B of the generic letter are met. This Paragraph B approach for generic letter positions does not require a relief request but the justification should be retained in the IST program. In that the generic letter does not supersede the regulations in any way, the option still exists to submit requests for relief from the Code for program-related positions in the generic letter. For plants not listed on Table 1 or 2 (i.e., plants that will be submitting a confirmation letter), any requests outside the scope of the generic letter that were submitted before April 3, 1989 are approved by the issuance of the Generic Letter. If a relief request is submitted after April 3 or a relief request submitted before April 3 is modified, the requested relief may not be implemented until receipt of staff approval. The date by which these relief request approvals are needed should be specified in the confirmation letter so that their review may be prioritized.

Verification of Generic Letter Implementation

Question 59

When and how is guidance going to be provided to the Regional offices on inspection and enforcement of the issues stated in the Generic Letter? (I #3, Dave Wallace, Fitzpatrick)

Regarding the approval of the IST Program scope and related relief requests, it appears that NRC is not planning to perform detailed review and is merely stating that their responsibility re. 10 CFR 50.55a is satisfied by the generic letter supplemented by plant site inspections. This eliminates the pre-approval discussions done previously; however little guidance is provided to give licensees' confidence that the subjective opinions of the various inspectors can be anticipated before the fact. It would help if there were some mechanism whereby a utility could receive an official opinion/determination with respect to program scope and relief request queries in a timely manner. (II #6, John Zudans, Florida Power & Light)

With respect to inspections, will there be an inspection module developed, or is this to be an "ad hoc" type of inspection? (II #27, Ron Jacobstein, Florida Power & Light)

To what extent is the NRC planning to make their guidance uniform policy for all inspections? It is very important that uniform policy be applied at all facilities, regardless of the composition of inspecting teams. (II #38, John Zudans, Florida Power & Light)

Many alternatives that are given seem vague and subject to interpretation. Who decides adequacy and what are the ramification of differences between licensees and the NRC? (III#23, Point Beach Nuclear Plant)

What guidance will Region/NRR auditors use in accessing IST Programs for Table 1 or 2 plants? Will they use the SER or the generic letter? (IV & V #3, T.F. Hoyle, Washington Nuclear 2)

Response

The NRC staff has been performing activities to provide assurance that application of the generic letter by the inspectors will be consistent. For example, a meeting to discuss the generic letter was held in Rockville, Maryland, in April 1989, and each NRC Region office was represented. A temporary instruction (TI) will be written by NRC/NRR, providing guidance to the regional inspectors on prioritized inspection activities for IST and the Generic Letter 89-04. It is intended that the TI will be completed in six to eight months. Periodic NRR/Region counterpart meetings will be held to ensure consistency on the IST subject matter. Additionally, the inspection teams are expected to be made up of NRC/NRR, NRC Region, and contractor personnel, thereby providing for consistent communication. These inspections will assist the staff in verifying the adequacy of the IST program rather than verifying adequacy by the traditional staff review. It is intended that the inspectors will rely on the generic letter, the temporary instruction, and the particular SER for Table 1 and 2 plants. These inspections will not be performed on an ad hoc basis. Although only relief requests will receive NRC review before their implementation, licensees may direct questions concerning interpretation of requirements on the IST program and procedures to the NRC staff through their Project Manager.

Question 60

If the SER does not constitute NRC concurrence that the generic letter requirements (at least those that are routinely addressed in the program submittal) are met, then how will issuance of SERs to Table 1 or Table 2 plants constitute NRC approval of the IST program? (II #19, Sid Burns, Alabama Power Company)

Will all SERs issued in the near future, or recently issued, incorporate all the issues in the generic letter? (II #41)

Response

It is recognized that the positions in Generic Letter 89-04 go beyond the areas covered by past SERs on inservice testing. Positions 4, 5, and 8 deal with procedural matters that are not reflected in the IST programs and SERs. Therefore, it cannot be expected that an SER would constitute concurrence that all of the generic letter positions have been met. The SERs for Table 1 and 2 plants explicitly contain approval only for relief requests. These SERs can be considered as providing IST program approval only in that the practice has been to perform a thorough review and identify problem areas that need resolution.

Updates and Revisions of the IST Programs

Question 61

If relief requests exist that do what one, or any, of the positions state, should these requests be retracted with the confirmation/resubmittal? (II #29, Jim Holton, Florida Power Corp.)

Do "changes to the program" include administrative changes such as referencing different procedures, or just intent of program? (II #32, Jim Holton, Florida Power Corp.)

In instances when a licensee modifies their IST program beyond that currently submitted to the NRC, [as discussed in] Paragraph D of the generic letter, and reviews the modification against the positions found in Attachment 1, is it required that the IST program modifications be submitted to the NRC? (III #14, M. H. Richter, Commonwealth Edison)

Our plant is on Table 1. We have revised the program to identify Generic Letter 89-04 as a reference and made some minor changes consistent with the letter. Do we need to resubmit the program? (III #26, Steve Bell, Illinois Power)

Are all future revisions to the IST program required to be submitted to the Commission? Section D of the generic letter is silent on this subject. (IV & V #5, T. F. Hoyle, Washington Nuclear 2)

Does the generic letter mean that program submittals are no longer required? Under what circumstances are submittals still required? (IV & V #12, Arkansas Nuclear One)

Should we provide changes to the NRC as soon as made even if numerous "trivial" or "typo" changes are being issued? What about the "complete and accurate" requirement in 10 CFR 50.9? (IV & V #30, Paul Croy, Southern California Edison/San Onofre)

Should updated plans document specific relief requests that were approved on a prior date? (IV & V #34, Alan Harris, Waterford 3)

Since programs are revised frequently and in a piece-meal fashion, does the NRC expect each change to be submitted as soon as it's made, or is once per year, once per two years, etc. adequate? (IV & V #35, San Onofre 1)

Response

The NRC staff should have the current IST program being implemented at each plant even if this means that a licensee sends multiple submittals to the NRC each year. The most up-to-date version of an IST program will not be used for the purpose of the staff performing complete program reviews as has been done in the past. Rather, it is needed to prepare for IST inspections and to assist in the review of relief requests. The staff would prefer to have a complete program rather than individual changed pages. The identification in the program of the mechanism for approval of specific relief requests would be particularly helpful. That is, the program should indicate whether the approval is (1) through a position in Generic Letter 89-04, (2) by virtue of the relief request being outside the scope of the positions in the Generic Letter and submitted before April 3, 1989, (3) through the mechanism described in Paragraph B in the generic letter, or (4) obtained using a relief request that will need staff approval by a specific date. Currently-approved relief requests that follow a generic letter position should not be retracted but the source of approval (i.e., the generic letter) should be identified in the IST program. Non-technical and minor typographical changes may be held until the licensee has collected several such changes. This is considered to meet the intent of 10 CFR 50.9 for complete and accurate information. For plants not listed in Table 1 or 2, revisions to the IST program should be sent when the confirmation letter is submitted.

Question 62

If valves are added to or removed from the system, does the change to the program require resubmittal? (II #32, Jim Holton, Florida Power Corp.)

Can components be deleted without prior NRC approval? (III #45)

Response

Neither the Commission regulations in 10 CFR 50.55a(g), in general, nor Generic Letter 89-04, in particular, require the licensee to obtain NRC approval on each test on every component in the IST program. As long as the program is consistent with the regulations, the ASME Code, and the Generic Letter, relief is not required. To amplify, deletions from or additions to the IST program do not necessarily require NRC approval. The burden is on the licensee to verify that their IST program is complete and all components that require IST are included and tested to the extent practical. If a particular component is deleted from the IST program, documentation of the reason in an appropriate place is recommended.

Question 63

Please clarify the intent of the last sentence of [Section D]: "The modified program should comply with the disposition of relief requests in any applicable SER based on a previously submitted IST program." The sentence quoted above seems to apply to Table 1 or Table 2 plants only. Also, the sentence seems to allow the use of an extension of a previously granted relief request. (IV &V #4, T. F. Hoyle, Washington Nuclear 2)

Response

Section D of the Generic Letter 89-04 applies to all plants. Previously approved relief requests remain valid. However, if a relief request has been denied in an SER, the SER usually provides information on the reason the relief request was denied and recommendations on appropriate actions for the licensee. The last sentence of Section D is indicating that these recommended actions should be followed.

Question 64

It is clear that if an NRC position is covered by Attachment 1, then the licensee must either comply with or follow the alternate provisions contained in Section B of the generic letter. But for program changes not covered by Attachment 1, [Section D] states that the provisions of 10 CFR 50.55a(g) should be followed. This infers that a relief must be submitted. Further, in accordance with the plant Technical Specifications, relief must be granted prior to implementation. (IV &V #4, T. F. Hoyle, Washington Nuclear 2)

Response

It is correct that, where an IST program change is proposed that is outside the scope of the positions in the Generic Letter and does not meet the Section XI requirements, the licensee must submit a relief request to the NRC for review. The program change may not be implemented prior to staff approval.

Question 65

For plants with SERs, can changes to NRC reviewed and approved programs be made without additional submittals to the NRC? What if changes are in accordance with the generic letter? (IV&V#13, Arkansas Nuclear 1 and 2)

Response

As described in the response to Question 61, licensees need to send any changes to their IST program to the NRC. If these changes are in conformance with Generic Letter 89-04, NRC review and approval are not necessary. The IST programs submitted to the NRC as a result of program changes should indicate the reasons for the changes and the relief requests, if any, that require staff review.

Relief Requests

Question 66

If a relief request issued for one unit has been approved, can, or will the turnaround time for approval of the same relief request on a second unit (for a two unit plant) be reduced? (II #18, Herbert P. Walker, Georgia Power/Vogtle Project)

For future relief requests outside the scope of Attachment 1, what is the perceived ability of the NRC regarding turnaround time? (II#23, Garry Galbreath, Duke Power)

Response

New relief requests will be evaluated on a priority basis. Therefore, the licensee should specify the date by which the relief is needed, and where possible, should provide additional information to assist in this review, such as "this relief request is identical to relief request number X in the Unit 1 IST program." The staff recognizes that, on occasion, there will be a need for rapid NRC response. The staff will make every available effort to be responsive to such needs.

Question 67

If revised relief request submittals are not considered approved, then do we continue working to the presently approved request? (II #30, Jim Holton, Florida Power Corp.)

Response

The approved relief request is controlling until the licensee receives approval of a revised relief request. As we have indicated above, if plant operations and ASME Code requirements dictate relief request approval by a certain date, the licensee should indicate that date in the submittal containing the relief request.

Question 68

Does a relief request that is grandfathered but no longer required still need approval? (II #44)

Response

By grandfathered relief request, we assume that the question is referring to a relief request not covered by the positions in Generic Letter 89-04 but submitted before April 3, 1989. Withdrawal of relief requests, regardless of the prior approval status, is permitted without NRC review, presuming the IST program remains consistent with the regulations, the ASME Code, or Generic Letter 89-04.

Question 69

Is a continuous feedback system required to provide a mechanism to reverify that relief requests are still valid based on ongoing maintenance and plant modification activities? (III #52)

Response

The licensee is expected to have a feedback system that will maintain the IST program as a living document that will be updated to be consistent with changes in plant configuration. If a particular relief request is no longer required because of changes in hardware, system design, or new technology, the licensee is expected to revise the program to withdraw the relief request. Conversely, if a system modification results in the addition of a component to the IST program, the feedback system should ensure that the Code requirements or Generic Letter 89-04 provisions are met, or that a relief request is submitted, as appropriate.

Question 70

Relief request requirements are changed in the Generic Letter. Previously approved relief requests are now being challenged because the NRC uses a different reviewer. This appears to be a backfit issue. (I #4, Dave Wallace, Fitzpatrick)

If relief was granted by the NRC for an item during the first interval, is the same relief granted during the second interval even though the relief is not in compliance with GL 89-04? (I#33, Joe Bashista, TMI-1)

In the 1st 10 Year submittal, an SER approved a relief request which is not consistent with the alternative positions in Generic Letter 89-04. Does the generic letter void previously approved alternatives/relief requests (via an SER) or may these alternatives/relief requests not consistent with Generic Letter 89-04 still be considered valid and so documented in the IST program? (III #31, Toledo Edison)

When will it be known what the staff's position is on SER approved relief requests that contradict Generic Letter 89-04 dictated testing? (III #33, Gary J. Roesner, Callaway Nuclear)

Response

We assume that the questions are not referring to interim reliefs but rather relief requests on which the NRC staff prepares an SER. Assuming that the reviewed information was complete, accurate, and remains up-to-date, an approved relief request may be currently followed even if it conflicts with the Generic Letter. These types of situations will be reviewed in preparation for inspections. Safety significant differences between the approved relief request and the Generic Letter will be discussed in an effort to obtain licensee agreement to adopt the Generic Letter position. Where agreement cannot be reached, the staff may consider initiation of backfit procedures. Relief requests are subject to review by the NRC staff at the ten-year update for consistency with current NRC regulatory positions, including those contained in Generic Letter 89-04. Reliefs that are inconsistent with the generic letter would likely not be approved for a succeeding ten-year interval.

Question 71

What is the long term status of the "relief" system? (III #22, Point Beach Nuclear Plant)

Response

The section of the Commission's regulations pertaining to the relief request system is 10 CFR 50.55a. This regulation is not, and cannot be, superseded by Generic Letter 89-04. A revision to this regulation is under consideration. With respect to the "relief" system as described in the regulation, the staff may, at some time in the future, issue additional guidance to provide a pre-approval mechanism much as the generic letter does in certain of its positions.

Question 72

To conform to generic letter positions, what does "document in the program" mean? Should relief requests be generated with the understanding that the generic letter grants them? Or does a statement included in the program describing how the deviation conforms to the generic letter suffice? (IV & V #21, Waterford 3)

Response

The IST program should include the deviation from the ASME Code that the licensee intends to take, and the basis for the change just as a program would normally contain. There should be sufficient information in the program to demonstrate that Generic Letter 89-04 is applicable to the situation in question and that the testing being performed conforms to the generic letter.

Question 73

Is the following statement correct? A relief request submitted prior to April 3, 1989 but not discussed in any SER and is not a subject of generic letter attachment 1 is approved for use without any further utility reviews. (III #49)

Response

Relief requests that were on the docket before April 3, 1989, for plants that are not in Table 1 or 2 in Generic Letter 89-04 and are topics that were not discussed in Attachment 1 are approved by this generic letter. Any relief requests outside of the Generic Letter positions that are submitted after April 3, 1989, will require staff review and approval before implementation. The response to Question 74 explains the basis for this approach. Other statements regarding utility's required actions for the review of implementing procedures additionally apply.

Question 74

What is the NRC's basis for stating that approval is by virtue of the generic letter for previously submitted relief requests when such reliefs could be outside the scope of the positions in the generic letter and have not undergone NRC review? (III #37, Brent Metrow, Illinois Dept. of Nuclear Safety)

Response

From the general knowledge of the relief requests, the NRC staff selected the technical issues considered the most significant to be addressed by Generic Letter 89-04. The NRC staff checked a sampling of the current IST programs to provide confidence that those issues not addressed in the Generic Letter were not highly safety significant. Additional issues that would require the NRC staff to perform a detailed regulatory analysis may be addressed in future generic guidance.

Question 75

Regarding a multi-unit site, if one unit has an approved SER which grants relief on items which do not meet all the criteria of the generic letter, can the approved SER provide a basis for the other unit to go ahead and implement the relief request prior to NRC re-review (assuming design differences do not exist between the two units)? (III #48)

Response

When relief is granted in an SER for one particular unit on a multiple unit site, that relief applies only to that one unit even if the other unit is essentially identical. If an SER is written for two (or more) units, the relief would apply to all units specified in the SER. The SER for one unit may not be used as a basis for implementing the request before staff approval. See also the response to Question 66.

Question 76

If an SER that is received by a plant on Table 1 after the generic letter was issued denies a relief, and another plant that is not getting an SER has the same relief request grandfathered (approved), is this fair? (II #42)

Response

Such situations will be considered by the NRC staff when preparing for plant inspections. Safety significant differences between the approved relief request and Generic Letter 89-04 will be discussed at that time to try to obtain licensee agreement to follow the generic letter. If agreement cannot be reached, the staff will consider the need to initiate backfit procedures.

Question 77

Does the first sentence of [the IST PROGRAM APPROVAL] section apply to Table 1 and Table 2 plants? The last sentence infers it does not. (IV&V#9, T. F. Hoyle, Washington Nuclear 2)

Response

The first sentence of the "IST PROGRAM APPROVAL" section of Generic Letter 89-04 states that "[t]his generic letter approves currently submitted IST program relief requests for licensees who have not received an SER provided that they (1) review their most recently submitted IST programs and implementation procedures against the positions delineated in Attachment 1 and (2) within 6 months of the date of this letter confirm in writing their conformance with the stated positions." This sentence applies only to plants not listed in Table 1 or 2.

Question 78

In the approval process, when an SER conditionally gives relief and requires further plan changes, is an SER supplement provided, or is relief approved by letter, or is the relief granted based on conformance to the SER stipulation? (IV & V #32, Alan Harris, Waterford 3)

Diablo Canyon's SER grants several relief requests with conditions. We are revising reliefs to meet these conditions. Will we need NRC approval of revised reliefs prior to implementation? (IV & V #39, John Arhar, Pacific Gas & Electric/Diablo Canyon)

Response

If the conditional approval specifically identifies what must be done to obtain relief, then conformance with the condition is complying with the relief. A revised program should be sent to the NRC stating that the conditions have been met. In that case, a follow-up SER would not be issued. Where the relief request is denied and the staff asks for more information (e.g., additional analysis or basis), then a specific request must be made to the staff for its review and approval before implementation by the licensee.

Recent and Upcoming SERs

Question 79

For a Table 1 plant, can changes be made to the IST program in accordance with the generic letter, even though the SER has not been received? (II #35, Al Koon, South Carolina Electric & Gas/Summer Nuclear Station)

Response

Any licensee may revise its IST program to conform to Generic Letter 89-04. The licensee should provide changes to the IST program to the NRC as discussed in the responses to Questions 61 and 65.

Question 80

Will the implementation schedule for procedure changes and hardware changes be specified in the SER? Will this schedule be similar to the generic letter; e.g., will the licensee have six months to effect procedure changes and 18 months/next refueling outage to make hardware changes? (IV & V #2, T. F. Hoyle, Washington Nuclear 2)

Response

The implementation schedule for procedure and hardware changes will be contained in the SER. The NRC staff expects the schedules to be similar to those in the Generic Letter 89-04. See also the response to Question 57.

Question 81

Before the SER is issued or for the first six months thereafter, is it permissible for the licensee to use its current IST program as submitted to the NRC? (IV&V#3, T. F. Hoyle, Washington Nuclear 2)

Response

Licensees should use the current version of their IST program. The generic letter, in effect, provides interim approval of the existing program for Table 1 licensees until the SER is issued.

Question 82

If a plant with an SER on its IST program has a 10 year review up coming, how should that be handled? Resubmittal? (III #35, Gary J. Roesner, Callaway Nuclear)

Response

A plant with an SER that is preparing a revision for the 10-year update should revise the program to be in conformance with the provisions of Generic Letter 89-04. The licensee does need to submit the program update to the NRC. The program should indicate which relief requests require NRC review and approval and which relief requests are already approved through the generic letter. Staff review and approval of the unapproved relief requests are required before the licensee implements the new program.

Alternatives to Positions in the Generic Letter

Question 83

Are the new criteria always to be used even if it is not applicable? Can it be partially implemented if the licensee feels the relief request is sufficiently justified by specific in house experience? (I #4, Dave Wallace, Fitzpatrick)

Response

Certain positions in the Generic Letter 89-04 are not fully applicable to all plants. For example, the components listed in positions 3 and 11 are not applicable to all plants. Further, Position 7 is applicable only to BWRs. Alternatives to the positions of the generic letter, or partial implementation as this question suggests, should be justified in accordance with Paragraph B of the letter. Specific in-house experience is only one of the sources of information that should be utilized when evaluating alternative testing, and is not a substitute for the criteria in Paragraph B of the generic letter.

Question 84

Will any deviations from the requirements in the Generic Letter be reviewed and an SER issued for those relief requests? (I #42)

Is a relief request required when only 2 or 3 of the 4 items identified in Generic Letter Item B, page 3, can be met? (I #45)

Generic Letter 89-04 states in Paragraph B, that when licensees are unable to comply with the positions of Attachment 1, evaluation of alternate testing should address [four criteria]. Is it mandatory for each instance to address all 4 of the above items? In some instances or situations, the above items may not apply, or only a portion may apply. When evaluating an alternate test to one of the Positions of Attachment 1 of Generic Letter 89-04, may the alternate test be implemented without prior NRC approval providing an evaluation is performed and documented and retained in the IST Program? Does the documented alternative test evaluation in the IST program have to be formally submitted to the NRC as an IST program revision, and if so, in what time frame? (III #12. M. H. Richter, Commonwealth Edison)

On Page 2 of Ted Sullivan's review, he indicated that the NRC will not issue SERs in Attachment 1 items and justified alternatives. Are the justified alternatives the 4 points on past component history? Can I use these 4 alternatives to justify a deviation from the Attachment 1 positions? If so, are these then approved by the generic letter? After issuing a confirmatory letter, can I go through the above process to get "automatic" or pre-approval of Attachment 1 exceptions in the future? Can the 4 points be used for non Attachment 1 items following a similar process? (III#32)

For relief requests not covered by this generic letter, is (in accordance with Technical Specification 4.0.5) specific written approval required prior to implementation? (IV&V#8, T.F. Hoyle, Washington Nuclear 2)

Response

Assuming that Section XI will not be followed, Paragraph B of the Generic Letter 89-04 provides guidance for the situation in which a licensee is unable to comply with one of the positions of the generic letter because of design considerations or personnel hazard (as opposed to inconvenience). In such a situation, a licensee may develop an alternative testing method provided an evaluation is performed that addresses four specific criteria. The alternate test would not be acceptable unless the data associated with those criteria are sufficient to justify its adequacy for detecting degradation and ensuring continued operability. Where the four criteria are satisfied, the alternate test is considered approved by the generic letter and may be implemented. The specific justification is expected to be documented in the IST program submitted to the NRC, but need not be documented in the form of a relief request. This documentation will be subject to review for completeness, accuracy, and applicability during NRC inspections.

If at some time, the circumstances change such that the justification obtained through Paragraph B is no longer valid, then the licensee must submit a relief request for staff review before continuing the alternate test. Paragraph B may also be used when future revisions to the IST program relating to the generic letter positions are prepared. If all four criteria cannot be met, then a relief request must be submitted to the NRC and the alternate test method cannot be implemented until staff approval is received. For technical issues outside the scope of the positions in the generic letter, the alternative provisions of Paragraph B may not be applied and, in these cases, a relief request must be submitted for NRC approval before implementation.

Question 85

Since 10 CFR 50.55a(g) is a top tier document, is it still permissible to use its provisions of the relief request process when the requirements of the Code/generic letter cannot be met? Must these relief requests be approved prior to implementation in accordance with plant Technical Specification 4.0.5? If a required test cannot be done, should the utility use the exigency provision? (IV&V#7, T. F. Hoyle, Washington Nuclear 2)

Response

The provisions of 10 CFR 50.55a(g) remain available for the licensee's use for submitting relief requests and obtaining approvals. In accordance with the Technical Specifications, approval of relief requests is required before implementation. Relief requests should indicate the date by which approval is needed. Generic Letter 89-04 is providing another method of receiving approval of deviations from the ASME Code requirements. The licensee may prepare a case to justify postponement of a particular test on the basis of exigency. At this point, we are unaware of any aspect of Generic Letter 89-04 that would qualify for the exigency provision.

Question 86

Was the generic letter issued as opposed to changing the regulation? Prior to regulation changes, will comments be solicited from the licensees? (IV & V #12, Arkansas Nuclear 1 and 2)

Response

Generic Letter 89-04 is not considered an alternative to the regulation but is a vehicle to obtain preapproved relief from certain ASME Code requirements. If the regulation is changed, the normal rulemaking process will be followed and comments will be solicited.

Requests for Additional Information (RAI)

Question 87

How do plants which have received requests for additional information (RAI) from the NRC but are not on the list of plants to receive an SER get RAI items resolved that are not addressed in the Generic Letter? (I#1, Dave Wallace, Fitzpatrick)

Does the Generic Letter or the RAI take precedence and which one must be complied with? (I#43)

We received 86 questions (RAI from NRC) of which some were general in terms. A couple dealt with justification wording in which the questioner recommended a more detailed justification, although the alternate method would remain the same. Would we have to make these recommended changes and resubmit, or can we leave them alone? If revision is more of an administrative wording issue, then are they considered to require an SER? (II #31, Jim Holton, Florida Power Corp.)

What do I do about an RAI that I received prior to the generic letter and issues in the RAI are outside Attachment 1? (II#43)

Response

There are a small number of plants that have received RAIs and that have not had an IST review meeting to discuss the RAI. Utilities in this category are plants not on either Table 1 or 2 and that are expected to respond to Generic Letter 89-04 with a confirmation letter. Utilities that have received RAIs do not need to respond explicitly to the RAIs, but should use them to assist in responding to the generic letter. The RAIs provide an indication of possibly weak or questionable aspects of an IST program. For those cases where the intent of an NRC question is unclear, licensees may obtain clarification through the NRC Project Manager.

Question 88

Some questions in a recent RAI are in conflict with previously approved relief requests. Which one must be complied with? (I #44)

Response

Previously approved relief requests remain valid despite what might appear to be a conflicting position in an RAI. This statement assumes that the previously approved relief was granted on the basis of accurate and complete information available to the NRC staff at that time.

Modification of the Generic Letter

Question 89

Is a NUREG to be issued on this Generic Letter to clarify underlying issues? (I #7, Dave Wallace, Fitzpatrick)

Response

There is no current plan to prepare a NUREG document to clarify any underlying issues with Generic Letter 89-04. These minutes will be sent to all licensees and attendees who provided their address.

Question 90

Will Generic Letter 89-04 be updated from time to time to provide additional positions on IST programs in areas such as the following? The ASME Section XI Code does not require leak testing for valves where leakage is continuously monitored, however, for PWR plants the NRC often requires leak testing for Category A valves such as the RCS accumulator/core flood discharge check valves which are monitored continuously for seat leakage. (III #11, Larry Campbell, Toledo Edison)

Response

The staff has no plan to issue a supplement to Generic Letter 89-04. Another generic letter on IST may be issued in the future, but would cover new topics or expand on the current scope of components covered by the IST program required by the ASME Code. The Code does require that valves whose leak tight integrity is important fo performance of their safety function be individually leak rate tested. From the staff's experience, most continuously monitored leakage detection systems do not verify the leaktight integrity of each valve in the flow path and the staff does not consider these systems to meet the Code requirements.

Backfit Concerns

Question 91

The Generic Letter states that "In cases where conformance with the stated positions would result in equipment modifications, the licensee should provide in his conformation letter a schedule for completing the required modifications." The Generic Letter goes on to state acceptable schedules for completion of these mods. Are these modifications subject to the provisions of 10 CFR 50.109 backfitting? (I#2, Dave Wallace, Fitzpatrick)

Please confirm that the NRC's opinion and present position is that the generic letter is not considered a backfit for all utilities. (II#17, K. Jacobs, New York Power Authority)

Does the staff intend to do a backfit analysis regarding this position? We currently have approved relief requests for the first Ten Year Interval in which the staff has found our lack of instrumentation acceptable. This applies to other positions as well. (II #34, Philip J. North, Duke Power)

Do the modifications that are needed to conform with the stated positions require a backfit. If modifications are necessary to comply with the stated positions, are relief requests necessary if it is deemed impractical to make the modifications? If not through relief, how do we deal with these issues? What if no maintenance history is available to substantiate relief? (IV & V #17, Arkansas Nuclear 1 and 2)

Defend or explain your basis for saying the generic letter does not require a backfit. (IV & V #26, Paul Croy, Southern California Edison/San Onofre)

Response

Generic Letter 89-04 was presented to the NRC's Committee to Review Generic Requirements (CRGR) as a backfit issue, and certain positions were identified as changes to past staff positions. As discussed with the CRGR, the staff determined that those positions in the generic letter that represented changes from previous staff positions were necessary in order to bring licensees into compliance with the Commission's regulations. Therefore, according to 10 CFR 50.109 (a)(4)(i), a backfit analysis was not required to justify issuance of the generic letter. If the positions in the generic letter cannot be met, the option discussed in Paragraph B may be available. Further, if the licensee will not be following the generic letter positions, Paragraph B of the letter, and the ASME Code, the licensee must submit to the NRC staff a request for relief from the ASME Code. Where a licensee is following a provision of its operating license or a particular exemption from the ASME Code that was granted by the NRC staff, a backfit analysis would need to be performed by the NRC staff before requiring any change to that licensee practice. With respect to the staff review of previously approved relief requests at the ten-year update of the IST program, however, a backfit analysis would not be necessary. See the response to Question 70.

Use of OM-6 and 10

Question 92

When addressing cold shutdowns, OM-10 uses statements like "sufficient duration" and "shall continue." When trying to implement these statements, operations personnel frequently ask what is the NRC's definition of a cold shutdown of sufficient duration. Is cold shutdown testing expected to be back to back tests or can 1 or 2 day breaks be acceptable (i.e. shall continue is not easily defined)? (I #39, Jeff Neyhard, Nine Mile Station)

In 1987 and early 1988, the NRC rejected a general relief request to use OM-6 criteria for flow and delta pressure for pumps. Can we now revise our program to use the criteria of OM-6 and OM-10? If the answer is yes, do we need a relief request? (I #21. Jeff Neyhard, Nine Mile Station)

What is the time frame for the 10 CFR 50.55a(g) change? Is the NRC willing to accept the currently approved OM-6/OM-10? (II#24, Garry Galbreath, Duke Power)

Will any of the guidance provided in the generic letter change with the implementation of Part 6 and Part 10 of O&M? (II #40, J. Zudans)

Once OM-6 and OM-10 are approved, will it be required to implement them immediately (within 6 months) or will they be implemented at the next program update? (III #27, Larry Hochman, Nutech)

Response

Rulemaking to reference ASME standards OM-6 and 10 in the regulations is underway at this time. It can be said, however, that, in some recent relief request evaluations, the use of the pump allowable range limits identified in OM-6 for flow rate and differential pressure has not been found acceptable to the staff. The staff has not completed its assessment of the interrelationship of Generic Letter 89-04 and OM-6 and 10. When appropriate references to OM-6 and 10 are incorporated in the regulations, these standards may be used by the licensee as the regulations permit the use of more recent referenced standards. We anticipate that rulemaking to reference these standards will be issued for public comment in the near future.

Solenoid-Operated Valves (SOVs)

Question 93

To perform position indication testing on solenoid operated valves, is a light check acceptable or must the position verification be performed by running the system or injecting air, etc. to prove valve position? (I #29, Jeff Neyhard, Nine Mile Station)

Is a remote position verification required for SOVs with no positive means available? (III #47)

Response

Verification of remote position indication by IWV-3300 is required to ensure that the indication accurately reflects actual valve position. This could take the form of a differential pressure test, flowrate measurement, or other change in some parameter that positively shows that the valve is in the indicated position. An indirect verification, using techniques such as radiography, may also be acceptable.

General Questions

Question 94

Please clarify what is meant by "one part of a broad effort" in the Background section of the Generic Letter. (I #11, Shafi Rokerya, New York Power Authority)

Response

Generic Letter 89-04 is part of a larger program to improve IST throughout the industry and to provide additional information and clarification on the subject to all affected parties. The joint ASME/NRC Symposium on IST held in Washington, D. C., in August 1989 is also part of this effort. Additional generic regulatory guidance may be prepared on other IST aspects. For a discussion of the "broad effort" that NRC is pursuing, refer to the summary of the presentation by Tad Marsh provided in these meeting minutes.

Question 95

How do the Generic Letter 89-04 requirements differ from the ASME requirements? (I #12, John Wiedemann, PSE&G)

Response

Generic Letter 89-04 is intended to provide fundamental information on the NRC's interpretation of certain Technical Specifications and ASME Code requirements, and to identify certain alternative testing that the NRC staff finds acceptable. The generic letter also goes beyond the ASME Code in that it covers procedural issues in addition to programmatic issues.

The generic letter may contain Code interpretations that differ from those of certain licensees. The one area that we are aware of in the generic letter that is different from the Code is contained in Position 8 on the starting point for the time period in Technical Specification action statements. This position is consistent with other Technical Specification starting points. This position is also articulated in the bases for certain of the Standard Technical Specifications.

Question 96

In a refueling outage that is greater than 3 months, how is the cold shutdown frequency handled? Can we perform the cold shutdown procedure once during the outage or do we perform the cold shutdown procedure every 3 months during the outage? (I#17, Jeff Neyhard, Nine Mile Station)

Response

When a component is required to be in service during the outage, the testing is expected to be performed quarterly during the outage. When a component is not required to be operable during an outage, the testing need not be performed quarterly. In accordance with IWV-3416 of the ASME Code, however, those valves must be tested within 30 days before return of the system to operable status. Further, as required by IWP-3400(a), pumps must be tested within one week after the plant is returned to normal operation.

Question 97

Is radiography on check valves an acceptable method for determining valve position? (I #25, Bill Kittle, PSE&G - Salem)

Response

Radiography may be utilized if it clearly indicates the actual position of the valve disk.

Question 98

Most plants have been given relief from measuring pump bearing temperatures per IWP-4310. Is it the policy of the NRC that this will continue to be an item of "generic" relief? (II #10, John Zudans, Florida Power & Light)

Response

It is true that some plants have been given relief from measuring pump bearing temperatures on the basis of the impracticality of measuring temperature for specific pump designs. This issue has not been treated as an item of "generic relief" because each relief request has been individually evaluated. For the foreseeable future, NRC will continue to evaluate these relief requests on a case-by-case basis.

Question 99

Where pump parameter measuring instruments do not meet the specific requirements of the Code but do satisfy the fundamental technical requirements for testing, would it be acceptable to allow relief? (II #12, John Zudans, Florida Power & Light)

Response

It would be difficult to answer this question without more specific information. There have been cases where relief requests in this area have been approved. In those cases, however, the basis for relief has been that the instrumentation has been adequate to meet the fundamental objective of detecting degradation. In relief requests of this type, the licensees should address the reason that the ASME Code requirements are not currently being met and the basis for concluding that the fundamental objectives of IST are being accomplished.

Question 100

The schedule for exercising manual valves should be extended to something less than once each quarter. Is this feasible? (II #13, John Zudans, Florida Power and Light)

Response

We are not aware of a basis for exercising manual valves at a frequency different from other valves. Because this subject is not specifically related to Generic Letter 89-04, it was not addressed at any length during the meeting. If the licensees are aware of reasons why the frequency should be changed, we recommend that this subject be explored with the ASME O&M Working Group on Valves.

Question 101

It has been said that some plants have excellent IST organizations. Who are they? (II #16, Charlie Dunkerly, Calvert Cliffs)

Response

Dresden is one example of a facility with a good IST organization.

Question 102

How do we handle cold shutdown justifications in the future? (II #20, Art Caudill, Georgia Power/Vogtle Project)

Response

Cold shutdown justifications were previously reviewed by NRR for adequacy. In the future, they will be reviewed during IST inspections. The cold shutdown justifications are expected to be described in the IST program the licensee provides to the NRC staff.

Question 103

After this meeting, what is the process for getting further questions answered regarding the generic letter? (II #21, Garry Galbreath, Duke Power)

Response

These meeting minutes will be distributed, which should answer most of the industry's questions. If after reading the meeting minutes you still have questions, you may contact the cognizant personnel through the NRC Project Manager.

Question 104

Does "needed to mitigate the consequences of an accident" mean an accident as described in Chapter 14 of the Final Safety Analysis Report (FSAR)? (II #36, Charlie Dunkerly, Calvert Cliffs)

Response

We assume that the question is directed to the chapter of the FSAR describing accident analyses performed by the licensee. Those analyses are intended to provide confidence that the public health and safety will be protected in the event of certain accidents and anticipated transients at a nuclear power plant. The term "accident" is also used in different sections of the Commission's regulations. For example, Appendix B to 10 CFR Part 50 establishes quality assurance requirements for the design, construction, and operation of "structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public." Part 100 describes structures, systems, and components that must be designed to remain functional during a "safe shutdown earthquake" as those necessary to ensure: "(1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part." As can be seen, the term "accident" is used by the Commission to describe a broad range of possible adverse events at a nuclear power plant. Therefore, although most of the accidents of concern to IST are addressed in the accident analyses chapter, licensees should be aware that there may be other accident analyses in the FSAR that need to be considered.

Question 105

This question is in reference to 10 CFR 50.55a(g)(4): "...to the extent practical within the limitations of design, geometry, and materials of construction of the components." In reviewing this wording, along with the statements of consideration, do you think this rule was intended to impose plant modifications as a result of meeting subsequent editions and addenda? That is, once the staff evaluates a licensee's determination of impracticality, will the NRC impose plant modifications as alternate requirements? (II #37, Mark Dryden, Florida Power & Light)

Response

The NRC staff in the Mechanical Engineering Branch of NRR has ;ad lengthy discussions with the NRC Office of the General Counsel on this matter. The current interpretation of the rule is that it is not intended to require a blanket imposition of all plant modifications that would be necessary to comply with subsequent editions and addenda. The rule does require an evaluation of the impact on the licensee, that is the impracticality of making the modifications, as part of an assessment of the requests for relief from the ASME Code requirements. The legal staff has stated that there is nothing in the regulations that relieves licensees from making all hardware modifications to the plant to comply with changes to IST requirements throughout a plant's life in later editions of Section XI. Some hardware modifications can be required. The difficult issue to resolve is how much may be required. For example, major equipment or piping modifications may be beyond the limitations of practicality in meeting subsequent editions of the Code. We, however, regard modifications such as the installation of instrumentation to be practical as used in 10 CFR 50.55a(g)(4).

Question 106

For plants that do not have operating licenses, 10 CFR 50.55 requires that you apply the codes that are in effect 12 months prior to plant startup. Where does the 6 month conformance letter stand for construction plants in this situation? (II #39, Jackie Jackson, Tennessee Valley Authority)

Response

There are only two plants expected to receive operating licenses for which the staff's review of the IST program has not been completed. These plants are Comanche Peak and Watts Bar. These two plants will be treated essentially as Table 1 plants in that a review will be completed and an SER issued. The reviews of the Comanche Peak and Watts Bar IST programs, however, may not be completed in the same time frame as the reviews for plants listed in Table 1. To obtain the scheduled completion dates for the IST program reviews, the Comanche Peak and Watts B· organizations should contact their respective NRC Project Managers.

Question 107

Currently, we only test the ICS pump suction check valves ICS 3A(B) to verify they open as part of the ICS pump test. Originally, the only safety function recognized was for the valves to open to provide · water source, the RWST, to the ICS pumps. During an independent review of the IST program, it was determined that these valves may also ha.e a safety function to close when the pumps are taking suction from the RHR system. These valves, if they failed open, could provide another flowpath (to the RWST) besides the normal flowpath to containment. This flowpath would also allow potentially contaminated water from the containment sump into the RWST (NOT DESIRABLE). As part of our company's in-house safety system functional inspection, it was determined that if these check valves failed open, adequate flow to the containment would still be achieved. We are also converting the manual valves upstream of 1CS 3A(B) into motor operated valves in order to prevent sump water form getting into the RWST. Do these check valves need to be leak tested? (III#17, Wisconsin Public Service Corp.)

Should Category A be applied to valves other than containment isolation valves (e.g., valves which isolate HVAC damper air accumulators: checks/SOVs)? (IV & V #27, Wayne Wolling, Gulf States Utility/River Bend)

Response

The NRC staff has a generic concern with the current practice of categorization of check valves. The ASME Code assigns all check valves as Category C. If seat leakage of a check valve is limited to a specified amount, the Code also requires that valve to be assigned to Category A. Whereas Category C check valves are required by the Code only to be exercised on a periodic basis, Category A/C check valves must be leak tested in addition to being exercised. The NRC staff has found that, in many instances, check valves are not being assigned to Category A/C despite the fact that credit is taken by the licensee for the check valve providing an essentially leak tight function. The categorization of a check valve is not dependent solely on the function performed by the valve, such as whether it is a containment isolation valve. When determining the proper categorization of a check valve, a licensee should take all applicable aspects into account. For example, the licensee should determine (1) whether the flow requirements for connected systems can be achieved with the maximum possible leakage through the check valve, (2) the effect of any reduced system flows resulting from the leakage on the performance of other systems and components, (3) the consequences of the loss of water from the system, (4) the effect that backflow through the valve may have on piping and components, such as the effect of high temperature and thermal stresses, and (5) the radiological exposure to plant personnel and the public caused by the leak. If any of the above considerations indicate that Category C testing may not be adequate, licensees should assign the check valve to Category A/C and should comply with the associated leak testing requirements.

Question 108

What is the NRC's opinion, per Generic Letter 89-04, of non-quantifiable demonstrations of performance? For example, a solenoid valve has no position indication that can be observed or timed, but bearing temperatures show no overheating. (III #24, Point Beach Nuclear Plant)

Response

The NRC staff is discouraging the use of qualitative criteria as an alternative to the Code required component testing. Licensees should strive to develop a quantitative method of determining the ability of a component to perform its required functions. This recommendation is based on the goal of IST to detect degradation prior to failure of the component. For specific examples, see the response to Question 1. With respect to the specific question, more details would be necessary before arriving at the acceptability of the suggested method.

Question 109

Should LaSalle County Station be on Table 2 of Generic Letter 89-04? If not, why? Zion Station underwent the same review 2 months after LaSalle and they appear on Table 2. (III #25, Roger Sagmoe, Commonwealth Edison Co.)

Response

Although the LaSalle nuclear power plant received an SER about a year ago, a significant revision to its IST program was subsequently submitted for NRC review. The NRC staff determined that a review of the IST program could not be completed in the necessary time frame. In the context of Generic Letter 89-04, LaSalle, therefore, has been classified as a plant that does not possess a current SER and will not be receiving an SER. As a result, LaSalle is expected to respond to the generic letter in accordance with the implementation provisions for plants not listed in Table 1 or 2.

Question 110

What additional NRC guidance can be provided on testing skid-mounted pumps and valves (i.e., diesel generator systems: lube oil pumps/valves, internal engine cooling; RCIC systems - condensate/vacuum pumps with only one source of power, etc.)? Most of these pumps and valves do not have the necessary test instrumentation to support ASME Section XI testing and do not fall within the scope statements of IWP and IWV. Will modifications need to be performed? (III #30, Roger Sagmoe, Commonwealth Edison Co.)

Response

The purpose of inservice testing is to provide assurance of the operability of components and to detect degradation in their performance. Where a particular component is integrated with other components in a system, it may be difficult to perform an individual test of that component. In specific cases for which individual testing is not feasible, an alternate test should be proposed by the licensee. In developing an alternate test, the licensee should attempt to develop quantitative criteria to evaluate the operability and condition of the component.

Question 111

Is temporary flow instrumentation (i.e., portable flow meter) permitted in lieu of a modification to install permanent flow instrumentation? If so, is relief required? (III #40)

Response

The staff does not interpret the ASME Code as excluding the use of portable flow rate instrumentation, such as ultrasonic. We have seen difficulty, however, in meeting the Code-specified accuracy requirements with these instruments.

Question 112

Is trending a requirement for pumps? Is it a requirement for valves? The Code and the regulations do not address this, nor does the generic letter. (IV&V#28, Wayne Wolling, Gulf States Utility/River Bend)

Response

We define "trending" as the analysis of test data to detect degradation of the tested component and to enable preventive maintenance to be performed before significant challenges to component operability occur. The ASME Code contains few requirements for trending of test data. For example, the ASME Code in IWV-3417(a) provides for more frequent stroke-time testing of power-operated valves where an increase in stroke time is seen from a previous test. The NRC staff allows a reference value to be used for this comparison in Position 6 of Generic Letter 89-04. In IWV-3427(b), the Code provides for more frequent testing, and possibly maintenance, where the leak rate of a large valve increases beyond a specified amount from one test to another. In Position 10 of the generic letter, the NRC staff explains its view that this provision of the Code may not be worthwhile and may be suspended. Although the ASME Code is weak in the area of trending, the NRC staff remains of the view that trending is a valuable tool in the IST program. The Commission's regulations can be interpreted to require efforts in this area. More explicit guidance for trending may be developed in the future. In the meantime, we recommend that licensees analyze IST data to take advantage of the benefits of trending.

INSERVICE TESTING
GENERIC LETTER 89-04
REGIONAL MEETINGS

LOGISTICS

- ATTENDANCE SHEETS IN BACK
- NAME TAGS
- CARDS FOR QUESTIONS - NAME, COMPANY, QUESTION
- MEETING MINUTES WILL BE PUBLISHED
- QUESTIONS - WE'LL ANSWER THEM ALL

SCHEDULE:

10:00-10:15	OPENING REMARKS - REGION MANAGEMENT
10:15-10:30	BACKGROUND ON GENERIC LETTER 89-04 T. MARSH
10:30-11:00	APPROACH OF GENERIC LETTER 89-04 - T. SULLIVAN
11:00-12:30	QUESTION/DISCUSSION SESSION I
12:30- 2:00	LUNCH/NRC STAFF CAUCUS
2:00 - 4:00	QUESTIONS/DISCUSSION SESSION II
4:00 - 4:30	BREAK/NRC STAFF CAUCUS
4:30 - 5:00	CLOSING REMARKS - NRC

OBJECTIVE — TO ASSESS OPERATIONAL READINESS OF SAFETY RELATED PUMPS AND VALVES

10 CFR 50.55A

- REQUIRES PUMPS AND VALVE IST PROGRAM IN ACCORDANCE WITH ASME CODE, SECTION XI
- UPDATE IST PROGRAMS TO THE CURRENT CODE EDITION AND ADDENDA EVERY 10 YEARS
- ALLOWS THE GRANTING OF RELIEF REQUESTS FOR CODE REQUIREMENTS THAT ARE IMPRACTICAL

STATUS

- FEW PLANTS HAVE RECEIVED SERs
- SOME OF THE ISSUED SERs ARE OUT OF DATE (SUPERSEDED BY LATTER SUBMITTAL)

PROBLEMS

- INADEQUATE TESTING REQUIREMENTS IN CODE
- NO WRITTEN NRC GUIDANCE ON IST
- HUGE VOLUME OF PROGRAMS/REVISIONS/RELIEF REQUESTS HUGE BACKLOG
- RELIEF REQUESTS IMPLEMENTED WITHOUT PRIOR NRC APPROVAL
- INSPECTION EFFECTIVENESS HAMPERED
- IST PROGRAM IMPLEMENTATION VARIES - SOMETIMES POOR

PURPOSE OF GENERIC LETTER (GL)

- PROVIDES GENERIC GUIDANCE ON ELEVEN SIGNIFICANT IST PROBLEM AREAS
- PROVIDES GUIDANCE ON DEVELOPING ACCEPTABLE IST PROGRAMS
- CLARIFIES APPROVAL STATUS OF IST PROGRAMS (I.E., RESOLVES TS 4.0.5 ISSUE)

FUTURE

NEW ASME STANDARDS O&M 6 PUMPS

O&M 10 VALVES

MODIFY 10 CFR 50.55A(G)

FURTHER GENERIC LETTERS

IST SYMPOSIUM - AUGUST 1 - 3, 1989

APPROACH USED IN GENERIC LETTER (GL) 89-04

<u>THREE GROUPINGS OF PLANTS</u>

TABLE 1 PLANTS

- SER NEARING COMPLETION
- SER CONSTITUTES APPROVAL

TABLE 2 PLANTS

- SER ISSUED ON CURRENTLY SUBMITTED PROGRAM
- SER CONSTITUTES APPROVAL

TABLE 1 AND 2 PLANTS

- DO NOT NEED TO RESPOND TO GL
- NEED TO ASSURE PROCEDURES CONSISTENT WITH GL

PLANTS NOT ON EITHER TABLE

- GL CONSTITUTES APPROVAL PROVIDED LICENSEES:
  - REVIEW PROGRAMS AGAINST ATTACHED POSITIONS, AND
  - CONFIRM CONFORMANCE OR JUSTIFY DEVIATIONS FROM ATTACHED POSITIONS IN SIX MONTHS, AND
  - MAKE ANY MODIFICATIONS WITHIN SPECIFIED TIME
- ALTERNATIVES TO ATTACHED POSITIONS MAY BE IMPLEMENTED PROVIDED:
  - MAINTENANCE AND DEGRADATION HISTORY EVALUATED
  - DEVIATION JUSTIFIED AND DOCUMENTED
- RESULTING IST PROGRAM TO BE PROVIDED TO NRC
- NRC WILL NOT ISSUE SERs ON
  - CONFORMANCE WITH ATTACHED POSITIONS
  - JUSTIFIED ALTERNATIVES TO ATTACHED POSITIONS
- NRC WILL ISSUE SERs ON
  - NEW RELIEF REQUESTS ON AREAS NOT COVERED BY ATTACHED POSITIONS

PROGRAM UPDATES/REVISIONS

- FOR PROGRAM CHANGES COVERED BY ATTACHED POSITIONS
  - SAME GUIDANCE AS ABOVE

- FOR PROGRAM CHANGES NOT COVERED BY ATTACHED POSITIONS
  - STAFF WILL EVALUATE PER 10 CFR 50.55A(G)

- RELIEF REQUESTS PREVIOUSLY APPROVED
  - WILL NOT BE REEVALUATED
  - APPROVAL REMAINS IN EFFECT

INSPECTION AND ENFORCEMENT

- INSPECTIONS TO BE CONDUCTED FOR CONFORMANCE WITH 10 CFR 50.55A, AS EXPLAINED IN GL
  - FOCUS ON ATTACHED POSITIONS
  - OTHER AREAS MAY BE INSPECTED

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