KMC

July 13, 1982

ROGER S. BOYD
Vice President

Mr. Robert L. Tedesco, Assistant Director
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Tedesco:

At the June 23 meeting of the LRG-I Tech Spec Group and representatives of NRR, we were advised that meeting minutes may not be prepared by NRR staff owing to the fact that there presently is no LRG Project Manager. That situation reduced the effectiveness of the meeting somewhat, and we thought it would not be helpful if there were no official minutes of the meeting. It is in that spirit that I am enclosing a copy of the LRG minutes for your use and distribution to representatives of the Licensing Guidance Branch.

The need for a Project Manager, of course, goes far beyond the necessary coordination of the Tech Spec Group. In the interests of conserving resources, it goes to the matter of all of the LRG Groups. We consider it essential to have a strong project management presence, at least on a part time basis. I would welcome the opportunity to discuss the matter further with you at your convenience, especially since our next Tech Spec Group meeting is scheduled for August 18-19, 1982, and we need to meet with the NRR staff on August 19 to discuss the outstanding Tech Spec issues that have been pending since March.

Sincerely,

Roger S. Boyd
KMC, Inc.

encl.

MEETING

NRC AND LRG-I

TECH SPEC SUBCOMMITTEE

DATE: JUNE 23, 1982

TIME: 9AM - 12:00

PLACE: GE OFFICES, BETHESDA, MARYLAND

AGENDA

9:00 - 9:05	INTRODUCTION	CHARLES ALM (CG&E)
9:05 - 10:00	OLD ISSUES - FEEDBACK FROM NRC	NRC
10:00 - 11:00	NEW ISSUES	CHARLES ALM & LRG-1 MEMBERS
11:00 - 11:45	A) OFFICE LETTER NO. 38 - HOW DO WE PROCEED? B) NUREG-0737 TECH SPECS - ALREADY COVERED?	NRC
11:45 - 12:00	SUMMARY AND MEETING CLOSE	CHARLES ALM

LRG-I STANDARD TECH. SPEC. ISSUES

ISSUE	DATE	SPEC. NO.	GENERAL DESCRIPTION	DISPOSITION
1	3/16/82	3/4.1.3	CONTROL ROD OPERABILITY	
2	3/16/82	3/4.7.5	SNUBBERS	
3	3/16/82	3/4.4.1	RECIRCULATION SYSTEM	
4	3/16/82	3/4.3.2	ISOLATION ACTUATION INSTRUMENTATION	
5	3/16/82	1.12	END-OF-CYCLE RPT SYSTEM RESPONSE TIME	
6	3/16/82	3.6.1.2	(CLARIFICATION) PRIMARY CONTAINMENT LEAKAGE	
7	3/16/82	3/4.3.9, 3/4.7.8	(CLARIFICATION) PLANT SYSTEMS ACTUATION INSTRUMENTATION	
8	3/16/82	3.4.3.2	(CLARIFICATION) REACTOR COOLANT SYSTEM -- OPERATIONAL LEAKAGE	
9	3/16/82	Table 4.3.1.1-1	(CLARIFICATION) APRM - FLOW BIASED SIMULATED THERMAL POWER	
10	3/16/82	3/4.4.6	(CLARIFICATION) REACTOR COOLANT SYSTEM - PRESS./TEMP. LIMITS	
11	6/23/82	3/4.4.7	MSIV CLOSURE TIMES	
12	6/23/82	3/4.2.2	APRM SETPOINTS	
13	6/23/82	3/4.5.3, 3/4.6.2.1	(COMBINE SPECS) ECCS - SUPPRESSION CHAMBER, DEPRESSURIZATION SYSTEM	
14	6/23/82	3/4.6.1.2	PURGE VALVE LEAK RATE TESTING	
15	6/23/82	3/4.5.2	ECCS - SHUTDOWN (PROPOSED BY CG&E: MAY 12, 1982 LETTER)	
16	6/23/82	4.7.7	VALVE POSITIONS IN FIRE PROTECTION SYSTEMS (PROPOSED BY DET. ED.: MAY 19, 1982 LETTER)	
17	6/23/82	Tables 3.3.1.2, 4.3.1.1-1	RPS RESPONSE TIMES (PROPOSED BY DET. ED.: MAY 19, 1982 LETTER)	
18	6/23/82	3/4.6.3.5	Primary Containment Isolation Valves (TIP)	

LRG TECHNICAL SPECIFICATION
WORKING GROUP

ISSUE: 11

Technical Specification 3/4.4.7 - Main Steam Isolation Valves

PROPOSED MODIFICATION:

See attachment.

JUSTIFICATION:

The term "closing time" in the Standard Technical Specification caused confusion in regard to just what was included in that term. This confusion was compounded by the discrepancies between the Startup MSIV Function and the Technical Specification.

The attached tech spec has been modified to reflect the actual requirements for the valve and, as such, is now consistent with the Safety Analysis and the Startup MSIV Function Test.

NSS:sem/B060111
6/1/82

REACTOR COOLANT SYSTEM

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing stroke times greater than or equal to 2.5 ~~(3)~~ and less than or equal to (5) seconds. The average stroke time of the fastest valve in each of the four steam lines shall be greater than or equal to 3.0 seconds

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more MSIVs inoperable:

1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

   a) Restore the inoperable valve(s) to OPERABLE status, or

   b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying ~~full closure between (3) and (5) seconds~~ compliance with the limiting conditions for operations above when tested pursuant to Specification 4.0.5.

DRAFT

ISSUE NO. 12

LRG TECHNICAL SPECIFICATION ISSUE

ISSUE

AFFECTED BASIS, DEFINITIONS AND SPECIFICATION

1. B 3/4 2.2 APRM Setpoints.

2. Definitions:

   a. 1.__ MAXIMUM FRACTION OF LIMITING POWER DENSITY

   b. 1.19 MAXIMUM TOTAL PEAKING FACTOR

   c. 1.__ FRACTION OF LIMITING POWER DENSITY

   d. 1.__ FRACTION OF RATED THERMAL POWER

3. 3/4 2.2 APRM Setpoints.

1. Basis 3/4 2.2

   Present text:

   a. "____ (TOTAL PEAKING FACTOR ___)"

   Suggested modification:

   Since it is the LHGR & MCPR which govern, delete <u>all</u> information referencing TOTAL PEAKING FACTOR. The text relating to LHGR then becomes generic.

NS:hjr/C050516-1*
5/5/82

a. The text presently reads:

"____ flow biased simulated thermal power upscale control rod block _____"

Suggested modification:

"____ flow biased neutron flux upscale control rod block _____"

Justification:

Simulated thermal power applies only to the scram trip setpoint and not to the rod block. Therefore, the suggested modification is more technically correct.

b. MFLPD See Definition 2a.

See modified basis 3/4 . text (attached)

a. 1.__ MAXIMUM FRACTION OF LIMITING POWER DENSITY

Suggested modification:

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY, CMFLPD.

Justification:

Since it is the CORE MFLPD (CMFLPD) which is used, it should be titled as such and would then read the same as the site computer printout.

NS:hjr/C050516-2*
5/5/82

b. 1.19 MAXIMUM TOTAL PEAKING FACTOR

Suggested modification:

Remove

Justification:

Justification is given in B3/4 2.2, 1a.

c. 1.__ FRACTION OF LIMITING POWER DENSITY

Suggested modification:

Remove 13.4 kW/ft LHGR

Justification:

This should be removed from the definition as the value is given in Technical Specification 3/4.2.4. This enables the parentheses to be removed from the total definition and within the definition, resulting in a generic definition for all plants, regardless of the value of LHGR.

d. 1. ____ FRACTION OF RATED THERMAL POWER

Suggested modification:

Remove the parentheses surrounding the total definition as this is now generic as a result of Basis 1a.

See modified text for definitions, (attached).

NS:hjr/C050516-3*
5/5/82

3. Specification 3/4 2.2

a. Present text:

$S \leq (0.66W + (54)\%)T$

Suggested modification:

$S \leq (0.66W + (51)\%)\ T.$

Justification:

The value (54)%, in the present text represents an allowable value, the value should represent the nominal setpoint, as in the modification.

b. PresentText:

"T is always ____ 1.0."

Suggested modification:

"Applied only for values of T less than 1.0."

Justification:

The present statement is untrue.

ACTION STATEMENT:

Present Text:

"____ limits* within 2 hours _____"

Suggested modification:

"____ limits* within six hours ____"

NS:hjr/C050516-4*
5/5/82

Justification:

If, because of proximity to the flow biased scram rod block lines, the required adjustment cannot be made, it becomes necessary to adjust the rod pattern. The total combined change of rod pattern plus must gain adjustment, cannot always be made in 2 hours. This then leads to an LER.

Operating plants have also found this to be the case and are seeking an adjustment to the requirements of from 2 to 6 hours.

We have found no justification for the two hours.

4.2.2

Present text:

"____ for each class of fuel _____".

Suggested modification:

Delete these words.

Justification:

As previously discussed, it is the single valued CORE MFLPD (CMFLPD)** used.

The balance of changes to this section have been discussed previously.

NS:hjr/C050516-5*
5/5/82

Foot Note:

Present Text:

"____ during power ascension up to 90% of RATED THERMAL POWER ____"

Requested modification:

Delete these lines.

Justification:

We have found no justification that would lead to this requirement. It was first established on Hatch Technical Specification; however, the allowed power ascension was to 95% of RATED THERMAL POWER.

See modified 3/4 2.2 specification text,(attached).

NS:hjr/C050516-6*
5/5/82

(MAXIMUM FRACTION OF LIMITING POWER DENSITY

1. The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY ~~(MFLPD)~~ CMFLPD shall be highest value of the FLPD which exists in the core.)

MAXIMUM TOTAL PEAKING FACTOR

1.19 The MAXIMUM TOTAL PEAKING FACTOR (MTPF) shall be the largest TPF which exists in the core for a given class of fuel for a given operating condition.

MINIMUM CRITICAL POWER RATIO

1.20 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core (for each class of fuel).

OPERABLE - OPERABILITY

1.21 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.22 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.23 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.24 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

FRACTION OF LIMITING POWER DENSITY

1.__ The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type. ~~(13.4 kw/ft)~~.

FRACTION OF RATED THERMAL POWER

1.__ The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.15 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.16 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.17 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.18 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, ie., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux ~~simulated thermal power~~-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

$$S \leq (0.66W + (\cancel{54}\ 51)\%)T$$
$$S_{RB} \leq (0.66W + (42)\%)T$$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of (108.5) million lbs/hr.
T = Lowest value of the ratio of ~~(design TPF, (2.43) for (8 x 8) fuel, divided by the MTPF obtained for any class of fuel in the core)~~ FRACTION OF RATED THERMAL POWER divided by the core MAXIMUM FRACTION OF LIMITING POWER DENSITY. ~~T is always less than or equal to 1.0.~~ CMFLD. Applied only for values of T less than 1.0.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (25)% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased Neutron Flux simulated thermal power-upscale scram trip setpoint and/or the flow biased ~~simulated thermal power~~-upscale control rod block trip setpoint less conservative than S or S_{RB}, as above determined, initiate corrective action within 15 minutes and restore S and/or S_{RB} to within the required limits* within ~~2~~ hours or reduce THERMAL POWER to less than (25)% of RATED THERMAL POWER within the next 4 hours.
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SURVEILLANCE REQUIREMENTS

4.2.2 The ~~(MTPF)~~ (FRTP and the ~~MFLPD) for each class of fuel~~ CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and control rod block trip setpoints verified to be within the above limits or adjusted, as required:
flow biased neutron flux upscale

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with (MTPF)(MFLPD) greater than or equal to (2.43) (FRTP).

*With ~~(MTPF)(MFLPD)~~ CMFLPD greater than the ~~(design TPF)~~ (FRTP) during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that the APRM readings are greater than or equal to 100% times ~~(MTPF) (MFLPD)~~, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

CMFLPD

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a ~~(TOTAL PEAKING FACTOR of (2.43) for (8 x 8) fuel)~~ (power distribution which would yield the design LHGR at RATED THERMAL POWER). The flow biased simulated thermal power-upscale scram setting and flow biased ~~simulated thermal power~~ Neutron flux upscale control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.06 or that > 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and ~~(peak, flux) (MFLPD)~~ Core maximum Fraction Limiting Power Density (CMFLPD) indicates a ~~(TOTAL PEAKING FACTOR greater than (2.43))~~ (higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded conditon). ~~(The method used to determine the design TPF shall be consistent with the method used to determine the MTPF.)~~

ISSUE NO. 13

LRG-I TECH SPEC GROUP

ISSUE 3

May 12, 1982

Affected Specifications:	3/4.5.1, 3/4.5.2, 3/4.5.3, 3/4.6.2.1, 3/4.7.4, and creates 3/4.3.7.12
Current Requirement:	Imposes instrumentation channel checks, channel functional tests, and channel calibrations on the ECCS, RCIC, and Suppression Chamber monitoring instruments.
Proposed Change:	The proposed change does not change current requirements. The change does restructure the technical specifications back to the original intended format. This change also reduces action and testing duplicated in specifications 3/4.5.3 and 3/4.6.2.1. Refer to the attached pages which are marked to show the requested change.
Justification:	Since the change request does not affect specification requirements, no justification is required. The intent of this change request is to remove the duplication of action and testing requirements in specifications 3/4.5.3 and 3/4.6.2.1 and to return the technical specifications to their original format structure.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 ECCS divisions 1, 2 and 3 and the automatic depressurization system (ADS) shall be OPERABLE with:

a. ECCS division 1 consisting of:

1. The OPERABLE low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

2. The OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

3. (At least) (7) OPERABLE ADS valves.

b. ECCS division 2 consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, each with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

2. (At least) (7) OPERABLE ADS valves.

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 1, 2* # and 3*.

*The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to (100) psig.

#See Special Test Exception 3.10.6.

EMERGENCY CORE COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For ECCS division 1, provided that ECCS divisions 2 and 3 are OPERABLE:

   1. With the LPCS system inoperable, restore the inoperable LPCS system to OPERABLE status within 7 days.

   2. With LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" to OPERABLE status within 7 days.

   3. With the LPCS system inoperable and LPCI subsystem "A" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCS system to OPERABLE status within 72 hours.

   4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. For ECCS division 2, provided that ECCS divisions 1 and 3 are OPERABLE:

   1. With either LPCI subsystem "B" or "C" inoperable, restore the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 7 days.

   2. With both LPCI subsystems "B" and "C" inoperable, restore at least the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

   3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*.

c. For ECCS division 3, provided that ECCS divisions 1 and 2 and the RCIC system are OPERABLE:

   1) With ECCS division 3 inoperable, restore the inoperable division to OPERABLE status within 14 days.

   2) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE:

   1) With LPCI subsystem "A" and either LPCI subsystem "B" or "C" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 Continued

LIMTING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2) With the LPCS system inoperable and either LPCI subsystems "B" or "C" inoperable, restore at least the inoperable LPCS system or the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

3) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*.

e. For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE and divisions 1 and 2 are otherwise OPERABLE:

1. With one of the above required ADS valves inoperable, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ (100) psig within the next 24 hours.

2. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤ (100) psig within the next 24 hours.

~~f. With an ECCS discharge line "keep filled" (pressure) (pump failure) alarm instrumentation channel inoperable, perform Surveillance Requirement 4.5.1.a.1 at least once per 24 hours.~~

~~g. With an ECCS header delta P instrumentation channel inoperable, restore the inoperable channel to OPERABLE status with 72 hours or determine ECCS header delta P locally at least once per 12 hours; otherwise, declare the associated ECCS inoperable.~~

f ~~h~~. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the useage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 Continued

SURVEILLANCE REQUIREMENTS

4.5.1 ECCS division 1, 2 and 3 shall be demonstrated OPERABLE by:

a. At least once per 31 days for the LPCS, LPCI and HPCS systems:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

~~2. Performance of a CHANNEL FUNCTIONAL TEST of the:~~

~~a) Discharge line "keep filled" (pressure) (pump failure) alarm instrumentation, and~~

~~b) Header delta P instrumentation.~~

~~3.~~ 2. Verifing that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. Verifing that, when tested pursuant to Specification 4.0.5, each:

1. LPCS pump develops a flow of at least (6598) gpm against a test line pressure greater than or equal to (452) psig.

2. LPCI pump develops a flow of at least (7666) gpm against a test line pressure greater than or equal to (111) psig.

3. HPCS pump develops a flow of at least (659) gpm against a test line pressure greater than or equal to (397) psig.

c. For the LPCS, LPCI and HPCS systems, at least once per 18 months:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 continued

SURVEILLANCE REQUIREMENTS (Continued)

~~2. Performing a CHANNEL CALIBRATION of the:~~

~~a) Discharge line "keep filled" (pressure) (pump failure) alarm instrumentation and verifying the:~~

~~1) High pressure setpoint and the low pressure setpoint of the:~~

~~(a) LPCS system to be (450) +( ), -( ) psig and (40) ± ( ) psig, respectively.~~

~~(b) LPCI subsystems to be (400) +( ), -( ) psig and (40) ± ( ) psig, respectively.~~

~~2) Low pressure setpoint of the HPCS system to be (40) ± ( ) psig.~~

~~b) Header delta P instrumentation and verifying the setpoint of the:~~

~~1) LPCS system and LPCI subsystems to be ± (1) psid.~~

~~2) HPCS system to be (0.5) ± (0.25) psid less than the normal indicated ΔP.~~

D ~~3.~~ Verifying that the suction for the HPCS system is (automatically) transferred from the condensate storage tank to the suppression chamber on a condensate storage tank low water level signal and on a suppression chamber high water level signal.

E ~~d.~~ At least once per 18 months for the ADS by:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

2. Manually opening each ADS valve (when the reactor steam dome pressure is greater than or equal to 100 psig*) and observing (the expected change in the indicated valve position) (that either:

   a) The control valve or bypass valve position responds accordingly, or

   b) There is a corresponding change in the measured steam flow.)

(*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.)

EMERGENCY CORE COOLING SYSTEMS

3/4 5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

a. The low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. Low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

c. Low pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

d. Low pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

e. The high pressure core spray (HPCS) system with a flow path capable of taking suction from one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

1. From the suppression chamber, or

2. When the suppression pool level is less than the limit or is drained, from the condensate storage tank containing at least (150,000) available gallons of water, equivalent to a level of ( )%.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION:

a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

b. With both of the above required subsystems/systems inoperable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel. Restore at least one subsystem/system to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.9.8 and 3.9.9. AND the conditions of specification 3/4.5.3B Are met when the cavity is flooded

(or being flooded from the suppression pool)

This change is requested by issue (X) 15

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS divisions shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1. ~~except that the header ΔP instrumentation is not required to be OPERABLE.~~

4.5.2.2 The HPCS system shall be determine OPERABLE at least once per 12 hours by verifying the condensate storage tank required volume when the condensate storage tank is required to be OPERABLE per Specification 3.5.2.e.

3/4.5.3 SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3 with a contained water volume of at least (142,160) ft^3, equivalent to a level of (26'10").

b. In OPERATIONAL CONDITION 4 and 5* with a contained water volume of at least ( ) ft^3, equivalent to a level of ( ), except that the suppression chamber level may be less than the limit or may be drained provided that:

1. No operations are performed that have a potential for draining the reactor vessel,

2. The reactor mode switch is locked in the Shutdown or Refuel position,

3. The condensate storage tank contains at least (150,000) available gallons of water, equivalent to a level of ( )%, and

4. The HPCS system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

#See Specification 3.6.2.1 for pressure suppression requirements.

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded (or being flooded from the suppression pool), the spent fuel pool gates are removed (when the cavity is flooded), and the water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

~~c. With one suppression chamber water level instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify the suppression chamber water level to be greater than or equal to (26'10") or (14' 0"), as applicable, at least once per 12 hours by (an alternate method).~~

~~d. With both suppression chamber water level instrumentation channels inoperable, restore at least one inoperable channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours and verify the suppression chamber water level to be greater than or equal to (26'10") or (14' 0"), as applicable, at least once per 12 hours by (at least one alternate method).~~

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying:

a. The water level to be greater than or equal to, as applicable:

1. (26'10") at least once per 24 hours.

2. ( )' at least once per (12) hours.

~~b. (At least) two suppression chamber water level instrumentation channels OPERABLE with the low water level alarm setpoint at greater than or equal to ( ) (or ( ), as applicable,) by performance of a:~~

~~1. CHANNEL CHECK at least once per 24 hours,~~

~~2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and~~

~~3. CHANNEL CALIBRATION at least once per 18 months.~~

4.5.3.2 With the suppression chamber level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, at least once per 12 hours:

a. Verify the required conditions of Specification 3.5.3.b to be satisfied, or

b. Verify footnote conditions * to be satisfied.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER#

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between (148,000) ft3 and (142,160) ft^3, equivalent to a level between (27' 10") and (26'10"), and a

2. Maximum average temperature of (95)°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) (105)°F during testing which adds heat to the suppression chamber.

b) (110)°F with THERMAL POWER less than or equal to (1)% of RATED THERMAL POWER.

c) (120)°F with the main steam line isolation valves closed following a scram.

This was not to be marked

b. Drywell-to-suppression chamber bypass leakage less than or equal ~~to 10% of the acceptable A/√K design valve of (0.03) ft²~~.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than (95)°F, restore the average temperature to less than or equal to (95)°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than (105)°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than (95)°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than (110)°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

3. With the suppression chamber average water temperature greater than (120)°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

#See Specification 3.5.3 for ECCS requirements.

CONTAINMENT SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

~~c. With one suppression chamber water level instrumentation channel inoperable and/or with one suppression pool water temperature instrumentation channel in any pair(s) of temperature instrumentation channels in the same sector inoperable, restore the inoperable channel(s) to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours.~~

~~d. With both suppression chamber water level instrumentation channels inoperable and/or with both suppression pool water temperature instrumentation channels in any pair(s) of temperature instrumentation channels in the same sector inoperable, restore at least one inoperable water level channel and one inoperable temperature instrumentation channel in each pair of temperature instrumentation channels in the same sector to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

c ~~e~~. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to (95)°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to (105)°F.

2. At least once per hour when suppression chamber average water temperature is greater than or equal to (95)°F, by verifying suppression chamber average water temperature to be less than or equal to (110)°F and THERMAL POWER less than or equal to (1)% of RATED THERMAL POWER.

3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to (95)°F, by verifying suppression chamber average water temperature less than or equal to (120)°F.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued

This is not to be marked

c. At least once per 18 months by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of (5) psi and verifying ~~that the A/√K calculated from~~ the measured leakage ~~is within the specified limit~~. If any drywell-to-suppression chamber ~~bypass leak test~~ fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

~~d. By verifying (at least) two suppression chamber water level instrumentation (channels)(divisions, with 2 channels per division) and at least sixteen suppression pool water temperature instrumentation channels, at least one pair in each suppression pool sector, OPERABLE by performance of a:~~

~~1. CHANNEL CHECK at least once per 24 hours,~~

~~2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and~~

~~3. CHANNEL CALIBRATION at least once per 18 months,~~

~~with the water level and temperature alarm setpoint for:~~

~~1. High water level ≤ ( ),~~

~~2. Low water level ≥ ( ), and~~

~~3. High water temperature ≤ ( ) °F.~~

PLANT SYSTEMS

3.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of (automatically) taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than (100) psig.

ACTION:

~~a. With a RCIC discharge line "keep filled" (pressure) (pump failure) alarm instrumentation channel inoperable, perform Surveillance Requirement 4.7.4.a.1 at least once per 24 hours.~~

~~b.~~ a. With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to (100) psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water,

~~2. Performance of a CHANNEL FUNCTIONAL TEST of the discharge line "keep filled" (pressure) (pump failure) alarm instrumentation, and~~

2 ~~3.~~ Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

3 ~~4.~~ Verifying that the pump flow controller is in the correct position.

b. At least once per (31) (92) days by verifying that the RCIC pump develops a flow of greater than or equal to (600) gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at (1000 + 20, - 80) psig.*

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation (and restart) and verifying that each automatic valve in the flow path actuates to its correct position, but may exclude actual injection of coolant into the reactor vessel.

2. Verifying that the system will develop a flow of greater than or equal to (600) gpm in the test flow path when steam is supplied to the turbine at a pressure of (150) ± (15) psig.*

3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal (and on a suppression pool water level - high signal).

~~4. Performing of a CHANNEL CALIBRATION of the discharge line "keep filled" (pressure)(pump failure) alarm instrumentation and verifying the:~~

~~a) High pressure setpoint to be ≤ ( ) psig.~~

~~b) Low pressure setpoint to be ≥ ( ) psig.~~

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

3/4.3.7.12 INSTRUMENTATION ECCS, RCIC AND SUPPRESSION CHAMBER INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.7.12 The monitoring instrumentation functions shown in Table 3.3.7.12-1 shall be OPERABLE with the minimum number of OPERABLE instrument channels per division and the alarm setpoints consistent with the values shown in the Alarm Setpoint column. The functions Suppression Chamber Water Level and Water Temperature consist of two divisions. All other monitoring functions consist of one Division.

APPLICABILITY: As Shown in TABLE 3.3.7.12-1

ACTION:

a) With one suppression chamber water level division and/or one suppression chamber water temperature division inoperable, restore the inoperable division to OPERABLE status within 7 days or verify suppression chamber water level or temperature as applicable to be within the limits of specification 3.5.3 and 3.6.2.1 once per 12 hours.

b) With both suppression chamber water level divisions and/or both suppression chamber water temperature divisions inoperable, restore at least one inoperable suppression chamber water level division and/or water temperature division to OPERABLE status within 8 hours, or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours.

c) With ECCS or RCIC line "keep filled" (pressure) (pump failure) alarm instrumentation channel inoperable, perform surveillance requirement 4.5.1a1 or 4.7.4a1 as applicable at least once per 24 hours.

d) With ECCS header delta P instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 72 hours or determine ECCS header delta P locally at least once per 12 hours; otherwise declare the associated ECCS inoperable.

SURVEILLANCE REQUIREMENTS

4.3.7.12 Each monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the Channel Check, Channel Functional Test, and Channel Calibration operation for the OPERATIONAL CONDITIONS and at the frequencies shown in TABLE 4.3.7.12-1.

TABLE 3.3.7.12-1

ECCS, RCIC, SUPPRESSION CHAMBER INSTRUMENTATION

	Monitoring Function	Minimum Number OPERABLE CHANNELS/Division	Alarm Setpoint	Applicable OPERATIONAL CONDITIONS
1)	SUPPRESSION CHAMBER			
	a) Water Level	(1)	≥( ) Low Level ≤( ) Hi Level	1,2,3,4,5[(a)]
	b) Water Temperature	(8)	≤(95°F) Hi Temp ≤(110°F) Hi Hi Temp	1,2
2)	ECCS			
	a) HPCS			
	1) discharge "keep filled"	1	≥( ) Lo Press	1,2,3,4[(b)],5[(b]
	2) Header P	1	(0.5) ± (0.25)PSID	1,2,3
	b) LPCS & LPCI			
	1) discharge "keep filled"	1	≥( ) Lo Press	1,2,3,4[(b)],5[(b]
	2) Header P	1	(0.0) ± (1)PSID	1,2,3
	3) Hi Press	1	≤( ) Hi Press	1,2,3
3)	Reactor Core Isolation Cooling			
	a) Keep Filled	1	≥( ) Lo Press	1,2,3[(c)]
	b) Hi Press	1	≤( ) Hi Press	1,2,3[(c)]

(a) The Suppression Chamber Water level instrumentation is not required OPERABLE when the Suppress Chamber is not required as a water source per specification 3.5.3.

(b) The Low Pressure Alarm is not required OPERABLE when the associated ECCS train is not required OPERABLE per specification 3.5.2.

(c) With reactor steam dome pressure greater than or equal to ( ).

TABLE 4.3.7.12-1

ECCS, RCIC, SUPPRESSION CHAMBER INSTRUMENTATION

Monitoring Function	Channel Check	Channel Functional Check	Channel Calibration	Applicable Operational Conditions
1) Suppression Chamber				
a) Water Level	D	M	R	1,2,3,4,5(a)
b) Water Temperature	D	M	R	1,2
2) EMERGENCY CORE COOLING				
a) HPCS				
1) discharge "keep filled"	N/A	M	R	1,2,3,4(b),5(b)
2) Header P	N/A	M	R	1,2,3
b) LPCS and LPCI				
1) discharge "keep filled"	N/A	M	R	1,2,3,4(b),5(b)
2) Header P	N/A	M	R	1,2,3
3) Hi Press Alarm	N/A	M	R	1,2,3
3) REACTOR CORE ISOLATION COOLING				
a) "keep filled"	N/A	M	R	1,2,3(c)
b) Hi Press Alarm	N/A	M	R	1,2,3(c)

(a) The Suppression Chamber Water Level instrumentation is not required OPERABLE when the Suppression Chamber is not required as a water source per specification 3.5.3.

(b) The Low Pressure Alarm is not required OPERABLE when the associated ECCS train is not required OPERABLE per specification 3.5.2.

(c) With reactor steam dome pressure greater than or equal to ( ).

ISSUE #4

PURGE VALVE LEAK RATE TESTING

Affected Specifications: 3/4.6.1.2

Current Requirement: Specification 4.6.1.8 imposes leak rate test requirements on containment purge valves, establishing individual leak rate limits for each valve and requiring testing more frequently than once per 24 months. Both of these items represent departures from the requirements of 10CFR50, Appendix J for Type C testing.

Proposed Changes: (see attachment) Exempt resiliant seat containment purge valves from inclusion in the combined leak rate total for Type B and C penetrations to be compared against the 0.60 L_a limit. This change is a departure from the provisions of 10CFR50, Appendix J.

Justification: In those cases where leakage characteristics of certain valves necessitated imposition of special leak rate limits and testing requirements (e.g. MSIVs, containment isolation valves that are hydrostatically tested, etc.), exemption from the normal Appendix J requirements was provided. The combined total provides a convenient way of evaluating leakage from a large number of valves which normally experience low leak rates. It is appropriate since overall containment leakage is the concern rather than individual penetration leakage.

At the present time, the standard technical specifications include specific leak rate requirements (0.05 L_a and 0.01 L_a) for the purge valves. Inclusion of the actual leakage in the Appendix J combined total is therefore no longer appropriate.

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, (0.635) percent by weight of the containment air per 24 hours at P_a, (40.4) psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* ~~(and valves which are hydrostatically leak tested per Table 3.6.3-1)~~, subject to Type B and C tests when pressurized to P_a, (40.4) psig. AND resiliant seat containment purge valves,

c. *Less than or equal to (11.5) (46) scf per hour for (any one) (all four) main steam line isolation valve(s) when tested at (P_t,) (20.2) psig.

d. A combined leakage rate of less than or equal to (1 gpm times the total number of)(3 gpm for all)(ECCS and RCIC) containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at (1.10) Pa, (44.4) psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* (and valves which are hydrostatically leak tested per Table 3.6.3-1), subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding (11.5) (46) scf per hour for (any one) (all four) main steam line isolation valve(s), or

d. The measured combined leakage rate for all (ECCS and RCIC) containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding (1 gpm times the total number of such valves)(3 gpm),

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steamline isolation valves* (and valves which are hydrostatically leak tested per Table 3.6.3-1), subject to Type B and C tests to less than or equal to 0.60 L_a, and

*Exemption to Appendix "J" of 10 CFR 50.

May 11, 1982

LRG-I TECH SPEC GROUP

ISSUE #5

Affected Specifications: 3/4.5.2

Current Requirement: From interpretation of the specification at least two trains of ECCS must be operable except during Operational Condition 5 after the vessel head is removed and the cavity is flooded.

Proposed Change: Change the *Note on specification 3/4.5.2 to read:

The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed (when cavity is flooded), water level is maintained within the limits of specification 3.9.8 and 3.9.9 and the conditions of specification 3/4.5.3B are met.

Justification: Five of the six ECCS train options of this specification require the suppression pool as their water source. Since specification 3/4.5.3 allows the suppression pool to be inoperable during the above conditions these specifications 3/4.5.2 and 3/4.5.3 contain a conflict unless this change is accepted.

Note: The number of this issue is to be assigned by Mr. Roger Boyd of KMC.

EMERGENCY CORE COOLING SYSTEMS

3/4 5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

a. The low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. Low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

c. Low pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

d. Low pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

e. The high pressure core spray (HPCS) system with a flow path capable of taking suction from one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

1. From the suppression chamber, or
2. When the suppression pool level is less than the limit or is drained, from the condensate storage tank containing at least (150,000) available gallons of water, equivalent to a level of ( )%.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION:

a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

b. With both of the above required subsystems/systems inoperable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel. Restore at least one subsystem/system to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, (or being flooded from the suppression pool) the spent fuel pool gates are removed, ~~the~~ (when the cavity is flooded), water level is maintained within the limits of Specification 3.9.8 and 3.9.9. And the conditions of specification 3/4.5.3B are met.

(4)

LRG-I TECH SPEC GROUP

ISSUE 16 DATE ______

Position of Valves in the Fire Protection Systems

Affected Specifications:

a. 4.7.7.1.1.c

b. 4.7.7.2.a

c. 4.7.7.3.1

d. 4.7.7.4

Current Requirements:

All the above specifications periodically require that "each valve, manual, power operated or automatic, in the flow path is in its correct position." Each of the above specifications concerns some type of fire suppression system.

Proposed Change:

Change the wording of the above specifications as follows:

> "each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position."

Justification:

As presently written, the Technical Specifications require that every 31 days a complete valve lineup be performed for the fire suppression systems. This will require an operator to enter areas of high radiation in order to verify the position of vent/drain valves on the fire system piping. This requirement is in effect even if the valves are locked or sealed. This seems overly restrictive from a Technical Specification viewpoint since the verification of valve position on ECCS systems does not require verification of valves that are locked or sealed in position. Therefore, it would seem appropriate that the fire protection valves should receive the same requirements.

ISSUE 16

SURVEILLANCE REQUIREMENTS

4.7.7.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the minimum contained water supply volume.

b. At least once per 31 days (on a STAGGERED TEST BASIS) by starting (each) ~~(the) electric motor driven~~ fire suppression pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path, that is not locked, sealed or otherwise secured in position, is in its correct position.

d. At least once per 6 months by performance of a system flush.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that each automatic valve in the flow path actuates to its correct position,

2. Verifying that each fire suppression pump develops at least (2500) gpm at a system head of (250) feet,

3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

4. Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to ___ psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.7.1.2 Each diesel driven fire suppression pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying the fuel day tank contains at least ( ) gallons of fuel.

2. Verifying the fuel storage tank contains at least ( ) gallons of fuel.

3. Starting the fuel transfer pump and transferring fuel from the storage tank to the day tank.

4. Starting the diesel driven pump from ambient conditions and operating for greater than or equal to 30 minutes on recirculation flow.

ISSUE 16

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING COND' ' FOR OPERATION

3.7.7.2 The following spray and sprinkler systems shall be OPERABLE:

a. (Plant dependent - to be listed by name and location.)

b.

c.

APPLICABILITY: Whenever equipment protected by the spray and/or sprinkler systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.2 Each of the above required spray and sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path, that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

ISSUE 16

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each (deluge) nozzle's spray area to verify that the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray (and) sprinkler header and verifying each open head spray (and) sprinkler nozzle is unobstructed.

ISSUE 16

PLANT SYSTEMS

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.3 The following low pressure and high pressure CO_2 systems shall be OPERABLE:

a. (Plant dependent - to be listed by name and location.)

b.

c.

APPLICABILITY: Whenever equipment protected by the CO_2 systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required CO_2 systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.3.1 Each of the above required CO_2 systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path *, that is not locked, sealed, or otherwise secured in position,* is in its correct position.

4.7.7.3.2 Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than ______ and pressure to be greater than _____ psig, and

b. At least once per 18 months by verifying:

1. The system valves and associated ventilation dampers and fire door release mechanisms actuate, manually and automatically, upon receipt of a simulated actuation signal, and

2. Flow from each (accessible) nozzle during a "Puff Test."

ISSUE 16

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.3.3 Each of the above required high pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying the CO_2 storage tank weight to be at least 90% of full charge weight.

b. At least once per 18 months by:

1. Verifying the system and associated ventilation dampers and fire door release mechanisms actuate, manually and automatically, upon receipt of a simulated actuation signal, and

2. Performance of a flow test through headers and nozzles to assure no blockage.

ISSUE 16

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.4 The following Halon systems shall be OPERABLE with the storage tanks having at least 95% of full charge weight (level) and 90% of full charge pressure:

a. (Plant dependent - to be listed by name and location.)

b.

c.

APPLICABILITY: Whenever equipment protected by the Halon systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.4 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path, that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight (level) and pressure.

c. At least once per 18 months by:

1. Verifying the system and associated ventilation dampers and fire door release mechanisms actuate, manually and automatically, upon receipt of a simulated actuation signal, and

2. Performance of a flow test through (accessible) headers and nozzles to assure no blockage.

ISSUE 17 6/22/82

Reactor Protection System Response Times - Flow Biased Simulated Thermal Power - Upscale

Affected Specifications:

a. Table 3.3.1-2, Item 2.b and **Footnote

b. Table 4.3.1.1-1, Footnote (h)

Current Requirements:

a. Technical Specification Table 3.3.1-2, Item 2.b requires that the response time of the Flow Biased Simulated Thermal Power - Upscale RPS Scram be determined to be 0.09 seconds, exclusive of the simulated thermal power constant.

b. Technical Specification Table 4.3.1.1-1, Footnote (h) requires that the 6 $\pm$ 1 second simulated thermal power time constant be adjusted or verified each refueling. (In most cases, this circuit cannot be adjusted.)

Proposed Change:

a. Instead of requiring that two (2) criteria be applied to the Flow Biased Simulated Thermal Power - Upscale RPS Scram, the Technical Specifications should be changed to require only one (1) criteria. Therefore, Table 3.3.1-2 would indicate this criteria and Footnote (h) would be deleted from Table 4.3.1.1-1.

Justification:

In order to do the present required testing, the following must be done:

1. The overall APRM string response time must be determined. This is done in accordance with the definition of RPS response time. This response time would include the 6 $\pm$ 1 second simulated thermal power time constant.

Justification (cont'd)

2. The 6 $\pm$ 1 second simulated thermal power time constant would have to be verified. This requires that the printed circuit board containing the RC network for the simulated thermal power time constant be removed from the APRM drawer and placed on an extender. The 6 $\pm$ 1 second time constant is then verified and the board replaced.

3. The time constant determined in (2) above is then subtracted from the overall response time from (1) above in order to determine if the 0.09 second criteria is being met.

The determination of the times as described above is not consistent with the determination of response times as required in the other Technical Specifications. If the philosphy that is used in all other response time testing in the Technical Specification is applied to the APRM Flow Biased Simulated Thermal Power - Upscale Scram, only the overall time described in (1) above would be necessary. If the overall criteria is established and this criteria is met, then the time that is used in the safety analysis will be verified.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	NA 6±1
b. Flow Biased Simulated Thermal Power - Upscale	< ~~(0.09)~~**
c. Fixed Neutron Flux - Upscale	< (0.09)
d. Inoperative	NA
(e. Downscale	NA
3. Reactor Vessel Steam Dome Pressure - High	< (0.55)
4. Reactor Vessel Water Level - Low, Level 3	< (1.05)
5. Main Steam Line Isolation Valve - Closure	< (0.06)
6. Main Steam Line Radiation - High	NA
7. (Primary Containment) (Drywell) Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	< (0.06)
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	< (0.08)#
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. (This provision is not applicable to Construction Permits docketed after January 1, 1978. See Regulatory Guide 1.18, November 1977.)

**~~Not including~~ measurements simulated thermal power time constant ONLY

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U(b), S S	S/U(c), W W	R R	2 3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor(f):				
a. Neutron Flux - Upscale, Setdown	S/U(b), S S	S/U(c), W W	SA SA	2 3, 5
b. Flow Biased Simulated Thermal Power - Upscale	S, D((g))	S/U(c), W	W(d)(e), SA, ~~(R(h))~~	1
c. Fixed Neutron Flux - Upscale	S	S/U(c), W	W(d), SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
(e. Downscale	S	W	SA	1, 2)
3. Reactor Vessel Steam Dome Pressure - High	(S)	M	(R)	1, 2
4. Reactor Vessel Water Level - Low, Level 3	(S)	M	(R)	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2
7. (Primary Containment) (Drywell) Pressure - High	(S)	M	(R)	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	(S)	M	(R)	1, 2, 5
9. Turbine Stop Valve - Closure	(S)	M	(R)	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	(S)	M	(R)	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least (1/2) decades during each startup and the IRM and APRM channels shall be determined to overlap for at least (1/2) decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER > 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
((g) Verify measured core flow to be greater than or equal to rated core flow.)
~~((h) This calibration shall consist of (verifying) (the adjustment as required, of) the 6 ± 1 second simulated thermal power time constant.)~~

ISSUE 18

TIP SHEAR VALVE TESTING

AFFECTED SPECIFICATIONS

4.6.3.5

CURRENT REQUIREMENTS

At the present time the following is required:

a. Once per 31 days the continuity of the TIP Shear Valves must be verified.

b. Once per 18 months an explosive squib from one of the TIP Shear Valves must be exploded.

PROPOSED CHANGE

Delete Technical Specification 4.6.3.5

JUSTIFICATION

Hanauer's letter of 3/26/82 to Denton gives the testing of the TIP Shear Valves as a low priority (see attached letter). Based on the NRC's letter, this testing requirement should be deleted from the Tech Specs due to the insignificance of these valves and the fact that the valves may be in a high radiation area.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shown in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow at greater than a (10) psid differential pressure.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing (at least one)(the) explosive squib(s) from (at least one)(the) explosive valve(s) (, such that each explosive squib in each explosive valve will be tested at least once per 36 months,) and initiating the explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and operating life, as applicable.

DELETE

Add to list of action items TIP

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MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Stephen H. Hanauer, Director
Division of Safety Technology

SUBJECT: PRELIMINARY RANKING OF NRR GENERIC SAFETY ISSUES

REFERENCE: NRR FY 1982-1984 Operating Plan, dated November 16, 1981

In accordance with the schedule established in the NRR FY 1982-1984 Operating Plan, the preliminary ranking of Generic Safety Issues and TMI Action Plan Issues has been completed by DST and is enclosed. A draft version of this ranking was issued to all NRR Divisions for review on March 4, 1982. Enclosures 1 and 2 represent the first phase of the development of an integrated priority list which is scheduled for completion by September 30, 1982.

Enclosure 1 is a complete list of current NRR Generic Safety Issues and TMI Action Plan Issues assigned to NRR for development. Based on the judgment of a group of senior NRR staff members, each issue has been assigned a preliminary ranking of either high, medium, or low priority after consideration was given to the relative safety significance of each issue and the cost of its solution. Other generic safety issues will be added to this list as they are identified.

Enclosure 2, which is incomplete, contains detailed evaluations of some of the issues listed in Enclosure 1. Because of insufficient manpower resources, DST has contracted Pacific Northwest Laboratory to evaluate those issues with a preliminary priority ranking of high or medium. This action has been taken in order that the evaluation of all issues will be completed by the end of FY 1982 as specified in the NRR Operating Plan. Evaluation of the remaining low priority issues will be completed by SPEB.

A copy of this memorandum and its enclosures is being sent to the ACRS with whom a meeting will be scheduled to discuss this prioritization and to obtain comments.

Original Signed by
Malcolm L. Ernst for

Stephen H. Hanauer, Director
Division of Safety Technology

8204280036

Enclosures:
1. NRR Generic Safety Issues
2. A Prioritization of NRR Generic Safety Issues

OFFICE	DST:SPEB	DST:SPEB	DST:AD:T	DST:D			
SURNAME	REmrit:kn	WMinners	MLErnst	SHanauer			
DATE	3/24/82	3/24/82	3/ /82	3/ /82			

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CONCLUSION

This issue was determined to be of LOW priority after preliminary screening of all generic issues was conducted.

*ISSUE 8: INADVERTENT ACTUATION OF SAFETY INJECTION IN PWRs

DESCRIPTION

Operator errors, instrument malfunction, and reactor transients and trips have been reported as the cause of inadvertent actuation of the safety injection system. At least forty cases of inadvertent actuation of safety injection have been identified in NUREG-0572.[4] Approximately one-fourth of the events sampled were due to operator error. The problem is repetitive in nature; at several facilities the problem has a long history. The vast majority of events occurred in Westinghouse Nuclear Steam Supply Systems, whereas plants supplied by other vendors had few or no reported events.

CONCLUSION

This issue was determined to be of LOW priority after preliminary screening of all generic issues was conducted.

*ISSUE 10: SURVEILLANCE AND MAINTENANCE OF TIP ISOLATION VALVES AND SQUIB CHARGES

DESCRIPTION

This issue apparently originated at a DOR proposal. It is discussed briefly in SECY-80-325:[8]

"The Transversing Incore Probe (TIP) monitoring system in a BWR penetrates the primary containment. In order to provide effective isolation capability when the TIP probes are in the reactor core, each line is equipped with an explosive (squib) isolation valve which will sever and seal the line when manually actuated. These valves would be used when containment isolation is needed and the TIP probe(s) could not be withdrawn from the core."

"At present, there are no surveillance requirements for these valves since they can only be destructively tested. Nevertheless, there is a finite "shelf-life" associated with the squib charges. Periodic replacement of the squib charges and circuit checks should be included in the Technical Specifications of each BWR plant, similar to the requirements for the explosive valves in the Standby Liquid Control System."

PRIORITY DETERMINATION

(a) Frequency Estimate

To have a radioactivity-releasing event, either of two sequences must take place:

(1) The probe must be inserted into the reactor, a leak must develop in the TIP tube, and the squib valve must fail.

(2) The TIP tube must leak, the ball valve must fail, and the squib valve must fail.

Good data on failure rates are not available. However, leaks have occurred in TIP tubes within the reactor. We will assume a frequency of 3×10^{-2}/R-Y. With about 50 tubes in each reactor, this is 6×10^{-4} leaks/tube-year.

A probe is inserted in each tube for about 5 minutes each month. This gives a probability of about 10^{-4} of a tube containing a cable at any given time.

No data are available for the ball valve failure rate. We estimate a worst-case upper bound of 10^{-3} failures per actuation and/or pressure impingement. (The probe is normally withdrawn and the ball valve is normally closed).

Finally, we estimate that, even without extra maintenance, the squib valve failure rate is no greater than 10%.

With these admittedly judgmental numbers, the frequency of sequence (1) is 3×10^{-7}/R-Y and the frequency of sequence (2) is 3×10^{-6}/R-Y.

(b) Consequence Estimate

This is a very small leak. The TIP guide tube has an ID of 0.272 inches and is roughly 50 feet long. For sequence (2), assuming the TIP tube snaps off completely in the reactor, a rough hand calculation estimates a leak rate of about one pound per second. (This is insignificant for LOCA analyses). At one μCi/g activity and assuming one week to shut down the plant and stop the leak, the release is only 300 Ci.

For sequence (1), the cable is in place with an 8-mil circumferential gap. Thus, its release is very low and, since its frequency is also lower than that of sequence (2), sequence (1) will not be considered further.

(c) Cost Estimate

No figures are available for costs. We will conservatively assume a $2,000 cost to each plant (roughly one man-hour per squib charge), plus 1 week of NRC staff time for each of 24 operating BWRs.

CONCLUSION

The LOW priority score resulting from these figures is, at most, 0.25 Ci/year/million dollars. This is almost an order of magnitude less than the lowest figure in SECY-81-513.[1] Thus it should not be considered further even if the cost is low.