UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
March 5, 1984

Docket No.: 50-410

APPLICANT: Niagara Mohawk Power Corporation (NMPC)

FACILITY: Nine Mile Point, Unit 2

SUBJECT: SUMMARY OF MEETING WITH NMPC TO DISCUSS ADMINISTRATIVE MATTERS CONCERNING CLOSING OUT OPEN ITEMS ON NINE MILE POINT, UNIT 2

On February 24, 1984, the NRC staff met with representatives from NMPC to discuss administrative matters concerning closing out open items on Nine Mile Point, Unit 2 (NMP-2).

During the meeting, NMPC was given a list of open items which have been identified by the staff and are expected to be included in the Draft SER for NMP-2. This list is included as Attachment 1.

NMPC outlined the steps to be taken by NMPC and their consultants in closing out open items. NMPC provided a schedule for responding outstanding questions from the NRC staff (Attachment 2).

NMPC was requested to review the open items on the list, and the Draft SER when it is issued, to group the open items for discussions at meetings with the NRC staff. These meetings are to be held to close out the open issues. In order to facilitate the closing out of open issues NMPC was requested to develop a faster system for submitting formal responses to open issues.

The questions to be used as a basis for the Mechanical Engineering Branch (MEB) SER review were provided to NMPC (Attachment 3). This meeting was tentatively scheduled for March 27-29, 1984, but will be postponed until early June.

The next Instrumentation and Controls meeting was scheduled for April 4-5, 1984.

Qualification testing of Diesel Generators was also discussed. Additional details of differences between NMP-2 and Susquehanna and Zion Diesel Generators was requested.

Attendants at the meeting were as follows:

A. Zallnick, NMPC
N. Rademacher, NMPC
M. Haughey, NRC
J. Lazevnick, NRC*
E. Tomlinson, NRC*

*diesel generator discussion only.

Original signed by:

Mary F. Haughey, Project Manager
Licensing Branch No. 2
Division of Licensing

Attachments:
As stated

cc w/ attachments:
See next page

DL:LB#2/PM	DL:LB#2/BC
MFHaughey:pt	ASchwencer
3/5/84	3/6/84

Nine Mile Point 2

Mr. Gerald K. Rhode
Senior Vice President
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

cc: Mr. Troy B. Conner, Jr., Esq.
Conner & Wetterhahn
Suite 1050
1747 Pennsylvania Avenue, N.W.
Washington, D. C. 20006

Mr. Richard Goldsmith
Syracuse University
College of Law
E. I. White Hall Campus
Syracuse, New York 13210

Mr. Jay Dunkleberger, Director
Technological Development Programs
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Ezra I. Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

Resident Inspector
Nine Mile Point Nuclear Power Station
P. O. Box 99
Lycoming, New York 13093

Mr. John W. Keib, Esq.
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Jay M. Gutierrez, Esq.
U. S. Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

OPEN ITEMS DRAFT SER

complete its review of these items before the operating license is issued. The staff will discuss the resolution of each of these items in a supplement to this report. These items are listed below and are discussed further in the sections of this report as indicated.

Issue		DSER Section
(1)	nearest population center	2.1
(2)	long-term diffusion estimates	2.3.5
(3)	seismic design of revetment ditch and flood protection berms	2.4.10
(4)	protection against PMP	2.4.2.2
(5)	protection of the main stack from wave forces from PMWS	2.4.10
(6)	adequacy of the ultimate heat sink	2.4.11.2
(7)	ground water level	2.4.12.2
(8)	analysis of postulated rupture of a liquid radwaste tank	2.4.13
(9)	recalculation of the changing stresses at the site, assuming shallower burial depths than in the original calculations	2.5.1, 2.5.2
(10)	an evaluation of the significance of the decoupled regional stress regimes in the Paleozoic and basement rocks measured in the site region	2.5.1
(11)	assessment of seismic or aseismic origin of sedimentary structures	2.5.1, 2.5.2

Issue		DSER Section
	CAPS	
(12)	monitoring program of the cooling tower fault designed to ascertain the strain or displacement rate on the fault	2.5.1, 2.5.2
(13)	magnitude of the fault movement for all seismic Category I structures in the power block	2.5.4.5.1
(14)	excavation, backfill and geological mapping data of the main stack	2.5.4.4.3
(15)	liquid faction potential analysis for the Category I electrical duct bank and manhole	2.5.4.7
(16)	update of slope inclinometer and rock extensometer data	2.5.4.10
(17)	dynamic stability of the slopes of the revetment ditch	2.5.6.2.3, 2.5.6.2.4
(18)	PMP - flood protection berm	2.5.6.3
(19)	turbine maintenance	3.5.1.3
(20)	adequacy of tornado missile protection for diesel generator exhaust outside air intakes for HVAC systems safety-related buried piping	3.5.2
(21)	effects of postulated pipe breaks	3.6.1
(22)	stress and cumulative usage factor limits and inspection requirements for piping inside the break exclusion zone	3.6.2
(23)	postulation of moderate energy cracks inside containment and of high-energy cracks	3.6.2

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Issue		DSER Section
(24)	postulation of pipe ruptures	3.6.2
(25)	feedwater isolation check valves	3.6.2
(26)	design of pipe rupture restraints	3.6.2
(27)	vertical floor flexibility in the seismic analysis	3.7.2
(28)	results of the concrete containment ultimate capacity analysis	3.8.1
(29)	containment response to SRV/pool dynamic loads	3.8.1
(30)	deviations from the applicable provisions of ASME Section III, Division 2	3.8.1
(31)	deviations from the applicable requirements of ACI 349 as amended by RG 1.142	3.8.3, 3.8.4, 3.8.5
(32)	SRV/pool dynamic loads on containment interior structure	3.8.3
(33)	consideration of upward seismic load effects in the foundation stability analysis of the screenwell building	3.8.5
(34)	structural audit action items	3.8.6
(35)	systems and locations to be monitored during the pre-operational testing program	3.9.2.1
(36)	acceptance criteria for observed or measured vibration levels	3.9.2.1

Issue		DSER Section
(37)	inclusion of all essential safety-related instrument lines in the vibration monitoring program	3.9.2.1
(38)	seismic design of HVAC systems	3.9.2.2
(39)	seismic methods used for the analysis of the safety-related piping in pipe tunnels	3.9.2.2
(40)	documentation of analysis for combined loads (LOCA and SSE)	3.9.2.4
(41)	methodology of combining loads	3.9.3.1
(42)	clarification of the BWR Mark II hydrodynamic loads	3.9.3.1
(43)	assurance that downcomers will not develop fatigue cracks	3.9.3.1
(44)	design of piping and supports in the wetwell area.	3.9.3.1
(45)	design of SRVs and attached discharge piping	3.9.3.2
(46)	design and construction of ASME Class 1, 2 and 3 component supports	3.9.3.3
(47)	stress categories and limits for core support structures and the applicable codes used for evaluation of the faulted condition	3.9.5
(48)	response to IE Bulletin 80-07	3.9.5
(49)	leak rate testing of isolation valves	3.9.6
(50)	preservice and inservice testing of pumps and valves	3.9.6

Issue		DSER Section
(51)	seismic and dynamic equipment qualification program	3.10
(52)	pump and valve operability assurance	3.10
(53)	dependability of containment isolation (purge valves)	3.10
(54)	performance testing of relief and safety valves (II.D.1)	3.10
(55)	qualification of accumulators on automatic depressurization system valves (II.K.3.28)	3.10
(56)	long-term operability of deep draft pumps	3.10
(57)	environmental qualification of equipment	3.11
(58)	irradiation fuel surveillance program	4.2
(59)	LPMS (loose parts monitoring system)	4.4.6 and (Table 4.4.0)
(60)	inadequate core cooling detection system (II.F.2)	4.4.7
(61)	pipe break in the BWR scram system	4.6
(62)	lead factors in surveillance capsules	5.3.1
(63)	P-T (pressure-temperature) curves	5.3.2, 5.3.3
(64)	ratio of neutron flux density of specimens in the surveillance capsule to peak neutron flux density at RPV	5.3.3
(65)	reactor coolant pressure boundary inservice inspection and testing	5.2.4

Issue		DSER Section
(66)	fracture prevention of containment pressure boundary	6.2.7
(67)	control room habitability	6.4
(68)	exceptions and deviations to RG 1.52, Rev. 2	6.5.1.5
(69)	fission product control systems	6.5.3
(70)	inservice inspection of Class 2 and 3 components	6.6
(71)	spent fuel storage pool materials surveillance	9.1.2
(72)	spent fuel storage pool materials surveillance	9.1.2
(73)	spent fuel pool design	9.1.2
(74)	light load handling system	9.1.4
(75)	heavy loads	9.1.5
(76)	failure of nonseismic buried pipe near safety-related buried pipe	9.2.1
(77)	backup nitrogen supply system	9.3.1
(78)	periodic air quality testing	9.3.1
(79)	flooding by rupture of nonseismic Category D piping, vessels, or tanks or by failure of a backflow prevention device in the drainage system	9.3.3
(80)	postaccident sampling (II.B.3)	9.3.2

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Issue		DSER Section
(81)	drainage of leakage water away from safety-related components or systems	9.3.3
(82)	design capability of the CB HVAC system (410.41 and 410.42)	9.4.1
(83)	protection against hydrogen accumulation in the battery rooms	9.4.1
(84)	outdoor temperatures assumed for sizing of the CB HVAC	9.4.1
(85)	spent fuel pool area ventilation system	9.4.2
(86)	tornado missle protection for diesel generator building louvers	9.4.4
(87)	diesel generator building HVAC system conformance to GDC 4	9.4.4
(88)	protection of essential electrical components from failure due to the accumulation of dust and particulate material	9.4.4
(89)	potential systems interaction	9.5.1.II.B
(90)	administrative controls	9.5.1.III
(91)	fire brigade and fire brigade training	9.5.1.IV
(92)	qualification of fire doors	9.5.1.V.A
(93)	floor drains	9.5.1.V.A

Issue		DSER Section
(94)	safe shutdown capability	9.5.1.V.B
(95)	alternate shutdown capability	9.5.1.V.B
(96)	emergency lighting	9.5.1.V.G
(97)	installation of fire detectors	9.5.1.VI.A
(98)	qualification of the electric fire pump	9.5.1.VI.A
(99)	valve supervision	9.5.1.VI.B
(100)	quality group classification information on the design of the turbine gland sealing system	10.4.3.5
(101)	protection of safety-related systems from flooding ~~from flooding~~ from a postulated failure of a circulating water expansion joint or line failure as a result of an SSE	10.4.5
(102)	parameters used for calculating liquid and gaseous source terms	11.1.2
(103)	assessment of the capability of liquid and gaseous radwaste systems for keeping the levels of radioactivity in effluents ALARA	11.2.1, 11.2.2, 11.3.1
(104)	assessment of charcoal absorber tank failure for 10 CFR 100 dose guidelines	11.3.1
(105)	process control program for the solid radwaste system	11.4.1, 11.4.2
(106)	compliance program to meet 10 CFR 61	11.4.2

Issue	DSER Section
(107) high-range noble gas monitor (II.F.1)	11.5
(108) airborne radioactivity levels (471.1)	12.2
(109) airborne radionuclide concentration in liquid radwaste handling area (471.3)	12.2
(110) conformance to RGs 1.8, 8.8, and 8.10 (471.4)	12.0
(111) dose rate criteria (II.B.2) (471.9)	12.3.2
(112) projected doses to individuals and dose rate maps (471.16)	12.3.2
(113) whole-body dose calculations (471.17)	12.3.2
(114) postaccident access and shield design review (471.19)	12.3.2
(115) crud buildup (471.19)	12.3.2
(116) postaccident vital area monitors (471.19)	12.3.2
(117) compliance with TMI II.B.2, shielding	12.3.2
(118) inhalation exposure (471.12)	12.4.2.2
(119) estimate of N-16 dose contribution (471.13)	12.4.2.2
(120) estimate of doses outside of plant structures (471.14)	12.4.2.2
(121) dose assessment (471.11)	12.4

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Issue	DSER Section
(122) separation of health physics and chemistry functions (471.21)	12.5.1
(123) qualifications of Superintendent, Chemistry and Radiation Management (471.21)	12.5.1
(124) qualifications of temporary RPMs and commitment to ANSI 3.1	12.5.1
(125) training of health physics technicians	12.5.1
(126) ANSI 18.1 qualified health physics technician	12.5.1
(127) initial training program	13.2.1.1
(128) requalification training program	13.2.1.2
(129) immediate upgrading of reactor operator and senior reactor operator training and qualifications	13.2.1.4
(130) administration of training programs	13.2.1.4
(131) STA training program	13.2.2
(132) emergency planning	13.3
(133) commitment to Section 5.3 of ANSI/ANS 3.2	13.5.1.1
(134) evaluation and development of procedures for transients (I.C.1)	13.5.2C
(135) upgraded emergency operating procedures	13.5.2

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Issue	DSER Section
(136) NSSS vendor review of procedures (I.C.7)	13.5.2
(137) pilot monitoring of selected emergency procedures for NTOLs (I.C.8)	13.5.2
(138) ATWS procedures	13.5.2
(139) licensed operator training program	13.2.1.1
(140) simulator training	13.2.1.1
(141) requalification	_____
(142) loss-of-air-supply tests (640.06)	14.2.7
(143) single-failure-proof cranes (NUREG-0612) and heavy load testing (NUREG-0554) (640.07)	14.2
(144) periodic testing of diesel generators (RG 1.108) (640.08)	14.2.7
(145) applicability of RG 1.140 to radwaste building exhaust (640.09)	14.2.7
(146) preoperational test abstracts (640.10, 640.13, 640.15, 640.16, 640.17, 640.19, 640.20, 640.21)	14.2.12
(147) protection of control room operators against accidental chlorine release (640.18)	14.2.12
(148) startup test abstracts (640.23, 640.24, 640.26, 640.27, 640.29)	14.2.12

Issue	DSER Section
(149) incorporation of specific testing identified into test abstracts (640.34)	14.2.12
(150) preoperational tests to be conducted after fuel load and tests to be exempted from prior notification (640.35)	14.2.12
(151) fuel handling accident	15.
(152) loss-of-coolant accident	15.
(153) leakage integrity from systems outside containment (III.D.1.1)	15.9.5
(154) DCDR Summary Report (I.D.1)	18.0
(155) SPDS safety analysis and implementation plan (I.D.2)	18.0
(156) Technical Specifications	16.0
(157) safeguards	----

1.9 Confirmatory Issues

At this point in the review there are some items that have essentially been resolved to the staff's satisfaction, but for which certain confirmatory information has not yet been provided by the applicant. In these instances, the applicant has committed to provide the confirmatory information in the near future. If staff review of the information provided for an item does not confirm preliminary conclusions, that item will be treated as open and the staff will report on its resolution in a supplement to this report.

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Issue	Section
(1) sharing of fuel handling and fuel storage	1.
(2) long-term diffusion estimates	2.3.5
(3) fuel rod fracturing	4.2
(4) Gadolinia thermal conductivity equation (incorporation in GESSAR II calculations)	4.2
(5) fire protection training	13.2.2
(6) full reactor isolation test abstract (640.28)	14.2.12
(7) test description for confirmatory in-plant tests of SRVs (640.30)	14.2.12
(8) loss of turbine generator and offsite power (640.32)	14.2.12

1.10 License Condition Items

There are certain issues for which a license condition may be desirable to ensure that staff requirements are met during plant operation. The license condition may be in the form of a condition in the body of the operating licenses or a limiting condition for operation in the Technical Specifications appended to the license.

Item	Section
(1) activation of the rack bar heating system when lake temperature drops (TS)*	2.4.7

*To be incorporated in the Technical Specifications appended to the license.

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Issue	Section
(2) Limiting conditions for operation, i.e., shutdown or system isolation when the final approved leakage limits are not met, also surveillance requirements, which will state the acceptable leak rate testing frequency (TS)	3.9.6
(3) fuel rod internal pressure criterion	4.2
(4) stability analysis prior to operation beyond Cycle 1	4.4.4
(5) crud deposition (TS)	4.4.5
(6) single loop operation (TS)	4.9
(7) natural circulation (TS)	4.9
(8) operation and surveillance of LPMS (TS)	4.9
(9) MSIV leakage	6.7
(10) operation while RHR is in pool cooling mode (TS)	9.1.3
(11) Following the first refueling outage the applicant shall have made commitments acceptable to the staff regarding the guidelines of Section 5.1.2 through 5.1.6 of NUREG-0612 (Phase II - 9 month responses to the NRC generic letter dated December 22, 1980)	9.1.5

1.11 Unresolved Safety Issues

Section 210 of the Energy Reorganization Act of 1974, as amended, reads as follows:

ATTACHMENT 2

STATUS OF RESPONSES TO BRANCH TECHNICAL QUESTIONS

TECHNICAL QUESTION AREAS

	INSTRUMENT AND CONTROLS* 421	ER-OLS	RADIOLOGIC 470, 471 EFFLUENT 460	PIPING 210 STRUCTURE/220 SEISMIC-230 GEOLOGY-231 HYDROLOGY-240 GEOTECH-241	EQUIPMENT QUAL. 270 271	QA 260	FIRE PROTECTION 280	POWER SYSTEM 430	CONTAINMENT REACTOR PHYSICS CORE PERFORM 480,491,492	REACTOR SYSTEM 440	AUXILIARY SYSTEM 110	OTHER 100,250,251 252,281,311 450,451,620 630,640,730	TOTALS
# OF QUESTIONS RECEIVED	46	51	46	103	13	51	33	118	66	49	51	106	733
RESPONSES COMPLETED		46	30	96	8	50	29	99	54	43	42	85	582
# OF OUTSTANDING RESPONSES	46	5	16	7	5	1	4	19	12	6	9	21	151

SCHEDULE FOR RESPONSE COMPLETION

NUMBER OF RESPONSES

1984													
JANUARY		1						10		1	1	3	16
MARCH		4	4	3	3			7	7		1	3	32
JUNE			5	2		1	3		5	4	4	5	29
SEPTEMBER			4	1	1		1	2			2	9	20
DECEMBER			2										2
1985													
MARCH				1							1		2
JUNE-DECEMBER			1		1					1		1	4

* Responses will be discussed in meetings with the commission scheduled to commence in February.

MECHANICAL ENGINEERING BRANCH

NINE MILE 2 SER QUESTIONS

SECTION 3.6.2

210.17 Branch Technical Position MEB 3-1 requires that certain stress and cumulative usage factor limits be met in the break exclusion zone. The criteria contained in the FSAR are not in compliance with these limits. Provide justification for the criteria used. In particular, address those cases for Equation (10) exceeding 2.4 S_m and the cumulative usage factor exceeding 0.1 for Class 1 piping.

210.18 Have occasional loads been considered in the evaluation of the sum of Equations (9) and (10) when comparing to the limits for Class 2 piping in the break exclusion area?

210.19 Provide assurance that 100% volumetric inservice examination of all pipe welds in the break exclusion area will be conducted during each inspection interval as defined in IWA-2400, ASME Code, Section XI.

210.20 Breaks in non-nuclear high energy piping not seismically analyzed (nor qualified) should be postulated at those locations which produce the greatest effect on an essential component or structure irrespective of the fact that the high stress or fitting criteria might not require a break to be postulated. Provide assurance that the above criteria have been met.

210.21 What criteria are used for postulating moderate energy leakage cracks inside containment?

210.22 Discuss how high energy leakage cracks are considered?

210.23 Discuss how pipe whip and jet impingement effects were determined for those postulated breaks in high energy piping that is not restrained.

210.24 Provide assurance that the tip deflection of a restrained whipping pipe does not adversely affect nearby safety-related components from performing their safety-related function.

210.25 Describe in more detail the design procedures and methodologies used in the jet impingement analyses. Specifically, address 1) the jet loads and jet configurations used for circumferential and longitudinal breaks, 2) how targets are determined, and 3) the acceptance criteria used to evaluate the effects on safety-related components and structures.

210.26 Provide the criteria used in the design of pipe rupture restraints including the auxiliary steel used to support the pipe rupture restraint. Provide assurance that the pipe rupture restraint and supporting structure cannot fail during a seismic event.

210.27 Provide the design criteria for pipe rupture restraints that also support piping.

210.28 In order to assure the pipe break criteria has been properly implemented, the Standard Review Plan requires the review of sketches showing the postulated rupture locations and summaries of the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative usage factor, and the calculated primary plus secondary stress range. The vast majority of this information in the FSAR is either preliminary or incomplete. Please provide a schedule for the completion of Tables 3.6A-2 through 3.6A-60 and Figures 3.6A-12 through 3.6A-39.

210.29 On page 3.6A-28 an amplification factor of between 1.0 and 1.1 to

account for pipe rebound is discussed. Provide justification for the use of an amplification factor of less than 1.1.

210.30 Provide justification for shape factors of less than unity as discussed on page 3.6A-35 of the FSAR.

210.31 Appendix 3C is very incomplete. Provide a schedule for its completion.

210.32 Provide a list of all instances where full break area opening times in excess of one millisecond were used. See page 3.6B-8.

210.33 Provide justification for using a thrust coefficient of less than 1.26 for saturated steam and 2.0 for subcooled water as discussed on page 3.6B-9.

210.34 Provide a list of all instances where mechanistic approaches were used to reduce break areas as discussed on page 3.6B-6.

210.35 Provide the basis for assuring that the feedwater isolation check valves can perform their function following a postulated break of the feedwater line outside containment.

210.36 Discuss the types of protection used to mitigate the effects of jet impingement on safety-related components and structures.

SECTION 3.9.2

210.37 Provide the acceptance criteria to be used in determining if the vibration loads observed or measured during the pre-operational

testing are acceptable. Specifically, address how the vibration amplitudes will be related to a stress load and what stress levels will be used for both steady-state and transient vibration.

It is the staff's position that all essential safety-related instrumentation lines should be included in the vibration monitoring program during pre-operational or start-up testing. We require that either a visual or instrumented inspection (as appropriate) be conducted to identify any excessive vibration that will result in fatigue failure.

Provide a list of all safety-related small bore piping and instrumentation lines that will be included in the initial test vibration monitoring program.

The essential instrumentation lines to be inspected should include (but are not limited to) the following:

a. Reactor pressure vessel level indicator instrumentation lines (Used for monitoring both steam and water levels).

b. Main steam instrumentation lines for monitoring main steam flow (used to actuate main steam isolation valves during high steam flow).

c. Reactor core isolation cooling (RCIC) instrumentation lines on the RCIC steam line outside containment (used to monitor high steam flow and actuate isolation).

d. Control rod drive lines inside containment (not normally pressurized but required for scram).

210.38 Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety-related systems and

components, it is requested that maintenance records for snubbers be documented as follows:

Pre-service Examination

A pre-service examination should be made on all snubbers listed in Tables 3.7-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9. This examination should be made after snubber installation but not more than six months prior to initial system pre-operational testing, and should as a minimum verify the following:

1. There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.
2. The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.
3. Snubbers are not seized, frozen or jammed.
4. Adequate swing clearance is provided to allow snubber movement.
5. If applicable, fluid is to be recommended level and is not leaking from the snubber system.
6. Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, re-examination of items 1, 4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

Pre-Operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250°F should be verified as follows:

a. During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

b. For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

c. Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepancies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

The above described operability program for snubbers should be included and documented by the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for the pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

210.39 Please provide a statement as to the compliance with NUREG-0619, "BWR Feedwater Nozzles and Control Rod Drive Return Line Nozzle Cracking".

210.40 Provide the basis used for the design of piping supports and anchors which separate seismically designed piping and non-seismic Category I piping. Include in your discussion the loads and load combinations used and how the local pipe wall stresses are considered.

210.41 Describe the design considerations given to assure that an

adequate number of modes have been used in the dynamic piping anlayses performed for:

1) seismic loadings,
2) SRV loadings,
3) LOCA loadings, and
4) hydraulic transients (e.g. steam and water-hammer.)

210.42 Explain how in the design process the reinforcement thickness of branch connections are determined for both internal pressure and mechanical loads and incorporated into the fabricated piping. Provide assurance that all branch connections decoupled from the main run piping on the piping analytical model are designed and fabricated to the required reinforcement area.

210.43 The staff finds insufficient information describing the design of safety-related HVAC ductwork and supports. Provide the design basis used for qualifying the HVAC ductwork and support structural integrity.

SECTION 3.9.3

210.44 Provide the basis for assuring that ASME Code Class 1, 2, and 3 piping systems are capable of performing their safety function under all plant conditions. Describe the methodology used to assure the

functional capability of essential piping system when service limits C or D are specified.

210.45 Provide a discussion of the design considerations used for safety and relief valve loads and piping reactions. Include in your discussion the basis for assuring that the valve end loads and the support arrangement for the affected piping are acceptable.

210.46 Describe those short-term and long-term actions being taken to preclude the occurrence of cracking in jet pump hold down beams as described in IE Bulletin 80-07.

210.47 Describe briefly the design considerations given to the piping stress analyses for the mainsteam piping and attached safety relief valve discahrge piping for the alternate shutdown cooling mode. Specifically address the capability of the spring hangers to accommodate the additional weight of water during this mode.

210.48 Provide assurance (or a commitment) that the design of all safety-related mechanical components and their supports can withstand the effects of safety-relief valve discharge laods as defined in NUREG-0802, "Safety/Relief Valve Quencher Loads = Evaluation for BWR Mark II and III Containments."

210.49 Provide assurance (or a commitment) that the design of all safety-related mechanical components and their supports can withstand the effects of loss-of-coolant accident loads as defined in NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria."

210.50 Provide the basis for assuring that a fatigue crack will not occur 1) in the safety relief valve discharge piping in the suppression pool wetwell airspace and 2) in the suppression pool downcomers. The staff requests that ASME Code Class 1 piping fatigue evaluation

be performed and should include all cyclic loadings due to normal operation, seismic, SRV discharge, LOCA chugging, and condensation oscillation loads (as appropriate).

210.51 Porvide for the staff review, typical examples of the amplified building response spectra used in the design of piping systems and including the following loadings:

a) seismic OBE
b) seismic SSE
c) SRV loads
d) LOCA-related loads

210.52 Briefly, describe the attenuation of the hydrodynamic loads in the plant and describe to what extent safety-related components are designed to these loadings in the various areas in the plant.

210.53 Describe the design considerations given to the piping in the suppression pool wetwell with respect to stability of the piping and its supports during a LOCA pool swell event.

210.54 Using the guidance of NUREG-0609, provide the methodology used and the results of the annulus pressurization (AP) analysis (asymmetric LOCA loads) for the reactor system and affected components including the following:

1. reactor pressure vessel and supports,
2. core supports and other reactor internals,
3. control rod drives,
4. ECCS piping attached to the reactor coolant system,
5. primary coolant piping, and
6. piping supports for affected piping systems.

The results of the above analysis should specifically address the effects of the combined loadings due to annulus pressurization and an SSE.

210.55 The staff review finds insufficient information regarding the design of component supports. Per SRP Section 3.9.3, our review includes an assessment of design and structural integrity of the supports. The review addresses three types of supports: (1) plate and shell, (3) linear, and (3) component standard types. For each of the above three types of supports, provide the following information (as applicable) for our review:

(a) Describe (for typical support details) which part of the support is designed and constructed as component supports and which part is designed and constructed as building steel (NF vs AISC jurisdictional boundaries).

(b) Provide the complete basis used for the design and construction of both the component support and the building steel up to the building structure. Include the applicable codes and standards used in the design, procurement, installation, examination, and inspection.

(c) Provide the loads, load combinations and stress limits used for the component support up to the building structure.

(d) Provide the deformation limits used for the component support.

(e) Describe the buckling criteria used for the design of component support.

210.56 The staff's review of your component support design finds that additional information is required regarding the design basis used for bolts.

(a) Describe the allowable stress limits used for bolts in equipment anchorage, component supports, and flanged connections.

(b) Provide a discussion of the design methods used for expansion anchor bolts used in component supports.

(c) Identify where in the plant high strength bolts have been used.

210.57 It is the staff's position that for the design of component supports, stresses produced by seismic anchor point motion of piping and the thermal expansion of piping should be categorized as primary stresses. Confirm that Nine Mile Point 2 meets this criteria.

210.58 Provide a discussion of the use of stiff pipe clamps as addressed in IE Information Notice 83-80.

210.59 Provide assurance that any snubbers used as a vibration arrestor has properly considered the cyclic loadings which might cause fatigue failure.

210.60 Valve discs are considered part of the pressure boundary and as such should have allowable stress limits. Provide these limits for our review.

210.61 Provide the stress categories and limits for core support/structures and include the applicable codes used for evaluation of the faulted condition.

SECTION 3.9.6

210.62 There are several safety systems connected to the reactor coolant

pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements which will state the acceptable leak rate testing frequency shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute (GPM)

for each valve to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

210.63 Provide a schedule for completion of your program for inservice testing of pumps and valves including any request relief from ASME Section XI requirements.

DATED: March 5, 1984

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