Consumers Power Company

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October 31, 1979

Director, Nuclear Reactor Regulation
Att Mr Dennis L Ziemann, Chief
Operating Reactors Branch No 2
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT - PROPOSED TECHNICAL SPECIFICATIONS CHANGE - PRIMARY SYSTEM LEVEL SET POINTS: ADDITIONAL INFORMATION CONCERNING

Consumers Power Company letter dated October 23, 1979 submitted a proposed change to Big Rock Point's Technical Specifications. The proposed change pertains to primary system level set points. The proposed change was intended to account for differences between instrument level set points and the actual level at which protective system actions occur caused by varying conditions of primary system pressure and temperature.

In telephone conversations with the NRC staff, Consumers Power Company was requested to provide additional information concerning the relationship between primary system conditions, level instrument reference leg temperature, protective system set points, and the level at which protective system action occurs. The requested information was provided to the NRC by telecopy on October 26 and October 30, 1979. This information is also presented as Enclosure 1 to this letter.

Consumers Power Company was also requested in the above-referenced telephone conversations to propose additional Technical Specifications changes to assure that primary system condition relationships were addressed for all primary system level set points discussed in Technical Specifications. Consumers Power Company was requested to propose specifications for set points which could be directly implemented and which would account for primary system condition effects and instrument errors while ensuring that protective system actions would occur as considered in plant safety analyses. The additional changes necessary to accomplish this are presented as Enclosure 2 to this letter. Big Rock Point Plant Review Committee and the Safety and Audit Review Board have reviewed these changes and concluded that they are acceptable from a safety standpoint.

David A Bixel (Signed)

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Nuclear Licensing Administrator

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Enclosure 1

ADDITIONAL INFORMATION PERTAINING TO BIG ROCK POINT PRIMARY SYSTEM LEVEL SET POINTS

The tables below summarize analyses performed to determine the effect of varying primary system conditions on Big Rock Point safety system actions. Reducing level instrument reference by temperatures (as discussed in proposed Technical Specifications change submitted October 23, 1979) affects the actual system water level at which protective system actions occur. The tables below demonstrate how this effect can be accounted for without resulting in action (under any assumed conditions) at a level below current Technical Specifications limits.

REACTOR VESSEL LEVEL

System Pres (Psia)	System Temp (°F)	Ref Leg Temp (°F)	Actual Vessel Level(1)	Scram Set Point(2)	Limit(3)
1350	582	250	610'-10"	610'-10"	>610'-5"
1350	582	100	610'-12"	610'-10"	>610'-5"
14.7	212	212	610'-6"	610'-10"	>610'-5"
14.7	212	100	610'-7"	610'-10"	>610'-5"

(1) The actual elevation of the water at the time of reactor scram.

(2) The reactor will scram when the level transmitter receives a certain differential pressure (dp). The transmitter is assumed to be calibrated so that the scram occurs when a dp is received that is equivalent to conditions of 1350 psia, 582°F, and Tref=250°F, and actual vessel level at 610'-10". Any change in operating conditions (ie, water density in the instrument columns) will be such that a level in the reactor different from 610'-10" will produce the dp required for reactor scram. The instrument is calibrated such that the scram dp will be received by the level transmitter when the actual level is greater than elevation 610'-5" (the lower limit of the range currently specified in the Technical Specifications and considered in analyses).

(3) Current Technical Specifications limits.

The reactor vessel level signals for scram, containment isolation, and RDS actuation are received from Yarway level sensors which are mounted such that the center line of the chamber corresponds to elevation 610'-6" or 2'-9" above the top of the active fuel. The chambers have a range of +9 to -9 inches from this point.

The reference leg temperature was conservatively assumed to be at a maximum of 250°F. The maximum containment temperature following an accident is 235°F. Therefore, accident conditions are not expected to create a more severe effect on level instrumentation than was already assumed. Further, the time constant of the Yarway reference leg has been calculated to be 20 to 30 minutes (Ref NEDO-24708, Section 2.3.2.6.4) and therefore any heating effects would occur slowly as compared to the accident or transient time scale.

Maximum ambient temperatures in the steam drum/reactor coolant pump cavity have been approximately 200°F based on representative data. Per discussion with Yarway application engineers, the maximum reference leg temperature should not exceed 50°F above ambient (250°F). It is believed that the actual reference leg temperature will be well below 50°F above ambient temperature.

STEAM DRUM LEVEL FOR RDS ACTUATION

System Pres (Psia)	System Temp (°F)	Ref Leg Temp (°F)	Actual Drum Level(1)	Actuation Set Point(2)	Limit(3)
1350	582	250	-17"	-17"	≥ -25"
1350	582	100	-10"	-17"	≥ -25"
14.7	212	212	-20"	-17"	≥ -25"
14.7	212	100	-16"	-17"	≥ -25"

(1) The actual water level in the steam drum when RDS is actuated, specified in inches below drum center line.

(2) The RDS will be actuated when the level switch senses a dp corresponding to 17" below drum center line. Because the level sensor is not temperature compensated, the actual drum level will be equal to the actuation set point at only the calibrated set of conditions. At all other operating conditions, the RDS will be actuated at a different actual drum level. It will always be actuated within the Technical Specifications limit of greater than or equal to 25" below drum center line.

(3) The RDS Technical Specifications requires the steam drum actuation set point to be at or above 25" below the steam drum center line.

The level signals for RDS actuation are received from Yarway level sensors which are mounted such that the center line of the chamber corresponds to the center line of the steam drum, and the chambers have a range of +30" to -30" from this point.

For comparison purposes, the table describing effect on low steam drum reactor scram which was included in the October 23, 1979 proposed Technical Specifications change is reproduced below. The assumed instrument reference leg temperature for each condition has been added for completeness.

LOW DRUM LEVEL SCRAM VS SYSTEM PARAMETERS WITH TEMPERATURE COMPENSATION REMOVED FROM STEAM DRUM LEVEL INSTRUMENTS

Psys (Psia)	Tref (°F)	Tsys (°F)	Actual Scram Point (In)(1)	Mass at Scram (Lbm)(2)	% Mo(3)
1,350	250	582	-8"	38,500	100
1,000	250	545	-10"	38,500	100
500	250	467	-12"	40,300	105
14.7	212	212	-16"	43,400	113

(1) Scram point is the actual drum level below center line.

(2) Mass in the system above the core at the corresponding low drum level.

(3) Percent mass above the core as compared to 100% power at 1,350 psia with drum level at 8" below center line.

The accident analysis remains valid.

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Enclosure 2

PROPOSED ADDITIONAL TECHNICAL SPECIFICATIONS CHANGES

The changes incorporated in the attached Technical Specifications tables are those necessary to ensure primary system level set points and are such that instrument inaccuracies due to varying primary system conditions, calibration errors, and instrument drift are accounted for. The proposed set points for Low Reactor Water Level and Low Steam Drum Level scram set points, Low Reactor Water Level core spray initiation set point, and Low Steam Drum Level and Low Reactor Water Level RDS initiation set points ensure that the required actions will occur, for all primary system conditions, at or before the time assumed in accident analyses. The footnote added to each table clarifies calibration requirements associated with these instruments.

Set points previously listed in terms of elevation above sea level have also been changed in the attached tables to refer to elevation above the top of active fuel. This change is intended to enhance operator training.

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6.1.2 Reactor Safety System During Power Operation

The following tabulation gives the arrangement of the reactor safety system that shall be effective during power operation:

Sensor and Trip Device	Trip Contacts in Each Channel	Coincidence in Each Channel	Scram Setting and Tolerance	Special Features	Instrument Ranges	Warning Annunciation Trip Set Point
High Reactor Pressure (4 Pressure Switches)	2	1 Out of 2	50±5 Psi Above Reactor Operating Pressure		100-1700 Psig	25±5 Psi Above Reactor Operating Pressure
Low Reactor Water Level (4 Level Switches)	2	1 Out of 2	≥3'1" Above Top of Active Fuel (Tolerance Limit -1")[1]	Closes containment sphere isolation valves. If reactor pressure is less than 200 psig, actuates core spray system. (Note: Spray water will not enter reactor vessel until reactor pressure drops below fire header pressure.)	Fixed Level Trip Point No Range	
Low Steam Drum Water Level (4 Level Switches)	2	1 Out of 2	≥8.0" Below Steam Drum Center Line (Tolerance Limit -0.5")[1]		-30" to +30" Water	-4" Below Steam Drum Center Line
Main Steam Line Backup Isolation Valve Closure (4 Position Switches)	2	1 Out of 2	[illegible] ±5 Percent of Full Closure		Position Switch Trains Adjustable Full Valve Travel	
High Condenser Pressure (4 Pressure Switches)	2	1 Out of 2	8.0±0.5" of Hg Absolute Pressure	Bypassed by pressure interlock as described in Section 6.1.3.	0 - 30" Hg Vac	

(1) Level instrument set points shall be as specified. Level instrument calibration shall be based on normal operating temperature (582°F) and pressure (1350 psia) and instrument reference leg temperature of 250°F or lower as measured to maintain an actual trip level greater than that assumed in accident analyses (2'8" above top of active fuel for reactor vessel level trips, and 38,500 lbm inventory above active fuel for reactor protective system trip). Set point drift between calibration checks within the specified tolerance limit shall be corrected but need not be reported under Specification 6.9.2.

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TABLES 11.3.1.4a AND 11.4.1.4a

Instrumentation That Initiates Core Spray

	11.3.1.4a Limiting Conditions for Operation			11.4.1.4a Surveillance Requirements	
Parameter	Trip System Logic	Limiting Set Point	Conditions for Operability	Instrument Trip Test	Instrument Calibration
Open Core Spray Valves					
Low Reactor Water Level[a]	One of Two for Each of Two Valves in Series	$\geq$3'1" Above Top of Active Fuel (Tolerance Limit -1")[c]	Power Operation and Refueling Operations[b]	Quarterly	Each Major Refueling
Primary Pressure Low[a]	One of Two for Each of Two Valves in Series	$\geq$200 Psig	Power Operation and Refueling Operations[b]	Quarterly	Each Major Refueling

Notes for Tables 11.3.1.4a and 11.4.1.4a

[a] Initiation of valve operation requires both low reactor water level coincident with low primary system pressure.

[b] The primary core spray system shall be available for use during refueling operations. The redundant core spray system shall be inoperable during refueling operations with the valves blocked or otherwise defeated (while the piping section from the valves to the reactor head is dismantled).

[c] Level instrument set points shall be as specified. Level instrument calibration shall be based on normal operating temperature (582°F) and pressure (1350 psia) and instrument reference leg temperature of 250°F or less as measured to maintain an actual trip level greater than that assumed in the accident analysis (2'8" above the top of active fuel for reactor vessel level trips). Set point drift between calibration checks within the specified tolerance limit shall be corrected but need not be reported under Specification 6.9.2.

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Tables 3.5.2.h and 4.5.2.h

Instrumentation That Initiates RDS Operation

3.5.2.H - Limiting Conditions for Operation				4.5.2.H - Surveillance Requirement		
Parameter	Minimum Operable Channels	Limiting Set Point	Conditions for Operability	Instrument Trip Test	Instrument Calibration	Protective Channel Trip
Low Steam Drum Level	3	Above or Equal to 17" Below Center Line (Tolerance Limit -5")(1)	At Power Levels Whenever the Reactor Is Critical With the Head On or When in Hot Shutdown	Monthly	Each Major Refueling	-
Fire Pump(s) Discharge Pressure	3	$\geq$100 Psig	Ditto	Monthly	Each Major Refueling	-
Low Reactor Water Level	3	$\geq$3'1" Above Top of Active Fuel (Tolerance Limit -1")(1)	"	Monthly	Each Major Refueling	-
120-Second Time Delay	3	$\geq$120 Seconds Following Low Steam Drum Level Signal	"	Monthly	Each Major Refueling	-
Input Channels A Through D	3	-	"	Monthly	-	-
Output Channels I Through IV	3	-	"	-	-	Monthly
*Fire Pump Start	1	-	"	Monthly	-	Monthly

*Reference Specifications 3.1.4 and 4.1.4 for Bases.

(1) Level instrument set points shall be as specified. Level instrument calibration shall be based on normal operating temperature (582°F) and pressure (1350 psia) and instrument reference leg temperature of 250°F or lower as measured to maintain an actual trip level greater than that assumed in accident analyses (2'8" above top of active fuel for reactor vessel level trips, and 25" below steam drum center line for RDS actuation). Set point drift between calibration checks within the specified tolerance limit shall be corrected but need not be reported under Specification 6.9.2.

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