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SUBJECT: Forwards response to request for addl info re request for Tech Specs change for testing of MSSVs.

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United States Nuclear Regulatory Commission
Attn: Document Control Desk
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING A REQUEST FOR A TECHNICAL SPECIFICATIONS CHANGE

Gentlemen:

NRC letter dated May 8, 1996, requested additional information regarding our Technical Specifications change request dated January 29, 1996, concerning testing of the Main Steam Safety Valves. The response to the request for addition information, enclosed, was requested to be provided within 30 days of receipt of the request. Since the NRC letter was received on May 17, 1996, the response is to be submitted by June 17, 1996.

Questions regarding this matter may be referred to me at (803) 857-1802.

Very truly yours,

R. M. Krich
Manager - Regulatory Affairs

JSK/klb
Enclosure
c: Mr. S. D. Ebneter, Regional Administrator, USNRC, Region II
Ms. B. L. Mozafari, USNRC Project Manager, HBRSEP
Mr. W. T. Orders, USNRC Senior Resident Inspector, HBRSEP

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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING A REQUEST FOR A TECHNICAL SPECIFICATIONS CHANGE

Request 1

You did not indicate in the [January 29, 1996], submittal that all postulated events were considered in the assessment. Confirm that all events, [Updated Final Safety Analysis Report (UFSAR)] Section 5.2.2 - Overpressure Protection, and [UFSAR] Chapter 15 - Accident Analyses, were considered.

Response 1

UFSAR Section 5.2.2 - Overpressure Protection and UFSAR Chapter 15 - Accident Analyses, were considered. Reactor Coolant System (RCS) overpressurization is considered by the Loss of Load accident (i.e., UFSAR Section 15.2.2) case biased to maximize RCS pressure.

All of the NUREG - 0800, "Standard Review Plan" (SRP), Revision 1 transients are considered as either transients that require analysis, or as transients that are bounded by other transients. The attached Table developed as part of the Operating Cycle 17 safety analysis for the reactor core reload shows the disposition for each of the applicable SRP events. For Cycle 17 and subsequent cycles bounded by the analysis conditions, the Siemens Power Corporation (SPC) PTSPWR/SLOTRAX plant response methodology for non-Loss of Coolant Accident (LOCA) transients was replaced by the SPC ANF-RELAP methodology. Also, the Steam Line Break, Small Break LOCA, and Large Break LOCA were reanalyzed due to an increase in the core power peaking limits. All of the events listed as "Analyze" in the attached table were analyzed using a +3% Main Steam Safety Valve (MSSV) setpoint shift.

The following SRP events were not reanalyzed for Cycle 17, but are bounded by previously performed analyses.

1. UFSAR Section 15.5.1, "Inadvertent Operation of the [Emergency Core Cooling System] ECCS During Power Operation."
2. UFSAR Section 15.6.3, "Radiological Consequences of Steam Generator Tube Rupture."
3. UFSAR Section 15.7.3, "Postulated Radioactive Releases Due to Liquid Tank Failure."

Review of the UFSAR Tables 15.0.9-1, "Plant Systems And Equipment Available For Transient And Accident Conditions," and 15.0.11-1, "Worst Single Failures," shows that the MSSVs do not perform a mitigating function for these three events.

Request 2

Specifically, identify which events were verified through evaluation and which were reanalyzed, and justify each method of verification.

Response 2

See Response 1.

Request 3

Outline the criteria used in each verification, whether by evaluation or reanalysis.

Response 3

For events that were explicitly analyzed that resulted in pressurization of the secondary system, we confirmed that the maximum pressure remained less than 110% of the design pressure. For events that were not analyzed and that were not bounded by an analyzed event, we confirmed that the MSSVs do not perform a mitigating function.

Request 4

Describe how you ensure that the +/-3% drift in the MSSV setpoint will not affect the MSSV's ability to prevent the maximum steam generator design pressure from being compromised.

Response 4

The event that produced the highest secondary pressure was the Loss of External Electrical Load Accident, UFSAR Section 15.2.2. Two cases were analyzed for this event, the RCS pressurization case and the Secondary Pressurization/Minimum Departure From Nucleate Boiling Ratio (DNBR) case.

The initial conditions, including the assumption that the MSSV setpoints had drifted by +3%, and control system operation for the Secondary Pressurization/Minimum DNBR case were selected to provide a conservative estimate of the minimum DNBR and peak secondary pressure. Examples of this conservative biasing are,

1. Rapid isolation of the turbine,
2. No reactor trip on turbine trip; and,
3. Delay of the reactor trip by activation of the pressurizer pressure control systems (i.e., pressurizer spray and pressurizer power operated relief valves).

The resulting peak secondary pressure with the MSSV setpoints at +3% was below the 110% design allowable pressure.

ATTACHMENT

Disposition Of Standard Review Plan Events Summary

For H. B. Robinson Steam Electric Plant, Unit No. 2 Cycle 17

Disposition Of Events Summary For
H. B. Robinson Steam Electric Plant, Unit No. 2, Cycle 17

UFSAR Event Designation	Event Name	Disposition	Bounding Event
15.1	**Increase In Heat Removal By Secondary System**		
15.1.1	Decrease in Feedwater Temperature	Bounded	15.1.3
15.1.2	Increase in Feedwater Flow	Bounded	15.1.3 15.4.1
15.1.3	Increase in Steam Flow	Analyze	
15.1.4	Inadvertent Opening of a Steam Safety Valve	Bounded	15.1.3/ 15.1.5
15.1.5	Steam Line Break	Analyze	
15.2	**Decrease in Heat Removal by the Secondary System**		
15.2.1	Steam Pressure Regulator Malfunction	N/A	BWR Event
15.2.2	Loss of External Load	Analyze	
15.2.3	Turbine Trip	Bounded	15.2.2
15.2.4	Loss of Condenser Vacuum	Bounded	15.2.2
15.2.5	Inadvertent Closure of Main Steam Isolation Valves	Bounded	15.2.2
15.2.6	Loss of Nonemergency AC Power	Bounded	15.2.7/ 15.3.1
15.2.7	Loss of Normal Feedwater	Analyze	
15.2.8	Feedline Break	Bounded	15.1.5
15.3	**Decrease In Reactor Coolant System Flow**		
15.3.1	Complete Loss of Forced Reactor Coolant Flow	Analyze	
15.3.2	Reactor Coolant Pump (RCP) Shaft Seizure (Locked Rotor)	Analyze	
15.3.3	RCP Shaft Break	Bounded	15.3.2

UFSAR Event Designation	Event Name	Disposition	Bounding Event
15.4	**Reactivity and Power Distribution Anomalies**		
15.4.1	Uncontrolled Rod Cluster Control Assembly (RCCA)Bank Withdrawal a Subcritical or Low Power Startup Condition	Analyze	
15.4.2	Uncontrolled RCAA Bank Withdrawal at Power	Analyze	
15.4.3	RCCA Misoperation		
	1) Single Rod Withdrawal	Analyze	
	2) Statically Misaligned	Analyze	
	3) Dropped Rod/Bank	Analyze	
15.4.4	Startup of an Inactive RCP at an Incorrect Temperature	Not Permitted	Technical Specifications
15.4.5	A Malfunction or Failure of the Flow Controller in a Boiling Water Reactor (BWR) Loop that Results in an Increased Reactor Coolant Flow Rate	N/A	BWR Event
15.4.6	Chemical and Volume Control System (CVCS) Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	Analyze	
15.4.7	Inadvertent Loading and Operation of Assembly in an Improper Position	Analyze	
15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	Analyze	
15.5	**Increase In Reactor Coolant Inventory**		
15.5.1	Inadvertent Operation of the Emergency Core Cooling System During Power Operation	Bounded by Previous Analysis	15.5.1
15.2	CVCS Malfunction that Increases Reactor Coolant System (RCS) Inventory	Bounded	15.4.2/ 15.4.6
15.6	**Decrease In Reactor Coolant Inventory**		
15.6.1	Inadvertent Opening of a Pressurizer Safety or Power Operated Relief Valve	Analyze	
15.6.2	Small Break Loss-of-Coolant Accident	Analyze	
15.6.3	Radiological Consequences of Steam Generator Tube Rupture	Bounded by Previous Analysis	15.6.3

UFSAR Event Designation	Event Name	Disposition	Bounding Event
15.6.4	Spectrum of BWR Steam Piping Failures Outside Containment	N/A	BWR Event
15.6.5	Large Break Loss of Coolant Accidents		
	1) Fuel Damage Limits	Analyze	
	2) Radiological Consequeces	Analyze	