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R. E. DENTON
GENERAL MANAGER
CALVERT CLIFFS

December 14, 1990

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 and 2; Docket Nos. 50-317 and 50-318;
License No. DPR 53 and DPR 69
Licensee Event Report 89-023, Revision 2

Gentlemen:

The attached report is being sent to you as required under 10 CFR 50.73 guidelines. LER 89-023, Revision 2 is submitted in response to the Nuclear Regulatory Commission (NRC) Safety Evaluation Report and Event Classification Assessment attached to NRC letter from D. G. McDonald to G. C. Creel, dated October 1, 1990. The primary purpose of this revision is to change the reporting classification from voluntary to required, per 10 CFR 50.73(a)(2)(v). Should you have any questions regarding this report, we will be pleased to discuss them with you.

Very truly yours,

RED/CDS/bjd
Attachment

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
R. A. Capra, NRC
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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Calvert Cliffs, Unit 1

DOCKET NUMBER (2): 0 5 0 0 0 3 1 7

PAGE (3): 1 OF 0 8

TITLE (4): Postulated Rupture in Non-Safety-Related Service Water Subsystem (SRW) Could Cause Failure of Both Safety-Related Subsystems

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 2	2 0	8 9	8 9	0 2 3	0 2	1 2	1 4	9 0	Calvert Cliffs, Unit 2	0 5 0 0 0 3 1 8
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	X 50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
Craig D. Sly, Compliance	3 0 1	2 6 0 - 4 8 5 8

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) — X NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

At 0900 hours on December 20, 1989 a design deficiency was identified at Calvert Cliffs that could result in unavailability of both safety-related (SR) Service Water (SRW) subsystems. A pipe rupture in the non-safety-related (NSR) SRW subsystem could result in rapid draining of both of the independent, SR SRW subsystems and ultimately the loss of the Emergency Diesel Generators (EDGs). The reported condition does not describe an actual event and was not contributed to by any actual component or system failures. At the time of discovery, Unit 1 was in cold shutdown and Unit 2 was defueled.

The root cause was a design deficiency. The SR subsystems were not adequately protected from a single failure of the NSR SRW subsystem from rendering them from performing their intended design function. The design basis requirements for the SRW system isolation valves were not well defined in our original Final Safety Analysis Report (FSAR). A seismic event with a simultaneous Loss of Offsite Power was not specifically addressed in our original FSAR.

Short term compensatory measures were established to justify plant startup and operation. The evaluation of the SRW and its potential vulnerabilities is within the scope of our Individual Plant Examination Project. An SRW design enhancement is planned to provide automatic closure of the isolation valves upon indication of a rupture in the NSR subsystem.

I. DESCRIPTION OF EVENT

On December 20, 1989 a design deficiency was identified at Calvert Cliffs that could have resulted in the loss of both independent, safety-related (SR) Service Water (SRW) subsystems. The design deficiency was recognized and documented in Non-Conformance Report (NCR) 8391. NCR 8391 postulates that a pipe rupture in the non-safety-related (NSR) portion of the SRW, without a Safety Injection Actuation Signal (SIAS) to shut the SR to NSR isolation valves, could cause the draining of both independent SR SRW subsystems. This would then lead to a loss of cooling water to the Emergency Diesel Generators (EDGs). Thus, a seismic event could rupture of the non-seismic NSR portion of the SRW and cause the loss of the non-seismic offsite power supplies and the loss of the EDGs.

At the time of the discovery, Unit 1 was in cold shutdown (MODE 6) with the Reactor Coolant System (RCS) partially drained, at atmospheric pressure, and 114 degrees Fahrenheit. Unit 2 was defueled, with the reactor vessel partially drained, and the RCS at atmospheric pressure and ambient temperature.

The SRW System is a closed-loop cooling system that removes heat from the main turbine-generator plant components, containment cooling units, spent fuel pool heat exchangers, and EDG heat exchangers, and transfers that heat to the Saltwater System. Two SR Auxiliary Building SRW subsystems and a single NSR Turbine Building SRW subsystem are needed for each Unit during normal plant operations.

Although the SRW piping configuration differs slightly between Unit 1 and Unit 2, each system is basically comprised of two independent, SR SRW subsystems serving Auxiliary Building heat loads and a single, NSR SRW subsystem serving Turbine Building heat loads. The subsystems operate in parallel to each other and are interconnected at certain points. For Unit 1 (Figure 1), the two SR Auxiliary Building subsystems are connected to the NSR Turbine Building subsystem by a common, NSR pipe located where the SRW System exits the Turbine Building and connects to the SRW suction header. For Unit 2 (Figure 2), the two SR Auxiliary Building subsystems are connected to the NSR Turbine Building subsystem by a common, NSR connection from the SRW discharge header where the SRW System enters the Turbine Building. As a result, the Turbine Building subsystem cross-connects the two SR Auxiliary Building subsystems.

The ability to isolate the Turbine Building subsystem from both Auxiliary Building subsystems is provided by dual, SR air operated isolation valves on the discharge header piping of each Auxiliary Building subsystem, and by check valves in the return piping from the Turbine Building subsystem to suction header of each Auxiliary Building subsystem. The isolation valves are located in the Auxiliary Building subsystem piping prior to connection with the Turbine Building subsystem piping. The SR check valves are located in the SR portions of the Turbine Building SRW subsystem return lines to the Auxiliary Building SRW subsystem suction header piping. The air operated isolation valves can be operated from the Control Room, and close automatically following receipt of a SIAS or loss of instrument air. Other than SIAS or loss of instrument air, there are no other automatic closure functions associated with the isolation valves.

Calculations have been performed assuming a worst-case, double-ended guillotine pipe break in the Unit 2 NSR Turbine Building SRW piping. The piping configuration for Unit 2 is less conservative than the Unit 1 configuration because the Unit 2 cross-connection occurs just downstream of the non-critical service water valves. The Unit 1 cross-connection is downstream of all Turbine Building loads. The calculations indicate that the SRW System could be drained before an operator could act to isolate the break under non-SIAS conditions. A double-ended guillotine type break is more conservative than is required under our licensing and design basis for a moderate energy line break (i.e., Seismic Category II ANSI B31.1). However, informal calculations indicate that even a moderately sized pipe break would result in a rapid loss of SRW inventory.

Updated Final Safety Analysis Report (UFSAR) Section 8.4.1.2 states that "the emergency diesel generators and their auxiliaries are designed to withstand Seismic Category 1 accelerations and are installed in Category 1 structures." The SR Auxiliary Building SRW subsystems supply cooling water to the EDGs and are considered to be auxiliary equipment to the EDGs.

The reported condition is a postulated scenario and does not describe an actual event and has not contributed to any actual component or system failures.

II. CAUSE OF EVENT

The root cause of this event is a design deficiency. The SR SRW subsystems were not adequately protected from a single failure of the NSR SRW subsystem rendering them incapable of performing their intended safety function. The design basis requirements for the SRW system as an auxiliary support system for the EDGs were not well defined in the original FSAR. The implied requirement for this automatic isolation mechanism was not considered when determining the required isolation signal sources for the isolation valves between the SR and NSR portions of the SRW system.

UFSAR Section 8.4.1.2 states that the EDGs and their auxiliaries are designed to withstand Seismic Category 1 accelerations and are installed in Category 1 structures. The SRW supplies cooling water to the EDGs and is considered "auxiliary equipment" for the EDGs. It was not realized that the scenario of concern was implicitly defined by our original FSAR.

III. ANALYSIS OF EVENT

The worst-case scenario involving the design deficiency is a seismically-induced catastrophic rupture of the NSR SRW subsystem concurrent with a LOOP. The rupture would render both EDGs inoperable due to the loss of all cooling water via the SRW system. Thus, the worst-case scenario would ultimately result in a station blackout.

The probability of a passive pipe failure as the initiating failure mechanism for the worst-case scenario is considered unlikely. Passive pipe failure risk as a whole is considered very low as detailed in NUREG/CR-4407, "Pipe Break Frequency Estimates for Nuclear Power Plants". A lack of any meaningful industry data concerning large ruptures of moderate energy piping systems has led us to conclude that a large passive rupture of our SRW system is a very unlikely initiator to the worst-case scenario described above.

Our Abnormal Operating Procedures (AOP) recognize a SRW subsystem pipe rupture as a possible event. AOP-7B, "Loss of Service Water", Sections IV and V, "Rupture of a Subsystem", provides the appropriate operator response to the different indications of a rupture in a SRW subsystem. AOP-7B instructs operators, among other things, to close the SR to NSR isolation valves. This action will prevent a rupture in the NSR subsystem from draining the SR subsystems.

The most likely initiating mechanism for a large rupture of NSR SRW subsystem piping is a severe seismic event. Considering the Seismic Class II design of the Turbine Building and the inherent ruggedness of steel piping [ref. ASME Code Case N-411], the occurrence of a severe seismic event would not necessarily entail such a large pipe rupture.

A pipe rupture in the NSR SRW subsystem, taken by itself, it not a significant contributor to risk because the only risk-related components cooled by the SR SRW are the containment air coolers and the EDGs. The SRW itself is not necessary for safe shutdown following a rupture unless a LOCA or LOOP also occurs.

Due to the low probability of a large passive SRW pipe failure, the small likelihood of a damaging earthquake, the procedural controls currently in place, and the unlikeliness of a concurrent LOOP or LOCA with the rupture, the described design deficiency did not have any significant affect on operational safety or endanger the health and safety of the public or plant personnel.

This event is considered reportable per 10 CFR 50.73(a)(2)(v), "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: (a) shutdown the reactor and maintain it in a safe shutdown condition; (b) remove residual heat; (c) control the release of radioactive material; or (d) mitigate the consequences of an accident."

IV. CORRECTIVE ACTIONS

In Supplement 1 to this LER we indicated that sufficient operator guidance was available to cope with a postulated loss of SR service water with a simultaneous LOOP. However, the following compensatory actions were established prior to the Unit 1 startup.

1. The frequency of leak-rate monitoring by Operations was increased for the Service Water System.

2. A change was made to the Alarm Manual to require immediate isolation of the NSR SRW header upon large rupture indications.

3. Operators were informed of the status of this issue.

In addition to the above, the procedural controls previously mentioned (AOP-7B) will ensure that operators promptly and properly respond to indications of SRW subsystem rupture. Periodic training on this procedure is covered in Licensed Operator requalification training.

As indicated in our August 24, 1990 letter to the NRC, our long-term plans are to enhance the SRW design by providing automatic isolation of the SR portion upon indication of a rupture in the NSR portion. Our schedule for completion of this enhancement is detailed in that letter.

In accordance with NRC Generic Letter 88-29, "Individual Plant Examination for Severe Accident Vulnerabilities", a systematic examination for identifying plant-specific vulnerabilities to severe accidents is currently in progress. The general methods, approach, and schedule for this program is described in our initial response to the Generic Letter.

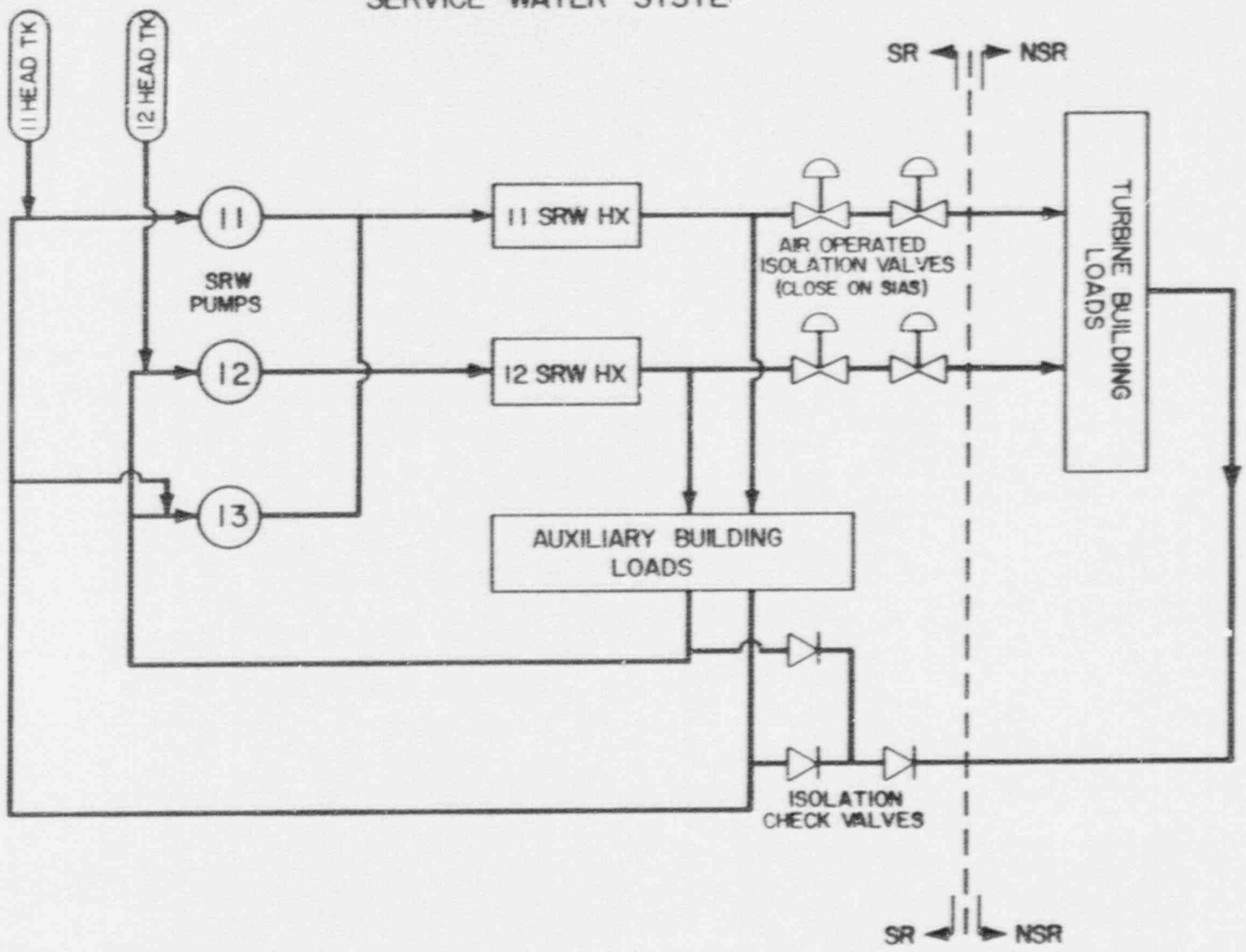
The evaluation of the SRW System and its potential vulnerabilities is within the scope of the IPE project. The IPE SRW System Analysis (draft) explicitly includes the failure effects of the common NSR Turbine Building SRW header. The determination of the common header's significance with regards to plant risk is pending completion of the IPE plant model. At that time, this potential vulnerability, as well as others from this and other plant systems, will be evaluated to determine whether potential improvements, both design and procedural, warrant implementation.

V. ADDITIONAL INFORMATION

A. Identification of Components Referred to in the LER.

Component/System	IEEE 803 Component ID Code	IEEE 805 System ID Code
Auxiliary Building		NF
Auxiliary Feedwater System		BA
Containment Coolers		BK
Control Room		NA
Emergency Diesel Generator		EK
Isolation Valve	ISV	
Reactor Coolant System		AB
Reactor Vessel	RCT	
Safety Injection System		JE/BQ/BP
Saltwater System		BS
Service Water System		BI
Spent Fuel Pool Cooling System		DA
Turbine Building		NM
Turbine Generator		TA

FIGURE 1
CALVERT CLIFFS UNIT 1
SERVICE WATER SYSTEM



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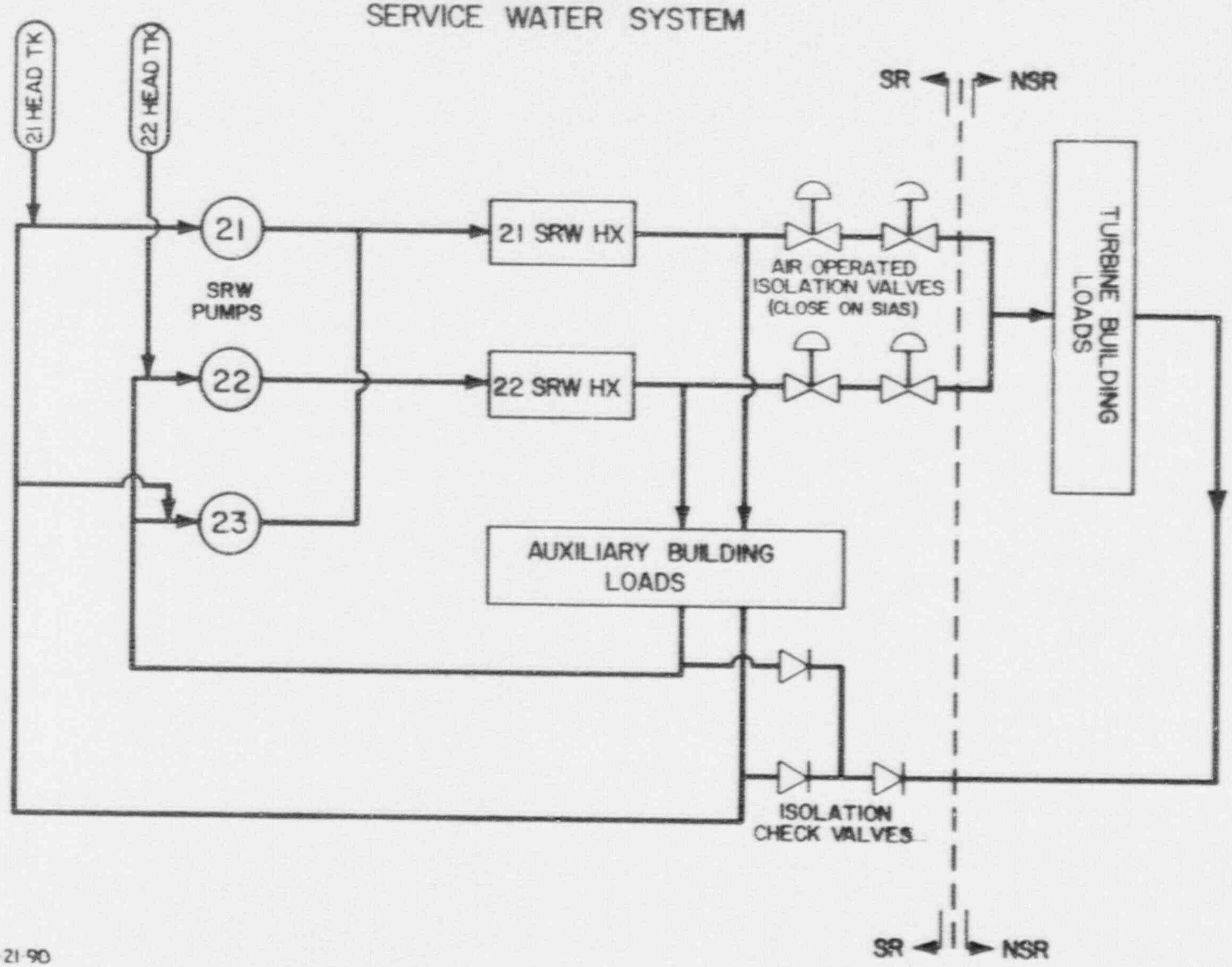
TEXT (If more space is required, use additional forms)

FIGURE 2

CALVERT CLIFFS UNIT 2
SERVICE WATER SYSTEM



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