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SOUTHERN COMPANY

December 16, 2013

Docket Nos.: 50-321
50-366

NL-13-2521

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
ISI Program Alternative HNP-ISI-ALT-18, Version 1

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.55a(a)(3)(ii), Southern Nuclear Operating Company (SNC) hereby requests NRC approval of an American Society of Mechanical Engineers (ASME) Section XI code alternative for the leakage examination of the Class 1 reactor vessel flange leak-off piping for both Units 1 and 2 of the Edwin I. Hatch Nuclear Plant. This System Leakage Test of Class 1 pressure retaining components is cited in Table IWB-2500-1, Examination Category B-P, Item No. B15.10, and is required to be performed during each refueling outage. Table IWB-2500-1, subparagraph IWB-5221(a) indicates that system leakage tests shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power. The requested alternative (HNP-ISI-ALT-18, Version 1.0) is necessitated by the impracticality of conformance to these specifications of the subject Code.

The proposed alternative examination for Class 1 reactor vessel flange leak-off piping pressure test at Plant Hatch is predicated on the ASME Code Case N-805 which has yet to be approved by the NRC. However, the NRC has previously approved the methods described in ASME Code Case N-805 in response to similar requests from operators of commercial reactors in lieu of testing at pressures corresponding to 100% rated reactor power.

Enclosure 1 of this letter provides a detailed discussion of the need and justification for the use of an alternate approach for the Class 1 leakage testing of the subject vessel flange leak-off piping in lieu of the methods described in IWB-5221(a) of the Code. An expedited approval of the use of the proposed alternative is requested by February14, 2014, to support plant restart following the Plant Hatch 1R26 Refueling Outage which is currently scheduled to begin February 3, 2014.

If you have any questions regarding this request, please contact Mr. G. K. McElroy at (205) 992-7369.

Respectfully submitted,

C. R. Pierce

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Regulatory Affairs Director

CRP/WEB

Enclosure: Request for Approval of Code Alternative

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President, & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. D. R. Vineyard, Vice President – Hatch
Mr. B. L. Ivey, Vice President – Regulatory Affairs
RType: CHA7000

U. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Regional Administrator
Mr. R. E. Martin, NRR Senior Project Manager – Hatch
Mr. E. D. Morris, Senior Resident Inspector – Hatch

ENCLOSURE

EDWIN I. HATCH NUCLEAR PLANT
ISI PROGRAM ALTERNATIVE HNP-ISI-ALT-18, VERSION 1

REQUEST FOR APPROVAL OF CODE ALTERNATIVE

ENCLOSURE

SOUTHERN NUCLEAR OPERATING COMPANY – HNP-1 and HNP-2
HNP-ISI-ALT-18, VERSION 1.0
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(ii)

Plant Site-Unit:

Edwin I. Hatch Nuclear Plant - Units 1 and 2

Interval-Interval Dates:

4th ISI Interval, January 1, 2006 through December 31, 2015

Requested Date for Approval and Basis:

Approval is requested by February 14, 2014, to permit performance of the proposed alternative pressure test during the 26th Refueling Outage of Hatch Unit 1. This proposed alternative would also apply to all the remaining refueling outages during the fourth ISI Interval.

ASME Code Components Affected:

Unit 1 and Unit 2 Reactor Pressure Vessel Flange Seal Leak-off Piping. The piping configuration between units is slightly different. A description of the piping follows.

Unit 1
NPS 1" and 3/8" Reactor Pressure Vessel Flange Seal Leak-off Piping
The 1" piping is A-312 TP 304 stainless steel, schedule 80. Design pressure is 600 psig at 850°F.
The 3/8" tubing is A-213 GR TP 304 or 316. Wall thickness is 0.065". Design pressure is 600 psig at 850°F.

Unit 2
NPS 1", 3/4"and 3/8" Reactor Pressure Vessel Flange Seal Leak-off Piping
The 1" piping is SA-106 GR B carbon steel, schedule160. Design pressure is 900 psig at 850°F.
The 3/4" piping is SA-106 GR B, schedule 160. Design pressure is 900 psig at 850°F.
The 3/8" tubing is SA-213 GR TP 304 or 316. Wall thickness is 0.065". Design pressure is 900 psig at 850°F.

Applicable Code Edition and Addenda:

ASME Section XI, 2001 Edition through the 2003 Addenda

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SOUTHERN NUCLEAR OPERATING COMPANY – HNP-1 and HNP-2
HNP-ISI-ALT-18, VERSION 1.0
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(ii)

Applicable Code Requirements:

System Leakage Test of Class 1 pressure retaining components per Table IWB-2500-1, Examination Category B-P, Item No. B15.10, to occur each refueling outage. As referenced in Table IWB-2500-1, subparagraph IWB-5221(a) indicates that system leakage tests shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power.

Background and Reason for Request:

Southern Nuclear recently determined, through the use of industry operating experience, that Hatch is susceptible to an issue identified in the industry regarding compliance with ASME Section XI for examination of the reactor vessel flange leak-off piping.

Hatch Units 1 and 2 both have reactor vessel flange leak-off piping that is used to detect leakage past the inner seal-ring. The reactor pressure vessel flanges are sealed with two concentric metal seal-rings designed to permit no detectable leakage through the inner or outer seal at any operating condition, including heating to operating pressure and temperature at a maximum rate of 100°F/h and cold hydrostatic pressure testing at the pressure specified in the ASME Code. To detect a lack of seal integrity, a 1 inch vent tap is located between the two seal-rings. A monitor line is attached to the tap to provide an indication of leakage from the inner seal-ring seal. This monitor connects to a pressure switch that will alarm to the main control room if pressure increases to 600 psig. The piping on both units is completely contained within the drywell. A typical configuration of the vessel flange leak-off piping in question is shown in Attachment 1.

IWB-5221(a) indicates that the system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power. IWB-5222(a) indicates that the pressure retaining boundary during the system leakage test shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. The visual examination shall, however, extend to and include the second closed valve at the boundary extremity.

Hatch Unit 1 and 2 configurations are such that the reactor vessel flange leak-off lines are not capable of being pressure tested at normal reactor operating pressure, 1060 psia, unless the inner o-ring seal fails or is intentionally failed. The only other viable option would be a design change to perform a Code compliant system leakage test. The option to intentionally fail the inner seal of the vessel flange to establish normal operating pressure and temperature on the leak-off piping is not considered a viable option due to the increased dose that would result from the need to replace the inner seal. Additionally, development and implementation of a modification that would facilitate installation of a means to isolate the piping (e.g., threaded plug) is not considered viable due to the potential introduction of foreign material into the vessel during implementation and during each subsequent installation of the isolation device. Both operations would subject individuals performing these tasks to considerable dose without a commensurate increase in public health and safety. Accordingly, these options are not considered to be viable.

Proposed Alternative and Basis for Use:

In accordance with the provisions of 10 CFR 50.55a(a)(3)(ii), Hatch proposes to examine the Class 1 portion of the leak detection system consisting of the accessible portions of the RPV head flange o-ring leak-off piping during each refueling outage. The leak-off piping shall be examined using the VT-2 visual examination method and will be performed by certified VT-2 examiners. The test shall be conducted at ambient conditions after the refueling cavity has been residing at its normal refueling water level of approximately 22 feet 4 inches above the reactor vessel flange for at least four (4) hours when the piping is subjected to the static head pressure that exists when the reactor cavity is filled and the closure head is removed. A static pressure of approximately 9.7 psig is expected to be experienced at the reactor pressure vessel head flange.

The Class 1 portion of piping originating from the reactor vessel flange is required to be examined. The Unit 1 Class 1 boundary stops on the downstream side of 1B21-F071 and 1B21-F063 and includes pressure switch 1B21-N002. The Unit 2 Class 1 boundary does not contain any boundary valves and includes pressure switch 1B21-N002. An excerpt from the Unit 1 and Unit 2 Nuclear Boiler System P&ID is contained in Attachments 2 and 3, respectively. The reactor vessel leak-off piping subject to the proposed alternate examination method is highlighted on the drawings. There are no sections of Class 2 piping associated with the reactor vessel flange leak-off line. Both the Unit 1 and Unit 2 piping are contained completely within the drywell.

For segments of the line that are inaccessible for direct VT-2 visual inspection, examination will include inspection of the surrounding areas below the line for evidence of leakage as permitted by IWA-5241(b) of the ASME Section XI Code, 2001 Edition through 2003 Addenda. A small portion of the piping, specifically the connection between the piping and the vessel is covered by insulation. A depiction of the piping covered by the insulation is shown in Attachment 4. The remaining portion of the piping is not insulated.

In lieu of the requirements of IWB-5222, a VT-2 visual examination of the accessible areas will be performed each refueling outage on the piping subjected to the static pressure from the head of water when the reactor cavity is filled for at least four hours. The reactor vessel flange seal leak-off piping is essentially a leakage collection/detection system and would only function as a Class 1 pressure boundary in the event of a failure of the o-rings, which isolates the flange seal leak-off piping from reactor coolant system operating pressure. Any significant leakage due to a failed o-ring would be expected to clearly exhibit water accumulation that would be discernible during the proposed alternate VT-2 visual examination that will be performed. The static head developed with the flange seal leak-off piping filled with water will allow detection of any gross indications in the leak-off piping.

If the inner o-ring should leak during the operating cycle, it will be identified through the alarm of a pressure switch in the main control room. Upon receiving an alarm, operator actions will involve monitoring: (1) drywell floor drain leakage per site procedures; (2) drywell dome area temperature on the remote shutdown panel; (3) drywell temperature per site procedures; (4) drywell pressure through multiple indicators; and (5) drywell fission products. If any monitoring actions indicate outer seal failure, operators are directed per the annunciator response procedure to Technical Specification 3.4.4, RCS Operational Leakage. Similarly, should the inner o-ring leak during the operating cycle and a thru wall leak of the reactor vessel flange leak-off piping exist, leakage will be detected in the same manner described above for leakage resulting from outer seal failure. In addition, there is no site-specific history of degradation associated with either unit's vessel flange leak-off piping.

Since there is reasonable assurance that the proposed alternate examination will detect gross indications of leakage should any exist from this piping and that the examinations will occur every refueling outage, SNC requests authorization to use the proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii) on the basis that compliance with the specified requirement would result in hardship or difficulty without a compensating increase in the level of quality and safety.

Duration of Proposed Alternative:

The alternative is requested for the current Fourth Inservice Inspection Interval, which began January 1, 2006 and is scheduled to end on December 31, 2015 for both Unit 1 and 2.

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HNP-ISI-ALT-18, VERSION 1.0
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(ii)

Precedents:

1. Arkansas Nuclear One, Unit 2, Fourth Inspection Interval Alternative, *Request for Relief from American Society of Mechanical Engineers (ASME) Code, Section XI – Request for Relief ANO2-ISI-015*, approved by the NRC in a letter dated June 27, 2013 (ADAMS Accession No. ML13161A241)

2. Callaway Plant, Unit 1, Third Inspection Interval Alternative, *Proposed Alternative to ASME Section XI Requirements for Leakage Testing of Reactor Pressure Vessel Head Flange Leak-off Lines (Relief Request I3R-14*, approved by the NRC in a letter dated August 13, 2013 (ADAMS Accession No. ML13221A091)

3. Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Third Inspection Interval Alternative, *Request for Relief from the American Society of Mechanical Engineers (ASME) Code, Section XI, Reactor Vessel Head Flange Seal Leak Detection Piping – Relief Request No. 49,* approved by the NRC in a letter dated April 4, 2013 (ADAMS Accession No. ML13085A254)

4. Diablo Canyon, Units 1 and 2, Third Inspection Interval Alternative, *Request for Approval of an Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Pressure Test Requirements for Class 1 Reactor Vessel Flange Leak-off Lines,* approved by the NRC in a letter dated September 12, 2013 (ADAMS Accession No. ML 13192A354)

5. Dresden, Units 2 and 3, Fifth Inspection Interval Alternative, *Request for Relief for Exemption from Pressure Testing Reactor Pressure Vessel Head Flange Seal Leak Detection System*, approved by the NRC in a letter dated September 30, 2013 (ADAMS Accession No. ML 13258A003)

6. Vermont Yankee, Fourth Inspection Interval Alternative, *Alternative to System Leakage Test for the Reactor Pressure Vessel Head Flange Leak-off Lines,* approved by the NRC in a letter dated March 1, 2013 (ADAMS Accession No. ML13055A009)

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SOUTHERN NUCLEAR OPERATING COMPANY – HNP-1 and HNP-2
HNP-ISI-ALT-18, VERSION 1.0
PROPOSED ALTERNATIVE IN ACCORDANCE WITH 10 CFR 50.55a(a)(3)(ii)

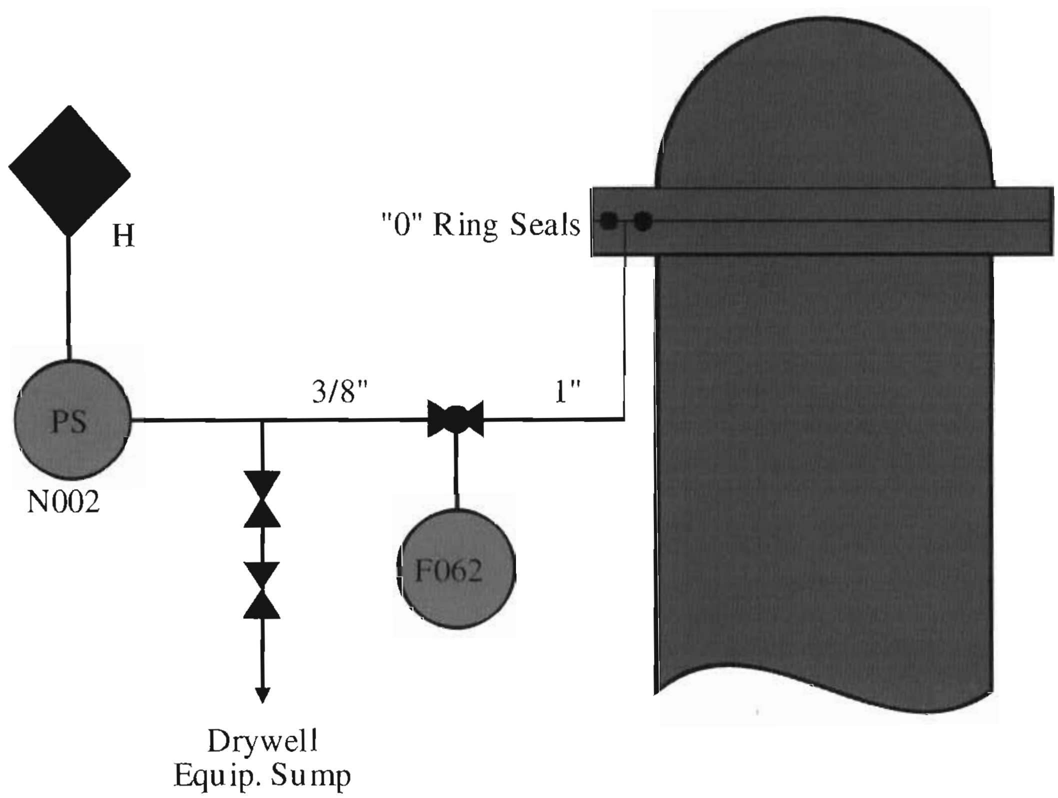
References:

1. ASME Code Case N-805, *Alternative to Class 1 Extended Boundary End of Interval or Class 2 System Leakage Testing of Reactor Vessel Flange O-ring Leak Detection System* was issued to the 2010 Edition of the ASME Section XI Code and is listed in Supplement 6 for Code Cases. However, Code Case N-805 has not been approved by the NRC and is not identified in Regulatory Guide 1.147, *Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1.*

Status:

Pending NRC approval.

ATTACHMENT 1
Simplified Schematic



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SOUTHERN NUCLEAR OPERATING COMPANY – HNP-1 and HNP-2
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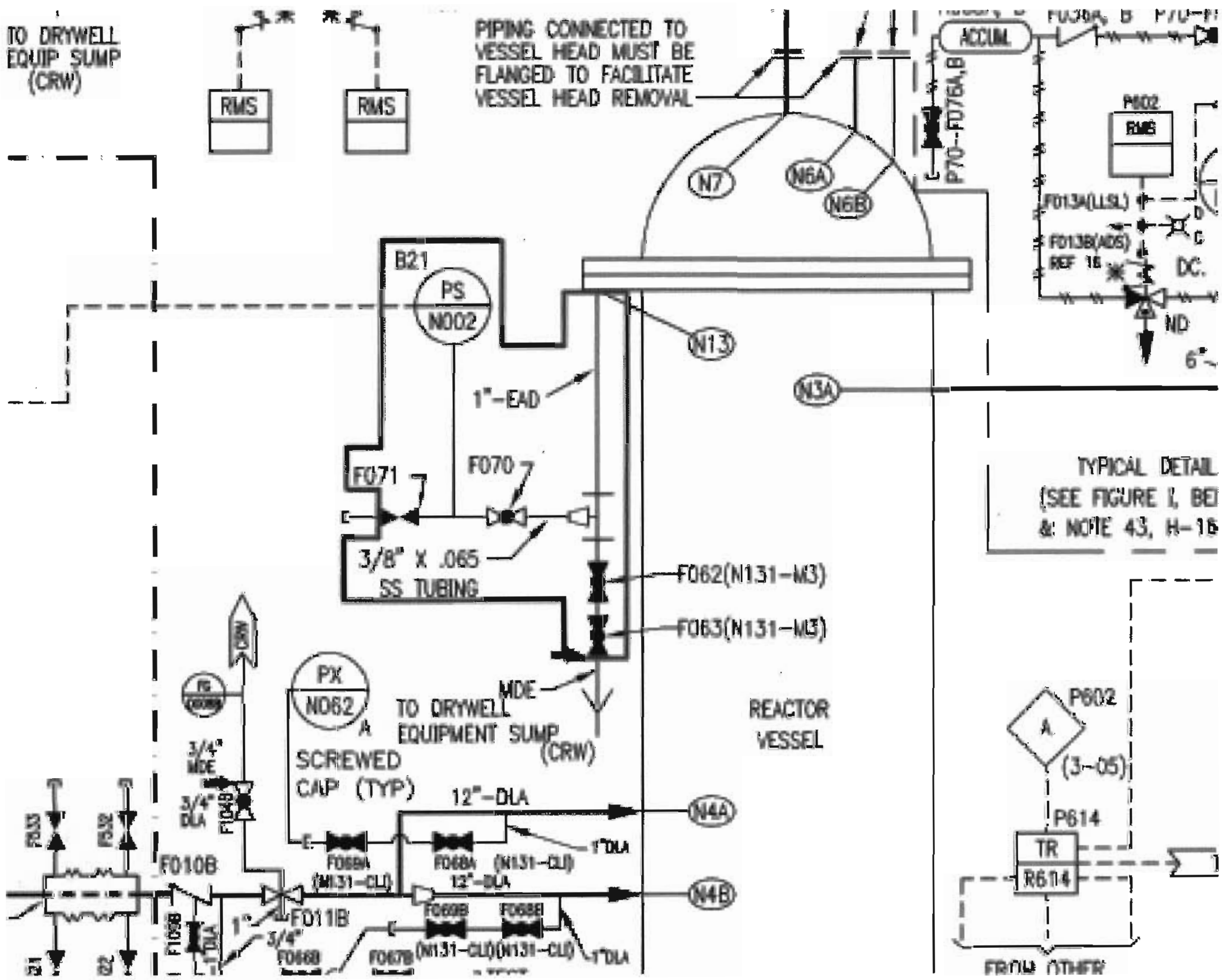
ATTACHMENT 2
Excerpt from Hatch Unit 1 P&ID H16062



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SOUTHERN NUCLEAR OPERATING COMPANY – HNP-1 and HNP-2
HNP-ISI-ALT-18, VERSION 1.0
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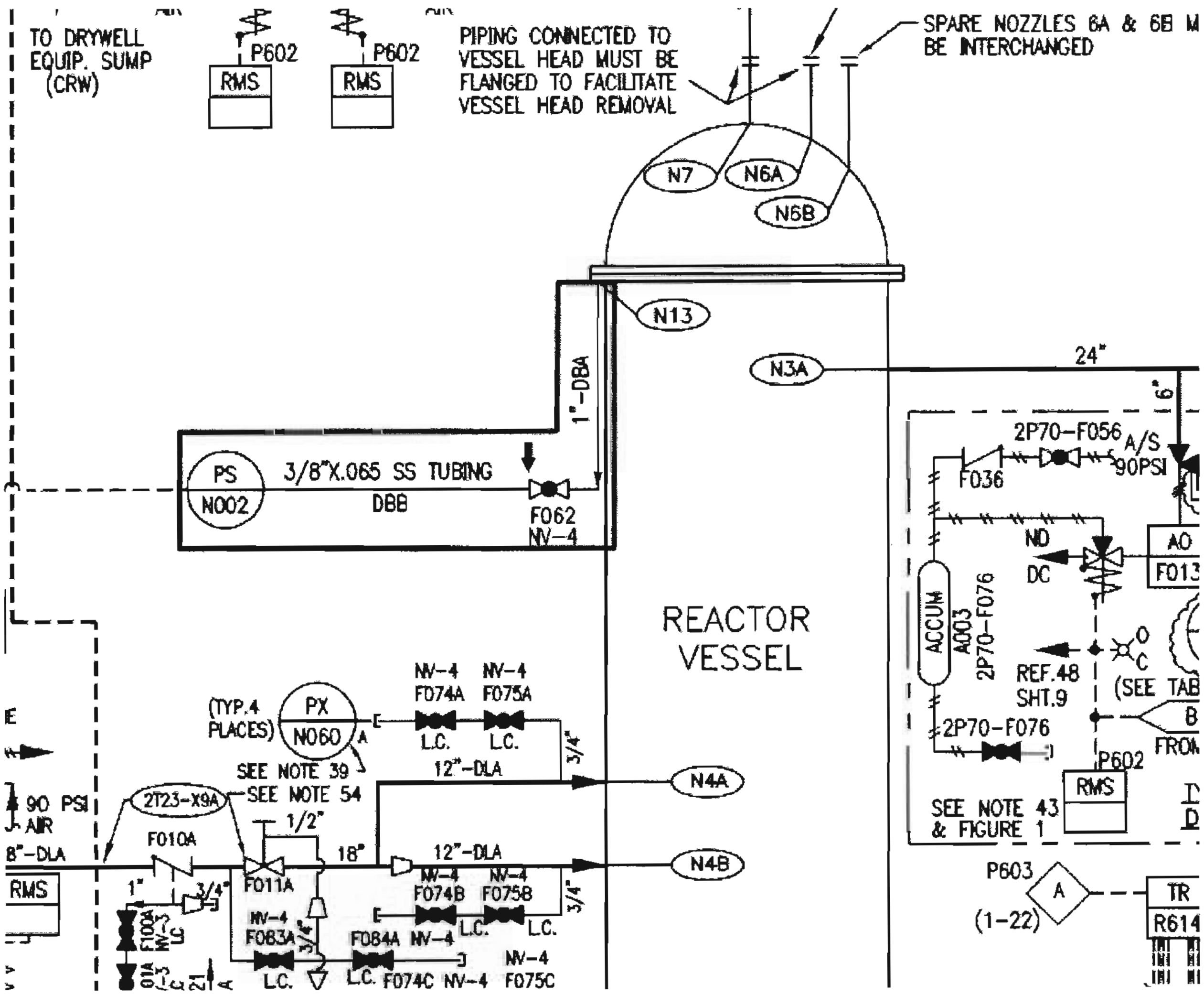
ATTACHMENT 3
Excerpt from Hatch Unit 2 P&ID H26000



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SOUTHERN NUCLEAR OPERATING COMPANY – HNP-1 and HNP-2
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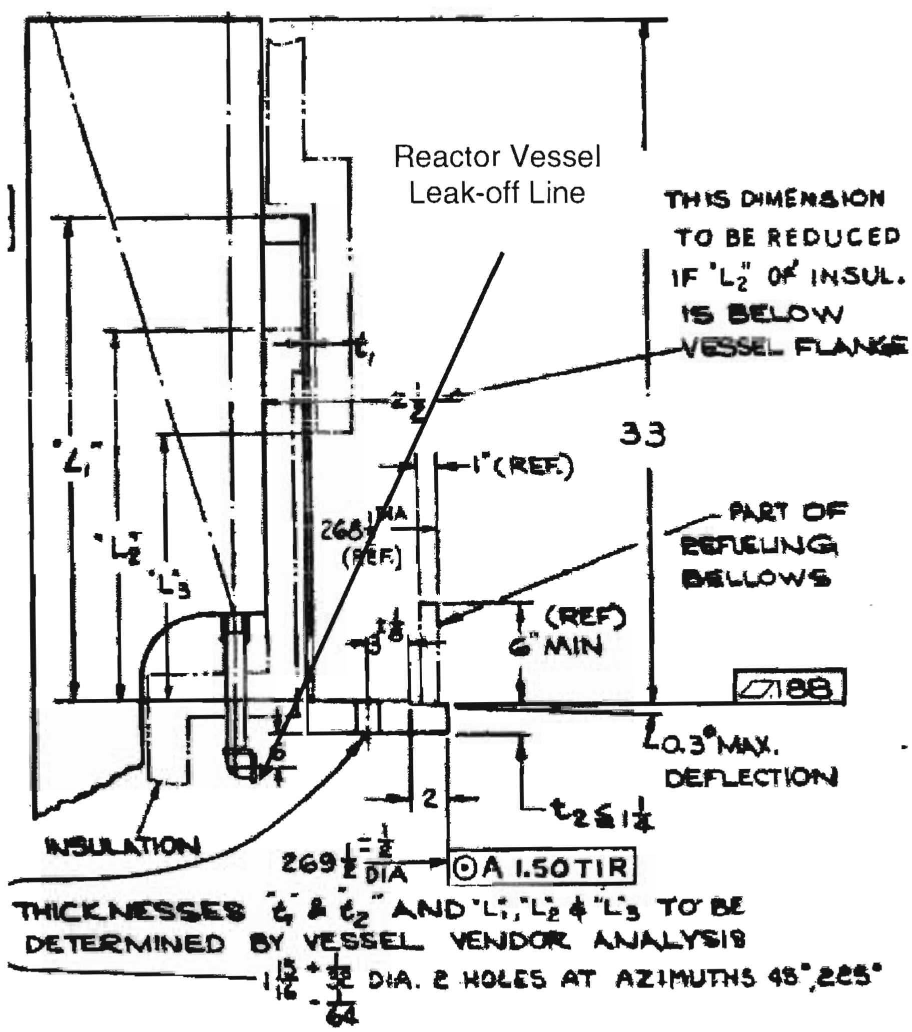
ATTACHMENT 4
Excerpt from Hatch Unit 1 Drawing S-15020
Depicting Piping Covered by Insulation

(Note: Unit 2 Typical)



VESSEL SHELL FLANGE