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RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director
EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards responses to NRC 810821 ltr requesting info re pressurized thermal shock.Detailed rept will be submitted by 820115.

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TITLE: Thermal Shock to Reactor Vessel

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	REAC SYS BR		1	1	REG FILE	05	1	1
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DUKE POWER COMPANY

CHARLOTTE, N. C. 28242

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

(704) 373-4249

October 20, 1981

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. D. G. Eisenhut, Director
Division of Licensing

Subject: Oconee Nuclear Station, Unit 1
Docket No. 50-269
Pressurized Thermal Shock

Dear Sir:

By letter dated August 21, 1981, the Staff provided its understanding of the pressurized thermal shock concern. Specific information was requested within 60 days and is included in the attachment to this letter.

As the Staff is aware, Duke Power Company is in the process of evaluating this concern for the Oconee 1 reactor vessel, using plant specific design details and realistic, yet conservative, analysis assumptions. A brief description of our planned efforts was provided during a meeting held with the Staff on September 22, 1981. It is our current intention to provide the Oconee 1 report by January 15, 1982, which is within 150 days of the August 21, 1981 Staff letter. This report will address the four possible actions discussed on page 4 of the Staff's letter as well as include the detailed design information requested in the enclosure to the Staff's letter. As the Staff is aware, all of the vessel materials information requested has been previously provided to the Staff in various B&W Topical Reports, submitted on the Oconee dockets, including BAW-1511P, "Irradiation-Induced Reduction in Charpy-Upper-Shelf Energy of Reactor Vessel Welds"; BAW-10046P, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G; BAW-10100A, "Reactor Vessel Material Surveillance Program - Compliance with 10CFR50, Appendix H, for Oconee Class Reactors"; as well as various surveillance capsule reports as referenced in Oconee Technical Specification 3.1.2, Pressurization, Heatup, and Cooldown Limitation. The Staff is also aware that Duke has provided much of the requested information to Oak Ridge National Laboratory in support of their effort, under contract W-7405-eng-26, to evaluate pressurized thermal shock on B&W NSSS design plants. However, the systems analysis information requested by the Staff is in the process of being prepared and will be included in our January submittal.

Duke Power Company is aggressively pursuing this issue and intends to submit in January a fully detailed and technically sound report that adequately addresses this concern.

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I declare under penalty of perjury that the statements set forth herein are true and correct to the best of my knowledge, executed on October 20, 1981.

Very truly yours,

A. C. Thies

A. C. Thies

RLG/php
Attachment

DUKE POWER COMPANY
OCONEE NUCLEAR STATION
UNIT 1

Response to NRC letter dated August 21, 1981
Request for Information

(1) Provide the RT_{NDT} values of the critical welds and plates (or forgings) in your vessel for:

(a) initial (as-built) conditions and location (e.g., 1/4 T) and
(b) current conditions (include fluence level) at the RPV inside carbon steel surface.

Response

Please see attached table.

(2) At what rate is RT_{NDT} increasing for these welds and plate material?

Response

Regulatory Guide 1.99, Revision 1, dated April 1977, is used to calculate the increase in RT_{NDT}. Capsule data has demonstrated this to be a conservative procedure. The Oconee 1 reactor vessel inner surface is accumulating fluence at the rate of 0.37E18 n/cm^2 per Effective Full Power Year (EFPY) at the peak fluence location and 0.33E18 n/cm^2/EFPY for the critical weld location.

(3) What value of RT_{NDT} for the critical welds and plate material do you consider appropriate as a limit for continued operation?

(4) What is the basis for your propsed limit?

Response

The establishment of limit values of RT_{NDT} for critical welds and plate material is not appropriate. The establishment of limit values for RT_{NDT} would indicate both a confidence of the validity of RT_{NDT} to conservatively predict the toughness of the materials as a function of service environment and an assurance that the material with the greatest index is the controlling material for any given analysis. This is not the case. Rather, it is suggested that the accident analysis evaluation as related to reactor vessel integrity be included within the currently established Appendix G to 10CFR50 by requiring periodic reevaluations over the life of the reactor vessel. This approach would recognize the changing technology used to evaluate reactor vessel integrity as well as accommodate the need to provide for changes in plant operations and safety related equipment and system upgrading.

Calculation of Reference Temperatures for Oconee Unit 1 Reactor Vessel Beltline Region Materials for 5.1 EFPY (As of October 1, 1981) at Inside Surface

Material Identification			Neutron Fluence 5.13 EFPY	Initial(3) RT_{NDT}	Adjusted RT_{NDT} (1)(2)(4) 5.13 EFPY
Weld or Heat Number	Type	Location	Inside Surface	^oF	^oF
AHR-54	SA508,C12	Nozzle belt	4.08E17	(+60)	84
SA1135	Weld	NB/US	4.08E17	(+20)	65
C2197-2	SA302-B	Upper Shell	1.94E18	(+40)	89
SA1073	Weld	US-Long.	1.53E18	(+20)	151
SA1229	Weld	UC(61% I.D)	1.94E18	(+20)	145
WF-25	Weld	UC(39% O.D)	N.A.	(+20)	-
SA1493	Weld	MS-Long.	1.86E18	(+20)	147
C3265-1	SA-302B	Mid.Shell	2.55E18	+20	71
C3278-1	SA-302B	Mid.Shell	2.55E18	(+40)	85
SA1585	Weld	Mid.Circ.	2.55E18	(+20)	126
C2800-1	SA302B	Low.Shell	2.55E18	(+40)	85
C2800-2	SA302B	Low.Shell	2.55E18	+20	65
SA1430	Weld	LS-Long	2.27E18	(+20)	160

(1) Weld Metal Chemistry per BAW-1511P, October 1980.
(2) Per BAW-1485, June 1978.
(3) Per BAW-10046P, March 1976.
(4) Per Regulatory Guide 1.99, Rev. 1, April 1977.

(5) Provide a listing of operator actions which are required for your plant to prevent pressurized thermal shock and to ensure vessel integrity. Include a description of the circumstances in which these operator actions are required to be taken. Included in this summary should be the specific pressure, temperature and level values for: a) high pressure injection (HPI) termination criteria presently used at your facility, b) HPI throttling criteria and instruction presently used at your facility, and c) criteria for throttling feedwater presently used at your facility. For each required operator action, give the information available to the operator and the time available for his decision and the required action. State how each required operator action is incorporated in plant operating procedures and in training and requalification training programs.

Response

The operator actions for preventing pressurized thermal shock and ensuring reactor vessel integrity are divided into two categories:

1) Actions for an overcooling event (steam line rupture, etc.)

2) Action during a loss of coolant accident.

Emergency procedures for overcooling and loss of reactor coolant (LOCA) both contain a caution statement which reads "Do not override Automatic Action of Engineered Safety features unless continued operation will result in unsafe plant conditions or will threaten reactor vessel integrity." The caution statement refers the operator to the pressure/temperature curve.

For the overcooling event, the Emergency Procedures require termination of uncontrolled cooldown and control of RCS Pressure and Pressurizer level. Uncontrolled cooldown is terminated by isolating the excessive steam flow and/or by terminating feedwater to the affected steam generator. With HPI actuation, the RCS pressure and pressurizer level are controlled by throttling HPI. Provided the RCS is $50^{o}F$ subcooled, HPI flow is throttled to maintain pressurizer level $\sim$220".

For the loss of coolant, the emergency procedures provide for HPI throttling and termination. Throttling of HPI flow is permitted, if the RCS is $50^{o}F$ subcooled, to keep the reactor coolant system (RCS) pressure/temperature in the appropriate region of the attached curve (Enclosure 1). If the High Pressure Injection system actuated because of low RCS pressure, it is required to remain in operation until one of the following criteria is satisfied:

1) The LPI System is in operation and flowing at a rate in excess of 1000 gpm in each line and the situation has been stable for 20 minutes; or

2) All hot and cold leg temperatures are at least $50^{o}F$ below the saturation temperature for the existing RCS pressure (see Enclosure 1), and the action is necessary to prevent the indicated pressurizer level from going off scale high.

As the Staff is aware, these actions were included in procedures as a result of IE Bulletin 79-05A. The Staff originally required that engineered safety features, including high pressure injection, not be overridden (IE Bulletin 79-05A, Item 4). In Duke's response to this (William O. Parker, Jr. letter dated April 10, 1979), it was stated that depending upon the nature of the transient, appropriate operator action may be required to alter the operation of the high pressure injection system to prevent an unsafe reactor coolant system condition. Subsequently, the Staff revised the required actions (IE Bulletin 79-05B, Item 2) to assure that reactor vessel integrity is not threatened.

The feedwater flow to the steam generators is normally controlled by an automatic level control system. The amount of feedwater supplied to the steam generator is a function of the difference between actual steam generator level and the level called for by the level control system. The operator can take manual control of the feedwater systems to limit the plant cooldown. One of the immediate actions required of the operator by the "Steam Supply System Rupture" emergency procedure is to isolate FW to the affected steam generator.

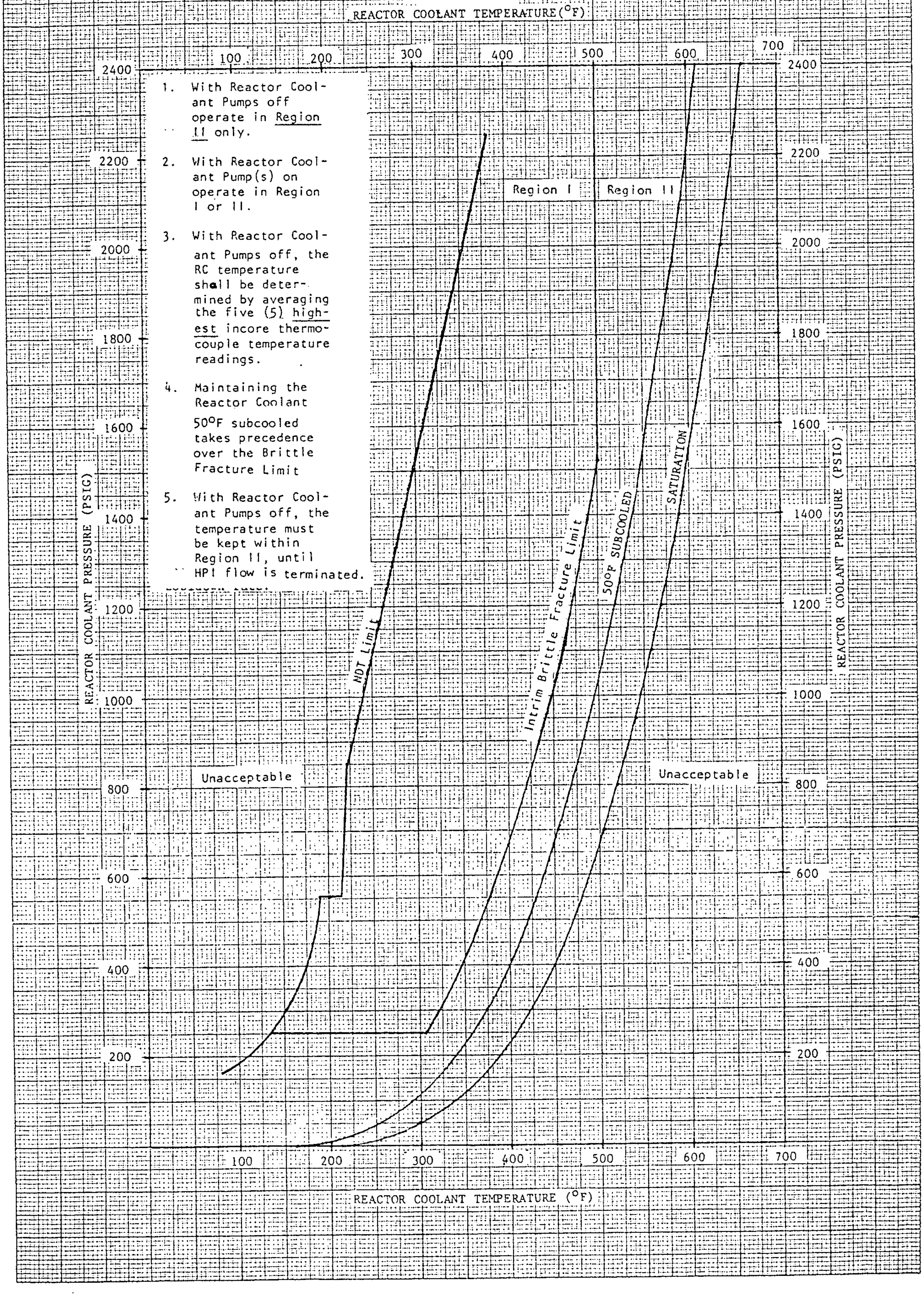
Furthermore, the operator has multiple indication of RCS pressure and pressurizer water level to assure that maximum RCS pressure is not exceeded by extended operation of the HPI system.

The operator receives information about plant status from instrument readouts and computer display in the control room. RCS pressure (wide range 0-2500 psi and narrow range 1700-2500 psi), RCS temperature (T_{Cold} 50^o-650^o, T_{Hot} 520-620^oF), pressurizer level (0-400"), steam generator level (operating range 0-100%) and feedwater flow (0-6 x 10^6 lb. m/hr.) are available on chart recorders in the control room. T_{Hot}, T_{Cold} and feedwater flow are also indicated on meters in the control room. The computer display provides the operator with all of the information supplied by chart recorders and meters. It also provides a digital readout of subcooling margin in each RCS loop and in the core.

The time available for the operator to respond is determined by the initiating event and the magnitude of the problem. The transient analyses currently in progress will be used to quantify the time available for the operator to take mitigative actions.

The information about HPI termination and throttling is incorporated into the emergency procedures for loss of coolant and steam line rupture. These procedures are reviewed by all operators during requalification training. Operators are also trained in plant response and use of the emergency procedures during training on the plant simulator in Lynchburg, Virginia.

ENCLOSURE I



REACTOR COOLANT TEMPERATURE (°F)
100
200
300
400
500
600
700
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
REACTOR COOLANT PRESSURE (PSIG)
1. With Reactor Coolant Pumps off operate in Region II only.
2. With Reactor Coolant Pump(s) on operate in Region I or II.
3. With Reactor Coolant Pumps off, the RC temperature shall be determined by averaging the five (5) highest incore thermocouple temperature readings.
4. Maintaining the Reactor Coolant 50°F subcooled takes precedence over the Brittle Fracture Limit
5. With Reactor Coolant Pumps off, the temperature must be kept within Region II, until HPI flow is terminated.
Region I
Region II
NDT Limit
Intrim Brittle Fracture Limit
50°F SUBCOOLED
SATURATION
Unacceptable
Unacceptable