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Safety Evaluation for Modification 1457, Rev. 5
Reactor Water Level Instrument Line Reroute
and Pressure Compensation
Peach Bottom APS Units 2 and 3
Doctype: 565

SUBJECT:

Modification 1457 removes the Yarway temperature compensated reference column, reroutes the associated reactor level measurement instrument lines through different drywell penetrations and adds reactor pressure compensation to the level measurement at Peach Bottom APS Units 2 and 3. This modification will improve the accuracy and reliability of the water level measurement under accident, transient, and normal operating conditions. It decreases the need for operator diagnosis and action in response to the weakness in the current installation.

CONCLUSION:

This modification affects safety related equipment. No unreviewed safety questions are involved. There are no significant hazard considerations. A change to the UFSAR is required and a change to the Technical Specification is recommended. The present level of fire protection is maintained and the ability to safely shut down the plant in the event of a fire is not reduced. This modification does not require a change to the plant operating license.

DISCUSSION: (General)

In response to a request made by the Nuclear Regulatory Commission (NRC), the BWR Owners' Group evaluated the measurement of reactor water level to assess the need for improvements in the instrumentation. This evaluation was documented in SLI-8211, "Review of BWR Reactor Water Level Measurement Systems." The evaluation found an extensive history of successful operation of reactor water instrumentation along with a few identifiable weaknesses. These weaknesses centered around the effects of high drywell temperature. The consequences of these effects are manageable but may require operator diagnostics and action may result in less than optimum remedial action. The BWR Owners' Group recommended that serious consideration be given to making improvements to overcome the effects of high drywell temperature on the reactor water level measurement. The NRC subsequently recommended that changes be made.

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Various potential modifications were studied for Peach Bottom. It was concluded that the Yarway temperature compensating reference columns will be removed and replaced with cold reference columns with piping that has a minimum elevation drop in the drywell. The removal of the Yarway columns will minimize the effects of high drywell temperatures on the reactor water level management. Electronic reactor pressure compensation will be installed to replace the Yarway temperature compensation and to further improve the measurement accuracy over the entire operating pressure range. The pressure compensated level measurement instruments will provide signals for indicating, recording and trip contacts for the input to the ECCS systems. The present ECCS Rosemount trip units are being replaced.

DISCUSSION: (Electrical)

Four independent compensation instrument systems provide four instrument channels A, B, C and D, for each of the two Peach Bottom units, Unit 2 and Unit 3. The A, B, C, and D channel compensation instruments each receive input signals from a wide-range level transmitter, a fuel-zone level transmitter and a reactor pressure transmitter. The following transmitters are used:

LT-2-3-72A, B, C and D
LT-2-3-73A and B
LT-2-3-73C and D*
PT-2-3-404C and D*
PT-2-3-404A and B

* Transmitters LT-2-3-111A and B and PT-2-3-52A and D are retagged to these designations.

Each pressure compensation instrument has two 4 to 20 ma level output signals. These signals provide two indication ranges of +60 inches to -165 inches (wide-range) and +60 inches to -325 inches (fuel-zone range) of reactor level. The analog output signals are provided through voltage to current converters that provide electrical isolation between the Class 1E compensation equipment and those indicators and recorders that are non-1E. These analog outputs will be connected to existing indicators and recorders as follows:

Unit	Channel	Range	Indicator/Recorder
2 & 3	A	+60 to -165	LI-2-3-85A & 85AX
2 & 3	B	+60 to -165	LI-2-3-85B & 85BX
2 & 3	C	+60 to -165	LR-2-3-110A
2 & 3	D	+60 to -165	LR-2-3-110B
2	A	+60 to -325	PR-2-3-404A
3	A	+60 to -325	LI-3-3-91**
2	B	+60 to -325	LI-2-3-91*
3	B	+60 to -325	PR-2-3-404B
2 & 3	C	+60 to -325	LR-2-3-110A
2 & 3	D	+60 to -325	LR-3-3-110B

* Unit 2 - Indicator LI-2-3-91B is retagged to LI-2-3-91 and indicator LI-2-3-91A is retagged to LI-2-3-113 by MOD 1029E.

** Unit 3 - Indicator LI-2-3-91A is retagged to LI-2-3-91 and indicator LI-2-3-91B is retagged to LI-2-3-113 by MOD 1029E.

In addition, four GE 180 meters will be installed for surveillance testing. These indicators tagged LI-2-3-73A, B, C and D and are installed such as to indicate the output of the fuel-zone range transmitter and I/E output for each channel.

Since the fuel-zone range level transmitter's variable leg uses the jet pump diffuser tap, this measurement is inaccurate whenever jet pump flow is present. Therefore, the fuel-zone range, +60 to -325 inches, outputs will be calculated from the wide-range transmitter signals when the reactor water level is above the bottom of the wide-range transmitter range. This will provide accurate level indication from the fuel-zone range indicators when there is jet pump flow. When the reactor water level is below the bottom of the wide-range transmitter range, this output will be calculated from the fuel-zone range transmitter signal and will provide accurate level indication to levels of -325 inches. At these levels, the recirc. pumps, the main source of flow through the jet pumps, are tripped.

Each compensation channel has sixteen Class 1E and four non-1E contact outputs for the pressure and the pressure compensated level trip points. These contact outputs provide system initiations and trips at levels and pressures that are within the Technical Specification limits and that are the same as the pressure compensated level indications. The following table enumerates the contacts and their functions:

No.	Class	Function	Process Action**	Technical Specification
1	1E	RCIC trip (A & B)	INCREASE	+45 In.
1	1E	HPCI trip (C & D)	INCREASE	+45 In.
2	1E	HPCI initiate	DECREASE	-48 In.
3	LE	RCIC initiate	DECREASE	-48 In.
4	1E	RHR/CORE SPRAY initiate	DECREASE	-160 In.
5	1E	ADS initiate	DECREASE	-160 In.
6	IE	RHR VALVE permit (A & B)	INCREASE	-226 In.
6	1E	Spare (C & D)		
7	1E	RHR/CORE SPRAY valve permit	DECREASE	+450 PSIG
8	1E	Close RECIRC. VALVE permit	DECREASE	+225 PSIG
9	1E	ADS Bypass time (A & C)	DECREASE	-160 In.
9	1E	Spare (B & D)		
10	1E	Spare		
11	1E	Spare		
12	LE	Spare		
13	LE	Spare		
14	1E	Spare		
15	1E	Spare		
16	1E	Spare		
17	Non-1E	ARI* and recirc. pump trip	INCREASE	+1107 PSIG
18	Non-1E	ARI* and recirc. pump trip	DECREASE	-48 In.
19	Non-1E	Alarm		
20	Non-1E	Spare		

* ARI - Alternate rod insertion.

** Contacts lose on listed process action.

The contact outputs of the compensation system will interface directly with the existing control circuits, in place of the removed auxiliary relay contacts. Relay coil suppression resistors will be added across the coil terminals of the existing HFA and HGA logic relays. Commercial-grade resistors have been analyzed for installation as Q components in this application. These resistors have a mean time between failure in excess of 300 years, and the probability of a resistor failing is small. The analysis determined that resistor failures will not prevent initiation of ECCS or an inadvertant trip of an operating ECCS system.

The compensation system is mounted in two new panels located in the cable spreading room. Each panel houses two channels: the A channel in the front on one panel with the C channel in the rear; and the B channel in front of the other panel with the D channel in the rear. Each channel has a power supply that has a 120 volt safeguard ac feed and a 125 volt dc feed. The outputs of the A anc C channel supplies are bused together, and the outputs of the B and D channel supplies are bused together to form redundant system. Each system channel consists of Foxboro Spec 200 equipment, with two dual-channel 4-20 ma. input cards, two Micro Spec 200 cards, one dual-channel 4-20 ma. output card, four class 1E four-relay contact output cards and one class 1E four-relay contact output card for isolation to non-1E circuits. This equipment is mounted in card racks in the panels and there are two racks in each system channel. A detachable hand station is provided for inputting configurations and checking and changing setpoints in the Micro Spec 200 cards. This station is not mounted in the panels and is not normally connected. A personal computer is required to configure the Micro Spec 200 cards through the hand station. Configuration of the equipment is not normally necessary and the personal computer and hand station are only required for maintenance and testing.

Reactor water level indication in the control room will remain on the same instruments in the same locations that are currently used. The tag numbers for the Unit 2 fuel-zone range indicators LI-2-3-91A and B are changed to LI-2-3-113 for the A indicator and LI-2-3-91 for the B indicator. Fuel-zone indication will be provided on recorder PR-2-3-404A for Unit 2. The tag numbers for the Unit 3 fuel-zone range indicators LI-2-3-91A and B are changed to LI-2-3-113 for the B indicator and LI-2-3-91 for the A indicator. Fuel-zone range indication will be provided on recorder PR-2-3-404B for Unit 3. This change is made in conjunction with MOD 1029E. The scale for both retagged fuel-zone range indicators is changed to +60 to -325 so that these indicators will indicate correctly on scale for all operating conditions. The indicator tagged LI-2-3-91 will normally receive compensated level signals based on the wide range transmitter signals and will automatically be switched to the fuel-zone range transmitter signals by the compensation system for levels below the range of the recirculation pump trip level.

The Rosemount units along with their auxiliary relays associated with the LT-2-3-72, LT-2-3-73 and PT-2-3-52 series of instruments will be removed from the C65 and C91 racks.

DISCUSSION: (Compensation)

Reactor pressure affects the density of the water and steam in the variable and reference water columns. The pressure effects on the water density cause errors in the reactor water level measurement. The level measuring differential pressure transmitters used with the present Yarway reference column arrangement were calibrated to provide accurate measurement at operating conditions of 1000 PSIG reactor pressure 135°F drywell temperature and 70° reactor building temperature. All deviations from these conditions introduced errors in the level measurement. The error that is introduced by variations in reactor pressure causes the measurement to indicate higher than the actual water level at reactor pressure less than the calibrated condition of 1000 PSIG. The Yarway column partially compensated for this error. The new pressure compensation instruments compensate the level measurement by approximating the steam table values for the density of water and steam as function of reactor pressure. These densities are then used with the differential pressure transmitter signals to calculate the actual reactor water level.

Errors caused by variations in drywell and reactor building temperatures are not compensated. Errors caused by drywell temperature variations are large with a Yarway reference column but will be essentially eliminated by the repiping that makes the variable leg and the reference leg have similar elevation drops inside the drywell. With this arrangement, the change in density of the water in the variable leg partially cancels the error due to change in density of the water in the reference leg. As a result, errors caused by drywell temperature changes are small. Errors due to reactor building temperature changes are also small. Calculation EE1457-1 indicates that the wide range errors are about 0.35 inches per 10 degrees temperature change. These error magnitudes are based on a comparison of calculated levels for reactor building temperatures of 70 degrees F. and 90 degrees F.

The fuel-zone range measurement is also affected by the flow through the jet pumps which increases the differential pressure measured by the fuel-zone transmitters. This effect causes the level measurement to indicate a higher than actual level when recirculation flow is greater than 45%. Hence, for levels above -165 inches the fuel-zone indication is based on the wide-range transmitter. Below -165 inches, the recirculation pumps will have tripped and jet pump flow will be low so the fuel-zone range indication and the level 0 (-226 inches) actuation are based on the fuel-zone range transmitter. This arrangement will minimize the effects of the jet pump flow and provide accurate fuel-zone level indication over the full range.

Five attached graphs show the real reactor level vs. reactor pressure at five indicated reactor water levels that correspond to the Technical Specification limits of +45 inches, -48 inches, -160 inches and -226 inches. One of the graphs for -160 inches of indicated level is data for a drywell temperature of 340° F. and is included for consideration of concerns in SIL-299, "High Drywell Temperature Effect On Reactor Vessel Water Level Instruments." The graph for -226 inches level is for a drywell temperature of 200° F. and provides data for consideration of the containment spray permissive. The other three graphs are for a drywell temperature of 135° F. expected during normal operations. The graphs are based on calculations assuming no subcooling, no flow related head losses and constant temperatures of the water in the drywell and reactor building portions of the measurement piping. These graphs show the relationships between indicated reactor levels and actual reactor levels and each curve is a constant indicated level. The calculation error is less that 1% of the level measurement range over a pressure range of 0 to 1000 PSIG. When the level compensation error is combined with the instrument accuracies, the resultant accuracy meets the requirements of all the instrument loops. The sources of these requirements are listed in Calculations EE1457-2.

The measurement accuracy and system response time for the compensation system have been calculated in Calculation EE1457-2 and are tabulated below. The required response times were established by modifications 273 and 439 which installed the Rosemount trip units, OPL4 data sheets for the plant transient analysis and NEDE 24222. The difference in the required and actual response times are evaluated in Calculation EE1457-2 and have been found acceptable since the measurement error due to the slower response time can be added to the instrument loop error and the combined error is within the specified accuracy. The contact numbers in this table correspond to the numbers in the above table where the contacts and their Technical Specification limits are listed.

No.	Function	Accuracy Req'd	Accuracy Actual	Response Time (Sec.) Req'd	Response Time (Sec.) Actual
1	RCIC trip (A&B)	3.5%	1.1%	0.5	0.61
1	HPCI trip (C&D)	3.5%	1.1%	0.5	0.61
2	HPCI initiate	3.5%	1.1%	0.5	0.61
3	RHR initiate	3.5%	1.1%	0.5	0.61
4	RHR/CORE SPRAY initiate	3.5%	1.1%	0.5	0.61
5	ADS initiate	3.5%	1.1%	0.5	0.61
6	RHR VALVE permit (A&B)	5.0%	1.1%	--	0.61
6	Spare (C&D)	--	--	--	0.61
7	RHR/CORE SPRAY VALVE permit	1.0%	0.4%	0.5	0.61
8	close RECIRC. VALVE permit	1.0%	0.4%	0.5	0.61
9	ADS Bypass Timer (A&C)	3.5%	1.1%	--	0.61
9	Spare (B & D)	--	--	--	0.61
10	Spare	--	--	--	0.61
11	Spare	--	--	--	0.61
12	Spare	--	--	--	0.61
13	Spare	--	--	--	0.61
14	Spare	--	--	--	0.61
15	Spare	--	--	--	0.61
16	Spare	--	--	--	0.61
17	ARI & RECIRC. Pump trip	1.0%	0.4%	0.45	0.61
18	ARI & RECIRC. Pump trip	3.5%	1.1%	0.5	0.61
19	Alarm	--	--	--	--
20	Spare	--	--	--	--

DISCUSSION: (Reference Leg Boil-Off)

Under severely abnormal conditions of high drywell temperature and low reactor pressure, the reference leg could boil-off. The elevation drop of the reference leg in the drywell will be 30 inches. Wide range indication will be at or below the bottom of the normal operating range when the actual level is just above the lower tap for the worst flashing conditions. This amount of error meets the acceptance criteria established for this event by the NRC in Generic Letter No. 84-23.

DISCUSSION: (ECCS ACTUATIONS)

As the level graphs show, level 8 (+45 inches) trips of the HPCI and RCIC turbines, and the main turbine stop valves and the feedwater turbines will occur at a higher and more conservative reactor water level with the compensation than with the Yarway measurement for reactor pressures less than 1000 PSIG. Since these trips occur on increasing level, the trip at a higher water level is conservative as considered from an ECCS aspect and will result in the HPCI and RCIC systems maintaining a greater volume of water in the reactor. This level will be close to +45 inches actual level for all reactor pressures.

Initiation of HPCI and RCIC and tripping of the recirculation pumps at level 2 (-48 inches) will occur at higher and more conservative reactor water levels with the pressure compensation than with the Yarway measurement for reactor pressure less than 1000 PSIG. This is conservative from an ECCS perspective in that these systems will maintain a greater volume of water in the reactor with the compensated measurement.

The HPCI system restarts on the resetting of the level 8 trip contact. The previous installation with the Rosemount Trip units had a minimum setting for this reset which corresponded to about a 1% reset band. The reset band with the pressure compensation system will be expanded to 5.2%. This wider band will cause the automatic cycling of the HPCI system to maintain reactor water level between +29 inches and the level 8 trip point. The pressure compensated measurement will be more accurate for pressures less than 1000 PSIG, than for the Yarway system measurement. The improved accuracy will result in an increased volume of water in the reactor vessel for a measured +29 inches of level. With the water level maintained above +29 inches, sufficient water volume is maintained within the reactor vessel for adequate core cooling. In addition, the number of trips and restarts of the HPCI system will be greatly reduced, and the reliability of the system will thus be increased.

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The changing of the reset deadband will not effect the operation of the RCIC system which restarts at level 2 (-48 inches). The level 8 RCIC trip contact will reset at +29 inches, well above the level 2 restart level.

The HPCI and RCIC level 8 trips also supply backup trip signals for the reactor feed pump turbines (RFPT) and the main turbine. These can be reset at the new +29 inch level. The resetting at this level will allow the starting of the RFPT's above the normal reactor water level, as an alternate water supply for the reactor vessel. The main turbine will also be available below the +29 inch level.

In accordance with SIL-299, level 1 actuation points had previously been increased to -130 inches to allow for reactor level measurement errors resulting from high drywell temperatures up to 340° F. Removal of the Yarway column and repiping of the reference column reduces the high drywell temperature error to 2.7 inches or 13% of the difference between the elevation drops of the reference column and the variable leg as predicted in SIL-299. The level 1 actuation point will be based on -160 inch limit and when the measurement accuracy is included in the set point, this will assure sufficient margin above the bottom instrument tap for the wide-range instruments and will assure actuation at elevated drywell temperatures. For measurements at the design basis operating temperature of 135° F., the -160 inch actuation point will assure operation above the Technical Specification limit. Graphs of -160 inches of indicated level vs. reactor pressure for 135° F. and 340° F. drywell temperatures are attached.

The Main Steam Line isolation function which is based on the level 1 (-160 inches) Technical Specification limit is not included in the functions provided by the pressure compensation instruments. With the removal of the Yarway reference column, this function as provided by LIS-2-3-99A, B, C and D will have no compensation. The limit for this actuation will also be set based on the -160 inch limit to be consistent with the other level 1 actuations. The graphs for -160 inches of indicated level for both 340° F and 135° F drywell temperatures show that the uncompensated level measurement as provided by the LIS-2-3-99A, B, C and D instruments is higher than the compensated level measurement for reactor pressures less than 1000 PSIG and is thus more conservative than the compensated level measurement. Therefore, MSIV isolation will occur slightly before the other level 1 actuations as reactor level decreases.

The containment spray permissive at level 0 (-226 inches) was previously calibrated to be accurate at 212°F drywell and reactor water temperatures and 0 PSIG reactor pressure. With the pressure compensation, the level measurement will be more accurate and will result in a spray permissive at the level 0 setpoint over the entire operating pressure range of the reactor. The graph for -226 inches of level shows a comparison between the compensated measurement and the unmodified measurement, which is shown for calibration at 212°F drywell and reactor temperature.

DISCUSSION: (Surveillance Testing)

The bases for the Technical Specifications establish a reliability goal of 0.993 for the instrument loops which initiate the emergency core cooling systems. The present Rosemount transmitter/trip unit combination is shown to meet this reliability goal through a set of overlapping surveillance tests. One test consists of comparing the indicated output of the four channels of measurement for a parameter. This assures the operation of the transmitter and trip unit input circuits. The Technical Specifications require this test to be performed once every 24 hours. The second test consists of substituting electrical signals for the transmitter input signals to the trip units. By varying these signals and observing the actuation of the contact outputs, the operation of the remainder of the circuit is verified. This second test is required once a month by the Technical Specifications. These test are in addition to the once a fuel cycle instrument calibrations.

The compensation equipment can be tested by a similar set of overlapping surveillance tests. Comparison of the eight indication outputs and the four fuel-zone transmitter loop indicators from the four channels will confirm the operation of the wide range, fuel-zone range and pressure transmitters, as well as, the operation of the compensation system input circuits and a majority of the Micro Spec 200 microprocessor card. This comparison should be performed at the existing 24 hour period.

The second overlapping test, consisting of substituting electrical signals for the transmitter signals into the input cards, will assure the operation of the remainder of the actuation circuit. In this test, the substituted signals would be varied while observing the actuation of the output contacts to assure the operation of the remainder of the Micro Spec 200 card and the contact of the relay output cards.

The two overlapping tests produce reliabilities for the wide range and fuel-zone actuations as shown in the following tables. The tables list the failure rate for each component, the test period for the component and the calculated reliability of each component. The individual component reliabilities are combined as a product to form the reliability of the actuation systems. The tables use 24 hour and conservative 2 month or 1461 hour test periods.

WIDE RANGE CONTACT OUTPUTS

Hardware Component	Failure Rate Failures/Hour	Test Period Hours	$R = e^{-FT}$
Transmitter+	3.0×10^{-6}	24	0.99993
Transmitter PT	3.0×10^{-6}	24	0.99993
2AI-1 (I/E)	0.8×10^{-6}	24	0.99998
2AI-2 (I/E)	0.8×10^{-6}	24	0.99998
CCA 1 Part 1*	10.4×10^{-6}	24	0.99975
CCA 1 Part 2*	3.2×10^{-6}	1461	0.99534
L2CR (relay)	0.2×10^{-6}	1461	0.99971
		Product	0.99462

* Micro Spec 200 card
\+ Wide range transmitter

FUEL-ZONE RANGE CONTACT OUTPUTS

Hardware Component	Failure Rate Failures/Hour	Test Period Hours	$R = e^{-FT}$
Transmitter+	3.0×10^{-6}	24	0.99993
Transmitter PT	3.0×10^{-6}	24	0.99993
2AI-1 (I/E)	0.8×10^{-6}	24	0.99998
2AI-2 (I/E)	0.8×10^{-6}	24	0.99998
CCA 1 Part 1*	10.4×10^{-6}	24	0.99975
CCA 1 Part 2*	3.2×10^{-6}	1461	0.99534
L2CR (relay)	0.2×10^{-6}	1461	0.99971
GE 180 Meter	1.0×10^{-6}	24	0.99998
		Product	0.99460

* Micro Spec 200 card
\+ Fuel-zone range transmitter

Since these reliabilities are greater than the 0.993 reliability goal established for ECCS actuation systems based on a conservative two month surveillance period, performing the surveillance tests every 24 hours and every one month as required by the Technical Specification will assure a reliable system. No change to the Technical Specifications testing requirements will be made for this modification.

A Design Analysis has been performed to establish these reliabilities for the compensation equipment.

DISCUSSION: (Mechanical)

The mechanical aspects of removing the Yarway temperature compensation column and rerouting the sensing lines are enumerated below:

1) The sensing line rerouting includes removing the two A & B Yarway temperature compensated reference columns, and the associated reference leg piping from the reference columns to drywell penetrations N-28 and N-29 in Unit 2, and penetrations N-28 and N-100B in Unit 3. The piping through the penetration will be capped on both the inside and outside of the penetration. The variable leg piping from the vessel nozzles to the existing Yarway reference columns will be reworked so that it slopes continuously down from the vessel nozzle to the remainder of the variable leg piping. Temporary supports (guide) shall be placed as close as possible to reactor nozzles N-16A and N-16B, to avoid any loading to the vessel nozzle during the piping rework. Upon completion of the rework, temporary supports will be removed and liquid penetrant examination will be performed on reactor nozzles N-16A and N-16B.

2) Installing two new condensing chambers which are not temperature compensated and routing the reference leg piping through drywell penetration N-26. The new condensing chambers will be fabricated from 3" dia. sch. 40 pipe, caps and 1" half coupling in accordance with ANSI B31.1 - 1980 edition. The butt welds in the condensing chamber shall be radiographed. The new condensing chambers will be located above their respective drywell penetrations to allow proper sloping of the reference column piping.

3) Separating the reference leg piping by a barrier to meet the criteria of ISA Standard 67.02-1980, "Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants." This means that two barriers will be installed in penetration N-26, and 18-inch penetration.

4) Moving the two fuel-zone water level reference legs from the current GEMAC cold reference legs to the new cold reference legs. These connections will be made in the reactor building. This change eliminates the need for repiping the existing cold reference legs.

5) Moving the restricting orifices from the bottom of the Yarway reference columns to just inside the drywell.

Containment isolation provisions of the original installation are being maintained. That is, the new lines will have a manually operated globe valve and an excess flow check valve installed outside of containment as close as practicable to the primary containment penetration. As noted above, the restricting orifices are being moved close to the drywell instead of remaining close to the vessel. This change in the orifice location has been reviewed against the recommendations of NRC regulatory guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment." This regulatory guide recommends that the orifices be located "as close as practical" to the reactor vessel. Analysis documented in SLI-8211 shows that there is a transient effect during flashing of water in the instrument line under certain postulated scenarios that can be avoided by locating the orifices close to the drywell penetration. It is believed that the best location is close to the drywell penetration.

Piping design for this modification is performed in accordance with applicable codes, and is shown on isometric and pipe support engineering drawings. Pipe stress analysis and pipe support design meet code requirements. Piping is reviewed for potential pipe break locations and the effects of postulated pipe breaks are considered. Inside containment, piping is routed with physical separation (on opposite sides of the containment or with barriers where required) to prevent a pipe break from affecting redundant equipment. Outside containment, a pipe break will be isolated by closure of containment isolation excess flow check valves. A pipe break inside the containment penetration is not considered credible since analysis shows this to be a low stress area and the area is shielded from missile impingement.

Pipe fabrication for this modification is performed in accordance with applicable codes with the following exception to the PBAPS-UFSAR: The geometry of the 3" diameter stainless steel caps purchased for the fabrication of the condensing chambers does not permit Volumetric examination as required by Appendix A (A.9) of the UFSAR for Peach Bottom. Liquid penetrant examination of these parts was performed in accordance with the requirements of the latest ASME code edition (Ref. Sect. NB2500 5A-652) for Nuclear Class 1.

During installation of the new piping or reworking of the existing portions, freeze plug procedures should be used as required and in accordance with the station requirements.

DISCUSSION: (General)

All design requirements applicable to the original installation were applied to this modification. These include, but are not limited to, conformance to applicable codes, separation requirements, seismic requirements, environmental qualification requirements, surveillance testing requirements, quality assurance, and preservice testing.

This modification makes no significant changes to bus loading. Each of the four channels has a power supply rated at 500 watts. Addition of this load is partially offset by a reduction in the load to existing power supplies for the Rosemont Trip Units.

The addition of the coil suppression resistors to the HFA and HGA relays will add a maximum of 1.04 AMPS to the 125 volt DC system during low level transients. As a conservative analysis, this load was considered to be added to the most heavily loaded and reduced the design margin by about 0.5% and is considered acceptable. The additional load during normal water level operation is 0.36 amps. Additional panel heat load and additional individual circuit fuse loading are negligible.

This modification does not change the ability to safely shut down the plant in the event of a fire as required by Appendix R 10CFR50. The Peach Bottom APS "Fire Protection Plan" provides for safely shutting down the plant in the event of a fire in the control room or the cable spreading room. Failure or inadvertent operation of equipment in the control room or cable spreading room due to damage caused by fire, fire suppression system actuation or fire fighting are addressed in the Fire Protection Plan. As outlined in the plan, the plant can be safely shut down from alternative control stations located outside the control room. All shutdown functions provided from these panels are independent of control room and spreading room equipment or are provided with isolation switches at the alternative control stations.

This modification represents a change to the plant as described in the UFSAR. UFSAR sections 7.1.6.1, 7.4, 7.8 and 8.4.5 require modification. A change to UFSAR Appendix A is in process (URCCF) No. CR-924) to accommodate the deviation to section A.9 noted above. | 5

10CFR50.59 Changes, Tests and Experiments

1. Technical Specification Tables 3.2.B, 3.2.F, 3.2.G, 4.2.A, 4.2.B and 4.2.F and related Bases 3.2 and 4.2 have been reviewed. No changes to the Technical Specifications are required. However, a change to the Table 3.2.F is recommended as follows:

Minimum No. of Operable Instrument Channels	Instrument	Type Indication and Range	Action
2	Reactor Water Level (wide-range)	recorder -165" to +60"	(10) (11)
2	Reactor Water Level (fuel-zone)	recorder -325" to +60"	(10) (11)

Since this modification results in instruments that have a larger range which is greater than the range currently listed on Table 3.2.F and envelopes the currently listed range, it is concluded that the change in the Technical Specification is not required prior to installation.

2. Based on the preceding discussions, it is concluded that:

   a) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis report is not increased. This modification improves the reactor water level measurement for both the wide-range and the fuel-zone ranges. The pressure compensation equipment replaces the

Rosemount trip units with Foxboro equipment that maintains or increases the level of reliability of the previous equipment and provides the same actuation functions as the previous equipment. The pressure compensation increases the accuracy of the control room water level indication for pressure less than the normal 1000 PSIG operating pressure, provides the operator with correct level indication and reduces the need for the operator to interpret the indications at lower reactor pressures. This reduces the probability of occurrence of an accident or malfunctions of equipment important to safety. The piping reroute reduces the effect of high drywell temperature on the water level measurement. The piping reroute meets the intention of NRC regulatory guide 1.11.

b) The possibility of an accident or malfunction of a different type than any previously evaluated in the safety analysis is not created. This modification only adds pressure compensation to the reactor water level measurement, changes the type of equipment used for wide and fuel-zone range reactor water level measurements and reroutes the level measurement instrument sensing lines. The new equipment provides the same actuation functions at the same levels as the previous equipment and increases or maintains the same level of reliability as the previous measurement equipment. The piping reroute does not create the possibility of an accident or malfunction of a different type than any previously evaluated since it merely reroutes existing instrument piping.

c) The margin of safety as described in the Technical Specification is not reduced. The existing actuations at levels 8, 2, 1 and 0 are all maintained. With the pressure compensation, the reactor water level generated actuations will occur near the designated setpoints for all operating reactor pressures instead of only at the calibration conditions as occurs with the present design. The reliability of the new instruments meets the reliability goals established in the bases for the Technical Specifications.

Therefore, an unreviewed safety question is not involved.

10CFR50.92 Significant Hazards Considerations

The NRC has provided guidance concerning the application of the standards for determining whether license amendments involve significant hazards considerations by providing certain examples (51FR7751).

An example of a modification involving no significant hazards considerations: (IX-2) "The repaired or replacement component or system does not result in a significant change in its safety function or a significant reduction in any safety limit (or limiting condition of operation) associated with the component or system." The proposed change to Table 3.2.F, increasing the reactor water level recorder range, fits this example of an action not involving a significant hazards consideration. The increased range of the recorder and recorder signal do not increase the magnitude of the measurement error beyond the required accuracy for the ECCS initiations.

Further, the proposed change does not involve a significant hazards consideration because operation of Peach Bottom APS with the recorder ranges changed would not:

1) Involve a significant increase in the probability or the consequences of an accident previously evaluated. The modification only recommends a change to the Technical Specification because of the increased range of the reactor water level recorders for the wide-range and fuel-zone range level measurements. The new ranges are wider than the previous ranges and include the upper and lower range limits previously specified in the Technical Specifications. The new range upper limit will match the upper limit of the narrow range measurement and will provide redundant measurements to +60 inches level. The process measurement errors for the expanded wide and fuel-zone ranges are within the required limits of the previous ranges. Since the accuracy with the expanded ranges meets the design accuracy and the same system functions are provided, a significant hazards consideration is not involved.

2) This modification does not create the possibility of a new or different kind of accident from any accident previously evaluated. The modification only recommends a change to the Technical Specification because of a change to the range of the reactor water level recorders for the wide-range and the fuel-zone range level measurements. The new level ranges are wider than the level ranges previously included in the Technical Specification and include the previous upper and lower range limits. The measurement accuracies are within the design limits.

3) This modification does not reduce the margin of safety. The recorder range changes included in this modification will provide the operator with wide-range and fuel-zone range measurement range upper range limits that correspond to the upper range limit of the narrow range measurement range. The measurement accuracy is within the limits required by the previous range.

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Responsible Engineer

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Lead Division Independent Reviewer

________________________ Date 12-3-87
Lead Division (Branch Head or Section Head)

________________________ Date 12-17-87
Non-Lead Division Responsible Engineer

________________________ Date 12-18-87
Non-Lead Division Independent Reviewer

________________________ Date 12/28/87
Non-Lead Division (Branch Head or Section Head)

________________________ Date 1/6/88
Nuclear and Environmental Section Head

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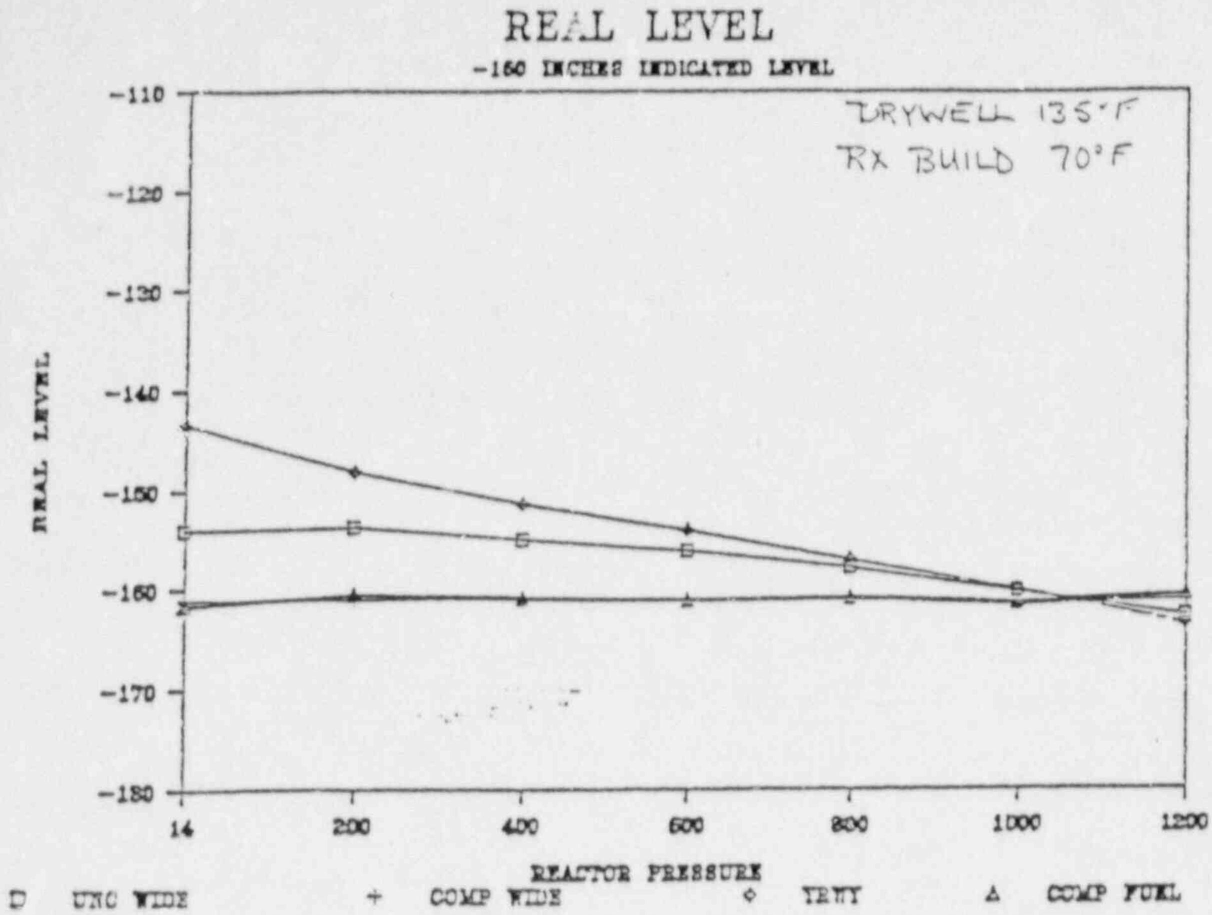
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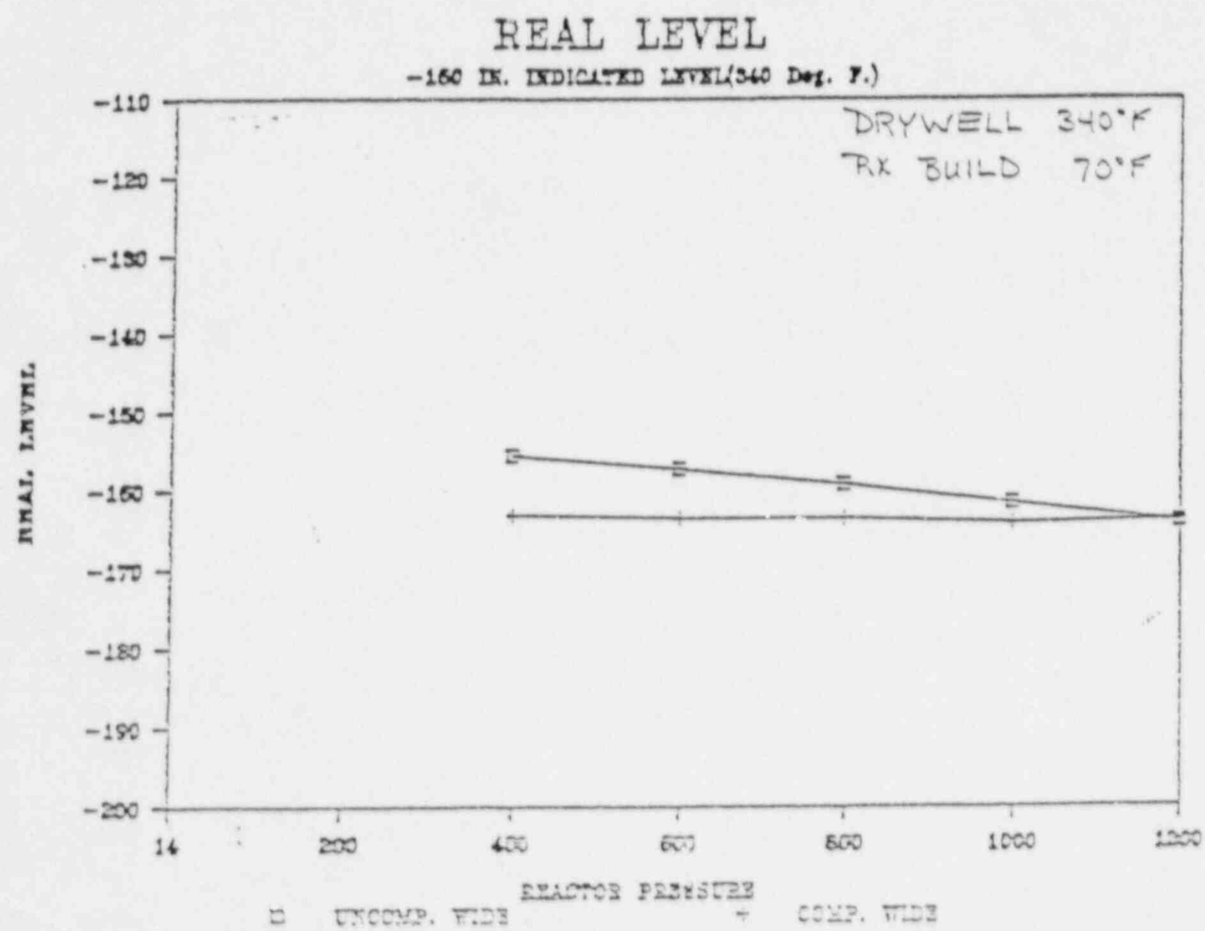
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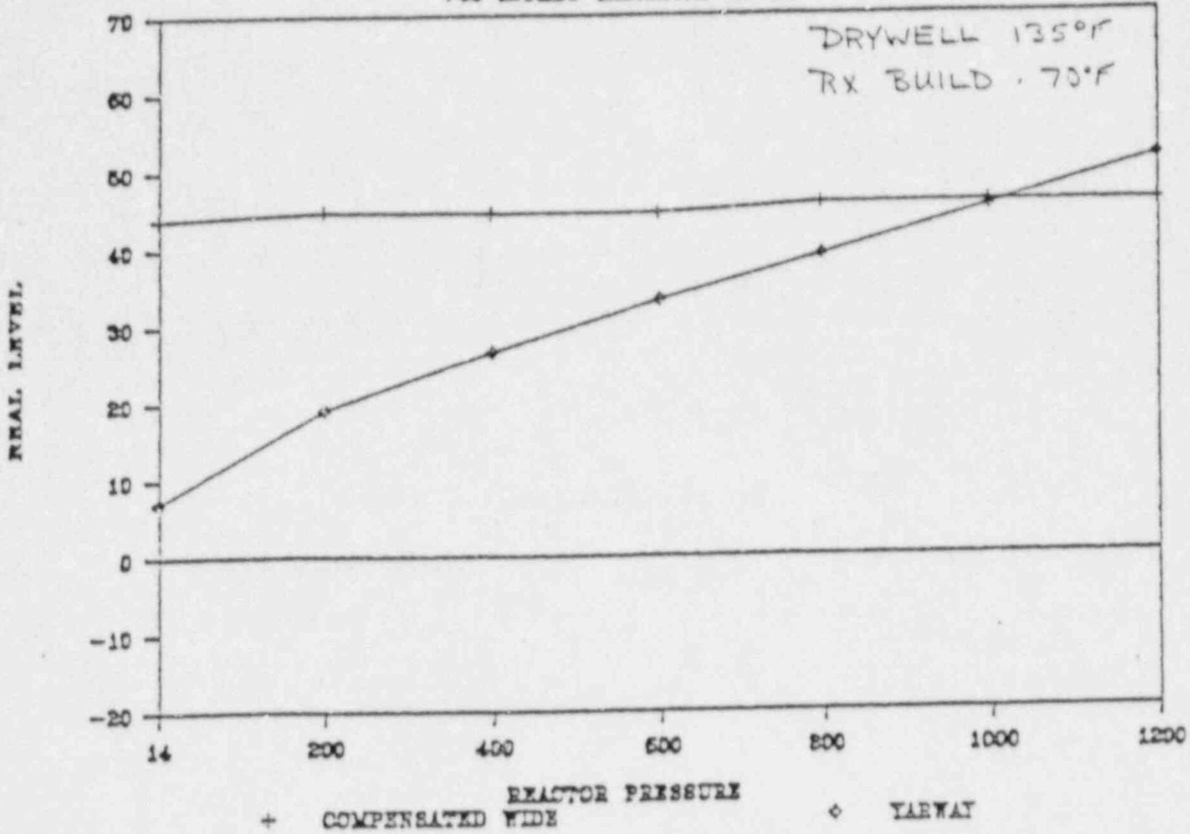

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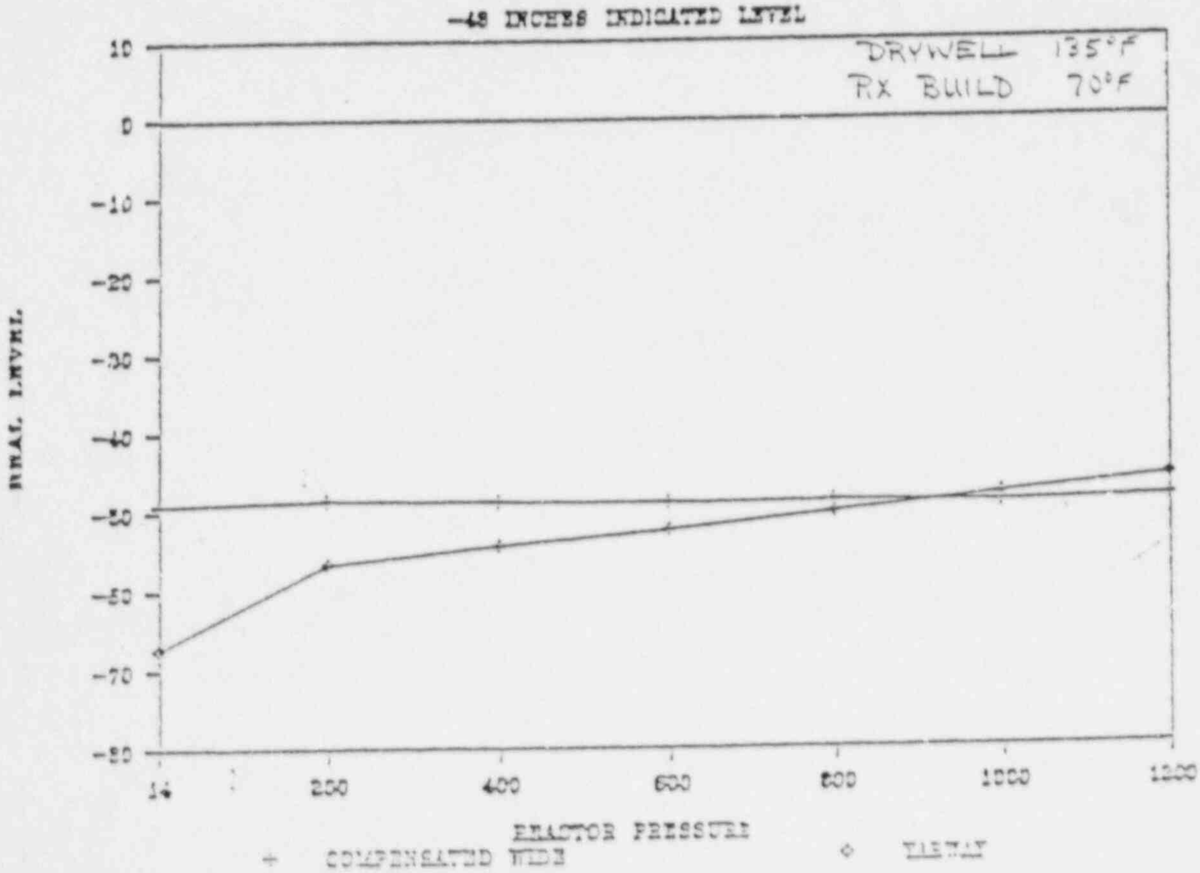

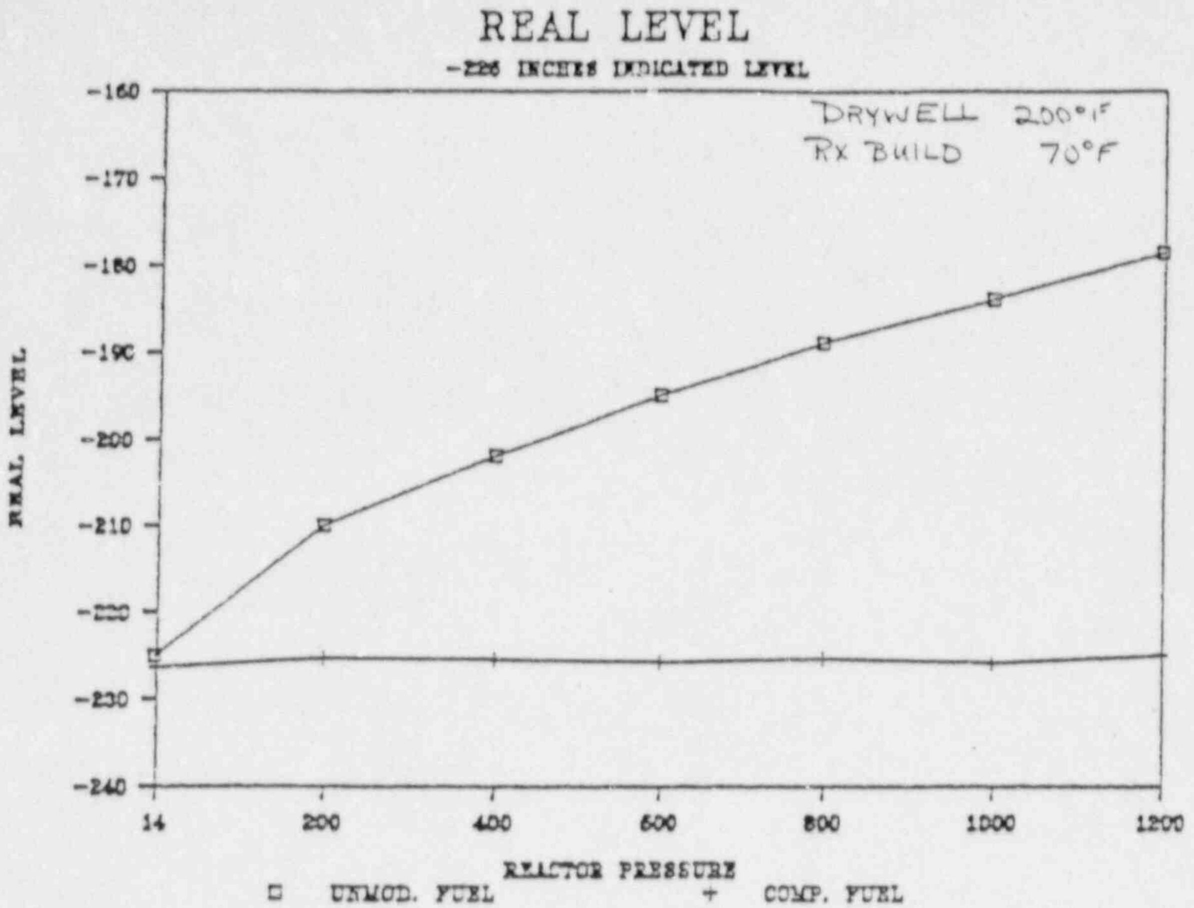
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