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50-287 Oconee Nuclear Station, Unit 3, Duke Power Co. 05000287
AUTH.NAME AUTHOR AFFILIATION
TUCKMAN,M.S. Duke Power Co.
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SUBJECT: Forwards proprietary & non-proprietry response to NRC 990317 RAI re TR DPC-NE-3005-P,which describes new Duke methodology for analyzing Oconee UFSAR Chapter 15 non-LOCA transients & accidents.Proprietary info withheld,per 10CFR2.790.

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Duke Energy

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April 19, 1999

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

ATTENTION: Document Control Desk

Subject: Oconee Nuclear Station
Docket Numbers 50-269, -270, -287
UFSAR Chapter 15 Transient Analysis Methodology,
Topical Report DPC-NE-3005-P, Revision 1

References: 1) Letter, M. S. Tuckman (Duke), to NRC, July 30, 1997

2) D. E. LaBarge (NRC) to W. R. McCollum (Duke), October 1, 1998

3) Letter, M. S. Tuckman (Duke), to NRC, February 1, 1999

By means of Reference 1, Duke Energy Corporation submitted Topical Report DPC-NE-3005-P which describes the new Duke methodology for analyzing the Oconee UFSAR Chapter 15 non-LOCA transients and accidents. NRC review and acceptance of this topical report were documented in a SER forwarded by Reference 2. Several issues requiring revisions to the topical report methodology were identified in Reference 2 and in a meeting with the NRC staff in Rockville on September 15, 1998. The requested revisions to the topical report were addressed by Duke in Revision 1 to DPC-NE-3005-P, which was submitted to the NRC by Reference 3. NRC letter dated March 17, 1999 contained seven questions on DPC-NE-3005-P, Revision 1. The Duke responses to these questions are contained in the attachments to this letter.

A001

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Some of the information contained in the attachments is considered proprietary. The proprietary information is that which is indicated by the bold brackets shown in Attachment 2. In accordance with 10CFR 2.790, Duke Energy Corporation requests that this information be considered proprietary. An affidavit supporting this request is included with this letter. A non-proprietary version is also included within as Attachment 1.

If there are any questions or if additional information is needed on this matter , please call J. S. Warren at (704) 382-4986.

M. S. Tuckman

M. S. Tuckman

Attachments (2)

<u>xc w/Attacment 1 only:</u>
L. A. Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

M. A. Scott
NRC Senior Resident Inspector (ONS)

<u>xc w/Attachment 1, Attachment 2, and Diskette:</u>
D. E. Labarge, NRC Senior Project Manager (ONS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

AFFIDAVIT

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and am authorized on the part of said Corporation (Duke) to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.

M. S. Tuckman (signature)

M. S. Tuckman

(Continued)

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is marked in Attachment 2 to Duke Energy Corporation letter dated April 19, 1999; SUBJECT: UFSAR Chapter 15 Transient Analysis Methodology, Topical Report DPC-NE-3005-P, Revision 1. This information enables Duke to:

(a) Respond to NRC requests for additional information regarding transient response of Babcock & Wilcox PWRs.

(bc) Simulate UFSAR Chapter 15 transients and accidents for Oconee Nuclear Station.

(cd) Perform safety evaluations per 10CFR50.59.

(de) Support Facility Operating Licenses/Technical Specifications amendments for Oconee Nuclear Station.

M. S. Tuckman

M. S. Tuckman

(Continued)

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman

M. S. Tuckman

(Continued)

M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M. S. Tuck

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 19TH day of April, 1999

Mary P. Nelms
Notary Public

My Commission Expires:
JAN 22, 2001

SEAL

bxc:

<u>w/Attachments 1 and 2:</u>
G. B. Swindlehurst
J. E. Burchfield
J. W. Sawyer
R. R. St. Clair
J. E. Smith
J. S. Warren
ELL

<u>w/o Attachments:</u>
C. J. Thomas

Attachment 1

Responses to Request For Additional Information
DPC-NE-3005P Revision 1

Q1. Pages 15-3 and 15-4 state that an inlet core mixing fraction of [ ] was used with the MSLB analyses and that the mixing fraction was obtained from tests performed on Oconee Unit 1. Discuss the mixing tests and how the mixing fraction was derived. Justify that this fraction is appropriate for MSLB analysis.

A1. [ ]

Q2. Page 15-4 states that the RETRAN transport delay model was disabled for the MSLB analyses since flow in the intact loop becomes stagnant so that use of the model is inappropriate. The staff understands that enthalpy transport was also disabled in the affected coolant loop that would not become stagnant. Consideration of enthalpy transport in the affected coolant loop might lead to an increased rate of core cooling. Provide the results of an evaluation showing the effect of considering enthalpy transport in the affected loop on core power and the departure from nucleate boiling ratio (DNBR).

A2. The RETRAN temperature transport delay model is used to more mechanistically treat temperature changes in piping volumes. The temperature transport delay option accounts for the fact that temperature changes move through piping as a front, while the finite difference homogeneous equilibrium model approach instantaneously mixes the incoming fluid with the volume contents. The temperature transport delay option establishes a mesh substructure with the volume to track temperature front movement. For the MSLB analysis, the temperature transport delay option is disabled in the primary loop due to flow reversals predicted in the intact loop. Experience with RETRAN has shown that the temperature transport delay model can give anomalous predictions during flow reversals. Turning off the temperature transport delay option in the affected loop will result in a more rapid propagation of the fluid cooled in the steam generator tube bundle to the core due to instantaneous mixing. This is conservative with respect to more rapidly decreasing the core moderator temperature, and thereby more likely to result in an earlier core return-to-power. Since boron injected by the ECCS systems terminates any return-to-power, and since ECCS inventory addition is subject to physical delays, having the return-to-power occur earlier in the event will lead to a more severe peak power statepoint. Sensitivity studies indicate that turning off the temperature transport delay option in the primary loop is a small penalty to the results of the MSLB analysis.

Q3. A postulated MSLB with off site power available would permit continued main feedwater flow and reactor coolant pump operation. Page 15-10 states that the reactor coolant pumps in the unaffected loop are assumed to trip at 100 seconds. This was stated to be conservative since reverse heat transfer was occurring in the steam generator, which was minimizing overcooling. Operators are trained to trip all reactor coolant pumps on loss of subcooling margin. Evaluate the MSLB for the case of continued main feedwater flow but with operator action to trip all reactor coolant pumps. Determine the effect on DNBR during any return-to-power.

A3. As described in DPC-NE-3005, Revision 1, in the MSLB return-to-power case the reactor coolant pumps (RCPs) in the unaffected loop are tripped at 100 seconds. To determine the effect of manually tripping all RCPs on a loss of the subcooled margin, the case which results in a return-to-power of approximately 13 %FP heat flux at 160 seconds, is reanalyzed. The additional case trips the RCPs in the broken SG loop at 160 seconds. The RETRAN boundary conditions for this additional case (core exit pressure, core inlet temperature, core inlet flow rate, and core average heat flux) are input to the VIPRE code to calculate the minimum DNBR. The minimum DNBR for this additional case is predicted to be 3.15 using the W-3S CHF correlation. It is noted that the Oconee emergency operating procedure does not allow the operators to trip the RCPs on loss of subcooled margin if the reactor power level is greater than 5%.

Q4. During a large main steam line break at Oconee the core flood tanks (CFTs) would be expected to discharge and add boric acid, which would act to reduce core power production. As the CFTs discharge, the cover gas will cool to a temperature lower than the liquid. RETRAN does not have a non-equilibrium CFT model. To assume equilibrium might lead to a higher CFT pressure and a greater discharge than would actually occur. Justify the CFT model in RETRAN is appropriate for analysis of main steam line breaks.

A4. The MSLB analysis uses the RETRAN separated volume model with air above subcooled liquid to model the CFTs. The initial conditions are 65 °F, 565 psia, 70.4% level and a conservatively low initial boron concentration of 1817 ppm. At the time of the peak power and minimum DNBR for the return-to-power case (approximately 160 seconds), the CFT conditions are 65 °F, 474 psia, and approximately 64% level. Note that the CFT level change is small due to the limited RCS depressurization below 565 psia. In reality the nitrogen cover gas will cool as the CFT depressurizes, and the RETRAN equations do not model this effect. However, since the initial temperature is assumed conservatively low at 65 °F, not much further temperature reduction is possible if this effect was accurately modeled. It should also be noted that the CFT injection lines are initially modeled with zero boron concentration.

Duke has benchmarked the RETRAN CFT model against Oconee CFT full blowdown tests with good agreement with data. The blowdown test is more challenging than the limited discharge transient that occurs during the MSLB. Based on the low initial CFT temperature, the limited CFT discharge that occurs during the MSLB analysis, and the results of the benchmarking analysis, the RETRAN CFT model is appropriate for MSLB analysis.

Q5. Page 15-12 states that the method of calculating boron reactivity feedback involves the use of the average core boron concentration and the average core boron worth. Boric acid from the CFTs and safety injection would reach the bottom of the core first, which would have a lower reactivity importance than the average core. Justify that using the average core boron concentration and average boron worth is appropriate.

A5. As described in DPC-NE-3005, Revision 1, in the MSLB return-to-power case the average core boron concentration and a conservative boron worth are assumed in modeling the reactivity due to ECCS boron injection. A differential boron worth of 120 ppm/%Δk/k is selected as an upper bound over the range of temperatures expected during the MSLB accident. The fact that the boron worth decreases with decreasing temperature is conservatively ignored. To determine the effect of assuming the core average boron concentration vs. the slightly lower boron concentration in the upper part of the core, the case which results in a return-to-power of approximately 13 %FP heat flux at 160 seconds, is reanalyzed. The additional case uses the boron concentration from the top two core volumes, rather than the core average, to determine the reactivity contribution from boron. The results show an increase in the core heat flux from 13.09% to 13.12%, which is essentially the same result.

Q6. Page 15-14 states that the limiting assumption with respect to maximizing overcooling and reactivity addition has been determined by analysis to be the case when the Integrated Control System (ICS) is controlling main feedwater. This case was found to cause more overcooling than the case that assumes failure of the ICS, which would permit unlimited main feedwater addition to the affected steam generator. Provide comparisons of the conditions within the affected steam generator secondary that would cause the ICS operating case to be the worst case. Compare steam generator level, pressure, temperature, and heat transfer coefficients that are calculated by the RETRAN code.

A6. MSLB analysis results show that the highest return-to-power occurs for those cases in which the ICS controls main feedwater (MFW) flow based on steam generator level, and that the 25 inch minimum post-trip level setpoint conservatively increased to 100 inches is the limiting level setpoint. The peak return-to-power for this case bounds the return-to-power predicted for the uncontrolled MFW case, in which it is assumed that the ICS fails to either control MFW flow or to trip MFW pumps on high SG level. Without ICS control the continued MFW flow results in steam generator overfill. The results of these two cases are presented in the following figures and discussion.

Figure 1 shows the feedwater flowrates to the affected steam generator. With the ICS controlling level to 100 inches, the MFW flow is throttled, whereas for the ICS failed both MFW and EFW flowrates are uncontrolled and much higher. Figure 2 shows the resulting steam generator level. With level control the level is approximately controlled to 100 inches, although the dynamic behavior of the transient causes instability in feedwater control and level response. Without level control the level continues to fill until going offscale at approximately 250 seconds. Figure 3 shows the steam generator liquid mass. With ICS level control the liquid mass is limited to 20,000-50,000 lbm, whereas the overfilled condition without ICS control reaches 180,000 lbm. The significant difference in the feedwater flow boundary condition for the two cases is evident.

Figure 4 shows the comparison of the steam generator pressures at the top of the tube bundle. There is very little difference in these pressures, which indicates that the secondary saturation temperatures are similar. Figure 5 shows the comparison of the total heat transfer across the steam generator tubes. During the initial phase in which the steam generator is blowing down, the heat transfer rates are the same or similar. Then the case with ICS controlling level is somewhat higher in the period of 50-200 seconds. From 200-400 seconds the case without ICS control is somewhat higher. From Figures 1-5 it can be concluded that the higher feedwater flow without ICS control is offset by the lower heat transfer per lbm of feedwater, since much of the feedwater exits the steam generator in the liquid phase. Most of the feedwater in the case with the ICS controlling to 100 inches is boiled before exiting the steam generator, and therefore a higher heat transfer per lbm occurs. Figure 6 shows the core neutron power response. The case with ICS feedwater control peaks at 13.09% power at 160 seconds, and the case without ICS control peaks at 6.9% power. This occurs before the time period during which the case without ICS control experiences higher heat transfer.

Figures 7-16 show the heat transfer coefficient and the heat transfer rate for [ ] steam generator tube conductors. The [ ] were selected to show the axial variation in the predictions. The elevation and height of each conductor above the lower tubesheet are as follows:

Conductor	Elevation Span (ft)
[	]

Under full power steady-state conditions, the SG tube bundle heat conductors exhibit well defined heat transfer modes. The lower bundle region (Conductors [ ]) contains a relatively large amount of liquid, and is in the nucleate boiling heat transfer mode. The middle bundle region (Conductors [ ]) contains a larger ratio of steam to liquid, and is in the forced convection vaporization heat transfer mode. The upper bundle region (Conductors [ ]) contains steam and is in the forced convection to superheated vapor heat transfer mode. During the blowdown phase of the MSLB, conditions within the SG undergo a violent change. The decompression of the steam generator causes liquid in the lower region of the tube bundle and in the downcomer to be transported to higher elevations within the tube bundle by entrainment in the high velocity steam flow. Due to the severe pressure reduction, flashing occurs in the downcomer and results in reverse flow through the aspirator port. The reverse aspirator port flow contributes to a second boiling region in the upper tube bundle. Upper tube bundle conductors, which are at the hottest temperatures, are periodically quenched with slugs of entrained liquid. For the ICS failure case, the SG gradually fills with liquid after the initial blowdown and experiences less boiling. The accumulation of subcooled liquid as the overfill progresses results in RETRAN predicting the forced convection to subcooled liquid heat transfer mode. For the ICS controlling to 100 inches case, MFW flow is unstable as the ICS attempts to maintain the level setpoint. During periods of high MFW flowrates the liquid levels within the tube bundle increase, causing heat transfer behavior similar to the initial blowdown. During periods of low MFW flowrates, the hotter tube bundle regions dry out. As the temperature differences between the primary and secondary decrease as the analysis progresses, heat transfer coefficients and rates decrease for all conductors for both cases. The integral effect of the heat transfer predicted for each tube bundle conductor is described in the preceeding paragraphs. The overall heat transfer rate, and therefore the consequences of the two MSLB cases are quite similar, with the case with ICS controlling to the level setpoint slightly more limiting.

Q7. To enable the NRC staff to perform audit calculations and sensitivity analyses if needed, please provide electronic copies (on computer disks) of the RETRAN and VIPRE input decks used for MSLB analysis.

A7. The RETRAN and VIPRE inputs decks on diskette have been mailed to the Oconee Project Manager.

References

[ ]

MAIN STEAM LINE BREAK

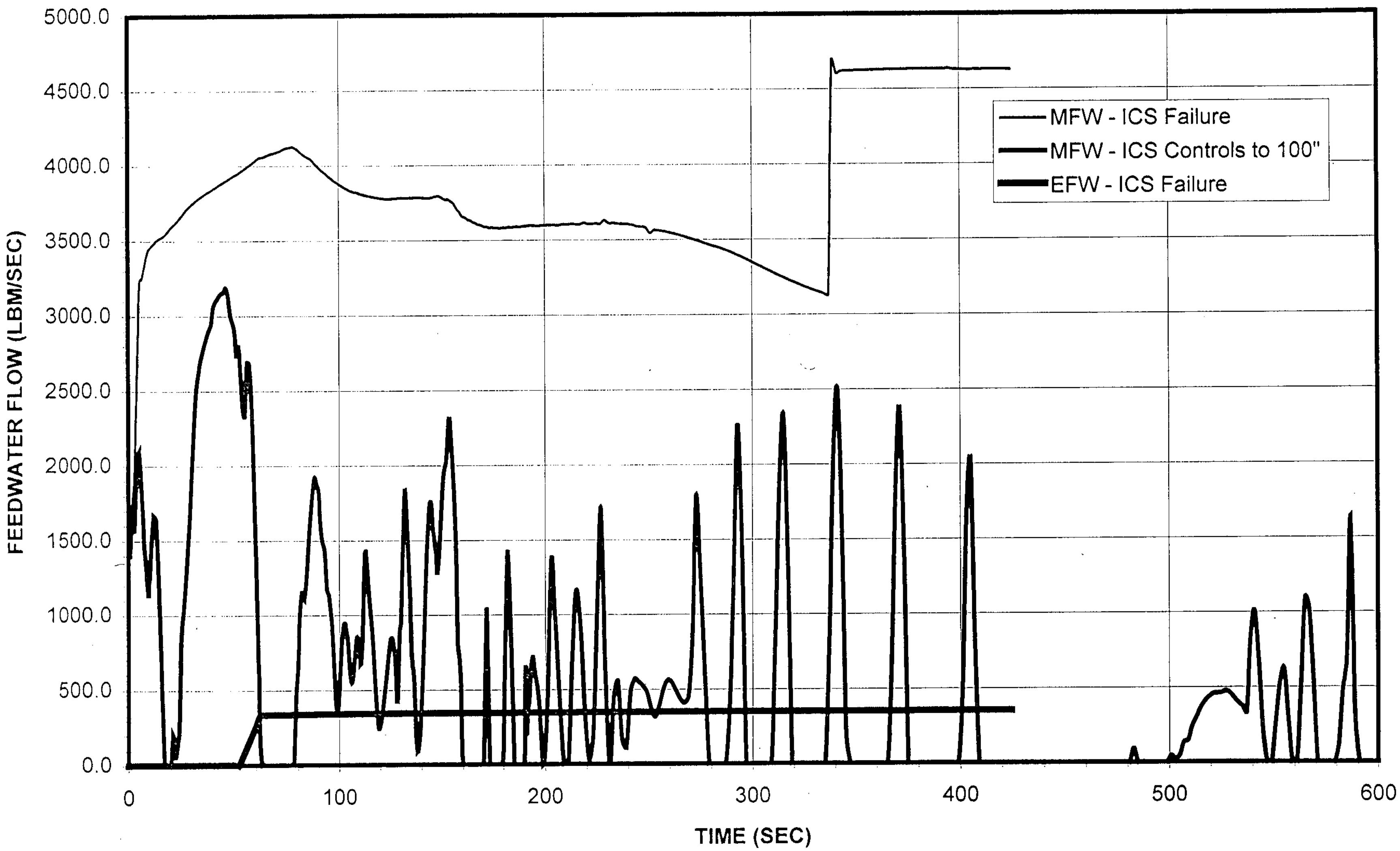

Figure 1

MAIN STEAM LINE BREAK

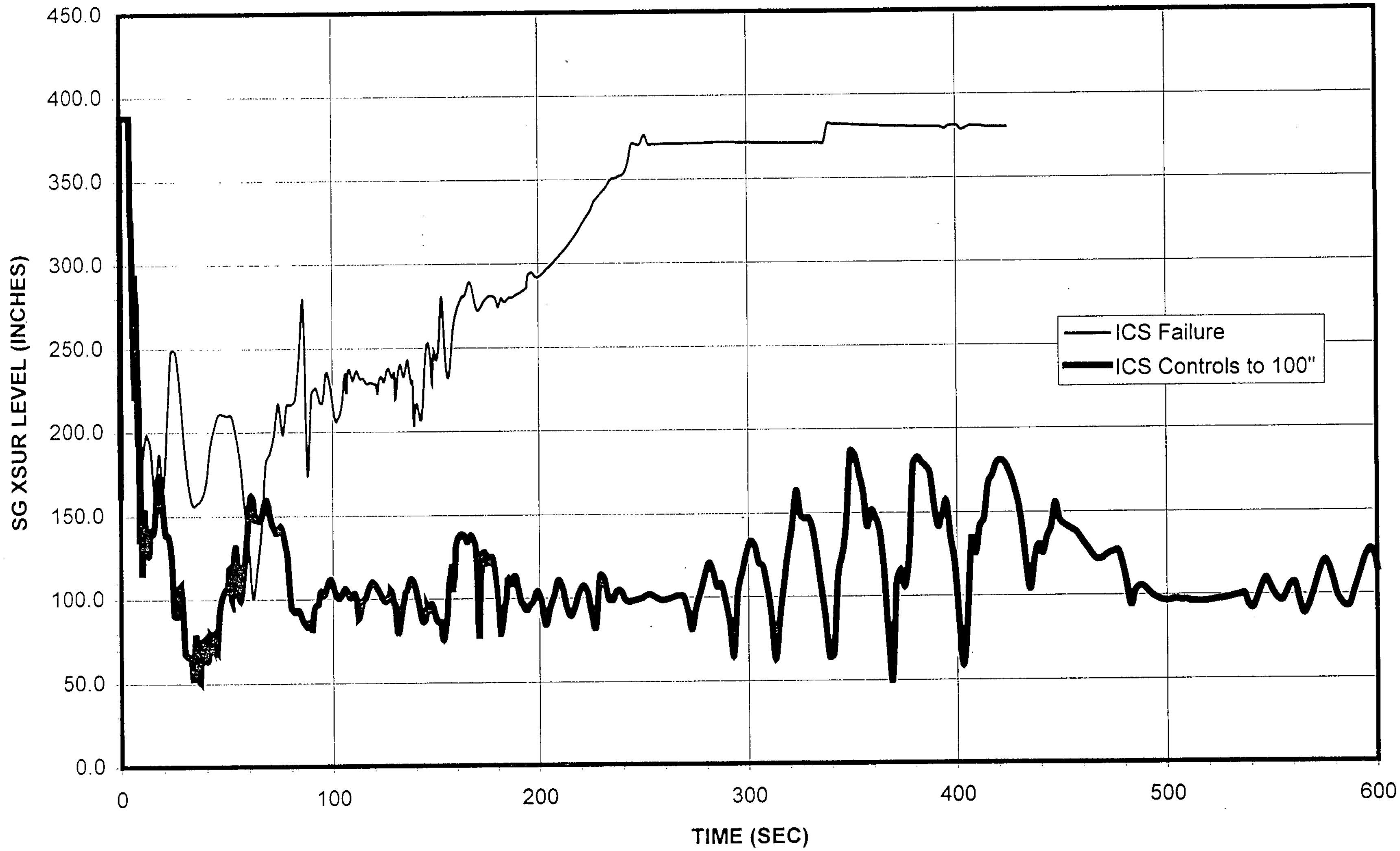

Figure 2

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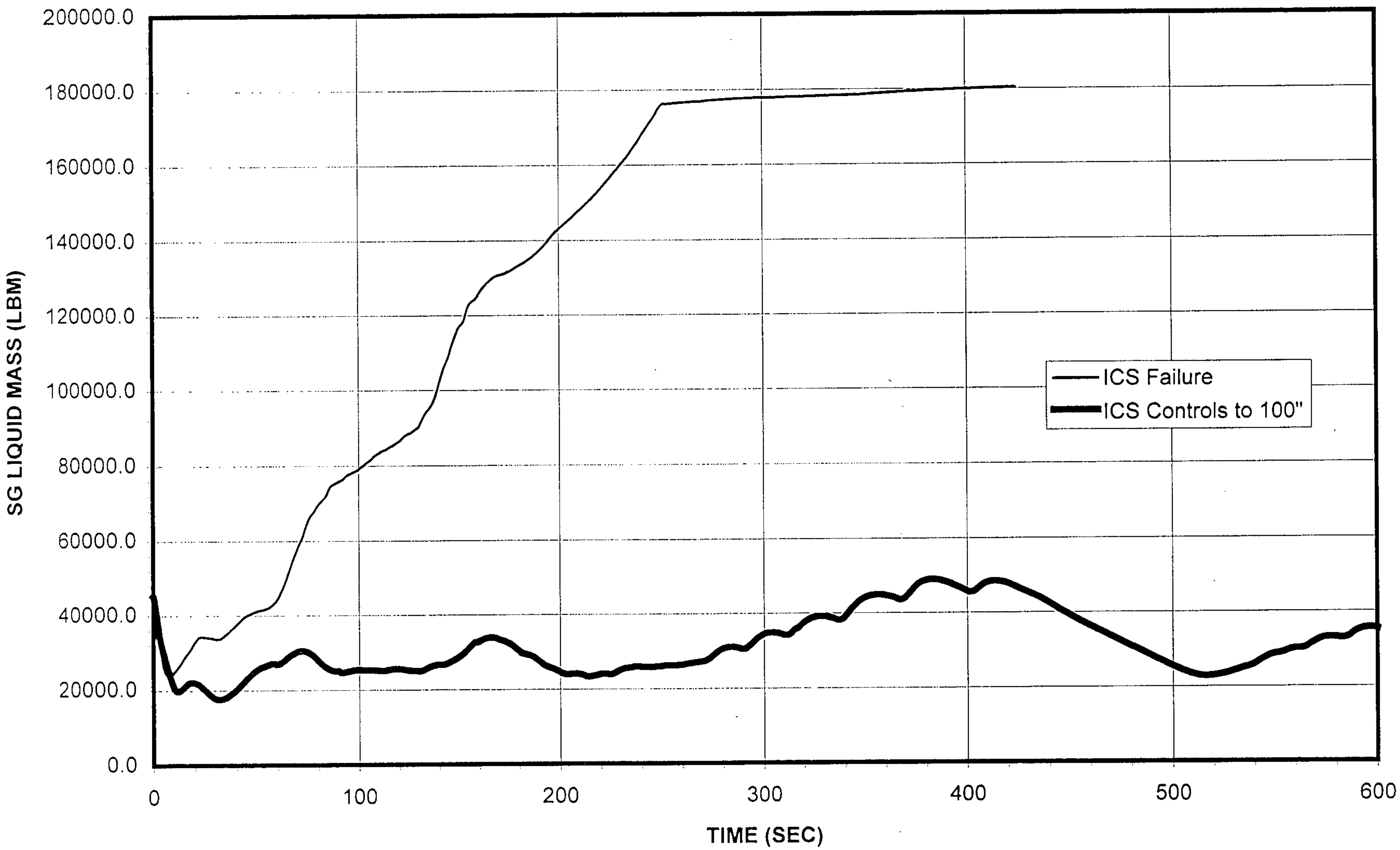

Figure 3

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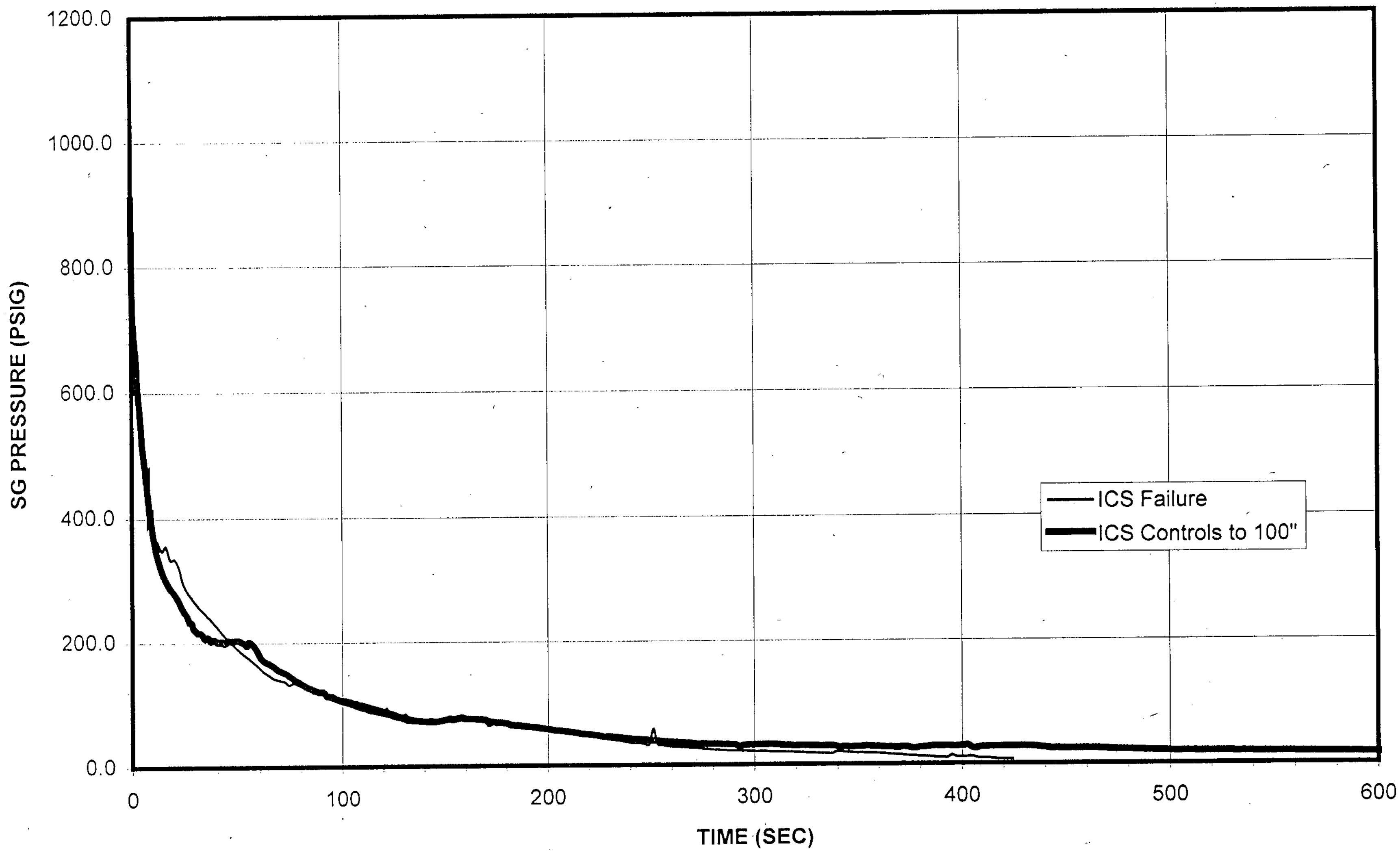

Figure 4

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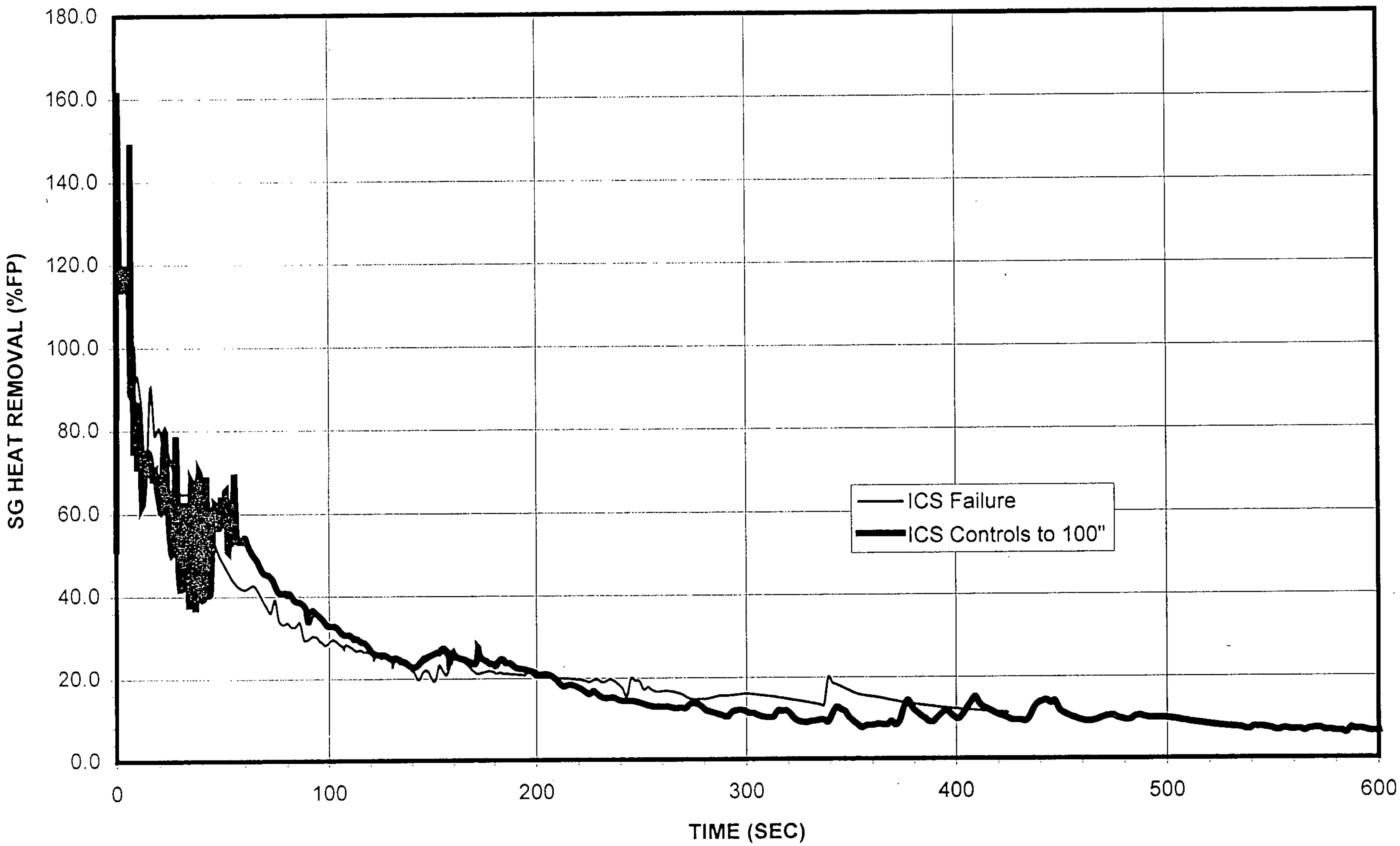

Figure 5

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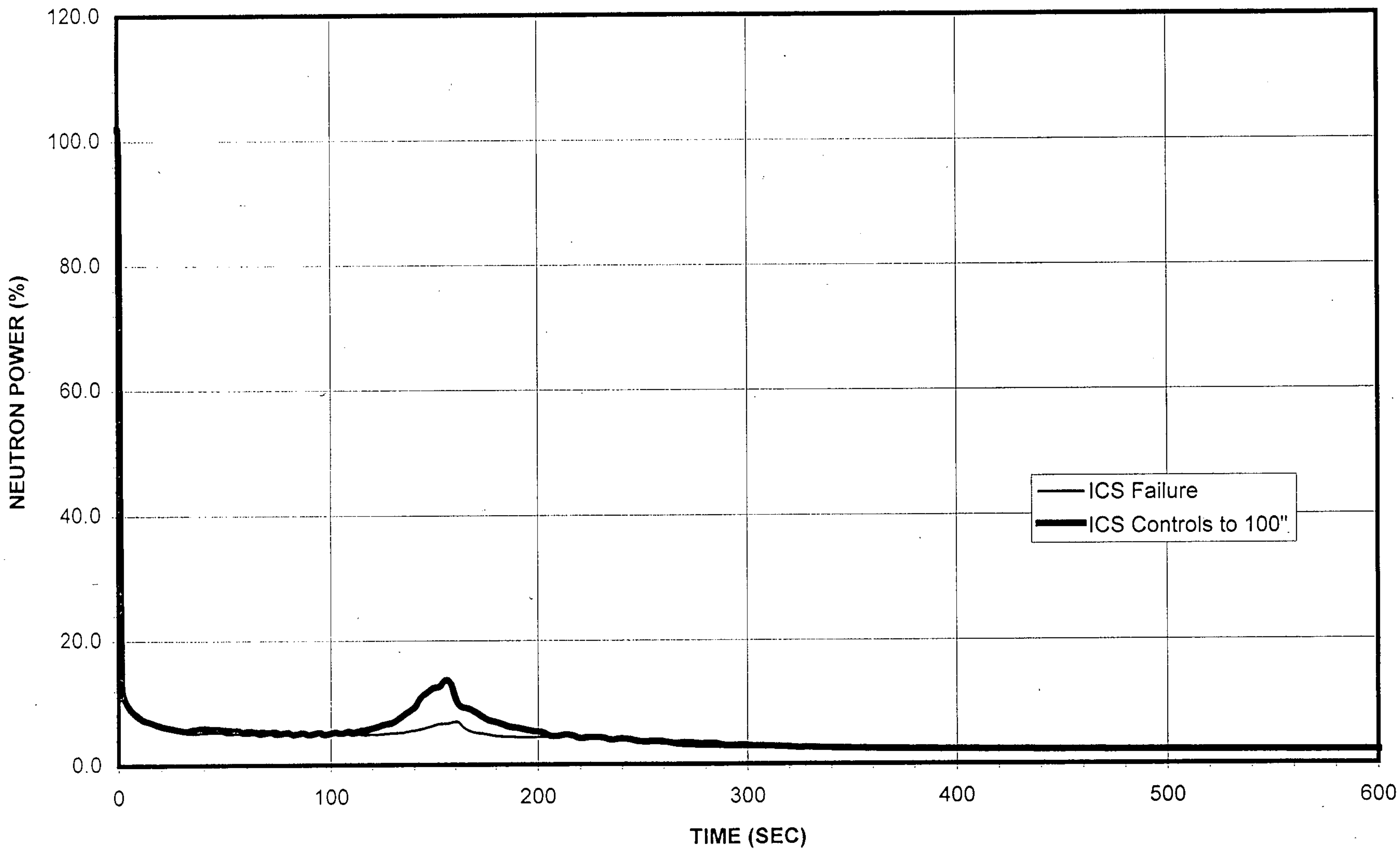

Figure 6

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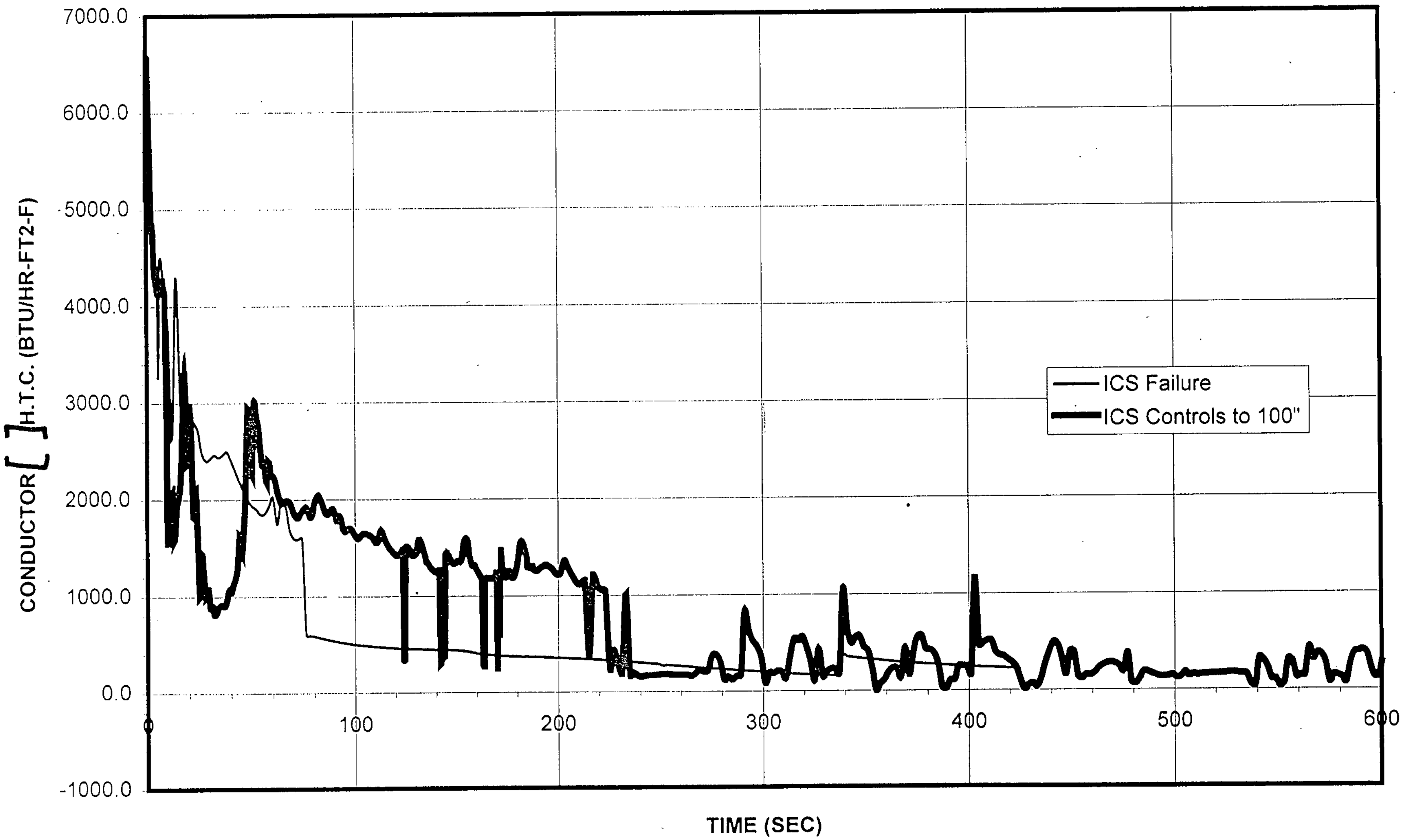

Figure 7

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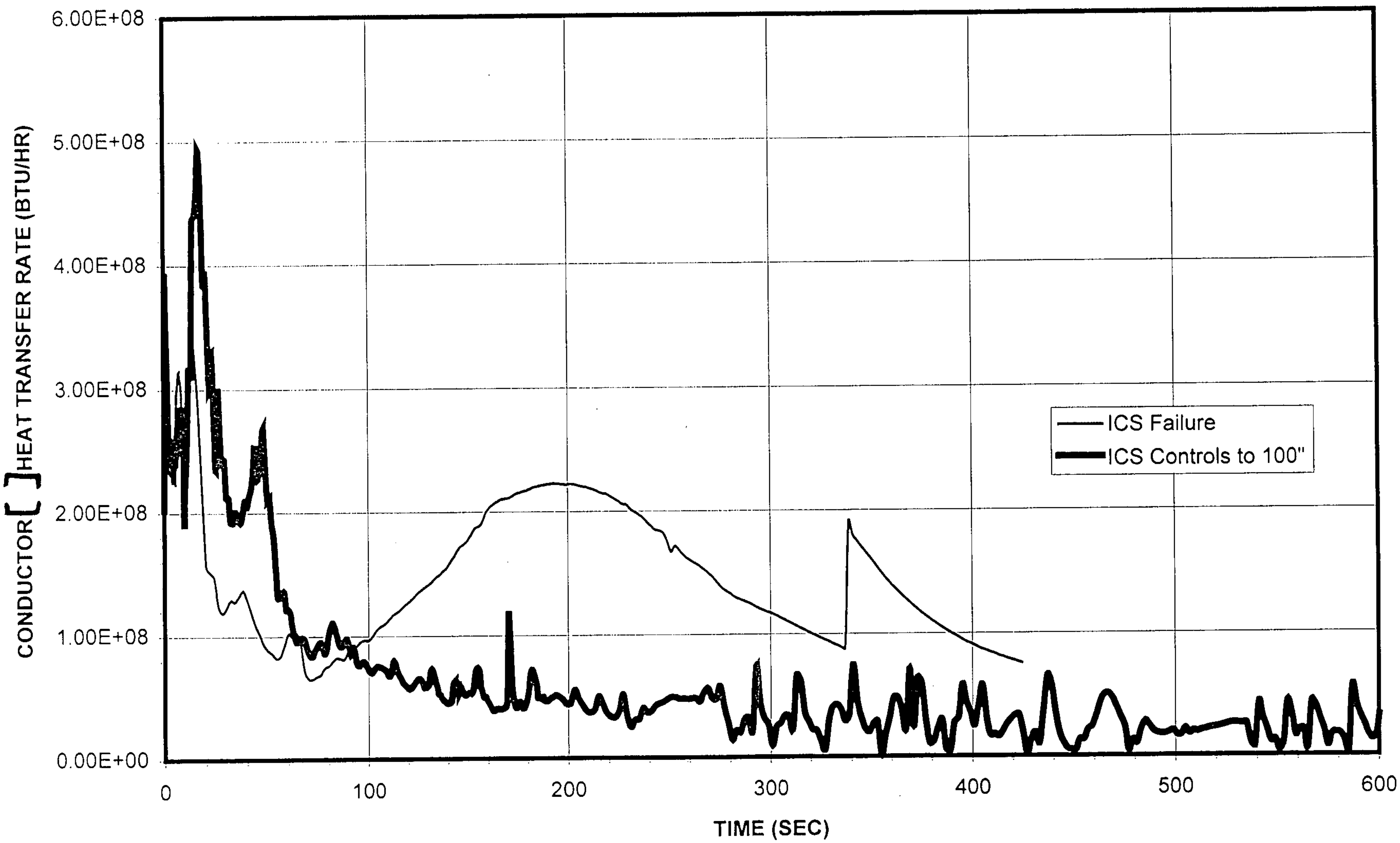

Figure 8

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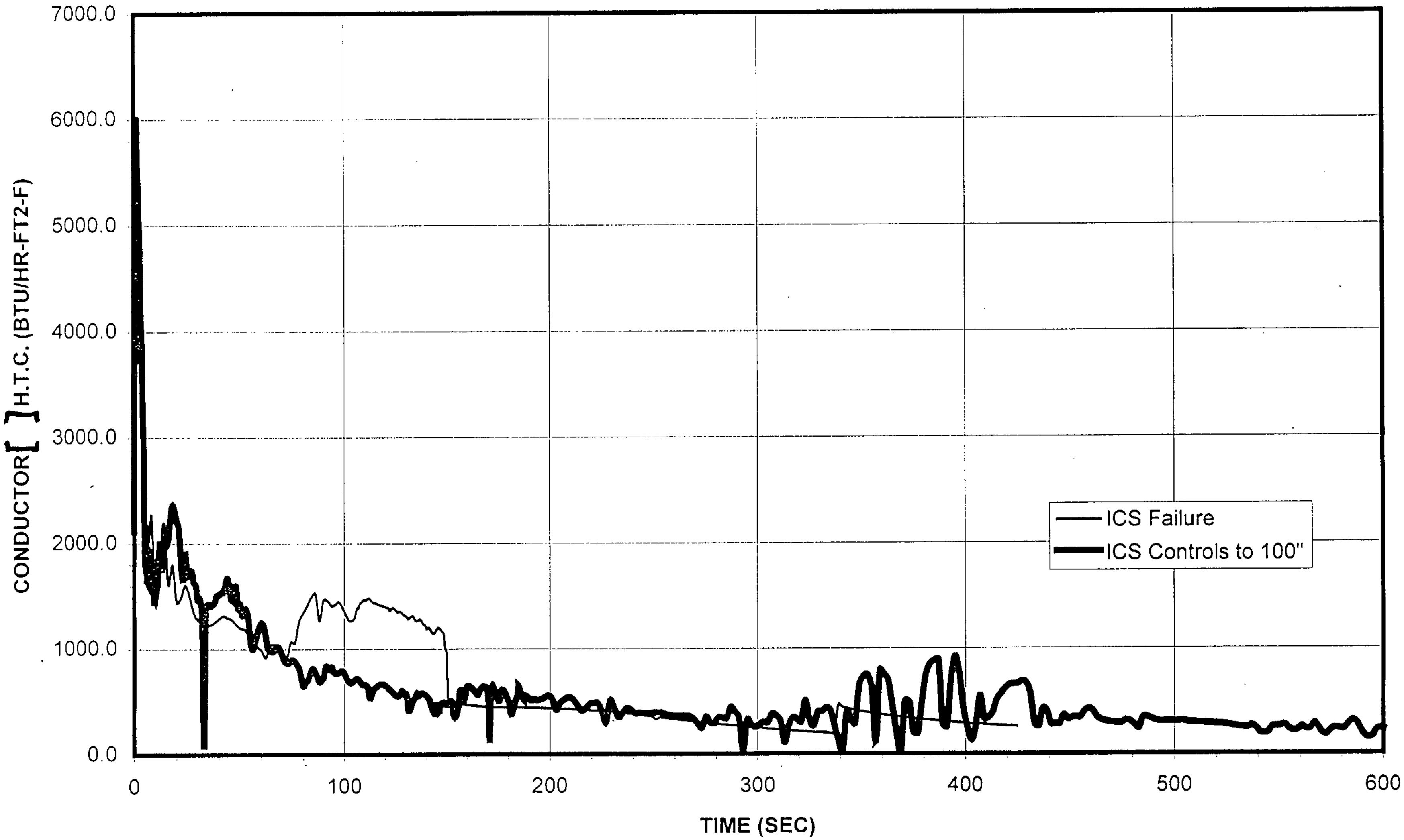

Figure 9

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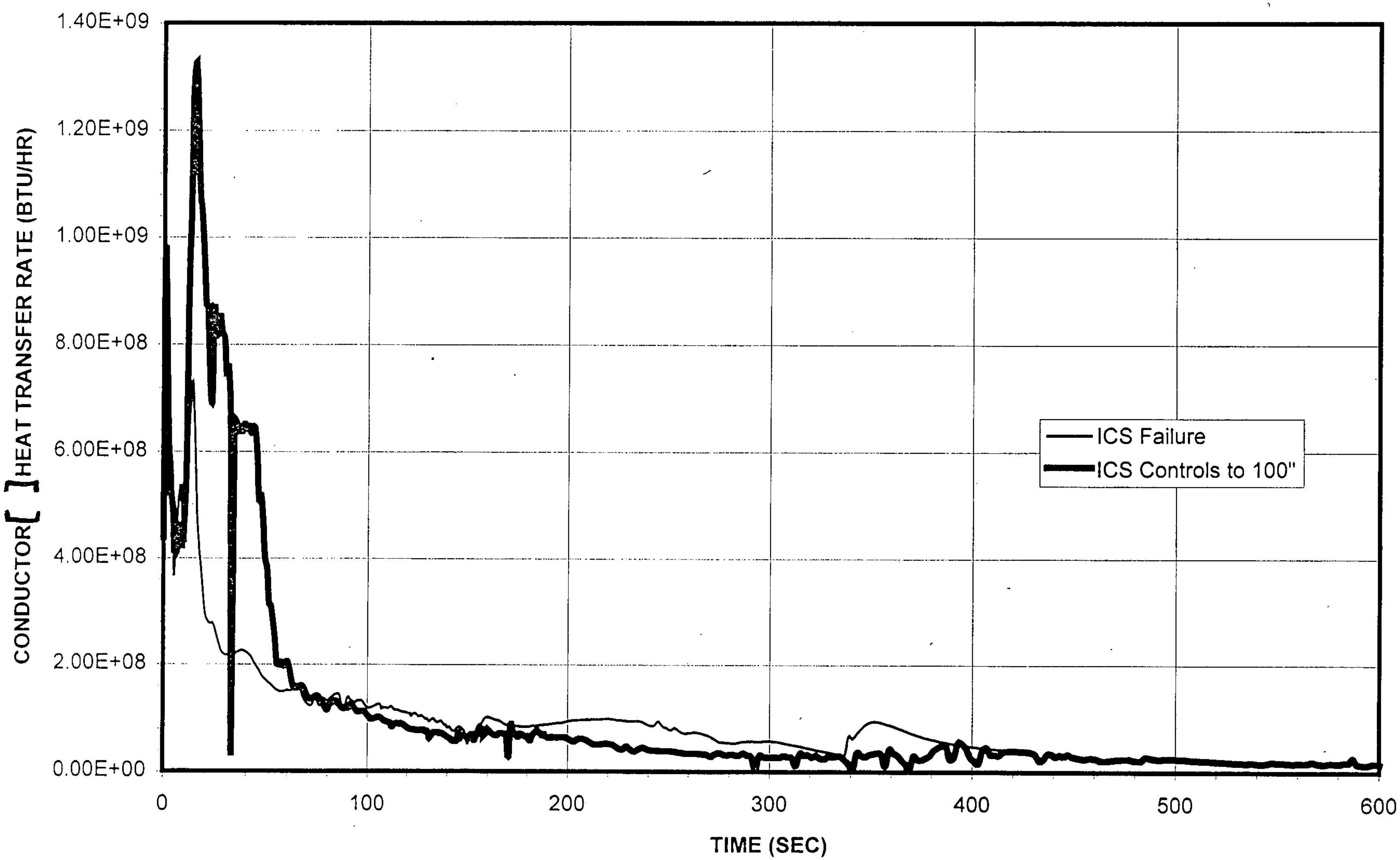

Figure 10

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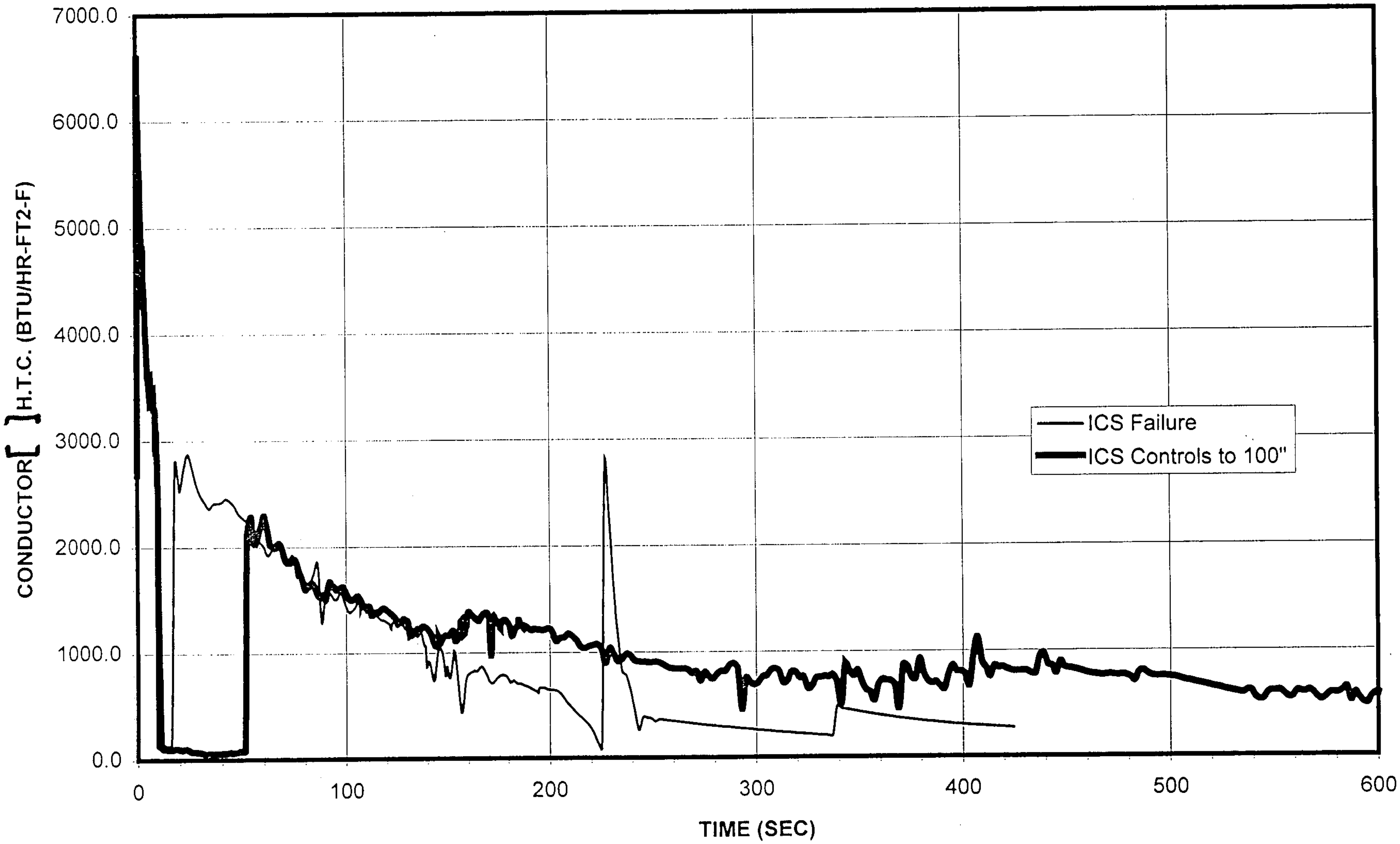

Figure 11

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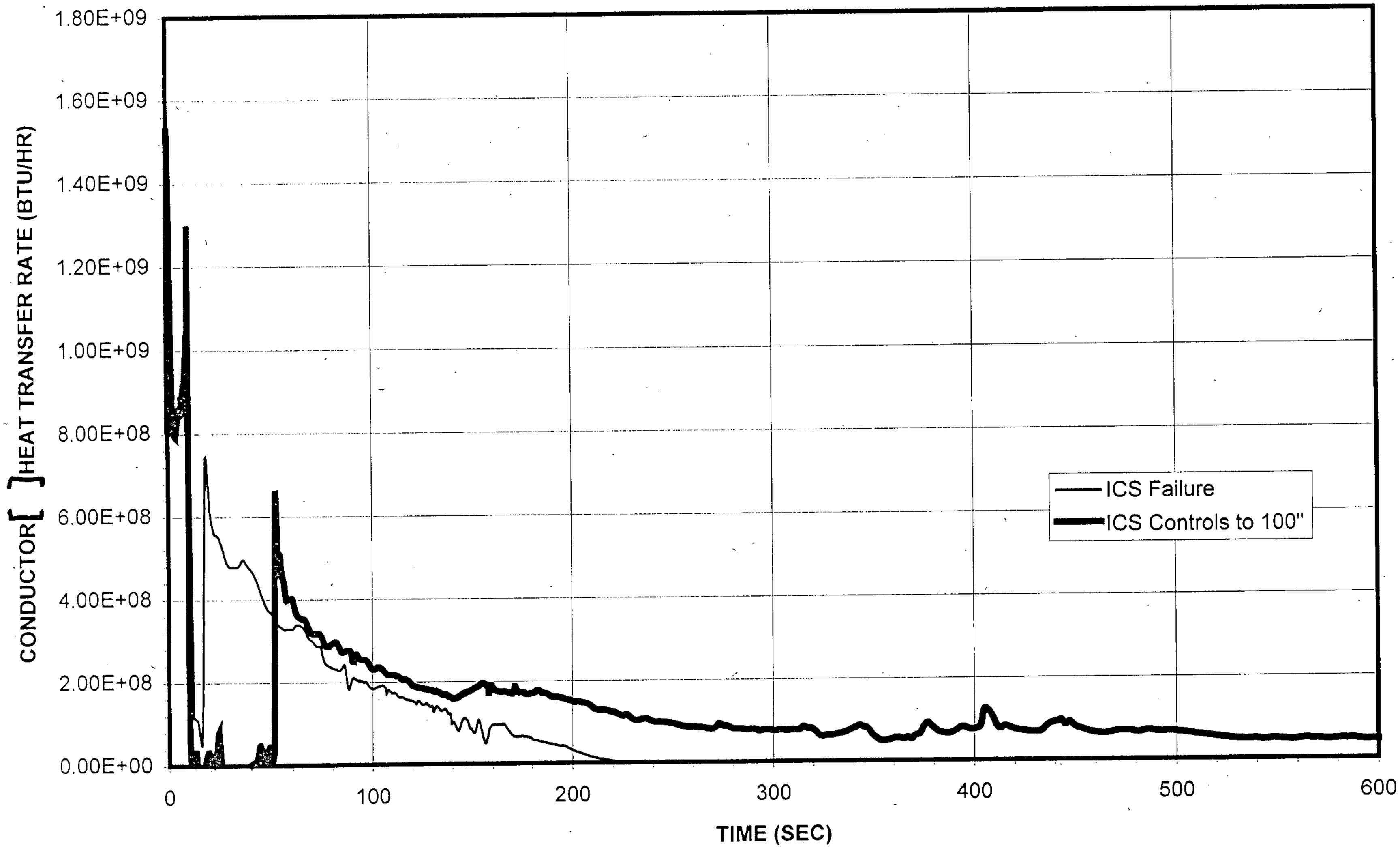

Figure 12

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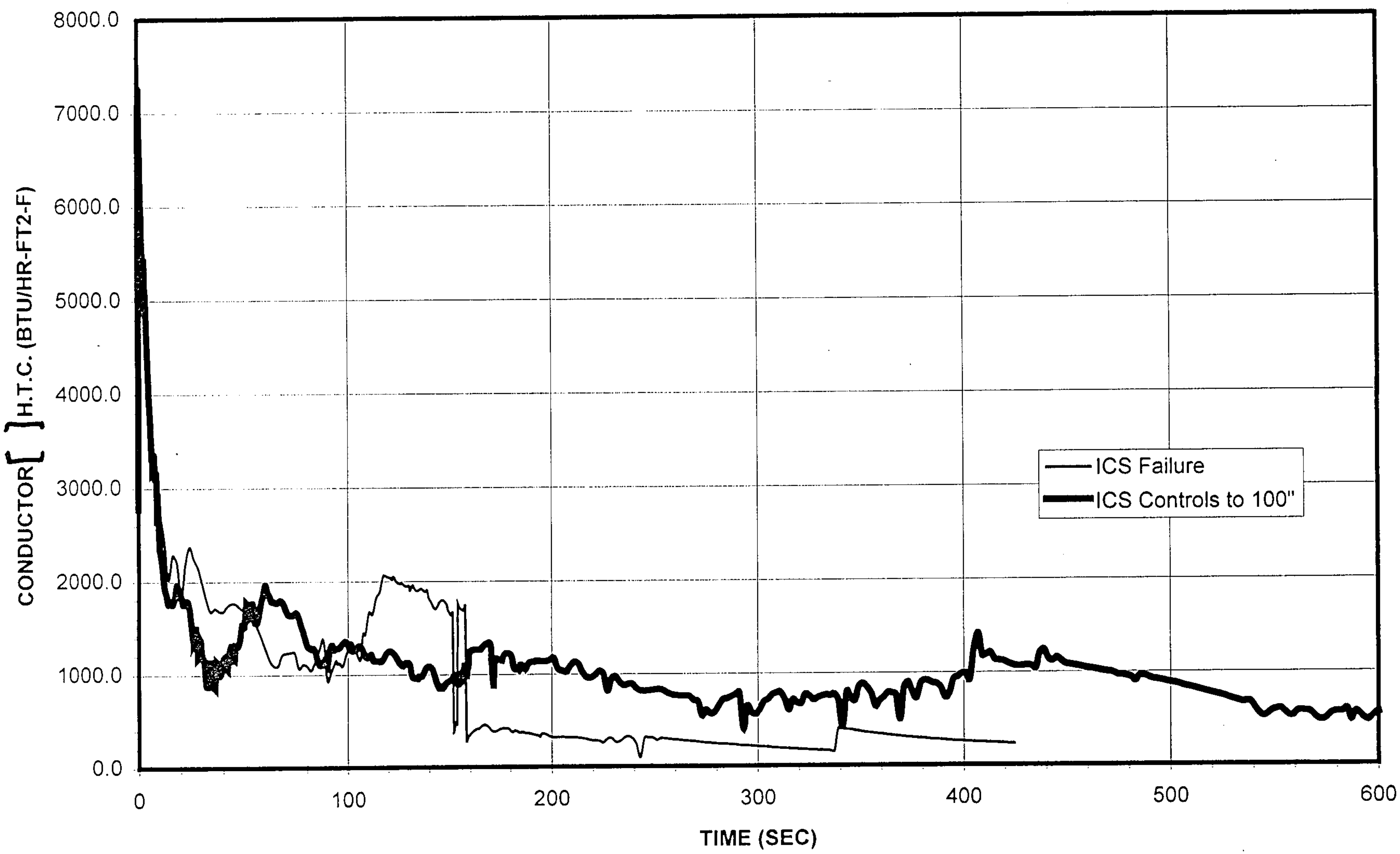

Figure 13

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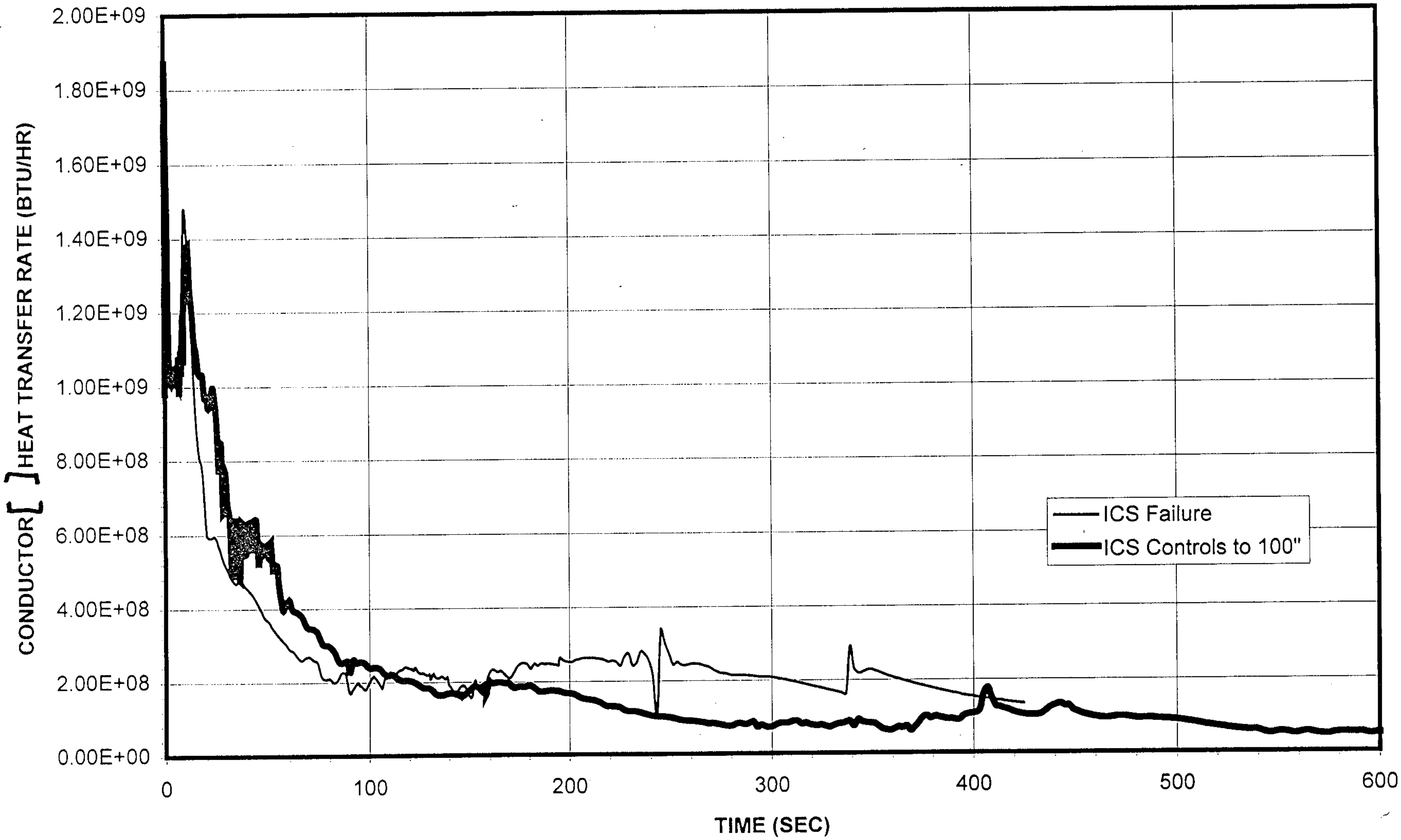

Figure 14

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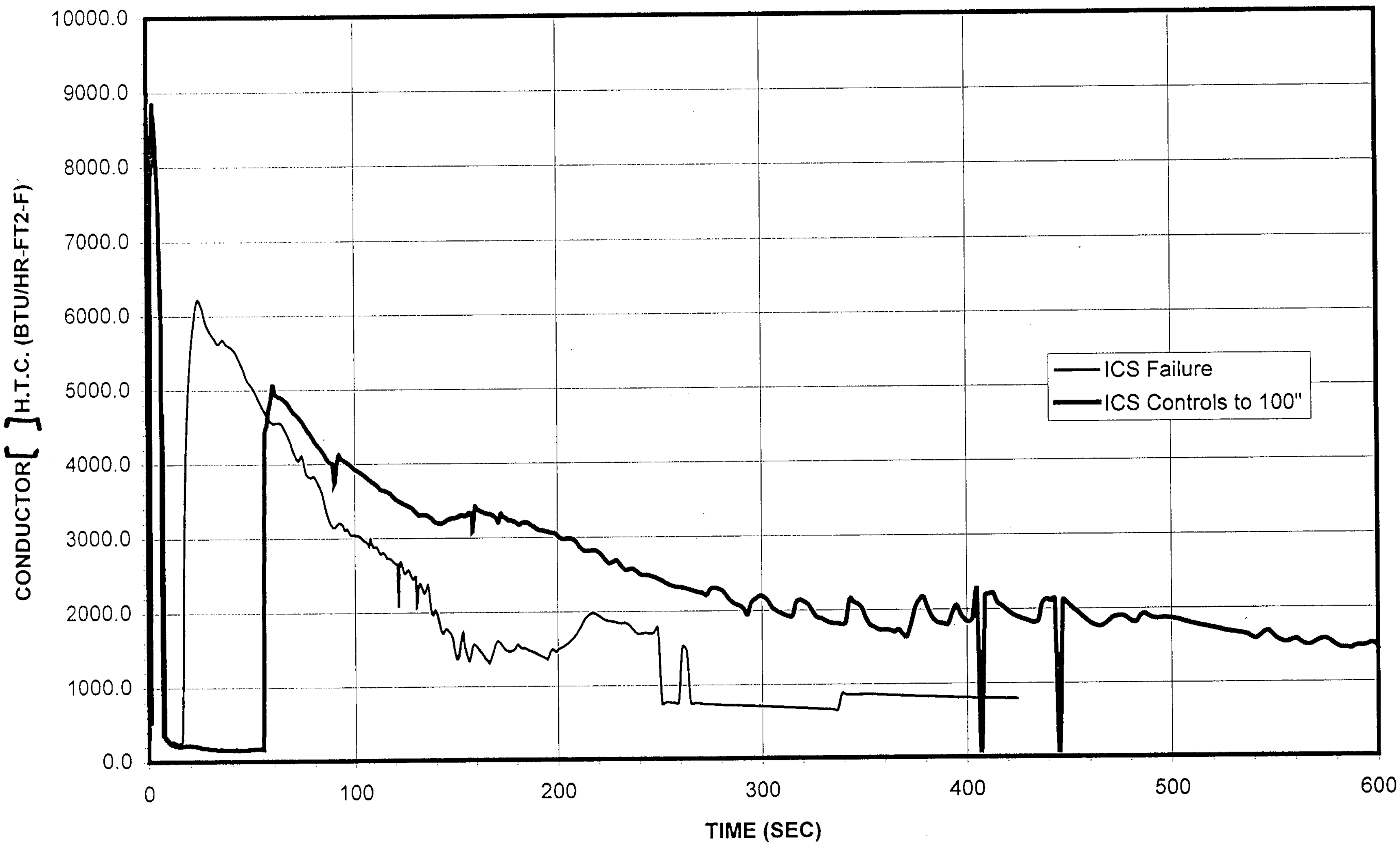

Figure 15

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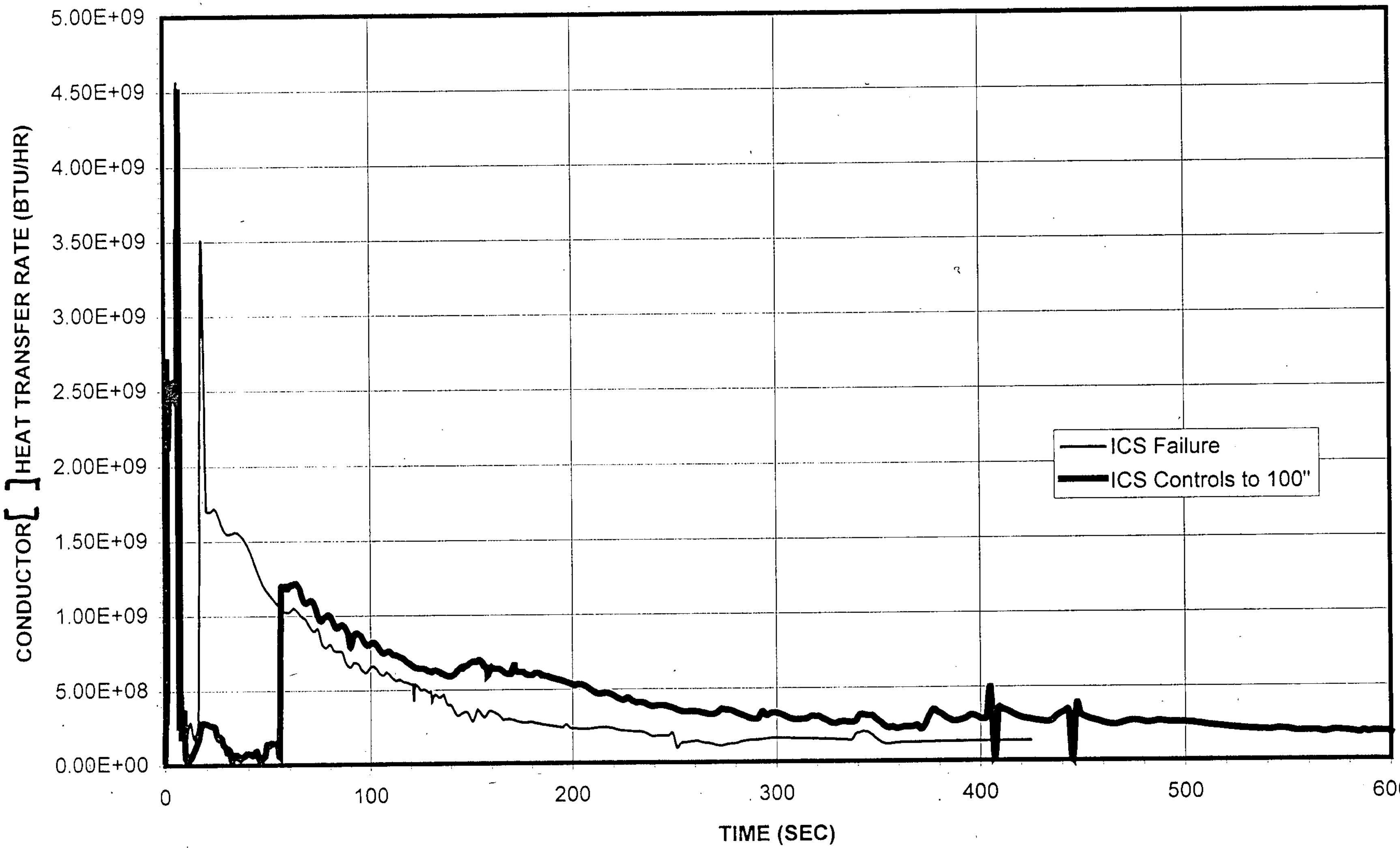

Figure 16